UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
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LISLE, IL 60532-4352

November 10, 2016

Mr. Bryan 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
05000461/2016003 AND 07201046/2016002

Dear Mr. Hanson:

On September 30, 2016, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Clinton Power Station. On October 14, 2016, the NRC inspectors discussed the results of this inspection with Mr. T. Stoner and other members of your staff. The enclosed report represents the results of this inspection.

Based on the results of this inspection, the NRC has identified five issues that were evaluated under the risk significance determination process as having very low safety significance (Green). The inspectors also evaluated one of the issues under the traditional enforcement process. The NRC determined that five 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 NRC Enforcement Policy. These NCVs are described in the subject inspection report. Further, the inspectors documented a licensee-identified violation which was determined to be of very low safety significance in this report. The NRC is treating this violation as an NCV consistent with Section 2.3.2 of the NRC Enforcement Policy.

If you contest the violations or significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555–0001, with copies to: (1) the Regional Administrator, Region III; (2) the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001; and (3) the NRC Resident Inspector at the Clinton Power Station.

In addition, if you disagree with the cross-cutting aspect assignment to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Clinton Power Station.

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

Sincerely,

/RA/

Karla Stoedter, Chief
Branch 1
Division of Reactor Projects

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

Enclosure:
IR 05000461/2016003; 07201046/2016002

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50–461; 72–1046
License No:	NPF–62
Report Nos:	05000461/2016003; 07201046/2016002
Licensee:	Exelon Generation Company, LLC
Facility:	Clinton Power Station
Location:	Clinton, IL
Dates:	July 1 through September 30, 2016
Inspectors:	W. Schaup, Senior Resident Inspector J. Wojewoda, Acting Resident Inspector J. Bozga, Reactor Inspector G. Edwards, Health Physicist N. Fields, Health Physicist C. Phillips, Project Engineer G. Roach, Senior Resident Inspector, Dresden S. Mischke, Resident Inspector, Illinois Emergency Management Agency
Approved by:	K. Stoedter, Chief Branch 1 Division of Reactor Projects

Enclosure

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SUMMARY

Inspection Report 05000461/2016003; 07201046/2016002; 07/01/2016 – 09/30/2016, Clinton Power Station, Unit 1; Maintenance Effectiveness, Operability Determinations and Functional Assessments, Plant Modifications, Follow-Up of Events and Notices of Enforcement Discretion and Other Activities.

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Additionally, the report covers the preoperational testing inspection of the Independent Spent Fuel Storage Installation at Clinton Power Station by the regional inspection staff. Five Green findings were identified by the inspectors. The findings involved Non-Cited Violations (NCVs), including one Severity Level IV violation of U.S. Nuclear Regulatory Commission (NRC) requirements. 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 February 4, 2015. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG–1649, "Reactor Oversight Process," dated July 2016.

Cornerstone: Initiating Events

- Green. The inspectors documented a self-revealed finding of very low safety significance and an associated non-cited violation (NCV) of Title 10 of the *Code of Federal Regulations* (CFR) Part 50, Appendix B, Criterion XVI, "Corrective Action," for the licensee's failure to take corrective actions to preclude repetition of a significant condition adverse to quality. After identifying intergranular stress corrosion cracking (IGSCC) on main steam flex hoses as the root cause of reactor coolant system (RCS) pressure boundary leakage in 2007 and concluding the leakage constituted a significant condition adverse to quality, the licensee's corrective action to prevent recurrence failed to prevent pressure boundary leakage at the same location in 2016. The licensee entered this issue into their corrective action program as action request (AR) 02670593. The affected hoses were replaced. The remaining hoses will be replaced in the next maintenance outage. The licensee is also developing a design change to address at least one of the three factors that contributes to IGSCC to ensure pressure boundary leakage does not occur at the main steam flex hoses.

  The inspectors determined that the licensee's failure to take corrective actions to preclude repetition for a significant condition adverse to quality was a performance deficiency. Specifically, the licensee's corrective action in 2007 to replace main steam flex hoses at 16 year intervals failed to preclude pressure boundary leakage due to IGSCC on the main steam flex hoses in 2016. 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 it was associated with the equipment performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. 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 the Initiating Events

cornerstone and determined to be of very low safety significance because the finding did not result in exceeding the RCS leak rate for a small loss of coolant accident (LOCA) and did not affect other systems used to mitigate a LOCA resulting in a total loss of their function. No cross cutting aspect was assigned because the performance deficiency was not indicative of current plant performance. (Section 4OA3)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance and an associated NCV of Technical Specification 3.7.6, "Main Turbine Bypass System," for the licensee's failure to meet the limiting conditions for operation and complete the associated required actions after making a deficient change to the turbine bypass valve surveillance testing frequency. Specifically, with the main turbine bypass system inoperable and without the Core Operating Limits Report (COLR) limits for thermal power, minimum critical power ratio (MCPR), and linear heat generation rate (LGHR) with the main turbine by pass system inoperable applied, thermal power was not reduced to less than 21.6 percent of rated thermal power within six hours. The licensee entered this issue into their corrective action program as AR 02690657. The licensee restored compliance by applying the COLR limits for reactor thermal power, MCPR and LGHR.

  The inspectors determined the failure to meet the limiting conditions for operation and complete the associated required actions prior to the end of the specified completion times was a performance deficiency. The performance deficiency was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was screened against the Mitigating Systems cornerstone and determined to be of very low safety significance because all of the associated questions in IMC 0609, Appendix A, were answered no. The inspectors determined this finding affected the cross-cutting area of human performance, in the aspect of change management, where leaders use a systematic process for evaluating and implementing change so that nuclear safety remains the overriding priority because the licensee's change management process was not fully utilized by senior management when evaluating and implementing a change to the turbine bypass valve surveillance testing frequency. (H.3) (Section 1R15)

- Severity Level IV. The inspectors identified a Severity Level IV NCV of 10 CFR 50.59 4(d)(1), "Changes, Tests, and Experiments," and an associated Green finding for the licensee's failure to perform a written evaluation which provided the bases for determining that changing the turbine bypass valve surveillance testing frequency from every 31 days, as specified in the Updated Safety Analysis Report, to once a year did not require a license amendment. The licensee has entered this issue into their corrective action program as AR 02720163. The licensee is currently evaluating the issue in accordance with their procedure for changes to the facility.

  The inspectors determined that the licensee's failure to perform a written evaluation to provide the basis for the determination that a change to the facility, a change to a procedure, or a change to a test or experiment did not require a license amendment was a performance deficiency. 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, it was associated with the procedure quality attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. 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 the Mitigating Systems cornerstone and determined to be of very low safety significance because all of the associated questions in IMC 0609, Appendix A, were answered no. Violations of 10 CFR 50.59 are dispositioned using the traditional enforcement process because they are considered to be violations that potentially impede or impact the regulatory process. The inspectors reviewed Section 6.1.d.2 of the NRC Enforcement Policy and determined this violation was Severity Level IV because the resulting changes were evaluated by the SDP as having very low safety significance. The inspectors determined this finding affected the cross-cutting area of human performance, in the aspect of consistent process, where individuals use a consistent, systematic approach to make decisions. The licensee made a decision to proceed with implementation of a change to the turbine bypass valve surveillance testing frequency after a plant oversight committee review in lieu of following their consistent, systematic process for evaluating changes to the USAR. (H.13) (Section 1R18)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a finding of very low safety significance and an NCV of 10 CFR 50.65 (b), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," for the licensee's failure to scope a non-safety related structure, system and component (SSC), whose function is used in one or more Emergency Operating Procedures (EOP) and whose failure could cause actuation of a safety-related system, into maintenance rule. Specifically, the licensee failed to scope the non-safety related fuel building pressure control function into their maintenance rule program. The licensee has entered this issue into their corrective action program as AR 02716300. The licensee is scoping the pressure control function of fuel building ventilation into maintenance rule.

  The inspectors determined that the licensee's failure to scope a non-safety related system whose function is used in one or more EOPs and whose failure caused the actuation of a safety-related system into maintenance rule 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 it affects the SSC and barrier performance attribute of the Barrier Integrity cornerstone and adversely affects the cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. 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 the Barrier Integrity cornerstone and determined to be of very low safety significance because the inspectors answered yes to the question "does the finding only represent a degradation of the radiological barrier function provided for the control room, or auxiliary building, or spent fuel pool, SBGT system (BWR)?". The inspectors determined this finding affected the cross-cutting area of human performance, in the aspect of avoiding complacency, where individuals recognize and plan for the possibility of mistakes, latent

issues, and inherent risk, even while expecting successful outcomes because the licensee identified water intrusion of the fuel building pressure sensing line was a longstanding, latent, known problem and failed to recognize and appropriately challenge how the function was scoped into maintenance rule. (H.12) (Section 1R12)

- Green. The inspectors identified a finding of very low safety significance and an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure of the licensee's design control measures to provide for the verifying or checking of the adequacy of design of the spent fuel pool liner. Specifically, calculations involving the liner had not been verified or checked to ensure the design basis requirements of American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Division 2, were included. The licensee documented this issue in the corrective action program as AR 02690744 and initiated actions to restore compliance.

  The inspectors determined that the failure of the calculation to include the design basis requirements was contrary to the design control requirements of 10 CFR Part 50 and was a performance deficiency. The performance deficiency was determined to be more than minor because if left uncorrected it could lead to a more significant safety concern if independent spent fuel storage installation loading was conducted. The inspectors determined the finding could be evaluated using the Significance Determination Process in accordance with IMC 0609, "The Significance Determination Process for Findings At-Power," Appendix A, Exhibit 3 – Barrier Integrity Screening Questions (Section D). Based on answering "No" to all the questions in Exhibit 3, Section D, the inspectors determined the finding to be of very low safety significance. The inspectors determined this finding affected the cross-cutting area of human performance in the aspect of design margin, where the organization operates and maintains equipment within design margins. Margins are carefully guarded and changed only through a systematic and rigorous process. Specifically, the licensee failed to ensure the SFP liner reflected the intended design margins of the design and licensing basis. (H.6) (Section 4OA5)

Licensee-identified Violations

Violations of very low safety or security significance or Severity Level IV that were identified by the licensee have been reviewed by the NRC. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and corrective action program tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

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

- On September 2, 2016, power was reduced to approximately 75 percent to perform a control rod pattern adjustment and to perform control rod, main steam isolation valve, turbine stop valve/combined intermediate valve and turbine control valve testing. The unit was returned to full power on September 6, 2016.

- On September 9, 2016, power was reduced to approximately 80 percent to prevent exceeding National Pollutant Discharge Elimination System thermal discharge limits to the lake. Reactor power was maintained between 76 percent and 94 percent until September 29, 2016, when the unit was returned to full power.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

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 Updated Safety Analysis Report (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 corrective action program (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:

- Emergency diesel generator ventilation;
- Switchyard; and
- Fire protection.

This inspection 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:

- Standby gas treatment train 'B' during maintenance on standby gas treatment train 'A'; and
- Division 2 emergency diesel generator during maintenance on Division 1 emergency diesel generator.

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.

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

b. Findings

No findings were identified.

.2 Semi-Annual Complete System Walkdown

a. Inspection Scope

On September 7, 2016, the inspectors performed a complete system alignment inspection of the low pressure core spray system to verify the functional capability of the system. This system was selected because it was considered both safety significant and risk significant in the licensee's probabilistic risk assessment. The inspectors walked down the system to review mechanical and electrical equipment lineups; electrical power availability; system pressure and temperature indications, as appropriate; component labeling; component lubrication; component and equipment

cooling; hangers and supports; operability of support systems; and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding WOs was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the CAP database to ensure that system equipment alignment problems were being identified and appropriately resolved.

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

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

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

- Fire Zone CB–1d, Radiation-chemistry lab and laundry – elevation 737';
- Fire Zone CB–3a, Control building division 1 and 2 auxiliary electrical equipment – elevation 781';
- Fire Zone CB–3e and f, Division 1 and 2 inverter rooms – elevation 781';
- Fire Zone F–1b, High pressure core spray pump room – elevation 712'; and
- Fire Zone M–3, Screen house 'B' (South) fire pump room – elevation 699'.

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; 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.

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

b. Findings

No findings were identified.

.2 Annual Fire Protection Drill Observation (71111.05A)

a. Inspection Scope

On August 3, 2016, the inspectors observed an unannounced fire brigade activation for a hydrogen fire at the turbine building general area, elevation 762'. Based on this observation, the inspectors evaluated the readiness of the plant fire brigade to fight fires. The inspectors verified that the licensee staff identified deficiencies, openly discussed them in a self-critical manner at the drill debrief, and took appropriate corrective actions. Specific attributes evaluated were:

- proper wearing of turnout gear and self-contained breathing apparatus;
- proper use and layout of fire hoses;
- employment of appropriate firefighting techniques;
- sufficient firefighting equipment brought to the scene;
- effectiveness of fire brigade leader communications, command, and control;
- search for victims and propagation of the fire into other plant areas;
- smoke removal operations;
- utilization of pre-planned strategies;
- adherence to the pre-planned drill scenario; and
- meeting drill objectives.

These activities constituted one annual fire protection inspection sample 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:

- Residual heat removal pump 'B' room.

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 August 17, 2016, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification training. The inspectors verified that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

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

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

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 September 2, 2016, the inspectors observed control room operators perform a down power to support a rod sequence exchange. On September 5, 2016, the inspectors

observed the control room operators return the unit to near rated thermal power. These activities required heightened awareness and were 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 inspection constituted one quarterly licensed operator heightened activity/risk sample as defined in IP 71111.11–05.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations

a. Inspection Scope

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

- 10 CFR 50.65 (a)(3) periodic assessment of maintenance rule program; and
- fuel building ventilation 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 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 inspection constituted two quarterly maintenance effectiveness samples as defined in IP 71111.12–05.

b. Findings

Failure to Scope Fuel Building Ventilation Pressure Control into Maintenance Rule

Introduction: The inspectors identified a finding of very low safety significance (Green) and an associated non-cited violation (NCV) of 10 CFR 50.65 (b), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," for the licensee's failure to scope a non-safety related structure, system and component (SSC), whose function is used in one or more Emergency Operating Procedures (EOP) and whose failure could cause actuation of a safety-related system, into maintenance rule. Specifically, the licensee failed to scope the non-safety related fuel building pressure control function into their maintenance rule program.

Description: On February 13, 2016, during routine venting of the drywell, Fuel Building Ventilation (VF) Exhaust Fan "A" tripped off due to indicated high secondary containment vacuum. Fuel Building Exhaust Fan "B" auto-started due to the degraded vacuum but was unable to maintain secondary containment vacuum greater than 0.25 inch water gauge as required by the TS. The licensee declared secondary containment inoperable and entered the limiting condition for operation action statement to restore secondary containment to an operable status within four hours. The licensee manually actuated the Standby Gas Treatment System (VG) in order to restore secondary containment to operable. The licensee performed an apparent cause evaluation (ACE) and determined that ice formed in the VF sensing line to atmosphere, resulting in the high differential pressure in secondary containment that tripped the fans.

The licensee previously identified the VF sensing line design was susceptible to moisture intrusion in late 2007. However, moisture intrusion events prior to 2007 had not resulted in the loss of secondary containment. Four years later, the licensee was required to manually actuate the VG system after both VF fans tripped due to moisture intrusion. The licensee generated maintenance actions to blow down the sensing line and replace the line filter following this event. In 2014, the licensee noted additional water intrusion issues and created another maintenance task to drain the sensing line on a periodic basis.

The inspectors reviewed the maintenance performed on VF and asked the licensee how the pressure function of VF was monitored under the maintenance rule to account for the multiple failures. The inspectors found that the pressure control function of VF, which is used to maintain secondary containment vacuum during normal operation, was never scoped into maintenance rule, and thus, the effectiveness of maintenance associated with the pressure control function was not monitored. The inspectors reviewed the questions for scoping SSCs into maintenance rule and determined the VF pressure control function is a non-safety related function whose failure could cause a reactor scram or actuation of safety related system. Section 8.2.1.5 of NUMARC 93–01,

Revision 4A, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," provides additional maintenance rule guidance stating, in part,

> "Licensees should consider the following SSCs to be within the scope of the rule: SSCs whose failure has caused a reactor scram or actuation of safety-related system at their site."

In addition to the NUMARC guidance, the inspectors also reviewed the Federal Register, Volume 56, No. 132, dated July 10, 1991, which states,

> "The objective of the final rule is to require the monitoring of the overall continuing effectiveness of licensee maintenance programs to ensure that: (1) Safety related and certain non-safety related structures, systems, and components are capable of performing their intended functions; and (2) for non-safety related equipment, failures will not occur which prevent the fulfillment of safety-related functions, and failures resulting in scrams and unnecessary actuations of safety related systems are minimized."

The VG system is a standby, safety related system that was manually actuated in February 2016 to maintain secondary containment vacuum after VF failed to perform its pressure control function. Therefore, by not monitoring the effectiveness of maintenance on the pressure control function of VF, the licensee failed to meet the intent of the maintenance rule that non-safety related equipment failures should minimize unnecessary actuations of safety related systems.

The inspectors discussed this issue with the licensee who documented the issue in AR 02716300. As part of the corrective action, the licensee assessed the scoping of the pressure control function of VF into maintenance rule. A Maintenance Rule Expert Panel was held on October 6, 2016, during which the licensee determined that the pressure control function of VF should be scoped into maintenance rule based on the following:

- The VF system is utilized in EOP–8, "Secondary Containment Control." The non-safety related VF system and the safety related VG system are used for secondary pressure control depending on if radiation mitigation is needed. The licensee determined that the function was a non-safety related function that is used in one or more EOPs.

- Additionally, if the VF system fails to maintain secondary pressure, the safety related VG system can be manually initiated by the station operators to maintain secondary containment negative pressure to atmosphere. Whenever the safety related VG system is manually or automatically initiated to maintain secondary pressure, the non-safety related VF system automatically isolates. This manual or automatic actuation of the VG system to maintain secondary containment is an actuation of a safety related system; therefore, failure of the non-safety related VF system will result in the actuation of the safety related VG system in order to maintain the TS required secondary containment differential pressure within the license requirements.

<u>Analysis</u>: The inspectors determined that the licensee's failure to scope a non-safety related system whose function is used in one or more EOPs and whose failure caused the actuation of a safety-related system into maintenance rule 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 it affects the SSC and barrier performance attribute of the Barrier Integrity cornerstone and adversely affects the cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. 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 the Barrier Integrity cornerstone and determined to be of very low safety significance because the inspectors answered yes to the question "does the finding only represent a degradation of the radiological barrier function provided for the control room, or auxiliary building, or spent fuel pool, SBGT system (BWR)?"

The inspectors determined that this finding affected the cross-cutting area of human performance, in the aspect of avoiding complacency, where individuals recognize and plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes. Specifically, the licensee identified water intrusion of the VF sensing line was a longstanding, latent, known problem and had multiple opportunities to recognize and appropriately challenge how the function was scoped into maintenance rule. (H.12)

Enforcement: Title 10 CFR 50.65 (b) states, in part, the scope of the monitoring program specified in paragraph (a)(1) of this section shall include safety related and non-safety related structures, systems, and components, as follows: (b)(2)(i) non-safety related structures, systems, or components that are relied upon to mitigate accidents or transients or are used in plant emergency operating procedures and (b)(2)(iii) non-safety related structures, systems, or components whose failure could cause a reactor scram or actuation of a safety-related system.

Contrary to the above, since the implementation of Maintenance Rule in 1996, the licensee failed to scope a non-safety related SSC, the secondary containment pressure control function of VF, that is used in plant EOPs and whose failure could cause actuation of a safety-related system into maintenance rule.

As corrective actions, the licensee is scoping the pressure control function of VF into maintenance rule. Because the violation was of very low safety significance and was entered into the licensee's corrective action program as AR 02716300, this violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. **(NCV 05000461/2016003–01: Failure to Scope Fuel Building Ventilation Pressure Control into Maintenance Rule)**

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:

- planned yellow during division 1 emergency diesel generator system outage window;
- emergent unplanned work on the secondary containment fuel building railroad bay inner door seal;
- emergent planned work in the switchyard to support the Brokaw line outage; and
- emergent yellow due to severe thunderstorm watch and entry into the tornado/high winds off normal procedure.

These activities were selected based on their potential risk significance relative to the Reactor Safety Cornerstones. As applicable for each activity, the inspectors verified 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.

These maintenance risk assessments and emergent work control activities constituted four 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:

- AR 02694534: 1SD–11 shutdown service water pump room ‘A’ watertight door does not operate;
- AR 02689812: division 1 emergency diesel generator bearing grease seepage;
- AR 02690657: NRC question COLR thermal limits removal;
- AR 02697915: Inner Railroad Bay Door degraded seal; and
- AR 02713680: NRC question on VG and service air pipe clearance in low pressure core spray room.

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 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 the licensee was identifying and correcting any deficiencies associated with operability evaluations.

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

b. Findings

Technical Specification Allowed Outage Time Exceeded for Main Turbine Bypass System

Introduction: The inspectors identified a finding of very low safety significance (Green) and an NCV of TS 3.7.6, "Main Turbine Bypass System," for failing to meet the limiting conditions for operation and take the required actions within the completion times. Specifically, with the main turbine bypass system inoperable and without the Core Operating Limits Report (COLR) limits for thermal power, minimum critical power ratio (MCPR), and linear heat generation rate (LGHR) with the main turbine by pass system inoperable applied, thermal power was not reduced to less than 21.6 percent of rated thermal power within six hours.

Description: On June 3, 2016, the control room staff was performing station procedure CPS 9072.01, "Steam Bypass Valve Tests," Revision 33, to satisfy TS Surveillance Requirement (SR) 3.7.6.1. The purpose of this surveillance requirement was to verify one complete cycle of each main turbine bypass valve every 31 days. While testing turbine bypass valve 4, the valve went from full shut to full open rapidly and remained full open. The operators entered the loss of feedwater off normal procedure and lowered reactor power to maintain power below rated thermal power. The maintenance department removed the amplifier test card in the test circuitry and turbine bypass valve 4 went shut.

On June 4, 2016, after troubleshooting efforts to determine if the testing could be completed were unsuccessful, the operations staff declared main turbine bypass valve 4 inoperable.

On June 6, 2016, operations documented in the station operations log that the 25 percent grace period for performing a SR allowed by limiting condition for operation (LCO) SR 3.0.2 had expired for completing the turbine bypass valve surveillance started on June 3, 2016. The operations staff documented that since turbine bypass valves 5 and 6 were not tested and the required testing interval for SR 3.7.6.1 was exceeded the valves were inoperable. With two or more turbine bypass valves inoperable, the main turbine bypass system is inoperable; however, the LCO for TS 3.7.6 can be met either by applying the COLR limits for reactor thermal power, MCPR and LGHR with the turbine bypass system inoperable or reducing power to less than 21.6 percent. The licensee chose to apply the COLR limits to meet the LCO.

On July 7, 2016, at 12:28 p.m., operations documented in the station operations log that due to a change per the surveillance frequency control program, the frequency of TS SR 3.7.6.1 was changed from every 31 days to every 12 months. The operations logs stated the last satisfactory performance of TS SR 3.7.6.1 was on April 28, 2016, and the surveillance frequency change makes the new surveillance due date April 28, 2017;

therefore, turbine bypass valves 5 and 6 have not exceeded the surveillance testing interval making the two valves operable. The licensee also removed the previously imposed COLR limits.

The following day during their review of the operations log, the resident staff questioned the operations staff as to why the turbine bypass valves were operable without completing the required testing. Technical Specification SR 3.0.1 states, in part, that failure to perform a surveillance within the specified frequency shall be failure to meet the LCO except as provided in SR 3.0.3. The TS Basis for SR 3.0.1 also states that surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to operable status. The inspectors determined that the licensee had failed to complete the TS required testing in accordance with SR 3.0.2 on June 6, 2016. After the challenge by the resident staff, the licensee applied the COLR limits for reactor thermal power, MCPR and LGHR to comply with TS 3.7.6 at 12:31 p.m. on July 8, 2016.

The inspectors reviewed the licensee's basis for changing the surveillance testing frequency and the decision to declare the bypass valves operable. The inspectors found the frequency change was approved by the plant oversight review committee (PORC). The PORC also discussed how the change would be implemented. After the inspectors challenged licensee management on the appropriateness of the surveillance testing frequency change, the managers determined they had not fully utilized Clinton Power Station's change management process. Specifically, licensee management had not engaged all the stake holders, primarily the licensed operators, when evaluating and implementing the change.

Analysis: The inspectors determined the failure to meet the LCO and not completing the associated required actions prior to the end of the completion times was a performance deficiency. 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, it was associated with the equipment performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. 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 the Mitigating Systems cornerstone and determined to be of very low safety significance because all of the associated questions in IMC 0609, Appendix A, were answered no.

The inspectors determined this finding affected the cross-cutting area of human performance, in the aspect of change management, where leaders use a systematic process for evaluating and implementing change so that nuclear safety remains the overriding priority. Specifically, the licensee's change management process was not fully utilized by senior management when they failed to engage all the stake holders, primarily the licensed operators, when evaluating and implementing a change to the turbine bypass valve surveillance testing frequency. (H.3)

Enforcement: Clinton Power Station TS 3.7.6, "Main Turbine Bypass System," requires either the main turbine bypass system shall be operable or the COLR limits for thermal power, MCPR and LHGR with the main turbine bypass system inoperable are in place with thermal power greater than or equal to 21.6 percent. Technical Specification 3.7.6

requires that if the LCO is not met that either the main turbine bypass system will be restored to operable or the COLR limits for thermal power, MCPR and LHGR with the main turbine bypass system inoperable are in place within two hours. If these required actions cannot be completed within two hours, then the unit must reduce thermal power to less than 21.6 percent within four hours.

Contrary to the above, between 12:28 p.m. on July 7, 2016 and 12:31 p.m. on July 8, 2016, the main turbine bypass system was inoperable and the COLR limits for thermal power, MCPR and LHGR with the main turbine bypass system inoperable were not applied. Additionally, when not meeting the LCO the required actions within two hours, the unit thermal power was not reduced to less than 21.6 percent within four hours.

As corrective actions, at 12:31 p.m. on July 8, 2016, the COLR limits for thermal power, MCPR and LHGR with the main turbine bypass system inoperable were applied by the licensee to restore compliance with the TS. Because the violation was of very low safety significance and was entered into the licensee's corrective action program as AR 02690657, this violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. **(NCV 05000461/2016003–02: Exceeded Technical Specification Allowed Outage Time for Main Turbine Bypass System)**

1R18 Plant Modifications (71111.18)

.1 Plant Modifications

a. Inspection Scope

The inspectors reviewed the following modification:

- Surveillance frequency change for main turbine bypass valves.

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

This inspection constituted one plant modification sample as defined in IP 71111.18–05.

b. Findings

Failure to Perform a 50.59 Screening for Changing the Frequency of Exercising the Turbine Bypass Valves

Introduction: The inspectors identified a Severity Level IV NCV of 10 CFR 50.59 4(d)(1), "Changes, Tests, and Experiments," and an associated Green finding for the licensee's failure to perform a written evaluation which provided the bases for the determination that changing the turbine bypass valve surveillance testing frequency from every 31 days, as specified in the USAR, to once a year did not require a license amendment.

Description: As discussed in Section 1R15 of this report, on June 6, 2016, the licensee documented turbine bypass valves 5 and 6 were inoperable since the surveillance test required by TS SR 3.7.6.1 had not been performed within the specified frequency (31 days) plus or minus 25 percent. On July 7, 2016, operations documented in the station operations log that due to a change of the surveillance frequency control program, SR 3.7.6.1 was changed from a 31 day frequency to a 12 month frequency. The log entry stated SR 3.7.6.1 was last performed on April 28, 2016. Due to the frequency change, the log entry stated the new due date as April 28, 2017. Based upon the frequency change, operations personnel concluded turbine bypass valves 5 and 6 had not exceeded the new required testing interval for SR 3.7.6.1 and declared the two valves operable.

After reviewing the operation logs on the following day, the resident staff questioned the operations staff as to why turbine bypass valves 5 and 6 were operable without completing the required testing. The inspectors also reviewed the licensee's approved evaluation and documentation developed to support the TS surveillance frequency change. During the inspectors' review, the inspectors noted the USAR had a requirement to exercise the main turbine bypass valves every 31 days. The inspectors requested the 50.59 screening and evaluation, if one had been determined to be required, for review. The licensee informed the inspectors that a 50.59 screening had not been performed and that they had only discussed the need for a USAR update during the PORC review.

A licensee may make changes to the facility as described in the final safety analysis report. However, the NRC requires licensees to maintain records of the changes, including a written evaluation which provides the bases for the determination that the change, test, or experiment does not require a license amendment. The inspectors determined that the licensee failed to maintain a record that provided the evaluation and basis for the determination that a license amendment was not required since the licensee had not performed any evaluation. This issue was documented in AR 02720163, and the licensee is currently performing a 50.59 screening.

Analysis: The inspectors determined that the licensee's failure to perform a written evaluation to provide the basis for the determination that a change to the facility, a change to a procedure, or a change to a test or experiment did not require a license amendment was a performance deficiency. 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 it was associated with the procedure quality attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems

that respond to initiating events to prevent undesirable consequences. 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 the Mitigating Systems cornerstone and determined to be of very low safety significance (Green) because all of the associated questions in IMC 0609, Appendix A, were answered no.

Violations of 10 CFR 50.59 are dispositioned using the traditional enforcement process because they are considered to be violations that potentially impede or impact the regulatory process. The inspectors reviewed Section 6.1.d.2 of the NRC Enforcement Policy and determined this violation was Severity Level IV violation because the resulting changes were evaluated by the SDP as having very low safety significance.

The inspectors determined this finding affected the cross-cutting area of human performance, in the aspect of consistent process, where individuals use a consistent, systematic approach to make decisions. The licensee made a decision to proceed with implementing a turbine bypass valve surveillance testing frequency change after a PORC review in lieu of following their consistent, systematic process for evaluating changes to the USAR. (H.13)

Enforcement: Title 10 CFR 50.59 4(d)(1) states, in part, that the licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments. 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 amendment.

Contrary to the above, on July 7, 2016, the licensee failed to maintain a record of the written evaluation that provided the bases for the determination that a change to the frequency for exercising the turbine bypass valves from every 31 days, as specified in the USAR, to once a year did not require a license amendment.

Specifically, the licensee failed to perform the 50.59 evaluation for this change. As corrective actions, the licensee is performing the required evaluation and has documented the issue in the corrective action program as ARs 02720163 and 02685337. In accordance with the Enforcement Policy, Section 6.1.d.2, the violation was classified as a Severity Level IV violation because the underlying technical issue was of very low safety significance. Because this violation was of very low safety significance, was not repetitive or willful, and was entered into the licensee’s corrective action program as ARs 02720163 and 02685337, it is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000461/2016003–03: Failure to Perform a 50.59 Screening for Changing the Frequency of Exercising the Turbine Bypass Valves)**

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

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

- testing of 1RIXPR009B control room air intake 'B' radiation monitor;
- testing of the control room 'B" train chiller;
- testing of the secondary containment railroad bay inner door seal;
- testing of 0FP05SA fire protection valve;
- testing of the division 1 emergency diesel generator; and
- testing of shutdown service water valve 1SX013D.

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 the problems were being corrected commensurate with their importance to safety.

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

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

a. Inspection Scope

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

- CPS 9080.03, “Diesel Generator 1c Operability – Manual & Quick Start Operability” (inservice test);
- CPS 9051.01, “HPCS Pump & HPCS Water Leg Pump Operability” (inservice test);
- CPS 9015.01, “Standby Liquid Control System Operability” (inservice test); and
- CPS 9069.01, “Shutdown Service Water Operability Test” (inservice test).

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 set points 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.

This inspection constituted four in-service test samples as defined in IP 71111.22, Sections –02 and–05.

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 August 31, 2016, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the main control room simulator and 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 corrective action program. 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 71114.06–06.

b. Findings

No findings were identified.

.2 Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of a routine licensee emergency drill on September 13, 2016, 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 corrective action program. 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 71114.06–06.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstones: Occupational Radiation Safety and Public Radiation Safety

2RS7 Radiological Environmental Monitoring Program (71124.07)

.1 Site Inspection

a. Inspection Scope

The inspectors walked down select air sampling stations and dosimeter monitoring stations to determine whether they were located as described in the Offsite Dose Calculation Manual (ODCM) and to determine the equipment material condition.

The inspectors reviewed calibration and maintenance records for select air samplers, dosimeters, and composite water samplers to evaluate whether they demonstrated adequate operability of these components.

The inspectors assessed whether the licensee had initiated sampling of other appropriate media upon loss of a required sampling station.

The inspectors observed the collection and preparation of environmental samples from select environmental media to determine if environmental sampling was representative of the release pathways specified in the ODCM and if sampling techniques were in accordance with procedures.

The inspectors assessed whether the meteorological instruments were operable, calibrated, and maintained in accordance with guidance contained in the Final Safety Analysis Report, U.S. Nuclear Regulatory Commission Regulatory Guide 1.23, “Meteorological Monitoring Programs for Nuclear Power Plants,” and licensee procedures. The inspectors assessed whether the meteorological data readout and recording instruments were operable.

The inspectors evaluated whether missed and/or anomalous environmental samples were identified and reported in the annual environmental monitoring report. The inspectors selected events that involved a missed sample, inoperable sampler, lost dosimeter, or anomalous measurement to determine if the licensee had identified the cause and had implemented corrective actions. The inspectors reviewed the licensee’s assessment of any positive sample results and reviewed any associated radioactive effluent release data that was the source of the released material.

The inspectors selected structures, systems, or components that involve or could reasonably involve a credible mechanism for licensed material to reach ground water, and assessed whether the licensee had implemented a sampling and monitoring program sufficient to detect leakage to ground water.

The inspectors evaluated whether records important to decommissioning, as required by Title 10 of the *Code of Federal Regulations*, Part 50.75(g), were retained in a retrievable manner.

The inspectors reviewed any significant changes made by the licensee to the ODCM as the result of changes to the land census, long-term meteorological conditions, or

modifications to the sampler stations since the last inspection. The inspectors reviewed technical justifications for any changed sampling locations to evaluate whether the licensee performed the reviews required to ensure that the changes did not affect its ability to monitor the impacts of radioactive effluent releases on the environment.

The inspectors assessed whether the appropriate detection sensitivities with respect to the ODCM where used for counting samples. The inspectors reviewed the quality control program for analytical analysis.

The inspectors reviewed the results of the licensee's inter-laboratory comparison program to evaluate the adequacy of environmental sample analyses performed by the licensee. The inspectors assessed whether the inter-laboratory comparison test included the media/nuclide mix appropriate for the facility. The inspectors reviewed the licensee's determination of any bias to the data and the overall effect on the radiological environmental monitoring program.

These inspection activities constituted one complete sample as defined in IP 71124.07–05.

b. Findings

No findings were identified.

.2 Groundwater Protection Initiative Implementation

a. Inspection Scope

The inspectors reviewed monitoring results of the groundwater protection initiative (GPI) to evaluate whether the licensee had implemented the program as intended and to assess whether the licensee had identified and addressed anomalous results and missed samples.

The inspectors evaluated the licensee's implementation of the minimization of contamination and survey aspects of the GPI and the Decommissioning Planning Rule requirements in 10 CFR 20.1406, and 10 CFR 20.1501.

The inspectors reviewed leak and spill events and 10 CFR 50.75(g) records and assessed whether the source of the leak or spill was identified and appropriately mitigated.

The inspectors assessed whether unmonitored leaks and spills where evaluated to determine the type and amount of radioactive material that was discharged. The inspectors assessed whether the licensee completed offsite notifications in accordance with procedure.

The inspectors reviewed evaluations of discharges from onsite contaminated surface water bodies and the potential for ground water leakage from them. The inspectors assessed whether the licensee properly accounted for these discharges as part of the effluent release reports.

The inspectors assessed whether on-site ground water sample results and descriptions of any significant on-site leaks or spills into ground water were documented in the

Annual Radiological Environmental Operating Report or the Annual Radiological Effluent Release Report.

The inspectors determined if significant new effluent discharge points where updated in the ODCM and the assumptions for dose calculations were updated as needed.

These inspection activities constituted one complete sample as defined in IP 71124.07–05.

b. Findings

No findings were identified.

.1 Problem Identification and Resolution

a. Inspection Scope

The inspectors assessed whether problems associated with the radiological environmental monitoring program were being identified by the licensee at an appropriate threshold and were properly addressed for resolution. The inspectors assessed the appropriateness of the corrective actions for a selected sample of problems documented by the licensee that involved the radiological environmental monitoring program.

These inspection activities constituted one complete sample as defined in IP 71124.07–05.

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 Safety System Functional Failures

a. Inspection Scope

The inspectors sampled licensee submittals for the Safety System Functional Failures performance indicator (PI) for the period from the first quarter 2015 through the second quarter 2016. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99–02, “Regulatory Assessment Performance Indicator Guideline,” Revision 7, dated August 31, 2013, and NUREG–1022, “Event Reporting Guidelines 10 CFR 50.72 and 50.73" definitions and guidance were used. The inspectors reviewed the licensee’s operator narrative logs, operability assessments, maintenance rule records, maintenance work orders, issue reports, event reports and NRC Integrated Inspection Reports for the period of January 1, 2015, through June 30, 2016, 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.

This inspection constituted one safety system functional failures sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.2 Reactor Coolant System Specific Activity

a. Inspection Scope

The inspectors sampled licensee submittals for the reactor coolant system specific activity PI for Clinton Power Station for the period from the third quarter of 2015 through the second quarter of 2016. The inspectors used PI definitions and guidance contained in NEI Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 2013, to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee's reactor coolant system chemistry samples, technical specification requirements, issue reports, event reports and NRC Integrated Inspection Reports 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. In addition to record reviews, the inspectors observed a chemistry technician obtain and analyze a reactor coolant system sample. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one reactor coolant system specific activity sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.3 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors sampled licensee submittals for the Occupational Exposure Control Effectiveness PI for the period from the third quarter of 2015 through the second quarter of 2016. The inspectors used PI definitions and guidance contained in NEI Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 2013, to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee's assessment of the PI for occupational radiation safety to determine if the indicator related data was adequately assessed and reported. To assess the adequacy of the licensee's PI data collection and analyses, the inspectors discussed with radiation protection staff the scope and breadth of its data review and the results of those reviews. The inspectors independently reviewed electronic personal dosimetry dose rate, accumulated dose alarms, dose reports, and the dose assignments for any intakes that occurred during the time period

reviewed to determine if there were potentially unrecognized occurrences. The inspectors also conducted walkdowns of numerous locked high and very high radiation area entrances to determine the adequacy of the controls in place for these areas. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one occupational exposure control effectiveness sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.4 Radiological Effluent Technical Specification/Offsite Dose Calculation Manual Radiological Effluent Occurrences

a. Inspection Scope

The inspectors sampled licensee submittals for the radiological effluent Technical Specification/Offsite Dose Calculation Manual radiological effluent occurrences PI for the period from the third quarter of 2015 through the second quarter of 2016. The inspectors used PI definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 2013, to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee's issue report database and selected individual reports generated since this indicator was last reviewed to identify any potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose. The inspectors reviewed gaseous effluent summary data and the results of associated offsite dose calculations for selected dates to determine if indicator results were accurately reported. The inspectors also reviewed the licensee's methods for quantifying gaseous and liquid effluents and determining effluent dose. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one Radiological Effluent Technical Specification/Offsite Dose Calculation Manual radiological effluent occurrences 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 corrective action program 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 Annual Follow-up of Selected Issues: Part 21 Report—STAAD Software Errors

a. Inspection Scope

The inspectors selected the following condition report for in-depth review:

- Action Request 2503008; Interim Part 21 Report – STAAD Software Errors; dated May 19, 2015.

The inspectors verified the following attributes during their review of the licensee's corrective actions for the above condition report:

- complete and accurate identification of the problem in a timely manner commensurate with its safety significance and ease of discovery;
- consideration of the extent of condition, generic implications, common cause, and previous occurrences;
- evaluation and disposition of operability/functionality/reportability issues;
- classification and prioritization of the resolution of the problem commensurate with safety significance;
- completion of corrective actions in a timely manner commensurate with the safety significance of the issue;
- effectiveness of corrective actions taken to preclude repetition; and
- evaluate applicability for operating experience and communicate applicable lessons learned to appropriate organizations.

The inspectors discussed the corrective actions and associated evaluations with licensee personnel.

Additionally, the inspectors performed a review to assess the adequacy of the concrete structure inside containment that was used to support the placement of the reactor vessel head during refueling outages. The inspectors reviewed USAR Section 12 to determine the structural code of record (American Concrete Institute 318-63) for the concrete structure inside containment at Elevation 711'-6". The inspectors also reviewed the licensee's classification for the load due to the placement of the reactor pressure vessel head during the refueling outage being considered a live load. The inspectors reviewed the design calculations for the concrete structure inside containment at Elevation 711'-6" which used the weight of the reactor vessel during initial construction of the plant for the design of the structural members. Based upon this, the inspectors concluded that the supports were adequate to support the weight of the

reactor vessel head. The inspectors reviewed the location of decay heat removal systems and their location with respect to the location of the reactor pressure vessel head during the refueling outage to ensure that the integrity of the heat removal systems would not be challenged if a seismic event were to occur while the reactor vessel head was within the laydown area.

This review constituted one in-depth problem identification and resolution inspection sample as defined in IP 71152.

b. Findings

No findings were identified.

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

.1 (Closed) Licensee Event Report 05000461/2016–006–00: Missed Surveillance Results in a Condition Prohibited by Technical Specifications

a. Inspection Scope

On June 10, 2016, the licensee submitted licensee event report (LER) 2016–006–00, which discussed failing to perform TS SR 3.3.2.1.2, functional testing of the 4-notch control rod withdrawal limit of the rod withdrawal limiter within one hour of resetting the high power set point (HPSP) during a power reduction if it has not been completed within the previous 92 days.

The inspectors independently reviewed the event, the TS, and station procedures. The inspectors determined that rod withdrawal limiter test had not been performed within one hour of resetting the HPSP because a senior reactor operator incorrectly determined the test did not need to be performed. This resulted in a licensee identified NCV of TS 3.3.2.1. The details of this violation are documented in Section 4OA7 of this report.

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

b. Findings

No findings were identified.

.2 (Closed) Licensee Event Report 05000461/2016–007–00: Main Steam Line Flex Hose Intergranular Stress Corrosion Cracking Identified During Refueling Outage

a. Inspection Scope

On May 17, 2016, with the plant in Mode 4 during a refueling outage, a system engineer identified water leaking from flexible hoses located on elbows of main steam line ‘B’ and ‘C’. These hoses are utilized as part of the main steam line flow instrumentation. Leakage from these hoses is pressure boundary leakage as described in TS. The licensee determined that the failure of the hoses was due to intergranular stress corrosion cracking (IGSCC) and was identical to an event that occurred in 2007.

The licensee replaced the damaged hoses and evaluated the hoses identified as part of the extent of condition as satisfactory for continued use. All hoses were tested as part of the post refueling outage reactor coolant system pressure test and successfully passed. The licensee documented this issue in AR 02670593.

The inspectors reviewed the event and identified a violation. The details of this review are documented below. This LER is closed.

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

b. Findings

Failure to Prevent Recurrence of a Significant Condition Adverse to Quality

Introduction: The inspectors documented a self-revealed finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the licensee's failure to take corrective actions to preclude repetition of a significant condition adverse to quality. Specifically, after identifying IGSCC on main steam line flex hoses as a significant condition adverse to quality and the root cause for pressure boundary leakage which resulted in a TS required plant shut down in 2007, the licensee's corrective action to prevent recurrence failed to prevent pressure boundary leakage at the same location in 2016.

Description: On May 17, 2016, a system manager performing a general walkdown in the drywell during the C1R16 refueling outage identified water leaking from main steam flex hoses 1B21–D372C and 1B21–D372E associated with the 'B' and 'C' main steam instrument lines. Leakage from these flexible hoses is pressure boundary leakage as described in the plant's TS. The licensee completed an extent of condition on all flexible hoses in the drywell and discovered no additional leakage.

The inspectors began their review with a similar event that occurred in 2007. In that instance, pressure boundary leakage occurred at 1B21–D372E and increased to the point that a reactor shutdown was required due to nearing reactor coolant system operational leakage limits. The licensee documented the 2007 event in AR 00641375, classified the AR document as a significant condition adverse to quality, performed a root cause evaluation for the event, and determined that the flex hoses had failed due to IGSCC. This type of cracking occurs when susceptible materials, like austenitic stainless steel, are coupled with stress applied in a corrosive environment. As a corrective action the licensee replaced all in-service flexible hoses that were susceptible to IGSCC. The licensee also established preventative maintenance activities to replace the susceptible hose assemblies every 16 years as a corrective action to prevent recurrence (CAPR).

The licensee performed a root cause evaluation for the 2016 event to determine why the flexible hoses failed again. The licensee's root cause determined that both 1B21–D372C and 1B21–D372E had failed because the corrective action to prevent recurrence assigned as part of the 2007 root cause evaluation failed to eliminate the factors that contribute to IGSCC. The inspectors independently reviewed the evaluation and supporting documentation and agreed with the licensee's assessment.

The licensee's immediate actions included replacing 1B21–D372C and 1B21–D372E and verifying there was no leakage at any flex hose locations in the drywell during the reactor pressure vessel test performed at the end of the C1R16 outage. The licensee is developing and will install a design change that eliminates or reduces at least one of the factors that contribute to IGSCC in the future.

Analysis: The inspectors determined that the licensee's failure to take corrective actions to preclude repetition of IGSCC induced pressure boundary leakage (a significant condition adverse to quality) 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 if left uncorrected pressure boundary leakage could become a more significant concern. Specifically, pressure boundary leakage is not allowed by TS and any leakage requires the plant to be shutdown. 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 the Initiating Events cornerstone and determined to be of very low safety significance because the finding did not result in exceeding the RCS leak rate for a small loss of coolant accident (LOCA) and did not affect other systems used to mitigate a LOCA resulting in a total loss of their function (e.g. Interfacing System LOCA).

No cross cutting aspect was assigned because the inspectors determined the performance deficiency was not indicative of current plant performance.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion XVI, states, in part, for significant conditions adverse to quality the measures taken shall assure that the cause of the condition is determined and corrective action taken to preclude repetition.

Contrary to the above, in 2007, the licensee failed to take corrective actions to preclude repetition. Specifically, the licensee's actions to replace main steam flex hoses every 16 years failed to preclude a repeat main steam line flex hose failure in 2016.

As corrective actions, the affected hoses were replaced and the remaining hoses will be replaced in the next maintenance outage. Additionally, the licensee is developing a design change for installation that will address at least one of the three factors that contributes to IGSCC to ensure pressure boundary leakage does not occur at the main steam flex hoses. Because the violation was of very low safety significance and was entered into the licensee's corrective action program as AR 02670593, this violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement policy. **(NCV 05000461/2016003–04: Failure to Prevent Recurrence of a Significant Condition Adverse to Quality)**

.3 (Closed) Licensee Event Report 05000461/2016–002–00: Trip of Fuel Building Ventilation Exhaust Fan Due to Moisture Formation Resulting in the Loss of Secondary Containment Vacuum

a. Inspection Scope

On April 13, 2016, the licensee submitted LER 2016-002-00 which discussed the VF Exhaust Fan "A" tripping due to indication of high SC vacuum during routine venting of the drywell. This resulted in SC vacuum degrading, eventually exceeding the TS limit of 0.25 inch vacuum water gauge.

At 2:08 a.m. on February 13, 2016, the VF Exhaust Fan “A” tripped and the “B” Exhaust Fan auto started. Secondary containment vacuum degraded to less than 0.25 inch vacuum water gauge and the licensee declared SC inoperable, entered LCO 3.6.4.1 and the associated four hour TS action to restore SC. The vacuum further degraded and the Secondary Containment Control EOP was entered when the vacuum reached 0 inch water gauge. The operators manually actuated the VG system at 2:56 a.m. and restored SC within TS limits at 2:57 a.m., within the required TS action time.

The licensee determined that ice formed in the low side of the VF “A” sensing line, causing an inaccurate SC vacuum reading, the Exhaust Fan “A” trip and the ultimate loss of secondary containment vacuum. Moisture had collected in the sensing line and froze due to the externally cold weather conditions. The licensee has subsequently modified the design of the sensing line to prevent moisture accumulation.

The inspectors reviewed the failure analyses, design basis documents, operating procedures, corrective actions and maintenance documents associated with the VF Exhaust Fan trip and loss of SC vacuum. During this review, the inspectors determined the licensee had not effectively controlled the performance of the VF system through maintenance due to maintenance rule scoping issues. This issue is documented in section 1R12 of this report. This LER is closed.

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

b. Findings

No findings were identified.

4OA5 Other Activities

.1 Review of 10 CFR 72.212 Evaluations at Operating Plants (60856.1)

a. Inspection Scope

(1) Control of Heavy Loads

An inspection was performed of the licensee’s control of heavy loads program that supports the initial loading of an Independent Spent Fuel Storage Installation (ISFSI) at the Clinton Power Station. The inspection included in-office and on-site reviews of plant design calculations including structural evaluations of the Fuel Handling Building (FHB) crane and crane support structure. The inspectors reviewed structural evaluations associated with the seismic design of the trolley girders, crane bridge girders, trolley rail, crane rails, crane rail clips, crane rail clip bolts, and crane support structure. The inspectors also reviewed inspection, testing, and maintenance documentation associated with the FHB crane, as well as documentation supporting the upgrade of the FHB crane to Single Failure Proof in accordance with NUREG-0554, “Single Failure Proof Cranes for Nuclear Power Plants,” and American Society of Mechanical Engineers (ASME) NOG–1–2004, “Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder),” which was previously documented in NRC Inspection Report 05000461/2016010; 07201046/2016001(DNMS).

The inspectors reviewed the structural and seismic qualification of the spent fuel pool (SFP) liner; SFP concrete structure; the concrete structure that supports decontamination of the spent fuel cask; the fuel handling building grade elevation; and fuel handling building airlock/truck bay concrete structure for cask placement to ensure compliance with the USAR requirements.

The inspectors, with the assistance of the Office of Nuclear Material Safety and Safeguards, reviewed the licensee's calculations for the vertical transfer configuration and fuel handling building structure to ensure the stability and structural integrity of an unrestrained stack-up. The dynamic nonlinear time history seismic and structural analysis was performed consistent with the NRC guidance in NRC Regulatory Issue Summary 2015–013, "Seismic Stability Analysis Methodologies for Spent Fuel Dry Cask Loading Stack-Up Configuration."

b. Findings

Spent Fuel Pool Liner Design Not Verified per Code

Introduction: A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified by the inspectors for the failure of the licensee's design control measures to provide for verifying or checking the adequacy of the design of the SFP liner.

Description: The process of safely moving spent nuclear fuel from the SFP into dry storage will place heavy loads on existing structures and components that need to be evaluated to ensure structural integrity.

The SFP liner is required to be Seismic Category I and Appendix B per the USAR Table 3.2–1, "Classification of Systems, Components and Structures." The SFP liner was designed to ASME Boiler and Pressure Vessel Code (BPVC), Section III, Division II, 1977, as delineated in USAR Table 3.8–4.

Engineering Change (EC) 394724, "ISFSI – Stability Analysis and Structural Impacts from Spent fuel Casks in the Fuel Building," Revision 0, includes evaluation/qualification of the cask loading pool liner. The cask loading pool liner must be able to safely withstand the applied loads imposed by the Holtec spent fuel transfer cask (HI-TRAC VW) during all design basis loading conditions. Section CC-3122 of ASME BPVC, Section III, Division II, states, in part, "The liner shall be designed to withstand the effects of imposed loads and to accommodate deformation of the concrete containment without jeopardizing leak tight integrity."

The inspectors reviewed Calculation No. HI–2146151, "Stability Analysis of HI-Trac VW in Fuel Cask Storage Pool and Structural Qualification of Liner at Clinton Power Station," Appendix C, Revision 4. In this calculation, the licensee used allowable bearing stress values from ASME BPVC, Section III, Appendix F, to evaluate the applied bearing stresses due to the placement of the spent fuel cask on the SFP liner. This evaluation was performed to establish that the applied stresses would be acceptable and the spent fuel cask would not penetrate the liner or cause overall slab instability. The licensee used acceptance limits from ASME BPVC, Section III, Appendix F.

The inspectors also reviewed 50.59 Evaluation No. CL–2015–E–015, "ISFSI-Stability Analysis and Structural Impacts from Spent Fuel Casks in the Fuel Building," dated March 15, 2016, to determine whether a 50.59 regulatory evaluation was performed for the use of ASME BPVC, Section III, Appendix F as a code of record. However, the inspectors identified that the change for the use of ASME BPVC, Section III, Appendix F in lieu of ASME BPVC, Section III, Division II was not explicitly reviewed in 50.59 Evaluation No. CL–2015–E–015.

Inspection Manual Chapter 0326, "Operability Determinations & Functionality Assessments for Conditions Adverse to Quality or Safety," dated December 3, 2015, Section C.11, states, in part, "When a degradation or nonconformance associated with piping or pipe supports is discovered, the licensee may use the criteria in Appendix F of Section III of the ASME Boiler and Pressure Vessel Code for operability determinations." In addition, the limits in Appendix F are "not intended to assure the safe operability or re-operability of the system either during or following the postulated event." The use of Appendix F is not consistent with the requirements stipulated in ASME BPVC, Section III, Division II, Section CC–3122.

Upon identification by the inspectors, the licensee documented this deficiency in AR 02690744, "Response to NRC Inquiry Did Not Answer Question," dated June 29, 2016. The licensee, in lieu of performing a regulatory evaluation to evaluate the usage of Appendix F, decided to revise the SFP liner structural and seismic analysis to demonstrate compliance with ASME BPVC, Section III, Division II.

Analysis: The inspectors determined that the licensee's failure to include the design basis requirements within Calculation No. HI–2146151 was contrary to the design control requirements of 10 CFR Part 50 and was a performance deficiency. In accordance with IMC 0612, "Issue Screening," Appendix B, dated September 7, 2012, the inspectors determined the performance deficiency was associated with the Barrier Integrity cornerstone attribute of Design Control. The performance deficiency was determined to be more than minor because if left uncorrected the performance deficiency could lead to a more significant safety concern for plant equipment if ISFSI loading was conducted. Specifically, compliance with ASME BPVC, Section III, Division II, for the SFP liner was required to ensure leak tight integrity of structures, systems, and components described in the USAR, when subjected to design loads as part of safe load handling of heavy loads near the SFP. The inspectors used IMC 0609, Attachment 4, "Initial Characterization of Findings," and Appendix A, "The Significance Determination Process for Findings at Power," dated June 19, 2012, to evaluate the performance deficiency. The finding was screened against the Barrier Integrity cornerstone and determined to be of very low safety significance (Green) because the inspectors answered no to the Exhibit 3 – Barrier Integrity Screening Questions applicable to the SFP.

The inspectors determined this finding affected the cross-cutting area of human performance in the aspect of design margin, where the organization operates and maintains equipment within design margins. Margins are carefully guarded and changed only through a systematic and rigorous process. Specifically, the licensee failed to ensure the SFP liner reflected the intended design margins of the design and licensing basis. (H.6)

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures be established to assure the applicable regulatory requirements and design basis are correctly translated into specifications, drawings, procedures, and instructions. It further requires, in part, that the design control measures provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculation methods, or by the performance of a suitable testing program. For Clinton, the design basis for the SFP liner is specified in ASME BPVC, Section III, Division II, as delineated in USAR Table 3.8–4.

Contrary to the above, on July 30, 2015 the licensee failed to ensure design control measures adequately provided for verifying or checking the adequacy of the design for the SFP liner.

Specifically, on July 30, 2015, in Calculation No. HI–2146151, "Stability Analysis of HI-TRAC VW in Fuel Cask Storage Pool and Structural Qualification of Liner at Clinton Power Station," Appendix C, Revision 4, the licensee failed to demonstrate "The liner shall be designed to withstand the effects of imposed loads and to accommodate deformation of the concrete containment without jeopardizing leak tight integrity," as specified in ASME BPVC, Section III, Division II. As corrective actions, the licensee decided to revise the SFP liner structural and seismic analysis to demonstrate compliance with ASME BPVC, Section III, Division II. This violation is being treated as an NCV, consistent with Section 2.3.2.a. of the Enforcement Policy, because it was of very low safety significance and was entered into the licensee's corrective action program as AR 02690744 **(NCV 05000461/2016003–05; NCV 07201046/2016002–01, Spent Fuel Pool Liner Design Not Verified per Code)**.

.2 Review of 10 CFR 72.212(b) Evaluations at Operating Plants (60856)

a. Inspection Scope

The inspectors evaluated the licensee's compliance with the requirements of 10 CFR 72.212 and 10 CFR 72.48. The inspection consisted of interviews with cognizant personnel and review of documentation. The licensee is required, as specified in 10 CFR 72.212(b)(1), to notify the NRC of the intent to store spent fuel at the Clinton Power Station ISFSI facility at least 90 days prior to the first storage of spent fuel. The licensee notified the NRC on April 14, 2016, of its intent to store spent fuel.

Written evaluations, prior to use, to establish that the terms, conditions, and specifications of a Certificate of Compliance (CoC) or an amended CoC have been met are required per 10 CFR 72.212(b)(5)(i). Written evaluations, prior to use, to demonstrate that the requirements of 10 CFR 72.104 have been met are required per 10 CFR 72.212(b)(5)(iii). "Clinton Power Station, Unit 1, 10 CFR 72.212 Evaluation Report for the HI-STORM FW MPC Storage System," Revision 0, dated August 30, 2016, documents these evaluations performed by the licensee prior to use of its 10 CFR Part 72 general license.

The inspectors reviewed and assessed the licensee's 10 CFR 72.212 Evaluation Report to verify that applicable reactor site parameters, such as fire and explosions, tornadoes, wind-generated missile impacts, seismic qualifications, lightning, flooding and temperature, had been evaluated for acceptability with bounding values specified in the Holtec HI-STORM FW Final Safety Analysis Report (FSAR) and associated analyses in accordance with 10 CFR 72.212(b)(6).

Per 10 CFR 72.212(b)(8), prior to use, the licensee is required to determine whether activities under the general license involve a change to the facility TS or require a license amendment. The licensee's 10 CFR 72.212 Evaluation Report documented this determination that a facility license amendment was necessary. The licensee received approval of License Amendment Request RS–16–019 from the NRC on August 17, 2016.

b. Findings

No findings were identified.

.3 Preoperational Testing of Independent Spent Fuel Storage Installations at Operating Plants (60854.1)

a. Inspection Scope

The inspectors reviewed documents, interviewed plant personnel, and performed in-field observations to assess the licensee's performance as it relates to the preoperational testing of an ISFSI. The inspectors reviewed ISFSI loading and unloading procedures to ensure they met the commitments and requirements as specified in the FSAR, the CoC, 10 CFR Part 72, and the TS. The inspectors verified that the loading and unloading procedures were prepared, reviewed, and approved in accordance with the licensee's administrative programs, and that the procedures ensure all required critical activities will be performed. The inspectors also reviewed the following documents: selected 72.48 reviews and selected 50.59 reviews related to ISFSI operations.

A review of corrective action reports related to ISFSI activities written during the inspection period indicated that the licensee was identifying and correcting conditions adverse to quality.

The inspectors performed an independent assessment that the licensee had adequately demonstrated its readiness to safely perform ISFSI loading and unloading operations.

(1) Dry Run Activities

During this inspection period, the licensee performed preoperational dry run activities in order to satisfy the ninth condition of the HI-STORM FW MPC Storage System CoC, Docket Number 072–01032, Amendment 0, Revision 1. NRC inspectors were onsite to observe dry run activities during the following dates: July 6–7, 2016, July 26–28, 2016, September 1, 2016, and September 8–9, 2016.

Specifically, the inspectors observed the licensee perform the following activities: performing pre-job briefs; welding and non-destructive examination of a mock-up multi-purpose canister (MPC); performing helium leak testing; conducting forced helium dehydration operations with a mock-up MPC; uploading, or transferring of an MPC simulator from the storage cask (HI-STORM) to the transfer cask (HI-TRAC); downloading, or transferring of the simulator from the HI-TRAC to the HI-STORM; transporting the HI-STORM and the simulator from the FHB to the ISFSI pad; moving a dummy spent fuel assembly from the SFP rack into an MPC; verifying the fuel loaded into an MPC; moving the HI-TRAC and the MPC from the SFP to the cask wash down pit; and performing radiation and contamination surveys.

The inspectors observed the licensee's oversight process, use of command and control, and control of both simulated and actual radiological hazards during dry run activities through both interviews with licensee personnel and procedural reviews.

(2) Fuel Selection

The inspectors reviewed the licensee's program associated with fuel characterization and selection for storage. The inspectors reviewed cask fuel selection packages to verify that the licensee was loading fuel in accordance with Appendix B of the CoC.

(3) Radiation Protection

The inspectors evaluated the licensee's radiation protection (RP) program pertaining to the operation of the ISFSI. The inspectors observed licensee RP technicians simulate dry run activities and interviewed both RP and other licensee personnel to verify their knowledge regarding the scope of the work and the radiological hazards associated with transfer and storage of spent fuel. The inspectors reviewed radiological surveys, both actual and simulated.

(4) Training

The inspectors reviewed the licensee's training program, which consisted of classroom and on-the-job training to ensure involved staff were adequately trained for the job they were responsible to perform. The inspectors interviewed licensee personnel to verify that they were knowledgeable of the scope of work that was being performed.

(5) Emergency Preparedness, Surveillance, Fire Protection, and Quality Assurance Activities

The inspectors reviewed selected licensee procedures to ensure that responsibilities for specific ISFSI activities have been defined and that these responsibilities have been integrated into the appropriate plant programs. The inspectors reviewed station emergency preparedness, surveillance, fire protection and quality assurance procedures to ensure that they meet the commitments and requirements as specified in the FSAR, the CoC, 10 CFR Part 72, and TS.

b. Findings

No findings were identified.

(6) (Open) Unresolved Item: Potentially Non-Conservative Changes Made to the Time-to-Boil Calculation in the FSAR under 10 CFR 72.48

Introduction: An unresolved item (URI) was identified by the inspectors relating to changes made to Section 4.5.3, "Maximum Time Limit during Wet Transfer Operations," of the HI-STORM FW MPC Storage System FSAR, Revision 2. Revision 2 of the FSAR is the licensing basis for the casks that the licensee proposes to load under Docket Number 72–1032, Amendment 0, Revision 1. These FSAR changes were made by

Holtec International via Engineering Change Orders, ECO–5018–25R0 and ECO–5018–48R1, and accepted by the licensee via 72.48 Screening CL–2016–S–007.

Description: Section 4.5.3 of the HI-STORM FW FSAR limits the amount of time for wet transfer operations, such that the water in the MPC is not permitted to boil. Revision 2 of the FSAR specifies, "The time limits are conservatively computed under an assumed adiabatic temperature rise of the cask with design heat load and understated thermal inertia of the cask defined in Table 4.5.3. The computed time limits are tabulated in Table 4.5.4."

The initial temperature utilized in Table 4.5.4 is described in Section 4.5.3 of Revision 2 of the FSAR as, "an initial (pool water) temperature." Section 4.5.3 also states, "Fuel loading operations are typically conducted with the HI-TRAC VW and its contents (water filled MPC) submerged in pool water. Under these conditions, the HI-TRAC VW is essentially at the pool water temperature."

ECO–5018–25R0 and ECO–5018–48R1 delineate several changes made to the time-to-boil calculation from the calculation as described in Section 4.5.3 of Revision 2 of the FSAR. Both ECOs were accepted by the licensee using 72.48 Screening CL–2016–S–007. Of particular regulatory concern are changes ECO–5018–25R0 [1], ECO–5018–48R1 [4], and ECO–5018–48R1 [1]. The changes ECO–5018–25R0 [1] and ECO–5018–48R1 [4] have been incorporated by the licensee into operating procedure HPP–2226–300, Revision 0, "MPC Sealing at Clinton." The inspectors noted that the all three of these changes had the potential effect of lengthening the time-to-boil limit in order to avoid performing alternate cooling.

Title 10 CFR 72.48(c)(2) states in part, "a general licensee shall request that the certificate holder obtain a CoC amendment pursuant to 72.244, prior to implementing a proposed change, test, or experiment if the change, test, or experiment would: ...(viii) Result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses."

The two key open questions regarding these changes are first whether or not 10 CFR 72.48(c)(2)(viii) applies to the time-to-boil calculation, i.e. whether or not the time-to-boil calculation is considered "a method of evaluation ...used in establishing the design bases or in the safety analyses." The second open question is whether or not the licensee is 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 (NMSS). This issue will be a URI pending further review by the inspectors after NMSS completes its review of the licensing concerns and provides the necessary inspection guidance.
(URI 05000461/2016003–06; URI 07201046/2016002–02, Potentially Non-Conservative Changes Made to the Time-to-Boil Calculation in the FSAR under 10 CFR 72.48)

4OA6 Management Meetings

.1 Exit Meeting Summary

On October 14, 2016, the inspectors presented the inspection results to Mr. T. Stoner, 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 inspection results for the Radiation Safety Program review with Mr. B. Kapellas, Plant Manager, on September 29, 2016;

- The results of the ISFSI stack-up and stability inspection (IP 60856.1) were presented on August 29, 2016, to Mr. B. Kapellas, Plant Manager, and members of the licensee management and staff; and

- The results of the ISFSI pre-operational inspection (IP 60854.1) were presented on September 9, 2016, to Mr. B. Kapellas, Plant Manager, and members of the licensee management and staff.

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.

4OA7 Licensee-Identified Violations

The following violation of very low significance (Green) or Severity Level IV was identified by the licensee and is a violation of NRC requirements which meets the criteria of the NRC Enforcement Policy for being dispositioned as an NCV.

- The following violation of very low significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of the NRC Enforcement policy for being dispositioned as an NCV.

  Clinton Technical Specification 3.3.2.1, “Control Rod Block Instrumentation” requires the control rod block instrumentation for each Function in Table 3.3.2.1–1 shall be operable. If one or more rod withdrawal limiter (RWL) channels is inoperable, the required action is to suspend control rod withdrawal. The completion time for the action is immediately. Contrary to the above, on April 13, 2016, both RWL channels became inoperable and the required action to immediately suspend control rod withdrawal was not completed.

  Specifically, on April 13, 2016, operations lowered reactor power below the control rod withdrawal limiter high power set point to remove the ‘B’ turbine driven reactor feed pump from service for repairs. Technical Specification Surveillance Requirement 3.3.2.1.2 requires a functional test of the 4-notch control rod withdrawal limit of the RWL within one hour of resetting the high power set point during power reduction if it has not been completed within the previous 92 days. The surveillance

was last performed on April 25, 2015, making the RWL channels inoperable below the high power set point. The licensee subsequently withdrew control rods to restore reactor power above the high power set point with the RWL channels inoperable.

The licensee discovered the missed surveillance on April 21, 2016, in preparations for lowering reactor power to place the 'B' turbine driven reactor feed pump into service after repairs. The surveillance was successfully performed after lowering power below the RWL high power set point. The licensee entered the issue into the CAP as AR 02659195.

The issue was determined to be more than minor because the performance deficiency affected equipment performance attribute of the mitigating systems cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was screened 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 the Mitigating Systems cornerstone and determined to be of very low safety significance (Green) because the finding did not affect a single reactor protection system trip signal to initiate a reactor scram, did not involve control manipulations that unintentionally added positive reactivity or result in a mismanagement of reactivity by operators.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

T. Stoner, Site Vice President
B. Kapellas, Plant Manager
D. Avery, Regulatory Assurance
R. Bair, Chemistry Manager
J. Crump, Site Dry Cask Project Manager
J. Cunningham, Maintenance Director
C. Dunn, Operations Director
C. Engelhardt, Acting Work Management Director
M. Friedmann, Emergency Preparedness Manager
S. Gackstetter, Engineering Director
M. Heger, Senior Manager Plant Engineering
N. Hightower, Radiation Protection Manager
N. Keen, Design Engineering
T. Krawyck, Senior Manager Plant Engineering
B. Marchese, ODCM Specialist
W. Marsh, Security Manager
S. Minya, Operations Training Manager
M. Pagel, Site Dry Cask Project Manager
J. Pfabe, Licensing Consultant
K. Pointer, Regulatory Assurance
J. Rogozinski, Senior Program Manager Dry Fuel Storage,
Clinton/Braidwood Nuclear Stations
D. Shelton, Regulatory Assurance Manager
S. Strickland, Shift Operations Superintendent
J. Ward, Organizational Effectiveness Manager

U.S. Nuclear Regulatory Commission

K. Stoedter, Chief, Reactor Projects Branch 1
W. Schaup, Clinton Senior Resident Inspector
J. Wojewoda, Acting Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000461/2016003-01	NCV	Failure to Scope Fuel Building Ventilation Pressure Control into Maintenance Rule (Section 1R12)
05000461/2016003-02	NCV	Exceeded Technical Specification Allowed Outage Time for Main Turbine Bypass System (Section 1R15)
05000461/2016003-03	SLIV /NCV	Failure to Perform a 50.59 Screening for Changing the Frequency of Exercising the Turbine Bypass Valves (Section 1R18)
05000461/2016003-04	NCV	Failure to Prevent Recurrence of a Significant Condition Adverse to Quality (Section 4OA3)
05000461/2016003-05; 07201046/2016002-01	NCV	Spent Fuel Pool Liner Design Not Verified per Code (Section 4OA5)

Closed

05000461/2016-006-00	LER	Missed Surveillance Results in a Condition Prohibited by Technical Specifications
05000461/2016-007-00	LER	Main Steam Line Flex Hose Intergranular Stress Corrosion Cracking Identified During Refueling Outage
05000461/2016-002-00	LER	Trip of Fuel Building Ventilation Exhaust Fan Due to Moisture Formation Resulting in the Loss of Secondary Containment Vacuum

Opened

05000461/2016003-06; 07201046/2016-002-02	URI	Potentially Non-Conservative Changes Made to the Time-to-Boil Calculation in the FSAR under 10 CFR 72.48 (Section 4OA5)

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

- WC-AA-107 Attachment 3; Plant System Readiness Review for FP [Fire Protection]; Revision 16
- WC-AA-107 Attachment 3; Plant System Readiness Review for SY [Switchyard]; Revision 15
- WC-AA-107 Attachment 3; Plant System Readiness Review for VD [Diesel Ventilation]; Revision 16
- CPS 3800.02; Area Operator Logs; Revision 19c
- CPS 1860.01C001; Operations Department Cold Weather Preparations Checklist; Revision 7d
- CPS 1860.01; Cold Weather Operation; Revision 8e
- Laboratory Report: Diesel Fuel; Cab #259 Trailer #125; Sampling Date October 27, 2015
- PES-P-006; Diesel Fuel Oil; Revision 11
- Area Rounds for SY
- Area Rounds for the 'A' and 'B' Fire Pump Rooms
- CPS 2016 Winter Readiness Challenge Process Presentation
- CPS E-15 Schedule Scope WW 1701
- AR 02711268; FMID Winter Readiness Issue with Hydrant 0FP177
- AR 01666153; 345KV SY GCB Door Seals Needing Inspection
- AR 02716282; Degraded Door Seal and Cabinet Lights in SY Breakers
- AR 01625219; Assessment of Leaks Associated with Fire Pump A Room

1R04 Equipment Alignment

- OP-AA-108-103; Locked Equipment Program; Revision 2
- CPS 3319.01E001; Standby Gas Treatment Electrical Lineup; Revision 11
- CPS 3319.01V001; Standby Gas Treatment Valve Lineup; Revision 8
- CPS 9067.03; Standby Gas Treatment System Operability; Revision 27b
- CPS 3313.01E001; Low Pressure Core Spray Electrical Lineup; Revision 11a
- CPS 3313.01V002; Low Pressure Core Spray Instrument Valve Lineup; Revision 8a
- CPS 3313.01V001; Low Pressure Core Spray Valve Lineup; Revision 13b
- CPS 3313.01; Low Pressure Core Spray; Revision 16e
- CPS 9052.02; Low Pressure Core Spray (LPCS) Valve Operability Checks; Revision 33a
- CPS 3506.01V002; Diesel Generator and Support Systems Instrument Valve Lineup; Revision 11b
- CPS 3506.01V001; Diesel Generator and Support Systems Valve Lineup Division 2; Revision 13a
- M05-1073; P&ID Low Pressure Core Spray (LPCS); Revision AH
- M05-1105; P&ID Standby Gas Treatment System (VG) sheet 1; Revision U
- M05-1035; Diesel Gen Aux System (DG) Exhaust Starting Air Exhaust & Combustion Sys; Revision AB
- M05-1036; P&ID Diesel Generator Fuel Oil System (DO); Revision S
- AR 02697345; Trend in Critical Component Failures
- AR 02700963; 5052-1H: Unexpected Division 2 VG Alarm

- AR 02713181; NRC ID: 1E21-F371 Locking Device Failed
- AR 02539116; ASME Qualification Documents not Obtained during Modification
- AR 02490436; Incorrect Breaker per EMD Breaker Plan Installed
- AR 02473666; Discrepancies with 1AP07EE Cubical During WO 1588365

1R05 Fire Protection

- OP-AA-201-003; Fire Drill Performance; Revision 16
- OP-AA-201-005; Fire Brigade Qualification; Revision 9
- OP-AA-201-003; Attachment 1; Fire Drill Record (U2016-15); September 6, 2016
- OP-AA-201-003; Attachment 3; Fire Drill Scenario (U2016-15); August 3, 2016
- CPS 1893.01; Fire Protection Impairment Reporting; Revision 20d
- CPS 1893.04M720; 762 Turbine: Turbine Auxiliaries Prefire Plan; Revision 6c
- CPS 1893.01M001; Fire Door Compensatory Measures; Revision 5f
- CPS 1893.04; Fire Fighting; Revision 18
- CPS 1893.04M001; Prefire Plan Cross Index; Revision 3c
- CPS 1893.04M002; Prefire Plan/Fire Zone Cross Index; Revision 3b
- CPS 1893.04M003; Prefire Plan Legend; Revision 1
- CPS 1893.04M321; 737 Control: Rad-Chemistry Lab and Laundry Prefire Plan; Revision 3
- CPS 1893.04M351; 781 Control: Auxiliary Electrical Equipment, Inverter and Battery Rooms Prefire Plan; Revision 7c
- CPS 1893.04M400; 712 Fuel: Basement Prefire Plan for HPCS Pump Room; Revision 5
- CPS 1893.04M802; 699 Screen House: 'B' (South) Fire Pump Room Prefire Plan; Revision 6
- Figure FP-2b; Fire Protection Features Auxiliary, Fuel Building and Containment Basement Floor Plan EL 707'-6" & 712'-0"
- T/NFPA Conformance Evaluation/2-1; Summary Conformance Table 2-1
- CR 1-86-05-089; 1VX12CA & 1VX12CB were unable to achieve design flows; June 24, 1985
- AR 02721345; NRC ID – Compressed Argon Cylinder Storage
- AR 02700456; Evaluate Correct Placement of no Radio Area Tape
- AR 02700454; Fire Marshall Identified Brigade Response Issue during Fire Drill 2016-15
- AR 02702044; Fire Marshall Identified NRC Question Regarding Fire Brigade Response to High Radiation Area
- AR 02700457; Fire Marshall Identified ½" WYE Appliance Missing from Fire Brigade Cart

1R06 Flooding

- CC-AA-309-1001; Internal Flooding Calculations; Revision 9
- CC-AA-309-1001; Design Analysis Major Revision Cover Sheet for Suppression Pool Equalization Levels (EC 380335); Revision 5
- ER-CL-450-1007; Clinton Surveillance Inspection Program for Flood Seals; Revision 1
- CPS 9610.01; Fire Rate Assemblies and Penetration Sealing Devices; Revision 28f
- M26-1000-02A; Auxiliary Building Piping Plan EL. 707'-6" Area 2; Revision H
- M05-1047; P&ID Auxiliary Building Drain System (RF); Revision N
- Drawing No. 119; Bisco-seal High Pressure Seal
- Drawing No. 106; 5" SF-60 Internal Conduit Seal
- Drawing No. 105; 9" SF-20 Internal Conduit Seal
- Drawing No. 102; 5" SF-60 Silicon Elastomer Pipe/Sleeve Thru Barrier
- AR 02690138; NRC Question: RHR B Penetration Flood Seals
- AR 02680762; NRC Question Documentation of Flood Penetration Seal Inspect
- AR 01197979; IER – Flood Seals do not have Periodic Inspection Program
- AR 00976295; ECCS Room Floor Drain Piping Connected to RW Pipe Tunnel

1R11 Licensed Operator Requalification Program

- OP–AA–101–111–1001; Operations Standards and Expectations; Revision 17
- OP–AA–300; Reactivity Management; Revision 9
- OP–CL–108–101–1003; Operations Department Standards and Expectation; Revision 35
- TQ–AA–150; Operator Training Programs; Revision 12
- TQ–AA–155; Conduct of Simulator Training and Evaluation; Revision 5
- CPS 3005.01; Unit Power Changes; Revision 43
- Reactivity Maneuver Guidance Sheet # C17-005
- Reactor Engineer's Evolution Plan # C17-005
- WO 01923734; WO for Cycle 17 Down Power Activities

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
- 10 CFR 50.65(a)(3) Periodic Assessment of Maintenance Rule Program; Clinton Power Station; May 18, 2015 through May 17, 2016; August 1 2016
- ACE 2625647; Differential Pressure in Fuel Building and Secondary Containment Loss
- EACE 1247016; Determine the Cause for Fuel Building Ventilation Differential Pressure Failure
- Maintenance Rule System Basis Document Fuel Building HVAC
- AR 02575694; PMRQ 191600-01 will Exceed Late Date due to Plant Conditions
- AR 02613542; VF Exhaust Fans Repeatedly Tripping
- AR 02725126; Perform EOC Review for New MRule Function VF-01
- AR 02716300; NRC Question About Basis for not Including VF Function in MR
- AR 02696469; MRule IR Reviews not Completed in 30 Days or Less

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 1
- Plan of the Day for September 15, 2016
- 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 4
- OP-AA-108-117; Protected Equipment Program; Revision 4
- 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
- CPS 4302.01; Tornado High Winds; Revision 21F

- AR 02692387; Entered Off-normal CPS 4302.01, Tornado / High Winds

1R15 Operability Evaluations

- OP–AA–108–115; Operability Determinations; Revision 16
- OpEval 02697915-02; Operability Evaluation for Degraded Seal on the Inner Railroad Bay Door to the Fuel Building; Revision 0
- CPS 1401.09; Control of System and Equipment Status; Revision 9b
- CPS 3512.03; 3D Monicore System; Revision 10
- CPS 9072.01; Steam Bypass Valve Tests; Revision 33
- EMD 036277; Support Load and Pipe Deflection Calculation for Piping Supported by Baldwin Associates; Revision 00
- EMD 037535; Interaction between Safety Related Piping and other Safety Related Plant Components with Comparable Stiffness; Revision 00
- EMD 035476; Fundamental Frequency and Deflected Shape Calculations; February 5, 1982
- EMD 022411; Formal Piping Stress Analysis; Revision 1
- EC 406561; Evaluation of Interaction between 1VG05AA and 1SA08B; Revision 0
- CL-15-006; Extend the Frequency of Steam Bypass Valve Testing from 31D to 1Y; Revision 0
- PORC Meeting Number 16-019; June 29, 2016
- WO 01674948; Door 1SD1-30 is Missing Small Portion of Seal; August 9, 2016
- AR 02697915; Inner Railroad Bay Door Seal Degrading
- AR 02713680; NRC Question on VG and SA Pipe Clearance in LPCS Room
- AR 02715239; Coating Eroded on 1VG05AA and 1SA08B
- AR 02677643; Number 4 Bypass Valve Stuck Full Open during CPS 9072.01
- AR 02677873; Number 4 Bypass Valve Stuck Full Open during CPS 9072.01
- AR 02690657; NRC Question COLR Thermal Limits Removal
- AR 02694218; Fleet Assessor Challenges Technical Input to Surveillance Frequency Extension
- AR 02715606; System Manager Identification: USAR Chapter 10 Review Observations
- AR 02717769; Number 4 Bypass Valve Stuck Open, not Evaluated for MRULE Impact
- AR 02689812; Division 1 Emergency Diesel Generator Bearing Grease Seepage
- AR 02694534; 1SD1-11 Shutdown Service Water Door does not Operate
- AR 02695275; NRC Question on Operability of Division 1 Shutdown Service Water during Maintenance

1R18 Plant Modifications

- LS-AA-104; Exelon 50.59 Review Process; Revision 10
- LS-AA-104-1000; Exelon 50.59 Resource Manual
- AR 02720163; USAR Change Package not Approved Prior to STI Change

1R19 Post-Maintenance Testing

- MA–AA–716–012; Post Maintenance Testing; Revision 20
- CPS 9065.02; Secondary Containment Integrity; Revision 30b
- CPS 9056.02D001; Secondary Containment Integrity Data Sheet; Revision 30a
- CPS 3506.01C005; Diesel Generator Start Log; Revision 1b
- CPS 9080.30A20 OP DG 1A Overspeed Trip Test; Revision 2
- CPS 3506.01D001; Diesel Generator 1A Operating Logs; Revision 5a
- CPS 9069.01; Shutdown Service Water Operability Test; Revision 49
- CPS 9069.01D001; SX System Operability Data Sheet; Revision 47

- CPS 9437.60; Main Control Room Air Intake Radiation 1RIX009B Channel Calibration; Revision 37f
- CPS 9437.60D001; Main Control Room Air Intake Radiation 1RIX-PR009B Channel Calibration Data Sheet; Revision 35a
- CPS 3402.01P001; Appendix A: VC Chiller Start-up Data Log
- CPS 8452.04D001; AH91/NH91 Hydramotor Disassembly/Assembly and Functional Testing Data Sheet; Revision 7
- CPS 9071.02; Diesel Fire Pump Capacity Test; Revision 40E
- CPS 9071.01; Diesel Driven Fire Pump Operability Test; Revision 40
- EC 406631; Division 1 Diesel Generator Turbocharger Impeller Blade Indications; Revision 0
- WO 01674948; Door 1SD1-30 is Missing Small Portion of Seal
- WO 01929251; Perform CPS 9065.02 Secondary Containment Integrity – Alternate VG Trains
- WO 01821786; 9080.30A20 OP DG 1A Overspeed Trip Test
- WO 01577918-01; 1AP07E: 74-SX1PA Relay has Over Heated Wires
- WO 01506250-03; EM PMT 1AP29E-4BR Verify Relay
- WO 01820910-04; EM/OP PMT – Run DG and Obtain Data
- WO 01369617-01; EM 9381.01SX2 Verify MOV Thermal Overload Bypass
- WO 01714027-03; EM PMT 1SX013D Thermal O/L Bypass Test
- WO 01714027-04; OP PMT 1SX013D Stroke/Verify Operation
- WO 01946509; OP SX Pump A Operability Test
- WO 01942405; Unexpected High Alarm on 1RIX-PR009B
- WO 01943799-01; Troubleshoot 0VC13CB Trip/Install Astromed Recorder
- WO 01943799-17; EM 1SX019B Prep Replacement Hydramotor
- WO 01943799-19; EM 1SX019B Rebuild/Replacement of Hydramotor
- WO 1711872-07; OP 0FP05SA PMT Valve
- AR 02698521; 1RIX-PR009B Unexpected Alert and High Alarm
- AR 02699502; Critique for Trip of VC B Chiller
- AR 02693083; NRC Question on Fire Pump Preconditioning

1R22 Surveillance Testing

- OP-AA-108-103; Locked Equipment Program; Revision 3
- CPS 9051.01; HPCS Pump and HPCS Water Leg Pump Operability; Revision 48
- CPS 1887.00; Administration of In-Service Inspection (ISI) and In-Service Testing (IST) Program Activities; Revision 8a
- CPS 9015.01; Standby Liquid Control System Operability; Revision 41c
- CPS 9015.01D001; SLC Pump and Valve Data Sheet: Revision 38a
- CPS 9080.03; Diesel Generator 1C Operability-Manual and Quick Start; Revision 34f
- Drawing M05-1052, P&ID Shutdown Service Water (SX): Revision AU
- Predefined History Report, Standby Liquid Control Operability Surveillance
- WO 01876960; 9069.01B20 OP SX Pump Oper Test (SX Pump B)
- WO 01891522; 9069.01B20 OP SX Pump Oper Test (SX Pump B)
- WO 01915452; 9069.01B20 OP SX Pump Oper Test (SX Pump B)
- WO 1915301-01; OP 9051.01 HPCS pump and HPCS Water Leg Pump Operability
- AR 2711246; SLC Pump B Test Switch Malfunctioning
- AR 2711234; 9015.01 Procedure Enhancement for Checking Oil on SLC
- AR 02692752; NRC Questions on Div 3 90 Degree Valve
- AR 02693066; NRC Question on 1SX006C
- AR 02693077; NRC Question on Procedure Enhancements

1EP6 Drill Evaluation

- Clinton Station Off-Year Exercise Full Manual; August 31, 2016
- Clinton 2015 Hostile Action Based NRC Graded Exercise Evaluation Report
- Clinton 2014 Off-Year Exercise Evaluation Report
- Clinton 2013 NRC Graded Ingestion Pathway Exercise Evaluation Report
- Clinton 2012 Off-Year Exercise Evaluation Report
- Clinton 2011 NRC Graded Exercise Evaluation Report

2RS7 Radiological Environmental Monitoring Program

- CY-AA-170-000; Radioactive Effluent and Environmental Monitoring Programs; Revision 006
- CY-AA-170-100; Radiological Environmental Monitoring Program; Revision 002
- CY-AA-170-1000; Radiological Environmental Monitoring Program and Meteorological Program Implementation; Revision 008
- CY-CL-170-301; Offsite Dose Calculation Manual; Revision 025
- EN-CL-6805-01; Operating Sediment Ponds and Filter House; Revision 000
- EN-CL-408-4160; RGPP Reference Material for Clinton Power Station; Revision 003
- EN-AA-408-4000; Radiological Groundwater Protection Program Implementation; Revision 006
- EN-AA-407; Response to Inadvertent Releases of Licensed Materials to Groundwater, Surface Water, Soil or Engineered Structures; Revision 007
- EN-AA-408; Radiological Groundwater Protection Program; Revision 000
- EIML-SPM-1; Sampling Procedures Manual, Environmental Incorporated Midwest Laboratory; Revision 015
- PI-AA126-1005-F-01; Check-In Self-Assessment Title, Pre-NRC Radiological Environmental Monitoring and Performance Indicator Verification (IPs 71124.07 and 71151); Revision 001
- NOS Objective Evidence Report Chemistry, Radwaste, Effluent and Environmental Monitoring Functional Area Audit; Dated June 17, 2016
- Annual Report on the Meteorological Monitoring Program at the Clinton Power Station, 2015, Prepared by Murray and Trettel, Incorporated
- Monthly Report on the Meteorological Monitoring Program at the Clinton Power Station, January 2016, Prepared by Murray and Trettel, Incorporated
- Monthly Report on the Meteorological Monitoring Program at the Clinton Power Station, February 2016, Prepared by Murray and Trettel, Incorporated
- Monthly Report on the Meteorological Monitoring Program at the Clinton Power Station, March 2016, Prepared by Murray and Trettel, Incorporated
- Monthly Report on the Meteorological Monitoring Program at the Clinton Power Station, April 2016, Prepared by Murray and Trettel, Incorporated
- Report of Analysis/Certificate of Conformance Prepared by Teledyne Brown Engineering, Incorporated; October 30, 2015
- Report of Analysis/Certificate of Conformance Prepared by Teledyne Brown Engineering, Incorporated; March 15, 2016
- Report of Analysis/Certificate of Conformance Prepared by Teledyne Brown Engineering, Incorporated; June 27, 2016
- Field Rotameter Calibration by Environmental, Incorporated Midwest Laboratory; January 21, 2015
- Pump Status by Environmental, Incorporated Midwest Laboratory; July 18, 2016
- Clinton Power Station Annual Radiological Groundwater Protection Program Report, January 1 through December 31, 2015, Prepared by Teledyne Brown Engineering Services, April 2016

- Clinton Power Station Annual Radiological Environmental Operating Report, January 1 through December 31, 2015, Prepared by Teledyne Brown Engineering Services, April 2016
- Exelon Generation, January 1, 2015, through December 31, 2015, Annual Radioactive Effluent Release Report, Clinton Power Station-Docket Number 50-461, Prepared by Clinton Power Station
- AR 02473280; Substitution of ODCM Vegetables at CL-114 Not Communicated
- AR 02721291; ODCM: Insufficient Vegetation for the September 2016 Sampling
- AR 02710845; ODCM: Insufficient Vegetation for the August 2016 Sampling
- AR 02687265; Insufficient Vegetation for the June 2016 Sampling at CL-115
- AR 02716261; 0UIX-PR050 Data Inputs Spiking
- AR 02584230; VG Stack Flow Monitor Reading Higher Than Normal
- AR 02597597; ODCM 30 Day Clock Expired on 0UIX-PR051 Channel 1
- AR 02615920; ODCM 2015 X/Q and D/Q Values Greater than 20%

4OA1 Performance Indicator Verification

- PI Summary; Reactor Coolant System Activity (RCSA); September 2015 through September 2016
- PI Summary; RETS/ODCM Radiological Effluent; September 2015 through September 2016
- PI Summary; Occupational Exposure Control Effectiveness; September 2015 through September 2016

4OA2 Identification and Resolution of Problems

- Calculation No. SDQ12-24DG05; Evaluation of Structural Steel Inside Drywell for the Impact of Additional Loads due to Installation of Permanent Shield Blankets Inside Drywell; Revision 10C
- Calculation No. IP-S-0237; Qualification of the Auxiliary Steel and Connection Details for Permanent Shielding Inside Drywell; Revision 0C
- Calculation No. SDQ12-23DG06; Evaluation of Structural Steel Inside Drywell for the Impact of Additional Loads Due to the Installation of Permanent Shield Blankets Inside Drywell; Revision 07A

4OA3 Follow-Up of Events and Notices of Enforcement Discretion

- CPS 9014.01; RPC System Withdrawal Limitation Test; Revision 28
- RCE 2670593; Water Leak at Flexible Hose for MS Elbows in Drywell
- ACE 2625647; Differential Pressure in Fuel Building and Secondary Containment
- ACE 2659195; Missed Required Technical Specification Surveillance
- EC 404886; Design Considerations Summary (ECP) and Work Planning Instructions; Revision 0
- EC 405699; Evaluation of MS Flex Hose Failures; Revision 000
- BWROG-TP-08-025; Reactivity Controls Review Committee; Revision 17
- AR 02722916; Incorrect Wind Speed used in the Wind Load Calculation
- AR 02670593; Water Leak at Flex Hose for MS Elbow Taps in DW
- AR 02661136; Discrepancies Identified in Voltages Recorded – 9082.02

4OA5 Other Activities

- EP-AA-1003; Addendum 3; Emergency Action Levels for Clinton Station; Revision 1

- MA-AA-716-001; Quality Material/Components Control and Identification/Segregation of Non-conforming Items; Revision 8
- MA-AA-716-008-1008; Reactor Services Refuel Floor FME Plan; Revision 12
- MA-AA-716-021; Rigging and Lifting Program; Revision 26
- NEI 96-07; Guidelines for 10 CFR 50.59 Implementation; Revision 1
- CPS 1019.05; Transient Equipment/Materials; Revision 23e
- CPS 1893.04; Fire Fighting; Revision 17e
- CPS 1893.04M410; 737 Fuel: Grade Level Prefire Plan; Revision 4c
- CPS 1893.04M420; 755 Fuel: Fuel Handling Floor Prefire Plan; Revision 2b
- CPS 1893.04M430; 781 Fuel: West Balcony Prefire Plan; Revision 4a
- CPS 1893.04M431; 781 Fuel: East Balcony Prefire Plan; Revision 2a
- CPS 9000.01; Control Room Surveillance Log; Revision 35d
- CPS 9000.01D001; Control Room Surveillance Log – Mode 1, 2, 3 Data Sheet; Revision 55b
- RP-CL-300-1002; HI-TRAC Radiation Survey; Revision 0
- RP-CL-300-1003; HI-STORM FW Radiation Survey; Revision 0
- RP-CL-300-1004; Independent Spent Fuel Storage Installation Radiation Survey; Revision 0
- RP-CL-300-1005; MPC Surface Contamination Verification; Revision 0
- SDQ15-23DG01; Fuel Handling Building General Design Grade Floor Elevation 737'-0" Design of Slabs; Revision 9D
- SDQ15-23DG02; Grade Floor Design of Beams; Revision 9B
- Specification No. K-2943; Certification of Specification for Stainless Steel Pool Liner Work Clinton Power Station – Unit 1 Illinois Power Company; October 25,1985
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- GQP-9.2; High Temperature Liquid Penetrant Examination and Acceptance Standards for Welds, Base Materials and Cladding (50-350F); Revision 9
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- HI-2135751; HI-STORM FW CoC Radiation Protection Program Dose Rate Limits for Clinton Power Station; Revision 0
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- HI-TP-003; Dry Fuel Storage System Preparation and Loading; Revision 0
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- Holtec 72.48 Screening 1136; Screening for ECO-5018-48; Revision 1
- HPP-2226-100; MPC Pre-Operation Inspection; Revision 0
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- Clinton Power Station ISFSI Organizational Chart; 05/18/2016
- Clinton Power Station ISFSI Organizational Chart; 07/22/2016
- Clinton Power Station Unit 1 10 CFR 72.212 Evaluation Report for the HI-STORM FW MPC Storage System; 08/30/2016
- 16-00204-1; ALARA Plan Dry Cask Storage and Support Activities – (6 Casks)
- 50.59 Screening No. CL-2014-S-036; ISFSI-Design & Install a VECASP in the Cask Washdown Pit; 03/27/2015
- 50.59 Screening No. CL-2015-E-015; ISFSI- Stability Analysis and Structural Impacts From Spent Fuel Casks in the Fuel Building; 03/15/2016
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- 72.48 Screening CL-2016-S-007; HI-STORM FW Time to Boil Calculation; Revision 0
- WO 01890939-46; ISFSI 2016 – Campaign Preps / Dry Run / NRC Demos; June 22, 2016
- AR 02603589; Math Error In Calculation IP-S-0315

- AR 02615443; Use of Incorrect Code For Allowable Stresses
- AR 02673434; EOC Did Not Address Entire Scope in Design Change
- AR 02690444; Informational IR Weld Porosity Indicated During DCS NRC Demo
- AR 02690744; Response to NRC Inquiry Did Not Answer Question
- AR 02713502; NRC Pre-op Dry Run #3 Observations for Upcoming SFLC
- AR 02713696; FME Plan Compliance Not Met

LIST OF ACRONYMS USED

ACE	Apparent Cause Evalution
ADAMS	Agencywide Document Access Management System
AR	Action Request
ASME	American Society of Mechanical Engineers
BPVC	Boiler Pressure Vessel Code
BWR	Boiling Water Reactor
CAP	Corrective Action Program
CAPR	Corrective Action to Prevent Recurrence
CFR	Code of Federal Regulations
CoC	Certificate of Compliance
COLR	Core Operating Limits Report
DNMS	Division of Nuclear Materials Safety
DRP	Division of Reactor Projects
EC	Engineering Change
EOP	Emergency Operating Procedure
FHB	Fuel Handling Building
FSAR	Final Safety Analysis Report
GPI	Groundwater Protection Initiative
HSPS	High Power Setpoint
IGSCC	Intergranular Stress Corrosion Cracking
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
ISFSI	Independent Spend Fuel Storage Installation
LCO	Limiting Condition for Operation
LHGR	Linear Heat Generation Rate
LER	Licensee Event Report
LLC	Limited Liability Corporation
LOCA	Loss of Coolant Accident
MCPR	Minimum Critical Power Ratio
MPC	Multipurpose Canister
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NMSS	Office of Nuclear Material Safety and Safeguards
NRC	U.S. Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual
PARS	Publicly Available Records System
PI	Performance Indicator
PM	Post-Maintenance
PORC	Plant Oversight Committee Review
RP	Radiation Protection
RWL	Rod Withdrawal Limiter
SC	Secondary Containment
SDP	Significance Determination Process
SFP	Spent Fuel Pool
SR	Surveillance Requirement
SSC	System, Structure, and Component
TS	Technical Specification
URI	Unresolved Item

USAR	Updated Safety Analysis Report
VF	Fuel Building Ventilation
VG	Standby Gas Treatment
WO	Work Order

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Sincerely,

/RA/

Karla Stoedter, Chief
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