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

February 8, 2018

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

SUBJECT: CLINTON POWER STATION—NRC INTEGRATED INSPECTION REPORT 05000461/2017004; 07201046/2017001 AND EMERGENCY PREPAREDNESS ANNUAL INSPECTION REPORT 05000461/2017501

Dear Mr. Hanson:

On December 31, 2017, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Clinton Power Station. On January 11, 2018, the NRC inspectors discussed the results of this inspection with Mr. B. Kapellas and other members of your staff. The results of this inspection are documented in the enclosed report. The U. S. Nuclear Regulatory Commission also completed its annual inspection of the Emergency Preparedness Program, which began on January 1, 2017, and the issuance of this letter closes Inspection Report 05000461/2017501.

Based on the results of this inspection, the NRC evaluated one self-revealed issue and two NRC-identified issues under the risk significance determination process as having very low safety significance (Green). The inspectors also evaluated an NRC-identified issue under the traditional enforcement process as having very low safety significance (Severity Level IV). The inspectors also determined that three violations are associated with these issues. Because the licensee initiated condition reports to address these issues, these violations are being treated as non-cited violations (NCVs), consistent with Section 2.3.2 of the Enforcement Policy. These NCVs are described in the subject inspection report.

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

If you disagree with a cross-cutting aspect assignment or a finding not associated with a regulatory requirement in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555–0001; with copies to the Regional Administrator, Region III; and the NRC Resident Inspector at the Clinton Power Station.

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

Sincerely,

/RA/

Karla Stoedter, Chief
Branch 1
Division of Reactor Projects

Docket No. 50–461; 72–1046
License No. NPF–62

Enclosure:
IR 05000461/2017004; 07201046/2017001;
05000461/2017501

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Letter to Bryan Hanson from Karla Stoedter dated February 8, 2018

SUBJECT: CLINTON POWER STATION—NRC INTEGRATED INSPECTION REPORT 05000461/2017004; 07201046/2017001 AND EMERGENCY PREPAREDNESS ANNUAL INSPECTION REPORT 05000461/2017501

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50–461; 72–1046
License No:	NPF–62
Report No:	05000461/2017004; 07201046/2017001; 05000461/2017501
Licensee:	Exelon Generation Company, LLC
Facility:	Clinton Power Station
Location:	Clinton, IL
Dates:	October 1 through December 31, 2017
Inspectors:	W. Schaup, Senior Resident Inspector E. Sanchez Santiago, Resident Inspector R. Baker, Senior Operations Engineer N. Fields, Health Physicist G. Hansen, Senior Emergency Preparedness Inspector M. Holmberg, Reactor Inspector D. Reeser, Operations Engineer G. Roach, Senior Resident Inspector (Dresden)
Approved by:	K. Stoedter, Chief Branch 1 Division of Reactor Projects

Enclosure

TABLE OF CONTENTS

SUMMARY

Inspection Report 05000461/2017004, 07201046/2017001; 10/01/2017 – 12/31/2017; 05000461/2017501; 01/01/2017 – 12/31/2017; Clinton Power Station; Follow-up of Events and Notices of Enforcement Discretion, Other Activities.

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Three Green findings were identified by the inspectors. Two of those findings involved non-cited violations of the U.S. Nuclear Regulatory Commission (NRC) requirements. The inspectors also evaluated an NRC-identified issue under the traditional enforcement process as having very low safety significance (Severity Level IV). 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 April 29, 2015. Cross-cutting aspects are determined using IMC 0310, "Aspects within the Cross-Cutting Areas," dated December 4, 2014. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated November 1, 2016. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG–1649, "Reactor Oversight Process," Revision 6.

Cornerstone: Initiating Events

- Green. A self-revealed finding of very low safety significance was identified for the licensee's failure to comply with the requirements of station procedure MA–AA–716–004, "Compression Fittings Inspection, Installation, Remake and Repair," Revision 3. Specifically, the licensee failed to properly assemble a joint that was part of the reactor recirculation (RR) pump motor 'B' oil level monitoring system that subsequently leaked requiring the plant operators to perform an unplanned power reduction to allow for identification and repairs of the leak. The licensee documented this issue in the corrective action program (CAP) as Action Request (AR) 04029024. As corrective actions, the licensee made repairs to the effected joint, inspected the remaining joints to ensure proper integrity, and filled the lower bearing reservoir.

- This issue was more than minor because it was associated with the equipment performance attribute of the Initiating Events cornerstone and impacted the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the leak on the reactor recirculation pump motor oil level monitoring system would have eventually resulted in the failure of the reactor recirculation pump causing a transient and upsetting plant stability. This finding was determined to be of very low safety significance because the event did not cause a reactor trip. The inspectors determined this finding affected the cross-cutting area of human performance in the aspect of work management, where the organization implements a process of planning, controlling, and executing work activities such that nuclear safety is the overriding priority. The work process includes the identification and management of risk commensurate to the work and the need for coordination with different groups or job activities. Specifically, the licensee failed to ensure that all personnel installing the compression fittings received an installer's brief prior to performing work on the reactor recirculation pump motor oil level monitoring system. [H.5] (Section 4OA3.1)

- Green. The inspectors identified a finding of very low safety significance and a non-cited violation of 10 CFR 50.65(a)(4), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," for the licensee's failure to assess and manage the increase in risk that may result from proposed maintenance activities. Specifically, the licensee failed to assess and manage the increase in risk associated with performing control rod scram time testing in Mode 1, prior to performing the activity. As corrective actions, the licensee assessed the increase in risk for performing control rod scram time testing at power and developed a risk mitigation plan that was used to complete the testing.

  This performance deficiency was determined to be more than minor because the finding was associated with the human performance attribute of the Initiating Events cornerstone and impacted the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the failure to assess risk and develop a risk mitigation plan for control rod time testing at power contributed to an automatic reactor scram. Using IMC 0609, Attachment 4, "Initial Characterization of Findings," Appendix K, "Maintenance Risk Assessment and Risk Management Significance Determination Process," issued May 5, 2005, and Appendix M, "Significance Determination Process Using Qualitative Criteria" dated April 12, 2002, the finding was screened against the Initiating Events cornerstone and determined to be of very low safety significance. Specifically, the inspectors and the Region III Senior Reactor Analyst (SRA) determined that Appendix K was not directly applicable to this finding because the licensee performs qualitative evaluations of maintenance activities that have the potential to cause a transient rather than quantitative evaluations. The SRA used insights from Appendix K to support a qualitative SDP evaluation using the principles of Appendix M. The SRA determined that the maintenance activity could only result in an uncomplicated reactor transient event and that the increased risk of a transient compared to the baseline risk of the plant was of very low safety significance. The SRA considered the conditional core damage probability of an uncomplicated transient in this evaluation, which was less than 1E–6, to conclude that the finding was Green. The inspectors determined this finding affected the cross-cutting area of human performance in the aspect of work management, where the organization implements a process of planning, controlling, and executing work activities such that nuclear safety is the overriding priority. The work process includes the identification and management of risk commensurate to the work and the need for coordination with different groups or job activities. While performing the work order risk screening for completing the control rod scram time testing while the reactor was shut down, the screener identified that a new screening would be needed if the testing was performed at power. However, no holds were placed on the work order to ensure the risk screening was completed. [H.5] (Section 4OA3.6)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance and a non-cited violation of 10 CFR 50, Appendix B, Criterion II, "Quality Assurance Program," for the licensee's failure to follow a procedure that implemented the Quality Assurance Program requirements. Specifically, the licensee failed to follow procedure PI–AA–125–1003, "Corrective Action Program Evaluation Manual," and identify the extent of condition for a lack of proficiency in applying the licensing basis

when performing 10 CFR 50.59 evaluations. The licensee documented this issue in their CAP as AR 04075581. The licensee planned an Updated Safety Analysis Report (USAR) Upgrade Project which reportedly would include a review of safety evaluations for USAR changes that dated back to 1986 and determined the scope of this project would be adequate to identify the extent of condition for this issue.

The inspectors determined that this issue was more than minor because if left uncorrected it had the potential to lead to a more significant safety concern. Specifically, because the extent of condition review was not adequate, there is a potential for other safety systems to have been adversely affected by a lack of proficiency in applying the licensing basis during safety related system changes. As a result, safety-related systems may not be able to perform intended safety functions as defined in the USAR. This issue would also adversely affect the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The finding was screened against all cornerstones and determined to be of very low safety significance because the finding met each of the applicable screening questions to be characterized as having very low safety significance. The inspectors determined this finding affected the cross-cutting area of human performance in the aspect of procedure adherence, which stated individuals follow processes, procedures, and work instructions. Specifically, when the NRC violation was documented in the CAP previously it was not appropriately classified in accordance with PI–AA–120 and was also incorrectly closed to an unrelated evaluation. This contributed to the failure to appropriately perform an extent of condition. [H.8] (Section 4OA5.1)

Cornerstone: Miscellaneous

- Severity Level IV. The inspectors identified a Severity Level IV non-cited violation of 10 CFR 72.48(d)(1), "Changes, Tests, and Experiments," for the licensee's failure to perform a written evaluation which provides the bases for the determination that changes do not require a Certificate of Compliance amendment pursuant to 10 CFR 72.48(c)(2). Specifically, the licensee accepted Engineering Change Order ECO–5018–25R0 [1], ECO–5018–48R1 [1], and ECO–5018–48R1 [4] on June 20, 2016, to the time-to-boil calculation as described in the HI-STORM FW Final Safety Analysis Report and incorrectly screened out performing an evaluation of those changes in accordance with 10 CFR 72.48. The licensee documented this issue in its CAP as AR 02714091 and AR 04081583. The licensee is performing a 10 CFR 72.48 evaluation for Engineering Change Order ECO–5018–25R0 [1] and ECO–5018–48R1 [4] while planning to revise the acceptance of ECO–5018–48R1 [1].

  The inspectors determined that the violation was of more than minor significance as the inspectors could not reasonably conclude that the above changes did not require prior NRC approval. The violation screened as a Severity Level IV non-cited violation using example 6.1.d.2 of the NRC Enforcement Policy. No cross-cutting aspect was identified since cross-cutting aspects are not assigned to traditional enforcement violations. (Section 4OA5.2)

REPORT DETAILS

Summary of Plant Status

The unit operated at or near full power during the inspection period with the following exceptions:

- On September 26, 2017, the operators lowered the unit power to approximately 75 percent due to nearing the National Pollutant Discharge Elimination System thermal discharge limits of 90 days per year of greater than 99 degrees Fahrenheit (ºF) at the discharge flume second drop structure. On September 26, 2017, the station reached 89 days of greater than 99 ºF. The unit returned to approximately full power on September 30, 2017.

- On December 9, 2017, operators initiated a manual reactor scram due to the trip of the 4160 Volt 1A1 breaker, 1AP07EJ, the 480 Volt transformer 1A and the A1 breaker. The unit subsequently entered Forced Outage C1F61. The cause of the breaker trip was determined to be a short to ground in transformer 1AP11E. Following replacement of the faulted transformer, the unit commenced startup on December 14, 2017, and synchronized to the grid on December 15, 2017. The unit returned to full power operation on December 17, 2017.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity and Emergency Preparedness

1R01 Adverse Weather Protection (71111.01)

.1 Winter Seasonal Readiness Preparations

a. Inspection Scope

The inspectors conducted a review of the licensee's preparations for winter conditions to verify that the plant's design features and implementation of procedures were sufficient to protect mitigating systems from the effects of adverse weather. Documentation for selected risk-significant systems was reviewed to ensure that these systems would remain functional when challenged by inclement weather. During the inspection, the inspectors focused on plant specific design features and the licensee's procedures used to mitigate or respond to adverse weather conditions. Additionally, the inspectors reviewed the USAR and performance requirements for systems selected for inspection, and verified that operator actions were appropriate as specified by plant specific procedures. Cold weather protection, such as heat tracing and area heaters, was verified to be in operation where applicable. The inspectors also reviewed CAP items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their CAP in accordance with station corrective action procedures. The inspectors' reviews focused specifically on the following plant systems due to their risk significance or susceptibility to cold weather issues:

- shutdown service water system and associated intake structure;
- offsite power transmission systems; and
- fire protection system.

This activity constituted one winter seasonal readiness preparations sample as defined in Inspection Procedure (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:

- Division 1 Shutdown Service Water; and
- High Pressure Core Spray system (HPCS) during maintenance on the Reactor Core Isolation Cooling (RCIC) 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, USAR, Technical Specification (TS) requirements, outstanding work orders (WOs), condition reports, and the impact of ongoing work activities on redundant trains of equipment in order 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 two partial system walkdown samples 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 (FZ) A2 A–2a, RCIC Pump Room—Elevation 707'-6";
- FZ C–2, Containment—Elevations 712'-0", 737'-0", 755'-0", 778'-0", 789'-1", 803'-3", 816'-7", 828'-3";
- FZ CB–3b, Division 4 Nuclear System Protection System Inverter Room—Elevation 781'-0";
- FZ CB–5c, Divisions 1 and 2 Cable Risers—Elevation 781'-0"; and
- FZ D–1, Division 3 Diesel Generator (DG) Fuel Tank Room—Elevations 712'-0", 719'-0".

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, 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 USAR, engineering calculations, and abnormal operating procedures to identify licensee commitments. The specific 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 corrective action 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 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:

- turbine building.

Documents reviewed during this inspection 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.

1R11 Licensed Operator Requalification Program (71111.11)

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

a. Inspection Scope

On December 13, 2017, the inspectors observed a crew of licensed operators in the plant's simulator during just-in-time training to support reactor plant start up. The inspectors verified that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and that 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;
- control board manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions.

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements.

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 December 13 and 14, 2017, the inspectors observed reactor plant start up. 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 (if applicable);
- correct use and implementation of procedures;
- control board (or equipment) manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions.

The performance in these areas was compared to pre-established operator action expectations, procedural compliance and task completion requirements.

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

b. Findings

No findings were identified.

.3 Annual Operating Test Results (71111.11A)

a. Inspection Scope

The inspectors reviewed the overall pass/fail results of the Annual Operating Tests and the Biennial Written Examinations required by Title 10 of the *Code of Federal Regulations* (CFR), Part 55.59(a), administered by the licensee from October 25, 2017, through December 8, 2017, as required by 10 CFR 55.59(a). The results were compared to the thresholds established in IMC 0609, Appendix I, "Licensed Operator Requalification Significance Determination Process (SDP)," to assess the overall adequacy of the licensee's Licensed Operator Requalification Training (LORT) Program to meet the requirements of 10 CFR 55.59. (02.02)

This inspection constituted one annual licensed operator requalification examination results sample as defined in IP 71111.11–05.

b. Findings

No findings were identified.

.4 Biennial Review (71111.11B)

a. Inspection Scope

The following inspection activities were conducted during the weeks of November 6 and November 13, 2017, to assess: (1) the effectiveness and adequacy of the facility licensee's implementation and maintenance of its systems approach to training (SAT) based LORT Program, put into effect to satisfy the requirements of 10 CFR 55.59; (2) conformance with the requirements of 10 CFR 55.46 for use of a plant referenced simulator to conduct operator licensing examinations and for satisfying experience requirements; and (3) conformance with the operator license conditions specified in 10 CFR 55.53. The documents reviewed are listed in the Attachment to this report.

- Licensee Requalification Examinations (10 CFR 55.59(c); SAT Element 4 as Defined in 10 CFR 55.4): The inspectors reviewed the licensee's program for development and administration of the LORT biennial written examination and annual operating tests to assess the licensee's ability to develop and administer examinations that are acceptable for meeting the requirements of 10 CFR 55.59(a).

  - The inspectors conducted a detailed review of one biennial requalification written examination versions to assess content, level of difficulty, and quality of the written examination materials. (02.03)
  - The inspectors conducted a detailed review of ten job performance measures and four simulator scenarios to assess content, level of difficulty, and quality of the operating test materials. (02.04)
  - The inspectors observed the administration of the annual operating test to assess the licensee's effectiveness in conducting the examination, including the conduct of pre-examination briefings, evaluations of individual operator and crew performance, and post-examination analysis. The inspectors evaluated the performance of one crew in parallel with the facility evaluators during two dynamic simulator scenarios, and evaluated various licensed crew members concurrently with facility evaluators during the administration of several job performance measures. (02.05)
  - The inspectors assessed the adequacy and effectiveness of the remedial training conducted since the last requalification examinations and the training planned for the current examination cycle to ensure that they addressed weaknesses in licensed operator or crew performance identified during training and plant operations. The inspectors reviewed remedial training procedures and individual remedial training plans. (02.07)

- Conformance with Examination Security Requirements (10 CFR 55.49): The inspectors conducted an assessment of the licensee's processes related to examination physical security and integrity (e.g., predictability and bias) to verify compliance with 10 CFR 55.49, "Integrity of Examinations and Tests." The inspectors observed the implementation of physical security controls (e.g., access restrictions and simulator I/O controls) and integrity measures (e.g., security agreements, sampling criteria, bank use, and test item repetition) throughout the inspection period. (02.06)

- Conformance with Operator License Conditions (10 CFR 55.53): The inspectors reviewed the facility licensee's program for maintaining active operator licenses and to assess compliance with 10 CFR 55.53(e) and (f). The inspectors reviewed the procedural guidance and the process for tracking on-shift hours for licensed operators, and which control room positions were granted watch-standing credit for maintaining active operator licenses. Additionally, medical records for seven licensed operators were reviewed for compliance with 10 CFR 55.53(l). (02.08)

- Conformance with Simulator Requirements Specified in 10 CFR 55.46: The inspectors assessed the adequacy of the licensee's simulation facility (simulator) for use in operator licensing examinations and for satisfying experience requirements. The inspectors reviewed a sample of simulator performance test records (e.g., transient tests, malfunction tests, scenario based tests, post-event tests, steady state tests, and core performance tests), simulator discrepancies, and the process for ensuring continued assurance of simulator fidelity in accordance with 10 CFR 55.46. The inspectors reviewed and evaluated the discrepancy corrective action process to ensure that simulator fidelity was being maintained. Open simulator discrepancies were reviewed for importance relative to the impact on 10 CFR 55.45 and 55.59 operator actions as well as on nuclear and thermal hydraulic operating characteristics. (02.09)

- Problem-Identification and Resolution (10 CFR 55.59(c); SAT Element 5 as Defined in 10 CFR 55.4): The inspectors assessed the licensee's ability to identify, evaluate, and resolve problems associated with licensed operator performance (a measure of the effectiveness of its LORT Program and their ability to implement appropriate corrective actions to maintain its LORT Program up to date). The inspectors reviewed documents related to licensed operator performance issues (e.g., licensee condition/problem identification reports including documentation of plant events and review of industry operating experience from previous 2 years). The inspectors also sampled the licensee's quality assurance oversight activities, including licensee training department self-assessment reports. (02.10)

This inspection constituted one biennial Licensed Operator Requalification Program inspection sample as defined in IP 71111.11–05.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations (71111.12Q)

a. Inspection Scope

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

- Division 1 Emergency Diesel Generator; and

- Reactor Recirculation System.

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 structures, systems, and components (SSCs)/functions classified as (a)(2), or appropriate and adequate goals and corrective actions for systems classified as (a)(1).

The inspector performed a quality review for the Division 1 Emergency Diesel Generator (EDG), as discussed in IP 71111.12, Section 02.02.

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.

This activity constituted one quarterly maintenance effectiveness sample and one quality control sample as defined in IP 71111.12–05.

b. Findings

No findings were identified.

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 due to planned maintenance on the Division 2 EDG;
- yellow risk due to planned maintenance on the Division 2 Shutdown service water system;
- yellow risk due to planned maintenance on the HPCS system; and
- yellow risk due to planned maintenance on the ‘A’ Residual heat removal system.

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 during this inspection are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

1R15 Operability Determinations and Functional Assessments (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- Action Request (AR) 04064255: 1SX027A Maximum Friction is High;
- AR 04055528: Diesel FP [Fire Pump] ‘A’ Oil Sample Results/EC [Engineering Change] 621388;
- AR 04054680: Standby Liquid Control Concentration Needs Attention; and
- AR 04037464: Valve Failed As Found Set Pressure Test.

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 USAR 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 corrective action 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 four 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 modification:

- EC 622359; Replacement of Dry Type Transformer 1AP11E2.

The inspectors reviewed the configuration changes and associated 10 CFR 50.59 safety evaluation screening against the design basis, the USAR, and the TS, as applicable, to verify that the modification did not affect the operability or availability of the affected system(s). 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 activity constituted one permanent plant modification samples as defined in IP 71111.18–05.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

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

- testing of the RCIC system;
- testing of the Division 1 EDG; and
- testing of the Division 1 EDG 16 cylinder cooling water heat exchanger.

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 USAR, 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 corrective action 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 activity constituted three post-maintenance testing samples as defined in IP 71111.19–05.

b. Findings

No findings were identified.

1R20 Outage Activities (71111.20)

.1 Forced Outage C1F61

a. Inspection Scope

The inspectors evaluated outage activities for a forced outage that began on December 9, 2017, and continued through December 15, 2017, when a loss of Division 1 power required operators to manually scram the reactor. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule.

The inspectors observed or reviewed the reactor shutdown and cooldown, outage equipment configuration and risk management, electrical lineups, control and monitoring of decay heat removal, control of containment activities, startup and heat up activities, and identification and resolution of the loss of power to Division 1.

This inspection constituted one other outage sample as defined in IP 71111.20–05.

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:

- CPS 9054.02: RCIC Valve Operability (Routine); and
- CPS 9015.01: Standby Liquid Control System Operability (Routine).

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 USAR, 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.

These inspections constituted two routine surveillance testing sample, as defined in IP 71111.22, Sections–02 and–05. In addition, the inspectors did not identify any performance degradation in the reactor coolant system (RCS) leakage for the entire

cycle. The RCS leak detection inspection sample was not performed as defined in IP 71111.22, Section–02.

b. Findings

No findings were identified.

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The regional inspectors performed an in-office review of the latest revisions to the Emergency Plan, Emergency Action Levels (EALs), and EAL Bases document to determine if these changes decreased the effectiveness of the Emergency Plan. The inspectors also performed a review of the licensee's 10 CFR 50.54(q) change process, and Emergency Plan change documentation to ensure proper implementation for maintaining Emergency Plan integrity.

The NRC review was not documented in a safety evaluation report, and did not constitute approval of licensee-generated changes; therefore, this revision is subject to future inspection. The specific documents reviewed during this inspection are listed in the Attachment to this report.

This EAL and Emergency Plan Change inspection constituted one sample as defined in Inspection Procedure 71114.04.

b. Findings

No findings were identified.

1EP6 Drill Evaluation (71114.06)

.1 Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of a routine licensee emergency drill on October 17, 2017, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the technical support center to determine whether the event classification, notifications, and protective action recommendations were performed in accordance with procedures. The inspectors also attended the licensee drill critique to compare any inspector-observed weakness with those identified by the licensee staff in order to evaluate the critique and to verify whether the licensee staff was properly identifying weaknesses and entering them into the CAP. As part of the inspection, the inspectors reviewed the drill package and other documents listed in the Attachment to this report.

This emergency preparedness drill inspection constituted one sample as defined in IP 71114.06–05.

b. Findings

No findings were identified.

.2 Training Observation

a. Inspection Scope

The inspector observed a simulator training evolution for licensed operators on October 18, 2017, which required emergency plan implementation by a licensee operations crew. This evolution was planned to be evaluated and included in performance indicator data regarding drill and exercise performance. The inspectors observed event classification and notification activities performed by the crew. The inspectors also attended the post-evolution critique for the scenario. The focus of the inspectors' activities was to note any weaknesses and deficiencies in the crew's performance and ensure that the licensee evaluators noted the same issues and entered them into the corrective action program. As part of the inspection, the inspectors reviewed the scenario package and other documents listed in the Attachment to this report.

This activity constituted one licensee's training evolution with emergency preparedness drill aspects as defined in IP 71114.06–05.

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 the period from the fourth quarter of 2016 through the third quarter of 2017. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the Nuclear Energy Institute Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, 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 the period of October 1, 2016, through September 30, 2017, 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 activity constituted one RCS leakage sample 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 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, adequate attention was being given to timely corrective actions, and adverse trends were identified and addressed. Some minor issues were entered into the licensee's corrective action program as a result of the inspectors' observations; however, they are not discussed in 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.

b. Findings

No findings were identified.

.2 Semi-Annual 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 4OA2.1 above, licensee trending efforts, and licensee human performance results. The inspectors' review nominally considered the 6-month period of July 2017 through December 2017, although some examples expanded beyond those dates where the scope of the trend warranted.

The review also included issues documented outside the 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 activity constituted one semi-annual trend review inspection sample as defined in IP 71152.

b. Observations and Assessments

Over the previous 6 months, the inspectors and the licensee identified a negative trend regarding fire protection procedure compliance and awareness. Notable focus areas for improvement included notifications to the control room, transient combustible control, fire door checks, fire system impairment control and housekeeping. The licensee planned to address this through a performance improvement program in accordance with station procedure ER–AA–610–1002, "Fire Protection Performance Indicators." The inspectors planned to monitor the effectiveness of the performance improvement program during routine plant walkdowns and quarterly fire inspections.

c. Findings

No findings were identified.

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153)

.1 Unplanned Reactor Down Power to Repair Reactor Recirculation Pump 'B' Motor Lower Bearing Oil Leak

a. Inspection Scope

The inspectors reviewed the licensee's response to an unplanned down power on July 8, 2017, to approximately 5 percent reactor power to investigate and determine the cause of a low oil level alarm indication for the 'B' RR pump motor lower bearing oil reservoir. The inspectors reviewed the operator actions associated with the event, station procedures, the causal analysis and the corrective actions taken by the licensee associated with this issue.

This event follow-up review constituted one sample as defined in IP 71153–05.

b. Findings

Reactor Down Power due to Reactor Recirculation Pump Motor Lower Bearing Oil Leak

Introduction: A self-revealed finding of very low safety significance was identified for the licensee's failure to comply with the requirements of station procedure MA-AA-716-004, "Compression Fittings Inspection, Installation, Remake and Repair," Revision 3. Specifically, the licensee failed to properly assemble a joint that was part of the RR pump motor 'B' oil level monitoring system that subsequently leaked requiring plant operators to perform an unplanned power reduction to 5 percent reactor power to allow for identification and repairs of the leak.

Description: On July 7, 2017, the licensee determined that there was a lowering oil level trend on the 'B' RR pump motor lower bearing oil reservoir that significantly exceeded the normal oil consumption rate and trend as compared to the 'A' RR pump. On the following day, the oil low level alarm indication was received in the control room confirming the condition. The licensee documented this issue in the CAP as AR 04029024. The licensee performed an unplanned down power to 5 percent reactor power to investigate and determine the cause of the low oil level.

After entry into the drywell, the licensee discovered a leak on a compression type fitting (Swagelok) that was part of the oil level and monitoring instrumentation of the 'B' RR pump motor installed during the previous refueling outage in May 2017. During the inspections, other connections were observed to have a film of oil but no confirmed leakage. During repairs, the connector was found cross-threaded between the fitting nut and the connector. In addition, the tubing was found loose and was not compressed on the connector ferrule as required to seal the joint.

The licensee made repairs to the effected joint, inspected the remaining joints to ensure proper integrity, and filled the lower bearing reservoir. The licensee inspected the 'A' RR pump motor to ensure a similar condition did not exist. Due to the as-found conditions, the licensee isolated the level monitoring systems on both RR pump motors to prevent future oil loss from the tubing to minimize operational challenges to unit operation. The reactor was then returned to full power.

The licensee performed a root cause investigation. The root cause investigation determined that the contractor personnel who installed the pump motor oil level monitoring system failed to comply with guidance provided in station procedure MA–AA–716–060, "Compression Fittings Inspection, Installation, Remake and Repair," regarding the elimination of connection side loading during installation. As a result, one of the connections was cross-threaded which resulted in the subsequent oil leak and the need to reduce reactor power to repair the leak. Additionally, the original work schedule had work being performed by one crew on days. This crew was provided a job specific installer's briefing for the assembly of Swagelok compression fittings on May 4, 2017. The briefing included direction on the proper techniques for installing compression fittings and for each participant, a hands-on confirmation of their ability to properly install a connection fitting. At a later date, the licensee revised the work schedule to allow work to be performed on both shifts (days and nights). Upon reviewing the worker shift assignments, installer's brief attendance sheets, and pre-job briefing records the licensee discovered that none of the contract pipe fitters on nights had attended the installer's briefing and the cross threaded joint had been made by one of those pipe fitters.

Analysis: The inspectors determined that the licensee's failure to comply with the requirements of station procedure MA–AA–716–004 was a performance deficiency. Specifically, the licensee failed to properly assemble a joint that was part of the RR pump motor 'B' oil level monitoring system that subsequently leaked requiring the plant operators to perform an unplanned power reduction to 5 percent reactor power to allow identification and repairs of the leak. The performance deficiency was more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated September 7, 2012, because the finding was associated with the equipment performance attribute of the Initiating Events cornerstone and impacted the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the leak on the reactor recirculation pump motor oil level monitoring system presented an operational challenge to unit operation and resulted in a plant shut down. Using IMC 0609, Attachment 4, "Initial Characterization of Findings," and Appendix A, "The Significance Determination Process for Findings At-Power," the finding was screened against the Initiating Events cornerstone and determined to be of very low safety significance (Green) because the event did not cause a reactor trip.

The inspectors determined this finding affected the cross-cutting area of human performance in the aspect of work management, where the organization implements a process of planning, controlling, and executing work activities such that nuclear safety is the overriding priority. The work process included the identification and management of risk commensurate to the work and the need for coordination with different groups or job activities. Specifically, the licensee failed to ensure that all contract pipefitters installing the compression fittings received an installer's brief that included specific procedure use and hands on confirmation of their ability to properly install a compression fitting prior to items being installed in the plant. [H.5]

Enforcement: The inspectors did not identify a violation of a regulatory requirement associated with this finding because this portion of the reactor recirculation system is not safety related component, and the procedure the licensee failed to follow was a self-imposed standard not subject to any regulatory requirements. The licensee has entered this issue into their CAP as AR 04029024. Because this finding does not involve a violation and is of very low safety significance, it is identified as a finding. **(FIN 05000451/2017004–01: Reactor Down Power due to Reactor Recirculation Pump Motor Lower Bearing Oil Leak)**

.2 (Closed) Licensee Event Report 05000461/2016–009–00: Trip of Fuel Building Fans Due to Damper Failure Results in Loss of Secondary Containment

a. Inspection Scope

On June 24, 2016, Clinton Power Station was operating at 99 percent power when the main control room received two fuel building trouble alarms. An unexpected damper closure resulted in secondary containment vacuum degrading, eventually exceeding the TS limit of 0.25 inch vacuum water gauge. Within 1 minute, secondary containment vacuum was restored within TS. The licensee determined that the 'A' exhaust fan isolation damper had failed to the closed position on loss of air due to failure of the associated air supply solenoid. The failure of the solenoid valve was due to a deformed and worn core which resulted in sticking. The failed solenoid valve was replaced and the fuel building ventilation system was restored. In addition, parts quality testing and preventative maintenance programs were established to replace fuel building ventilation solenoid valves on a periodic basis. The inspectors reviewed the apparent cause evaluation, equipment procurement information, and maintenance strategies associated with the failed component. The inspectors also reviewed the response procedures and operator actions associated with this event. The inspectors did not identify a performance deficiency associated with this issue. Documents reviewed are listed in the Attachment to this report.

This licensee event report (LER) is closed.

This event follow-up review constituted one sample as defined in IP 71153–05.

b. Findings

No findings were identified.

.3 (Closed) Licensee Event Report 05000461/2016–010–00 and 01: Failure to Complete Technical Specification Required Actions Within the Completion Time Resulted in a Condition Prohibited by Technical Specifications

a. Inspection Scope

On October 14, 2016, Clinton Power Station determined that a condition prohibited by TS 3.7.6, "Main Turbine Bypass System," had occurred associated with a failure to complete a TS-required action within the associated completion time. Specifically, on June 6, 2016, the TS limiting condition for operations surveillance requirement 3.0.2, grace period of 1.25 times the normal required frequency of 31 days for completing testing of the main steam bypass valves per surveillance requirement 3.7.6.1 was exceeded. Turbine bypass valves 4, 5, and 6 were not tested within the required frequency. The licensee declared the valves inoperable as required by TS surveillance requirement 3.0.1, and the applicable required actions were completed. The turbine bypass valves were tested on September 28, 2016, and the system was declared operable.

An non-cited violation (NCV) of the TS was documented by the inspectors in NRC Integrated Inspection Report 05000461/2016003 (ML16315A053). This LER is closed.

This event follow-up review constituted two samples as defined in IP 71153–05.

b. Findings

No additional findings were identified.

.4 (Closed) Licensee Event Report 05000461/2017–002–00 and 01: Failure of the Division 1 Diesel Generator Ventilation Fan Load Sequence Relay Circuit During Concurrent Maintenance of RHR Division 2 Results in an Unanalyzed Condition

a. Inspection Scope

On March 7, 2017, an area operator detected a clicking sound coming from safety related unit substation 1A. The sound was emanating from relay 427X2–41A (X2 relay) every 10 seconds. The relay is an Agastat time delay relay that provides the signal to reset the load shed and resequencing circuit for the Division 1 EDG room vent fan. It was determined that the vent fan would not respond to a demand signal nor could the fan be started locally. The licensee declared the EDG inoperable on March 9, 2017.

The inspectors reviewed the licensee's root cause report, maintenance and operating procedures, design requirements, reportability requirements, and corrective actions. As a result of this review a violation of the Unit 1 TS and 10 CFR 50, Appendix B, Criterion III, "Design Control" was documented by the inspectors in NRC Inspection

Report 05000461/2017010 (ML17331B161), “Clinton Power Station – Final Significance Determination of a White Finding with Assessment Followup and Notice of Violation.”

This LER is closed.

This event follow-up review constituted two samples as defined in IP 71153–05.

b. Findings

No additional findings were identified.

.5 (Closed) Licensee Event Report 05000461/2017–004–01: Main Steam Isolation Valve Local Leak Rate Test Limit Exceeded During Refueling Outage

a. Inspection Scope

During the refueling outage (C1R17) on May 12, 2017, the licensee tested its main steam isolation valves (MSIVs) and discovered the as-found leakage for main steam line ‘D’ exceeded the TS 3.6.1.3, “Primary Containment Isolation Valves,” surveillance requirements. An investigation determined the as-found condition of the MSIVs did not reveal any damage, only what the licensee classified as normal wear indications. The licensee reported this event due to principal plant safety barriers being seriously degraded under the provision of 10 CFR 50.73(a)(2)(ii)(A). The inspectors reviewed Revision 0 of this LER and identified a violation as documented in Section 1R15 of NRC Integrated Inspection Report 05000461/2017003 (ML17313A039).

The licensee submitted Revision 1 to the LER on November 22, 2017, to state the event was also reportable as a condition prohibited by TS under the provision of 10 CFR 50.73(a)(2)(i)(B). The inspectors identified the licensee’s failure to report this issue as a condition prohibited by TS and dispositioned it as a minor violation as described in Section 4OA3.6 of NRC Integrated Inspection Report 05000461/2017003. The inspectors reviewed the updated LER and did not identify any additional issues of concern.

This LER is closed.

This event follow-up review constituted one sample as defined in IP 71153–05.

b. Findings

No additional findings were identified.

.6 (Closed) Licensee Event Report 05000461/2017–005–00 and 01: Automatic Reactor Scram During the Performance of Scram Time Testing as a Result of an Invalid Oscillating Power Range Monitor Growth Rate Trip

a. Inspection Scope

On May 30, 2017, while at 28 percent reactor power, an automatic scram occurred during the performance of control rod scram time testing. The automatic scram signal was initiated from the oscillation power range monitor growth rate algorithm. The control room operators ensured the plant was stable and completed shut down in accordance with station procedures. The inspectors responded to the site to assess the plant

conditions post-SCRAM as well as the operators' response to the event. The inspectors also reviewed the licensee operating logs, operating and maintenance procedures, design requirements, corrective action documents, and root cause report. As a result of this review, the inspectors identified a violation of NRC requirements.

This LER is closed.

This event follow-up review constituted two samples as defined in IP 71153–05.

b. Findings

Failure to Assess and Manage Risk Associated with the Performance of Control Rod Time Testing

Introduction: The inspectors identified a finding of very low safety significance and an associated NCV of 10 CFR 50.65(a)(4), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," for the licensee's failure to assess and manage the increase in risk that may result from the proposed maintenance activities. Specifically, the licensee failed to assess and manage the increase in risk when performing control rod scram time testing in Mode 1, prior to performing the activity. As corrective actions, the licensee assessed the increase in risk associated with performing control rod scram time testing at power and developed a risk mitigation plan that was used to complete the testing.

Description: On May 30, 2017, the control room operations staff was conducting control rod scram time testing in accordance with procedure CPS 9813.10, "Control Rod Scram Time Testing," when an automatic reactor scram was initiated from the oscillation power range monitor growth rate algorithm. The operating crew placed the mode switch in the shutdown position, entered the appropriate off-normal and emergency procedures, and placed the plant in a safe and stable shutdown condition. The issue was documented in the CAP as AR 04016563.

Investigation into the automatic scram determined that the event did not occur due to actual core thermal hydraulic instability and that core thermal limits had not been challenged by the event. The licensee's casual evaluation determined that the cause of the event was that the oscillating power range monitor's growth rate algorithm (GRA) trip function design was unable to distinguish between plant response to system perturbations and the onset of thermal-hydraulic instabilities. The root cause stated that in the seconds leading to the scram, when control rod 12–17 was being continuously withdrawn, a pressure perturbation associated with the turbine was occurring simultaneously. The effect on reactor pressure, and therefore reactor power, from the turbine pressure perturbation coincident with local power changes associated with the continuous withdrawal of the control rod resulted in sequential pressure and power peaks occurring while the pressure regulator compensated for the changes. The licensee also found that similar operating experience was available from the LaSalle and Hope Creek plants regarding the performance of control rod scram time testing at power. Specifically, these plants found that the oscillating power range monitor is unable to distinguish between the local power change from a control rod withdrawal and a power change due to an oscillation. Lastly, the root cause evaluation stated that the procedural guidance had waiting periods for control rod drive mechanism temperature limitations but not for power stabilization. As a result, it was a common practice to continuously withdraw the scrammed rod and immediately restore it to the fully withdrawn position.

The licensee's interim corrective actions included raising the oscillation power range monitor trip set points and implementing an operating strategy when performing scram time testing in the oscillation power range monitor trip enabled region.

After the inspectors reviewed the licensee's root cause investigation report, the inspectors asked the licensee how they had assessed and managed the increased risk associated with the scram time testing at power since the activity was normally performed with the reactor in a shutdown condition. The licensee provided the inspectors a copy of the Attachment 8, "Risk screening/mitigation plan," that was performed as part of procedure WC–AA–104, "Integrated Risk Management." The inspectors noted that the risk screening/mitigation plan provided by the licensee assumed the control rod scram time testing would be performed in Mode 4 with the reactor shut down. In addition, the risk screening/mitigation plan documented that if the control rod scram time testing was performed with the reactor at power operating in Mode 1 or 2, then a reactivity management plan hold would be required per procedure WC–AA–104 to allow for the development of a risk mitigation plan for performing the testing at power. The inspectors asked for the risk mitigation plan for performing the testing at power and the licensee determined that one had not been performed. During discussions with the licensee about why the mitigation plan had not been performed, it was determined that a WO task was not created to direct the work group supervisor to develop the risk mitigation plan. This was documented in AR 04073063.

Title 10 of the CFR, Section 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," Paragraph (a)(4) states, in part, that before performing maintenance activities, the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. When the inspectors reviewed the licensee's process for screening operational risk per station procedure WC–AA–101, "On-line Work Control Process," it was noted that if maintenance activities cause a significant increase in the likelihood of an initiating event, the operations staff is required to determine if the activity would be a high risk evolution. If it is a high risk evolution, the licensee reflects this qualitatively through the use of the Paragon online risk model high risk evolution trigger that toggles online risk to yellow. The inspectors determined if this evaluation had been performed prior to performing scram time testing at power, online risk would have been yellow. In addition, the licensee would have been required to put risk mitigation actions in place to manage the increase in risk that resulted from the test.

Analysis: The inspectors determined that the failure to assess and manage the increase in risk that may result from a proposed maintenance activity prior to performing the activity, as required by 10 CFR 50.65 (a)(4), was a performance deficiency. The performance deficiency was determined to be more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated September 7, 2012, because the finding was associated with the human performance attribute of the Initiating Events cornerstone and impacted the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the failure to assess risk and develop a risk mitigation plan for performing control rod scram time testing at power contributed to an automatic reactor scram.

Using IMC 0609, Attachment 4, "Initial Characterization of Findings," Appendix K, "Maintenance Risk Assessment and Risk Management Significance Determination

Process," issued May 5, 2005, and Appendix M, "Significance Determination Process Using Qualitative Criteria" dated April 12, 2002, the finding was screened against the Initiating Events cornerstone and determined to be of very low safety significance (Green). Specifically, the inspectors and the Region III Senior Reactor Analyst (SRA) determined that Appendix K was not directly applicable to this finding because the licensee performed qualitative evaluations of maintenance activities that have the potential to cause a transient rather than quantitative evaluations. The SRA used insights from Appendix K to support a qualitative SDP evaluation using the principles of Appendix M. The SRA determined that the maintenance activity could only result in an uncomplicated reactor transient event, and that the increased risk of a transient compared to the baseline risk of the plant was of very low safety significance. The SRA considered the conditional core damage probability of an uncomplicated transient in this evaluation, which was less than 1E–6, and concluded that the finding was Green.

The inspectors determined this finding affected the cross-cutting area of human performance in the aspect of work management, where the organization implements a process of planning, controlling, and executing work activities such that nuclear safety is the overriding priority. The work process includes the identification and management of risk commensurate to the work and the need for coordination with different groups or job activities. Specifically, while performing a risk screening for the work order to perform the control rod scram time testing while the reactor was shut down, the screener identified that the testing activity would be required to be screened again if the activity was performed at power. However, a reactivity management plan hold was not placed on the work order in accordance with station processes resulting is a failure to assess the risk and develop a risk mitigation plan for performing the activity at power. [H.5]

Enforcement: Title 10 of the CFR, Paragraph 50.65(a)(4) requires, in part, that before performing maintenance activities (including but not limited to surveillance testing, post-maintenance testing, and corrective and preventative maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activity.

Contrary to the above, on May 27, 2017, prior to commencing control rod scram time testing at power, the licensee failed to assess and manage the increase in risk that may result from the proposed maintenance activity. The failure to assess and manage the risk associated with performing this testing contributed to an automatic reactor scram. As corrective actions, the licensee assessed the increase in risk for performing control rod scram time testing at power and developed a risk mitigation plan that was used to complete the testing at power. Because this finding was of very low safety significance and was entered into the CAP as AR 04073063, this violation is being treated as an NCV in accordance with Section 2.3.2 of the NRC Enforcement Policy.
(NCV 05000461/2017004–02: Failure to Assess and Manage Risk Associated with the Performance of Control Rod Time Testing)

4OA5 Other Activities

.1 Follow Up Inspection for Three or More Severity Level IV Traditional Enforcement Violations in the Same Area in a 12-Month Period (92723)

a. Inspection Scope

The inspectors assessed the licensee's evaluation of eight Severity Level (SL) IV non-cited violations, which occurred within the area of impeding the regulatory process, issued between October 1, 2015, and December 30, 2016. These violations were documented in NRC Inspection Reports as: (1) NCV 05000461/2015004–01; (2) NCV 05000461/2016001–03; (3) NCV 05000461/2016001–04; (4) NCV 05000461/2016002–07; (5) NCV 05000461/2016002–08; (6) NCV 05000461/2016002–09; (7) NCV 05000461/2016003–03; (8) NCV 05000461/2016009–03.

The inspection objectives were to provide assurance that:

- The licensee understood the causes of the multiple SL IV traditional enforcement violations;
- The licensee identified the extent of condition and extent of cause of multiple SL IV traditional enforcement violations; and
- The licensee's corrective actions to address the traditional enforcement violations were sufficient to address the causes.

The inspectors reviewed: (1) the various licensee CAP documents including Corrective Action Program Evaluation (CAPE) 03970173, "Evaluation of Two Additional SL IV Violations in Preparation for NRC IP 92723 Inspection," and Apparent Cause Evaluation (ACE) 02680308, "SL IV Violations Cause Evaluation;" (2) the licensee's Check-In Self-Assessment Report 03962672, "Pre IP 92723 Inspection—Check-In Self-Assessment;" (3) the licensee's procedures that implemented the CAP; and (4) the licensee's condition reports associated with the violations.

b. Findings

Failure to Identify the Extent of Condition for an Inadequate 10 CFR 50.59 Evaluation

Introduction: The inspectors identified a finding of very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion II, "Quality Assurance Program," for the licensee's failure to follow a procedure that implemented the Quality Assurance Program requirements. Specifically, the licensee failed to follow procedure PI–AA–125–1003, "Corrective Action Program Evaluation Manual," to identify the extent of condition for a lack of proficiency in applying the licensing basis when performing 10 CFR 50.59 evaluations.

Description: On March 9, 2017, the licensee completed CAPE 03970173 "Evaluation of Two Additional SL IV Violations in Preparation for NRC IP 92723 Inspection," that included the evaluation of eight NRC SL IV Violations within the scope of this inspection in accordance with procedure PI–AA–125–1003, "Corrective Action Program Evaluation Manual," Revision 4. In this CAPE, the licensee referenced ACE 02680308, "SL IV Violations Cause Evaluation," completed on July 8, 2016, which evaluated six of the

eight SL IV Violations within the scope. In these assessments, the licensee documented the causes and actions to determine the extent of condition for these NRC violations and the scope of each evaluation including NRC NCV 05000461/2016002–07. In this violation, the licensee failed to comply with 10 CFR 50.59(b)1 for an inadequate safety evaluation completed for a USAR change issued in 1989. The change was associated with the RT system operating procedure 3303.01 “Reactor Water Cleanup (RT).” Specifically, the licensee had changed this RT system operating procedure to allow bypassing both divisions of flow instruments, which resulted in bypassing of an RT system safety function (isolation). The licensee attributed the cause for this violation to their lack of understanding of the NRC regulations and a lack of guidance in the 10 CFR 50.59 implementing procedure. In CAPE 3970173, the licensee identified this error as a legacy item because the NRC regulations for 10 CFR 50.59 were revised in 1999, and the licensee’s procedure that implemented the 10 CFR 50.59 process had undergone multiple revisions since this time period.

The inspectors determined that the licensee had not performed an extent of condition review associated with NRC violation 05000461/2016002–007 for the lack of proficiency in applying the licensing basis when performing 10 CFR 50.59 evaluations that occurred between 1989 and 1999. Specifically, in ACE 02680308, the licensee action to address the extent of condition for three 10 CFR 50.59 violations included a review of the last 10 years (e.g. from 2006) of 10 CFR 50.59 evaluations associated with seismic issues. This type of review could not determine an extent of condition for the inadequate 10 CFR 50.59 evaluation of the USAR change associated with the RT system operation, because this evaluation was not related to a seismic issue and occurred at a much earlier timeframe (e.g.1989). Therefore, the inspectors determined that the licensee had not followed procedure PI–AA–125–1003, step 4.4.1.3, that required the assigned evaluator of a CAPE to determine the extent of condition. Specifically, no review had been assigned by the licensee in the CAPE or ACE to determine the extent of condition for inadequate safety evaluations completed from 1989 through 1999, when the licensee identified inadequate procedure guidance was in place for conducting 10 CFR 50.59 evaluations. Without completing an extent of condition review, the inspectors were concerned that other inadequate safety evaluations or screenings may exist that resulted in inappropriate USAR changes that adversely affected plant operating procedures. The licensee entered this issue into the CAP as AR 04075581. Based on other completed licensee corrective actions that included reviews of plant operating procedures to identify those which may allow disabling system safety functions, the inspectors did not identify an operability concern for safety systems. Additionally, the licensee planned to perform a USAR Upgrade Project which reportedly would include a review of safety evaluations for USAR changes that date back to 1986. The inspectors determined that the scope of this project would identify the extent of condition for this issue.

The licensee also performed a work group evaluation to determine the cause for the failure to perform an extent of condition for the identified issue. The licensee determined that when the AR was initially generated to address the identified 10 CFR 50.59 violation, they incorrectly closed the issue to an evaluation that did not address the identified performance deficiency. Because they believed the issue had already been addressed, they classified it as a significance level 4/D issue. Per procedure PI–AA–125, “Issue Identification and Screening,” NRC violations should be classified as significance level 3 issues. Significance level 3 issues would be evaluated to determine what type of causal product would be appropriate. Therefore, inappropriately classifying

the issue and closing it to an unrelated evaluation contributed to the failure to perform an appropriate cause or extent of condition evaluation.

Analysis: The inspectors determined the failure to follow a Quality Assurance Program implementing procedure and determine the extent of condition in accordance with procedure PI–AA–125–1003, "Corrective Action Program Evaluation Manual," was contrary to 10 CFR 50, Appendix B, Criterion II, and was a performance deficiency. Specifically, the licensee failed to identify the extent of condition for a lack of proficiency in applying the licensing basis when performing 10 CFR 50.59 evaluations. Although, this issue could potentially affect each of the Reactor Safety cornerstones, the inspectors elected to evaluate the more-than-minor aspect under the Mitigating Systems cornerstone because the actual inadequate safety evaluation identified by the NRC adversely affected a mitigating system (e.g. RT system) operating procedure.

The performance deficiency was more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated September 7, 2012, because if left uncorrected, the performance deficiency has the potential to lead to a more significant safety concern. Specifically, because an adequate extent of condition review was not completed, there was a potential for other safety systems to have been adversely affected by a lack of proficiency in applying the licensing basis for safety systems, and therefore, safety-related systems may not be able to perform intended safety functions as defined in the USAR. This issue would also adversely affect the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The lack of proficiency in applying the licensing basis with respect to 10 CFR 50.59 evaluations has the potential to affect all cornerstones, therefore, the inspectors assessed each set of cornerstone screening questions within IMC 0609, Appendix A, based on guidance in IMC 0609 that states, in part, "If more than one cornerstone is affected, the screening questions in all the affected cornerstones apply." Using IMC 0609, Attachment 4, "Initial Characterization of Findings," and Appendix A, "The Significance Determination Process for Findings at Power," issued June 19, 2012, the finding was screened against all cornerstones and determined to be of very low safety significance (Green) because the finding met each of the applicable screening questions to be characterized as having very low safety significance.

The inspectors determined this finding affected the cross-cutting area of human performance in the aspect of procedure adherence, which stated individuals follow processes, procedures, and work instructions. Specifically, when the NRC violation was documented in the CAP, it was not appropriately classified in accordance with procedure PI-AA-120, and was also incorrectly closed to an unrelated evaluation. This contributed to the failure to appropriately perform an extent of condition. (H.8)

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion II, "Quality Assurance Program," requires in part "a Quality Assurance Program which complies with the requirements of this appendix. This program shall be documented by written policies, procedures, or instructions and shall be carried out throughout plant life in accordance with those policies, procedures, or instructions."

The Exelon Quality Assurance Program was described in "Quality Assurance Topical Report (NO–AA–10)," Revision 92. Chapter 16 of NO–AA–10 states, in part, that "the

Company uses procedures to describe the method used to: Review and evaluate the conditions to determine the cause, extent and measures needed to correct and prevent recurrence."

Procedure PI–AA–125, "Corrective Action Program (CAP) Procedure," Revision 5, implemented the requirements of the Quality Assurance Program as described in Chapter 16 of NO–AA–10. Step 4.3.4 of PI–AA–125 stated in part that "PERFORM Class 'B' Cause Evaluations in accordance with PI–AA–125–1003, "Corrective Action Program Evaluation Manual."

Step 4.4.1.3 of PI–AA–125–1003, Revision 4 stated in part that "Determine the extent of condition. Evaluate similar components on the other train and unit, or similar situations to determine the extent of the problem."

Contrary to the above, on March 9, 2017, the licensee failed to carry out the quality assurance program in accordance with their written policies, procedures or instructions. Specifically, the licensee failed to determine the extent of condition for the problem associated with an inadequate safety evaluation identified by the NRC on August 10, 2016, as NCV 05000461/2016002–07, as required by procedure PI–AA–125. Specifically, in CAPE 03970173, the licensee evaluated NCV 05000461/2016002–07 within the scope of the problem and failed to determine the extent of condition for this problem. Because this violation is of very low safety significance and was entered into the CAP as AR 04075581, this violation is a NCV consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000461/2017004–03: Failure to Identify the Extent of Condition for an Inadequate 10 CFR 50.59 Evaluation)**

c. Observations

The licensee's procedures that implemented the CAP did not require an evaluation of the effectiveness of a corrective action (CA), unless the CA was associated with a root cause level investigation. For the level of investigations (CAPE and ACE) completed for the eight NRC SL IV NCVs discussed above, no action was required to be assigned to monitor the effectiveness of the CAs. For example, the licensee identified CA No. 15 in CAPE 03970173 to conduct a review in order to identify and correct errors in the USAR. This review was identified by the licensee as the USAR Upgrade Project, and the planned scope included a review of docketed licensing related correspondence and the associated USAR Sections to confirm the accuracy of the plant license basis documented in the USAR. The licensee reportedly intended to commit substantive resources to this project and it has the potential to generate an increased number of USAR changes over the next year. The inspectors noted that any updates to the USAR would be incorporated by licensee staff using procedures LS–AA–107 "UFSAR / FPR Update Procedure," Revision 12 and LS–AA–107–1001, "UFSAR / FPR Update T&RM," Revision 7, that were recently revised with guidance intended to improve the quality of the content within the USAR. However, the licensee elected not to assign a review to evaluate the effectiveness of the USAR Upgrade Project nor the recently revised USAR update procedures. Although not required, the inspectors considered the lack of an effectiveness review assignment for this CA a missed opportunity to validate that this CA was effective at preventing recurrence of similar problems.

.2 Operation of an Independent Spent Fuel Storage Installation at Operating Plants (60855.1)

a. Inspection Scope

During a pre-operational inspection of the Independent Spent Fuel Storage Installation (ISFSI) in 2016, the inspectors identified an unresolved item (URI) associated with changes that the licensee had made to Section 4.5.3, "Maximum Time Limit During Wet Transfer Operations," of the HI-STORM FW Multi-Purpose Canister (MPC) Storage System Final Safety Analysis Report (FSAR), Revision 2. Revision 2 is the licensing basis for the six casks that the licensee had loaded to Certificate of Compliance (CoC) number 1032, Amendment 0, Revision 1. These FSAR changes were made by Holtec International via Engineering Change Orders, ECO–5018–25R0 and ECO–5018–48R1 and subsequently accepted by the licensee via 72.48 Screening CL–2016–S–007.

The URI contained two questions regarding these changes: firstly, does 10 CFR 72.48(c)(2)(viii) apply to the time-to-boil calculation, and secondly, is the licensee permitted under 10 CFR 72.48 to make these changes without prior NRC approval.

These questions were discussed with the Division of Spent Fuel Management staff in the Office of Nuclear Material Safety and Safeguards. The Office of Nuclear Material Safety and Safeguards reviewed the inspectors' licensing concerns and provided necessary regulatory guidance. The inspectors reviewed relevant documents to determine if the licensee has demonstrated compliance with applicable commitments and requirements as specified in the FSAR, the CoC, 10 CFR Part 72, and the TS.

b. Findings

The NRC staff has determined that the prohibition on water boiling in the MPC cavity, following the guidance in NUREG-1536, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility—Final Report," and as stated in Section 4.5.3 of the HI-STORM FW FSAR, is a part of the design basis for the HI-STORM FW system. This determination is based on the definition of design bases in 10 CFR 72.3 which refers, in part, to "restraints derived from generally accepted state-of-the-art practices for achieving functional goals." Guidance provided in Nuclear Energy Institute (NEI) document 96-07, Appendix B, "Guidelines for 10 CFR 72.48 Implementation," Section B3.5 also describes design bases, in part, as "Values . . . chosen by the licensee from an applicable code, standard or guidance document as reference bounds for design to meet design bases functional requirements."

This prohibition on water boiling in the MPC cavity is an underlying assumption in various parts of the HI-STORM FW FSAR. For example, the corrosion evaluation in Section 8.12.2.1 assumes that the MPC will only be in contact with water if the temperature of the water is below boiling, and the FSAR does not postulate a scenario where the water in the MPC cavity boils. It is important to note that for an individual cask design, both that the design criteria and design bases as listed in Section 2.2 of NUREG–1536 may not be exhaustive, and that a particular cask FSAR may not explicitly identify all of the design bases and design criteria that correlate to Chapter 2 of NUREG–1536.

The NRC staff has determined that the time-to-boil calculation performed in the FSAR is not part of the design basis, nor is it used to establish the design basis. However, the staff has also determined that the time-to-boil calculation is a method of evaluation (MOE) used in a safety analysis, in that it ensures that the design basis, including the prohibition on boiling, is maintained and states when additional actions (i.e., initiating forced water circulation to remove decay heat) would be required to maintain the design basis. Method of Evaluations are discussed in Section B3.10 of NEI 96–07, Appendix B. Section B3.12 of NEI 96–07, Appendix B, describes safety analyses which include, "UFSAR analyses that demonstrate the design and performance of structures, systems, and components important to safety during normal operations and expected operational occurrences."

Under normal conditions for short term operations, the calculated time-to-boil should be sufficient to complete wet transfer operations. The FSAR states that, "in the unlikely event that the maximum allowable time ... is found to be insufficient to complete all wet transfer operations, a forced water circulation shall be initiated and maintained to remove the decay heat from the MPC cavity." The NRC staff considers this initiation of forced water cooling to be an off-normal or expected operational occurrence for short term operations. It is important to note that a cask FSAR may contain safety analyses besides those explicitly correlated with Chapter 12 of NUREG–1536.

Since the time-to-boil calculation is a MOE used in a safety analysis of the HI-STORM FW system, the NRC staff were able to conclude that the criteria of 10 CFR 72.48(c)(2), in particular criterion (viii), are applicable to changes made to the time-to-boil calculation in the HI-STORM FW system. Therefore, in accordance with 10 CFR 72.48(d)(1), "The licensee ... shall maintain records of changes in the facility or spent fuel storage cask design, of changes in procedures, and of tests and experiments made pursuant to paragraph (c) of this section. These records must include a written evaluation which provides the bases for the determination that the change, test, or experiment does not require a ... CoC amendment pursuant to paragraph (c)(2) of this section."

Of the three changes accepted by the licensee that are the subject of this URI, the NRC staff's position is that Engineering Change Order ECO–5018–25R0 [1] and ECO–5018–48R1 [4] are changes to the input parameters of the time-to-boil calculation and not changes to elements of a method. This position follows the guidance of NEI 96–07, Appendix B, Section B3.8, which states that "Input parameters are those values derived directly from the physical characteristics of SSC or processes in the ISFSI facility or cask design, including flow rates, temperatures, pressures, dimensions or measurements (e.g., volume, weight, size, etc), and system response times." Section B3.8 further states that "Changes to methods of evaluation described in the UFSAR (see Section B3.10) are evaluated under criterion 10 CFR 72.48(c)(2)(viii), whereas changes to input parameters described in the UFSAR are considered changes to the ISFSI facility or cask design that would be evaluated under the other seven criteria of 10 CFR 72.48(c)(2), but not criterion (c)(2)(viii)."

The NRC staff's position is that Engineering Change Order ECO–5018–48R1 [1] is a non-conservative change to an element of a method, and that this change under 10 CFR 72.48(c)(2)(viii) requires NRC approval prior to implementing the proposed change. This position follows the guidance of NEI 96-07, Appendix B, Section B4.3.8., "Does the Activity Result in a Departure from a Method of Evaluation Described in the UFSAR Used in Establishing the Design Bases or in the Safety Analyses?"

(1) Failure to Perform an Evaluation in Accordance with 10 CFR 72.48 for Changes Made to the Time-to-Boil Calculation

Introduction: The inspectors identified a Severity Level IV NCV of 10 CFR 72.48(d)(1) for the licensee's failure to perform a written evaluation which provides the bases for the determination that the changes made to the time-to-boil calculation do not require a CoC amendment pursuant to 10 CFR 72.48(c)(2).

Description: The above discussion delineates that the position of the NRC staff is that 10 CFR 72.48 is applicable to the time-to-boil calculation as described in the HI-STORM FW FSAR. Therefore, changes made to the time-to-boil calculation must be evaluated by the licensee against the appropriate 10 CFR 72.48(c)(2) criteria. It is the NRC staff's position that Engineering Change Order ECO–5018–25R0 [1] and ECO–5018–48R1 [4] must be evaluated against 10 CFR 72.48(c)(2)(i) through (vii) and that Engineering Change Order ECO–5018–48R1 [1] must be evaluated against 10 CFR 72.48(c)(2)(viii). On June 20, 2016, the licensee performed a 72.48 screening for these changes but did not perform a written evaluation documenting a valid basis for the determination that these changes to the time-to-boil calculation did not require a CoC amendment.

The licensee had incorporated Engineering Change Order ECO–5018–25R0 [1] and ECO–5018–48R1 [4] into operating procedures HPP–2226–300, "MPC Sealing at Clinton," Revision 0, and HPP–2226–200, "MPC Loading at Clinton," Revision 0. However, despite the incorporation of these changes into the licensee's loading campaign, there were no operational incidents associated with the changes made to the time-to-boil calculation. The licensee entered this issue into its CAP as AR 02714091 and as AR 04081583.

Analysis: The inspectors determined that the licensee's failure to perform a written evaluation in accordance with 10 CFR 72.48(c)(2) for changes made to the time-to-boil calculation was a performance deficiency. This violation was determined to be more than minor because the inspectors could not conclude that the changes made to the time-to-boil calculation would not require a CoC amendment in accordance with 10 CFR 72.48(c)(2) criteria (i) through (viii).

In accordance with Section 2.2 of the NRC Enforcement Policy and IMC 0612, Appendix B, "Issue Screening," ISFSI facilities are not subject to the SDP and are not subject to the Reactor Oversight Process. Violations identified at ISFSIs are assessed using traditional enforcement. Traditional enforcement violations are not assessed for cross-cutting aspects.

Consistent with the guidance in Section 1.2.6.D of the NRC Enforcement Manual, if a violation does not fit an example in the enforcement policy violation examples, it should be assigned a severity level: (1) commensurate with its safety significance, and (2) informed by similar violations addressed in the violation examples. The inspectors determined that the violation could be classified as a Severity Level IV, informed by NRC Enforcement Policy example 6.1.d.2, in that the licensee failed to perform a 10 CFR 72.48 evaluation, similar to a 10 CFR 50.59 evaluation, which resulted in a condition having low safety significance.

Enforcement: Title 10 CFR 72.48(d)(1) states that "the licensee and certificate holder shall maintain records of changes in the facility or spent fuel storage cask design, of changes in procedures, and of tests and experiments made pursuant to paragraph (c) of this section. These records must include a written evaluation which provides the bases for the determination that the change, test, or experiment does not require a license or CoC amendment pursuant to paragraph (c)(2) of this section."

Contrary to the above, on June 20, 2016, the licensee accepted Engineering Change Order ECO–5018–25R0 [1], ECO–5018–48R1 [4], and ECO–5018–48R1 [1] to the time-to-boil calculation, but did not perform a written evaluation that provided the bases that these changes did not require a CoC amendment in accordance with 10 CFR 72.48(c)(2).

The licensee has entered this issue into its CAP as AR 02714091 and AR 04081583. The licensee is not currently performing ISFSI loading operations and is not scheduled to perform loading operations until 2018. Because this issue was of very low safety significance (Severity Level IV) and has been entered into the licensee's CAP, this violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. **(NCV 05000461/2017004–04: Failure to Perform an Evaluation in Accordance with 10 CFR 72.48 for Changes Made to the Time-to-Boil Calculation)**

Unresolved Item 05000461/2016003–06; 07201046/2016002–02, "Potentially Non-Conservative Changes Made to the Time-to-Boil Calculation in the FSAR under 10 CFR 72.48," is closed.

4OA6 Management Meetings

.1 Exit Meeting Summary

On January 11, 2018, the inspectors presented the inspection results to Mr. B. Kapellas 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

Interim exits were conducted for:

- The results of the follow-up inspection for three or more SL IV traditional enforcement violations in the same area in a 12-month period were presented to Mr. B. Kapellas, Plant Manager, on November 17, 2017.
- The results of the biennial LORT Program inspection were presented to Mr. B. Kapellas, Plant Manager, and other members of the licensee's staff, on November 17, 2017. The licensee acknowledged the issues presented.
- The results of the Emergency Preparedness Program inspection were presented over the phone to Mr. M. Friedmann, Emergency Preparedness Manager, on December 19, 2017.
- The results of the ISFSI URI closure (IP 60855.1) were presented to Mr. T. Stoner and other members of the licensee management and staff on December 19, 2017. The licensee acknowledged the information presented.

- The inspectors discussed the completed 2017 LORT annual operating test and biennial written examination results with Mr. R. J. Frederes, Operations Training Manager, on December 20, 2017.

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

T. Stoner, Site Vice President
B. Kapellas, Plant Manager
K. Pointer, Regulatory Assurance
N. Santos, Regulatory Assurance
R. Bair, Work Management Director
J. Cunningham, Maintenance Director
T. Dean, Training Director
C. Dunn, Operations Director
M. Friedmann, Emergency Preparedness Manager
M. Heger, Senior Manager Design Engineering
T. Krawyck, Engineering Director
W. Marsh, Organizational Effectiveness Manager
R.J. Reynolds, Operations Training Manager
F. Paslaski, Radiation Protection Manager
D. Shelton, Regulatory Assurance Manager
R. Champley, Shift Operations Superintendent
D. Koons, Chemistry Manager
J. Wilson, Senior Manager Plant Engineering
A. Sigemund, Security Manager

U.S. Nuclear Regulatory Commission

K. Stoedter, Chief, Reactor Projects Branch 1

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000461/2017004–01	FIN	Reactor Down Power due to Reactor Recirculation Pump Motor Lower Bearing Oil Leak (Section 4OA3.1)
05000461/2017004–02	NCV	Failure to Assess and Manage Risk Associated with the Performance of Control Rod Time Testing (Section 4OA3.5)
05000461/2017004–03	NCV	Failure to Identify the Extent of Condition for an Inadequate 10 CFR 50.59 Evaluation (Section 4OA5.1)
05000461/2017004–04	NCV	Failure to Perform an Evaluation in Accordance with 10 CFR 72.48 for Changes Made to the Time-to-Boil Calculation (Section 4OA5.2)

Closed

05000461/2017004–01	FIN	Reactor Down Power due to Reactor Recirculation Pump Motor Lower Bearing Oil Leak (Section 4OA3.1)
05000461/2017004–02	NCV	Failure to Assess and Manage Risk Associated with the Performance of Control Rod Time Testing (Section 4OA3.5)
05000461/2017004–03	NCV	Failure to Identify the Extent of Condition for an Inadequate 10 CFR 50.59 Evaluation (Section 4OA5.1)
05000461/2017004–04	NCV	Failure to Perform an Evaluation in Accordance with 10 CFR 72.48 for Changes Made to the Time-to-Boil Calculation (Section 4OA5.2)
05000461/2016003–06; 07201046/2016002–02	URI	Potentially Non-Conservative Changes Made to the Time-to-Boil Calculation in the FSAR Under 10 CFR 72.48 (Section 4OA5.2)
05000461/2016–009–00	LER	Trip of the Fuel Building Fans Due to Damper Failure (Section 4OA3.2)
05000461/2016–010–00; 05000461/2016–010–01	LER	Failure to Complete TS Required Actions Within the Completion Time Resulted in a Condition Prohibited by TS (Section 4OA3.3)
05000461/2017–002–00; 05000461/2017–002–01	LER	Failure of the Division 1 Diesel Generator Ventilation Fan Load Sequence Relay Circuit During Concurrent Maintenance of RHR Division 2 Results in an Unanalyzed Condition (Section 4OA3.4)

05000461/2017–004–01	LER	Main Steam Isolation Valve Local Leak Rate Test Limit Exceeded During Refueling Outage (Section 4OA3.5)
05000461/2017–005–00; 05000461/2017–005–01	LER	Automatic Reactor Scram During the Performance of Scram Time Testing as a Result of an Invalid Oscillating Power Range Monitor Growth Rate Trip (Section 4OA3.6)

Discussed

None.

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

- AR 04046530; Cold Weather Preparation Procedure Needs Revised
- AR 04035941; EOID: Fire Pump A, OFP01PA Block Not Warm Heater Not Working
- AR 04065808; Winter Readiness Issue Exhaust Fan Louver Will Not Close
- AR 04043401; Cold Weather Oil Filter Cover is Broken
- WC–AA–107; Seasonal Readiness; Revision 18
- CPS 1860.01; Cold Weather Operation; Revision 9
- CPS 3113.01; Circulating Water; Revision 39

1R04 Equipment Alignment

- CPS 3309.01V002; High Pressure Core Spray Instrument Valve Lineup; Revision 9
- CPS 3309.01V001; High Pressure Core Spray Valve Lineup; Revision 11b
- CPS 3309.01E001; High Pressure Core Spray Electrical Lineup; Revision 8a
- WO 04640788; EOID—EM 1E22S004102–F01 & F04 1C1 Pt Fuses Found Blown
- WO 04659310; IM T-Shoot/Rework DG1C Annunciator Panel
- WO 01688707; IM 9433.36 RTT #HPCS RTT (All Channels)
- AR 04031683; Operator Inexperience Led to Delayed Surveillance Performance
- AR 03962315; Enhancement to 9051.01
- AR 03991611; Total HPCS Response Time Not Recalculated Following 9051.06
- AR 03997051; MCR Annunciator 5064–8D, Not Open Excess Flow Check Valve
- AR 04031746; Tubing Needs Replaced from 1E22R001 to Tent Valve
- AR 04035306; NRC Question of Oil Accumulation on 1E22F001
- Diagram No. M05–1074; P&ID High Pressure Core Spray (HP); Revision 0

1R05 Fire Protection

- CPS 1893.04M351; 781 Control: Aux. Elect. Equip., Inverter & Battery Rooms Prefire Plan; Revision 7c
- Calculation No. 01FP14; Analysis of the Effects of Initiation of Fire Suppression Systems on Safe Shutdown Equipment at CPS
- Drawing No. M05–1115; P&ID Essential Switchgear Heat Removal (VX); Revision M
- Drawing No. A21–1036; Hollow Metal Door Details, Sheet 1; Revision E
- Drawing No. M05–1115; P&ID Essential Switchgear Heat Removal (VX); Revision L
- Drawing No. M05–1113; P&ID Turbine Bldg. HVAC (VT); Revision Q
- Drawing No. M01–1109; General Arrangement Control Bldg. Floor Plan El. 781’-0”; Revision D
- CPS 1893.04M003; Prefire Plan Legend; Revision 1
- EC 384533; Revise Calculation 01FP14 to Strengthen Analysis for Fire Zone CB–3A; Revision 0
- CPS 1893.04M354; 781 Control: Div 1 & 2 Cable Risers Prefire Plan; Revision 5a
- Drawing No. M01–1109; Figure 11—Cable Tray, General Arrangement Control Building Floor Plan, El. 781’-0”; Revision 15

- Drawing No. M01–1109; Figure FP-13b, Fire Protection Features Control Building Floor Plan, El. 781'-0"; Revision 15
- Drawing No. M01–1109; Figure FP–13a, Fire Zone Boundaries Control Building Floor Plan, El. 781'-0"; Revision 8
- AR 04067481; NRC ID 'A' Halon Extinguisher Not Inspected for September
- AR 00568315; CPS Extinguisher Inspection Frequency Exceeds OSHA
- CPS 1893.04M500; 712 Diesel Generator: Div 3 Diesel Fuel Tank Room Prefire Plan; Revision 5
- CPS 3822.13; Visual Inspections of Portable Fire Extinguishers; Revision 4f
- Drawing No. M05–1103; P&ID Diesel Gen. Room Ventilation (VD); Revision P
- Drawing No. M01–1109; Figure FP-9b Fire Protection Features Control & Diesel Gen. Building Floor Plan, El. 719'-0"; Revision 14
- Drawing No. M01–1105; Figure FP-8b Fire Protection Features Control & Diesel Generator Building Basement Floor Plan, El. 702'-0" & El. 712'-0"; Revision 14
- Drawing No. M14–1063, Sheet 2; Diesel Gen. & HVAC Building Basement Floor Vent. Plan, El. 712'-0" Oil Storage Tank Room; Revision L
- Operation Logs for October 23, 2017
- CPS 1893.01; Fire Protection Impairment Reporting; Revision 20e
- CPS 1893.04M103; 707 Auxiliary: RCIC Pump Room Prefire Plan; Revision 5
- Drawing CPS Figure FP–1 Fire Protection Legend; Revision 13
- Drawing CPS Figure FP–2b Fire Protection Features Auxiliary Fuel Building and Containment Basement Floor Plan, El. 707'-6" & 712'-0"; Revision 10
- Drawing CPS Figure FP–2a Fire Zone Boundaries Auxiliary Fuel Building and Containment Basement Floor Plan, El. 707'-6" & 712'-0"; Revision 10

1R06 Flood Protection Measures

- CC–AA–309–1001; Internal Flooding Calculations; Revision 9

1R11 Licensed Operator Requalification Program

- CPS 3313.01H001; LPCS Manual Initiation – Shutdown Hard Card; Revision 0
- HU–CL–104–1011001; CPS Specific Procedure Use & Compliance; Revision 5
- OP–AA–102–106; Operator Response Time Program; Revision 4
- OP–AA–105–102; NRC Active License Maintenance; Revision 13
- OP–CL–102–106–1001; Operator Response Time Master List at CPS; Revision 8
- OP–CL–105–102–1001; Operations Department-Personal Indoctrination, STA/IA Certification/STA Proficiency and Non-Licensed Field Supervisor (NL-FS) Qualification; Revision 2
- TQ–AA–150; Operator Training Programs; Revision 14
- TQ–AA–150–J202; LORT Annual Exam Development Job Aid; Revision 0
- TQ–AA–155; Conduct of Simulator Training and Evaluation; Revision 6
- TQ–AA–201; Examination Security and Administration; Revision 17
- TQ–AA–201–F–01; Printing or Copying Exam Material Checklist; Revision 5
- TQ–AA–201–F–03; Exam Security Briefing Checklist; Revision 2
- TQ–AA–201–F–04; NRC Exam Development Room Checklist; Revision 3
- TQ–AA–306; Simulator Management; Revision 8
- AR 04076047; TRGN—JPM Administration
- AR 02739165; Pre-NRC IP 71111.11; Pre-Inspection Assessment
- AR 03943566; CDBI-Observations of Piloted TCA7 Performance
- AR 04065253; 2017 INPO Site Exit: 5 AFIs & 8 Strengths (Potential)

- AR 02716391; (ZTP) 9170.02, VC B Valve Operability Test Observations
- AR 02704778; T–18 Exit of Summary GAPS Identified LF.1-7
- AR 02659195; Missed Required Surveillance
- AR 02739386; Green NCV for a Condition Prohibited by Tech Spec 3.7.6
- AR 04010227; Informal Communication Methods Utilized to Establish Compensatory Measures
- AR 04029079; Laminated Copies of Procedure Forms in Simulator Not Up to Date
- AR 02729359; Clinton Candidate for Elevation
- 2017 Clinton Power Station (CPS) Licensed Operator Requalification (LOR) Program Biennial Written Examination for Crews C and G (RO and SRO); October 2017
- 2017 CPS LOR Program Annual Examination JPM Number 273; Revision 4
- 2017 CPS LOR Program Annual Examination JPM Number 264; Revision 2
- 2017 CPS LOR Program Annual Examination JPM Number 471; Revision 1
- 2017 CPS LOR Program Annual Examination JPM Number 058; Revision 3
- 2017 CPS LOR Program Annual Examination JPM Number 246; Revision 4
- 2017 CPS LOR Program Annual Examination JPM Number 221; Revision 2
- 2017 CPS LOR Program Annual Examination JPM Number 259; Revision 4
- 2017 CPS LOR Program Annual Examination JPM Number 244; Revision 4
- 2017 CPS LOR Program Annual Examination JPM Number 281; Revision 3
- 2017 CPS LOR Program Annual Examination JPM Number 230; Revision 3
- 2017 CPS LOR Program Annual Simulator Evaluation Scenario ESG-LOR-12: Revision 5
- 2017 CPS LOR Program Annual Simulator Evaluation Scenario ESG-LOR-19: Revision 2
- 2017 CPS LOR Program Annual Simulator Evaluation Scenario ESG-LOR-85: Revision 5
- 2017 CPS LOR Program Annual Simulator Evaluation Scenario ESG-LOR-92: Revision 3
- Individual and Crew Simulator Grading Packages; Week 4; November 17, 2017
- Crew Simulator Evaluation Forms; Week 4; November 16, 2017
- Scenario Based Testing Packages for 2017 Clinton Power Station Licensed Operator Requalification Program Annual Simulator Evaluation Scenario ESG–LOR–12; October 2017
- Scenario Based Testing Packages for 2017 Clinton Power Station Licensed Operator Requalification Program Annual Simulator Evaluation Scenario ESG–LOR–92; October 2017
- OP–AA–105–102 Attachment 1; Active License Tracking Logs for 3rd Quarter 2017
- OP–AA–105–102 Attachment 2; Reactivation of License Logs for 5 Licensed Operators; April 24, 2015 through August 25, 2017
- Remedial Training Notification and Action on Failure – Written Biennial Exam Failure; October 30, 2017
- TQ–AA–1002–F004; Training Review Committee Agenda/Minutes; August 2, 2017
- TQ–AA–306–F–06; BWR Critical Conditions for Cold Startup; June 5, 2017
- TQ–AA–306–F–07; BWR Power Coefficient of Reactivity and Control Rod Worth; June 5, 2017
- TQ–AA–306–F–08; BWR Xenon Worth; June 5, 2017
- TQ–AA–306–F–09; BWR Site Specific Shutdown Margin and Reactivity Anomaly Tests; June 5, 2017
- SPVG 1.02; Simulator Stability [Steady State Tests]; June 7, 2017
- SPVG 1.02; Simulator Stability [Drift Test with Heat Balance] September 7, 2017
- SPVG 5.01; Manual SCRAM; June 19, 2017
- SPVG 5.06; Main Turbine Trip from Maximum Power Level without [not resulting in] an Immediate SCRAM [approx. 25% power]; August 10, 2017
- SPVG 5.07; Maximum Rate Power Ramp Down to Approximately 75% then Back Up to 100%; June 19, 2017
- SWR 132592; 3D Monicore/MFLCPR Response to Control Rod Movement or Core Flow Changes; October 10, 2016

- SWR 132617; Main Generator Operation with Negative MVARs; October 17, 2016
- SWR 132737; SRM Period Annunciator when not Expected; December 12, 2016
- SWR 132834; SCRAM Discharge Volume Drain Down Timing; January 23, 2017
- SWR 133000; 0RIX–PR008 & PR012 Radiation Monitors Indicated Low Flow when not Expected; March 6, 2017

1R12 Maintenance Effectiveness

- ER–AA–310; Implementation of Maintenance Rule; Revision 9
- ER–AA–310–1001; Maintenance Rule Scoping; Revision 4
- ER–AA–310–1002; Maintenance Rule Functions – Safety Significance Classification; Revision 3
- ER–AA–310–1003; Maintenance Rule – Performance Criteria Selection; Revision 4
- ER–AA–310–1004; Maintenance Rule – Performance Monitoring; Revision 13
- ER–AA–310–1005; Maintenance Rule – Dispositioning Between (a)(1 and (a)(2); Revision 7
- ER–AA–310–1006; Maintenance Rule – Expert Panel Roles and Responsibilities; Revision 5
- AR 04042400; RR System Exceeds Maintenance Rule Reliability Criteria
- Exelon Maintenance Rule—Failure Classification Form; Issue Report Number 04029024
- AR 04036400; Unclear Definition of Maintenance Rule Functional Failure FO

1R13 Maintenance Risk Assessments and Emergent Work Control

- AD–AA–3000; Nuclear Risk Management Process; Revision 1
- AD–AA–3000; Nuclear Risk Management Process; Revision
- ER–AA–600; Risk Management; Revision 7
- ER–AA–600–1011; Risk Management Program; Revision 14
- ER–AA–600–1012; Risk Management Documentation; Revision 12
- ER–AA–600–1014; Risk Management Configuration Control; Revision 7
- ER–AA–600–1042; On-line Risk Management; Revision 9
- OP–AA–108-117; Protected Equipment Program; Revision
- WC–AA–101; On-Line Work Control Process; Revision 26
- WC–AA–104; Integrated Risk Management; Revision 23
- WC–AA–101; On-Line Work Control Process; Revision 26
- WC–AA–101–1006; On-Line Risk Management and Assessment; Revision 2

1R15 Operability Evaluations

- AR 04037464; Valve Failed as Found Set Pressure Test
- AR 04055528; Diesel Fire Pump ‘A’ Oil Sample Off-Normal Results
- EC 621388; Functional Evaluation of OFP01PA Engine 9–27–2017; Revision 0

1R18 Plant Modifications

- EC 622359; Replacement of Dry Type Transformer 1AP11E2; Revision 2
- WO 01695780; Automatic Trip of Breaker 1AP07EJ

1R19 Post Maintenance Testing

- WO 01916716; Inspect/Clean Heat Exchanger, Boroscope/Eddy Current 100%
- WO 01916887; 9080.30B20 OP DG 1B Overspeed Trip Test

1R22 Surveillance Testing

- AR 04066721; NRC Question on 9054.02
- AR 04066846; 9054.02 RCIC Valve Op Procedure Enhancement
- CPS 9054.02; Reactor Core Isolation Cooling Valve Operability Checks; Revision 43
- CPS 9054.02D001; RCIC Valve Operability Data Sheet; Revision 41b
- Drawing M05–1079, Sheet 2; P&ID Reactor Core Isolation Cooling (RCIC) (RI); Revision 0
- Drawing M05–1079, Sheet 1; P&ID Reactor Core Isolation Cooling (RCIC) (RI); Revision 0
- PMRQ 00191635–01; Perform UT Testing to Check for Accumulated Air-RCIC

1EP4 Emergency Action Level and Emergency Plan Changes

- EP–AA–120; Emergency Plan Administration; Revision 20
- EP–AA–120–1001; 10CFR50.54(q) Change Evaluation; Revision 9
- EP–AA–1003; Exelon Nuclear Radiological Emergency Plan Annex for Clinton Station; Revisions 26 and 27
- EP–AA–1003 Addendum 3; Emergency Action Levels for Clinton Station; Revisions 1 and 2
- 10 CFR 50.54(q) Qualified Evaluator List; Dated June 6, 2017
- 50.54(q) Evaluation No. 16–94; EP–AA–1003, “Exelon Nuclear Radiological Emergency Plan Annex for Clinton Station” (Revision 26) Evaluation / Assessment Review; Dated July 22, 2016
- 50.54(q) Evaluation No. 16–109; EP–AA–1003, “Exelon Nuclear Radiological Emergency Plan Annex for Clinton Station” (Revision 26) Evaluation / Assessment Review; Dated July 22, 2016
- 50.54(q) Evaluation No. 17–34; EP–AA–1003, Addendum 3, “Emergency Action Levels for Clinton Station” (Revision 2) Evaluation / Assessment Review; Dated September 6, 2016

4OA1 Performance Indicator Verification

- ER–AA–2008; Mitigating Systems Performance Index Monitoring and Margin Evaluation; Revision 4
- CC–AA–102; Design Input and Configuration Change Impact Screening; Revision 29
- CL–MSPI–01; Clinton MSPI Basis Document; Revision 10
- CPS 9000.01; Control Room Surveillance Log; Revision 35e
- CPS 9000.01D001; Control Room Surveillance Log—Mode 1,2,3 Data Sheet; Revision 55c
- CPS 3315.02; Leak Detection; Revision 14e

4OA2 Problem Identification and Resolution

- AR 04036685; 702 Mechanical Maintenance Incorrect Use of OP–AA–201–006 Section 4.3.4
- AR 04031190; Housekeeping Issues Floor Drain Evaporator Recirculation Pump C.A
- AR 04001400; Material Stored Outside 828 CT PAL
- AR 04001386; Material Stored Outside SX Pump Rooms
- AR 0399133; Typographical Errors Identified in EC 393108
- AR 03998666; Material Storage 737 RW
- AR 03998664; Material Storage Stores Area
- AR 03996665; Black Backpack Left Unattended in Division 3 DG
- AR 03982603; Drums Blocking Electrical Cabinet and Gai-Tronics
- AR 03971902; Work Area/Parts Staging Outside Waste Sludge ‘A’
- AR 03967627; 755 Fuel Building Housekeeping
- AR 04019916; Fire Door not Challenged
- AR 04013757; Free-Air Cable Attached to FP Piping
- AR 04018372; Free-Air Cables Attached to FP Piping

- AR 04011233; Fire Door not Challenged
- AR 03999732; Free-Air Cable Attached to FP Piping
- AR 04001084; Deep Fryer Left on while not in Use
- AR 03991638; Free-Air Cable Attached to FP Piping
- AR 03977922; Free-Air Cable Attached to FP Piping
- AR 03998665; Material Storage 702 RW
- AR 03970965; Coaching on Fire Door Checks
- AR 03970433; Travel Path Impeded
- AR 03966647; Temp Heaters used in Oil Storage Warehouse
- AR 04023795; RP Supplies Left in Pump Alley
- AR 03964737; Missed Hourly Fire Impairment
- AR 04067481; Halon Extinguisher not Inspected for September
- AR 04065792; Fire Door not Secured
- AR 04062478; Free-Air Cable Attached to FP Piping
- AR 04060446; HH 28 and 29 Missing Breakaway Lock/Seal
- AR 04047495; Free-Air Cable Attached to FP Piping
- AR 03989068; Material Storage Questions Require Follow Up

4OA3 Event Followup

- AR 04029024; EOID: RR 'B" Motor Lower Oil Level Trending Down 1LR–RR001
- MA–AA–716–060; Compression Fittings Inspection, Installation, Remake and Repair; Revision 3
- PI–AA–125–1001; RR 'B' Motor Lower Oil Level Trending Down 1LR–RR001; Revision 3
- Log Entries from 07/07/2017
- PI–AA–125–1001; Automatic Reactor Scram; Revision 3
- AR 04073063; Risk Mitigation Plan Not Performed
- WO 04581915; 9813.01R20 OP Scram 10% Control Rods Per 9813.01
- WC–AA–104; PMRQ OP Scram 10% Control Rods Per 9813.01; Revision 23
- WC–AA–101; On-Line Work Control Process; Revision 27
- PI–AA–125–1001; Automatic Reactor Scram; Revision 3
- CPS 9813.01; Control Rod Scram Time Testing; Revision 41c
- AR 04020663; Operators Manually Tripped Turbine During Scram Actions
- AR 04016987; Informal Benchmarking BWR 6 OPRM Trips
- EC 619930; Clinton Core Stability Evaluation; Revision 0

4OA5 Other

- ACE 02680308; Severity Level IV Violations Cause Evaluation; July 8, 2016
- ACE 02652522; Bypassing Both Divisions of RT Leak Detection is Reportable; May 2, 2016
- AR 03962672; Pre-NRC Level 4 Traditional Enforcement (IP 92723)
- AR 03970173; Evaluation of Two Additional Severity Level IV Violations in Preparation for NRC IP 92723 Inspection
- AR 03960085; NRC SLIV/NCV 2016003–03: No 50.59 for Surveillance Frequency Change
- AR 02741851; NRC CDBI Identified USAR Update Needed
- AR 02728304; NRC Exit Need Steam Bypass Valves Past Reportability Review
- AR 02720163; USAR Change Package Not Approved Prior to STI Change
- AR 02706233; NRC SLIV NCV 2016002–09 Failure to Update the USAR CD/FW Sys
- AR 02706228; NRC SLIV NCV 2016002–08 Failure to Update USAR Supp Pool Temp
- AR 02706222: NRC SLIV NCV 2016002–07 No License Amendment for RTLD Bypass
- AR 02685337; NRC Inadequate 1989 Safety Evaluation RT LD Bypass

- AR 02690657; NRC Question COLR Thermal Limits Removal
- AR 02680308; NRC SLIV Violations; Recommend ACE from NCAP EVAL IR 2664298
- AR 02676514; NRC SLIV 2016001–04, Failure to Report within 60 days
- AR 02664276; Discrepancy Found in USAR During NRC Review
- AR 02656128; NRC ID: Potential USAR Revision Required
- AR 02652522; Bypassing Both Divs of RT LDS is Reportable
- AR 02651634; NRC Followup Questions from IR 2642391—Tech Spec LDI
- AR 02645140; NRC Potential Issue on Bypassing RT LD
- AR 02627077; SLIV 2015004–01, Failure to Update the USAR—Hydrogen Water Chemistry
- AR 02619114; NCV 2015003–04, Failure to Enter TS Actions during OpDRVs
- AR 02594259; NRC Inspector Concerned About USAR 5.4.15 Level of Detail
- AR 02511697; SLIV Violations Cause Evaluation
- CAPE 03970173; Evaluation of Two Additional Severity Level IV Violations in Preparation for NRC IP 92723 Inspection; March 9, 2017
- LS–AA–107; UFSAR/FPR Update Procedure; Revision 12
- LS–AA–107–1001; UFSAR/FPR Update T&RM; Revision 7
- PI–AA–120; Issue Identification and Screening Process; Revision 6
- PI–AA–125; Corrective Action Program (CAP) Procedure; Revision 4
- PI–AA–125; Corrective Action Program (CAP) Procedure; Revision 5
- PI–AA–125–1003; Corrective Action Program Evaluation Manual; Revision 4
- PI–AA–125–1003; Apparent Cause Evaluation Manual; Revision 3
- PI–AA–125–1004; Effectiveness Review Manual; Revision 2
- PI–AA–125–1006; Investigation Techniques Manual; Revision 3
- Self-Assessment Report 03962672, “Pre IP 92723 Inspection—Check-In Self-Assessment,” Report Date March 24, 2017
- 3303.01; Reactor Water Cleanup (RT); Revision 36b
- RRTI-2226-011; Holtec Response to Request for Technical Information; September 13, 2016
- 72.48 Screening CL-2016-S-007; HI-STORM FW Time to Boil Calculation; June 20, 2016
- AR 02714091; NRC Identified a URI at the Sept. 9, 2016 DCS Project Exit
- AR 04081583; Potential NRC SL IV Violation for Time to Boil 72.48 Screen
- ECO–5018–25R0; April 23, 2014
- ECO–5018–48R1; September 3, 2015
- Holtec 72.48 Screening 1136R1; Screening for ECO–5018–48, Revision 1; March 11, 2016
- HPP–2226–200; MPC Loading at Clinton; Revision 0
- HPP–2226–300; MPC Sealing at Clinton; Revision 0
- NEI 96–07, Appendix B; Guidelines for 10 CFR 72.48 Implementation; March 5, 2001

LIST OF ACRONYMS USED

^{o}F	Degree Fahrenheit
ACE	Apparent Cause Evaluation
ADAMS	Agencywide Document Access Management System
AR	Action Request
CA	Corrective Action
CAP	Corrective Action Program
CAPE	Corrective Action Program Evaluation
CFR	*Code of Federal Regulations*
CoC	Certificate of Compliance
DG	Diesel Generator
EAL	Emergency Action Level
EC	Engineering Change
EDG	Emergency Diesel Generator
FSAR	Final Safety Analysis Report
FZ	Fire Zone
GRA	Growth Rate Algorithm
HPCS	High Pressure Core Spray
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Issue Report
ISFSI	Independent Spent Fuel Storage Installation
LER	Licensee Event Report
LORT	Licensed Operator Requalification Training
MOE	Method of Evaluation
MPC	Multi-Purpose Canister
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
PI	Performance Indicator
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
RR	Reactor Recirculation
RT	Reactor Water Cleanup
SAT	Systems Approach to Training
SDP	Significance Determination Process
SRA	Senior Reactor Analyst
SL	Severity Level
SSC	System, Structure, and Component
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
USAR	Updated Safety Analysis Report
WO	Work Order