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

November 7, 2017

EA–17–176

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

SUBJECT: CLINTON POWER STATION—NRC INTEGRATED INSPECTION REPORT 05000461/2017003

Dear Mr. Hanson:

On September 30, 2017, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Clinton Power Station. On October 19, 2017, the NRC inspectors discussed the results of this inspection with Mr. T. Stoner and other members of your staff. The results of this inspection are documented in the enclosed report.

Based on the results of this inspection, the NRC has identified five issues that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has also 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 Enforcement Policy. These NCVs are described in the subject inspection report.

A violation involving a failure to set secondary containment during Operations with the Potential to Drain the Reactor Vessel (OPDRV) was identified. Specifically, from May 9, 2017, through May 28, 2017, Clinton Power Station performed OPDRV activities without setting secondary containment, which is a violation of Technical Specification (TS) 3.6.4.1. The NRC issued EGM 11–003, "Enforcement Guidance Memorandum on Dispositioning Boiling Water Reactor Licensee Noncompliance with Technical Specification Containment Requirements During Operations with a Potential for Draining the Reactor Vessel," Revision 3, on January 15, 2016, allowing for the exercise of enforcement discretion for such OPDRV-related TS violations, when certain criteria are met. The NRC concluded that Clinton Power Station met these criteria. Therefore, I have been authorized, after consultation with the Director, Office of Enforcement, and the Regional Administrator, to exercise enforcement discretion and refrain from issuing enforcement for the violation.

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

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

This letter, its enclosure, and your response (if any) will be made available for public inspections and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document room in accordance with 10 CFR 2.930, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

/RA/

Laura Kozak, Acting Chief
Branch 1
Division of Reactor Projects

Docket No. 50–461
License No. NPF–62

Enclosure:
Inspection Report 05000461/2017003

cc: Distribution via LISTSERV®

Letter to Bryan Hanson from Laura Kozak dated November 7, 2017

SUBJECT: CLINTON POWER STATION—NRC INTEGRATED INSPECTION REPORT 05000461/2017003

DISTRIBUTION:
Jeremy Bowen
RidsNrrDorlLpl3
RidsNrrPMClinton Resource
RidsNrrDirsIrib Resource
Cynthia Pederson
Kenneth O'Brien
Richard Skokowski
Allan Barker
Carole Ariano
Linda Linn
DRSIII
DRPIII
ROPreports.Resource@nrc.gov

ADAMS Accession Number: ML17313A039

OFFICE	RIII		RIII					
NAME	KLambert for RSkokowski:lg		LKozak					
DATE	11/07/2017		11/07/2017					

OFFICIAL RECORD COPY

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50–461
License No:	NPF–62
Report No:	05000461/2017003
Licensee:	Exelon Generation Company, LLC
Facility:	Clinton Power Station
Location:	Clinton, IL
Dates:	July 1 through September 30, 2017
Inspectors:	W. Schaup, Senior Resident Inspector E. Sanchez Santiago, Resident Inspector M. Doyle, Reactor Engineer J. Wojewoda, Reactor Inspector J. Havertape, Resident Inspector L. Rodriguez, Reactor Inspector M. Domke, Reactor Inspector G. Edwards, Health Physicist S. Mischke, Resident Inspector, Illinois Emergency Management Agency
Approved by:	L. Kozak, Acting Chief Branch 1 Division of Reactor Projects

Enclosure

TABLE OF CONTENTS

SUMMARY

Inspection Report 05000461/2017003; 07/01/2017–09/30/2017; Clinton Power Station; Operability Determinations and Functional Assessments; High Radiation Area and Very High Radiation Area Controls; Identification and Resolution of Problems; Follow-up of Events and Notices of Enforcement Discretion.

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Five Green findings were identified by the inspectors. The findings involved Non-Cited Violations (NCVs) of the 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 November 1, 2016. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG–1649, "Reactor Oversight Process," Revision 6.

Cornerstone: Initiating Events

- Green. The inspectors documented a self-revealed finding of very low safety significance and an associated NCV of Technical Specification (TS) 5.4.1, "Procedures," for the licensee's failure to establish sufficient instructions in station procedure Clinton Power Station (CPS) 3103.01, "Feedwater (FW)," Revision 31e, for changing modes of operation for the nuclear steam supply system. Specifically, the station procedure did not provide instructions requiring the locking out the flow control valves (FCVs) to prevent a reactor recirculation FCV runback while changing the feedwater pump lineup resulting in an unexpected plant transient and 9.2 percent change in reactor power. The licensee entered this issue into their corrective action program (CAP) as Action Request (AR) 04007861. As corrective actions, the licensee revised their CPS 3103.01 procedure to require that the FCVs be locked out prior to shifting reactor feed water pumps.

  The performance deficiency was more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated September 7, 2012, because the finding was associated with the procedure quality 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. Specifically, the failure to have adequate procedures for shifting feedwater pumps during a plant shutdown on May 7, 2017, resulted in an unexpected recirculation pump run back and a 9.2 percent change in reactor power. 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 event did not cause a reactor scram. The inspectors determined this finding affected the cross-cutting area of human performance in the aspect of conservative bias, where individuals use decision making practices that emphasize prudent choices over those that are simply allowable and a proposed action is determined to be safe in order to proceed, rather than unsafe in order to stop. Specifically, the procedure provided for the option to lockout the reactor recirculation flow control valves if deemed necessary during a shift of the reactor feedwater pumps and the operations crew did not make the prudent choice of locking out the valves before determining that it was safe to proceed. [H.14] (Section 4OA3.1)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance and an associated NCV of Title 10 Code of Federal Regulations (CFR) Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the licensee’s failure to demonstrate compliance with the requirement as prescribed in procedure ER–CL–330, “CPS Snubber Program,” Revisions 1 and 2. Specifically, the licensee failed to perform engineering evaluations to determine the cause of failure of snubbers that did not satisfy their functional testing acceptance criteria. The licensee entered this issue into their CAP as ARs 04015242 and 04041302. As corrective actions, the licensee evaluated the components affected by the failed snubber and determined that no operability issues existed.

  The performance deficiency was determined to be more-than-minor in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Screening,” because it was associated with the Mitigating Systems cornerstone attribute of Protection against External Factors and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability for mitigating systems to respond to initiating events. Specifically, compliance with ER–CL–330 would ensure the failed snubber was evaluated for the cause of failure, to ensure the licensee identified other snubbers that may have been vulnerable to the same type of deficiency. This would ensure that any potential undesired loading on the piping system could be avoided and the affected safety-related residual heat removal and reactor water cleanup piping systems could continue to perform their design function of maintaining the pressure boundary and structural integrity following a postulated design basis seismic event. The inspectors determined the finding could be evaluated using the Significance Determination Process in accordance with IMC 0609, Appendix A, “The Significance Determination Process for Findings At-Power,” dated June 19, 2012, Exhibit 2, “Mitigating Systems Screening Questions,” for the Mitigating Systems cornerstone and then Exhibit 4, “External Events Screening Question.” The finding screened as having very low safety significance because in each instance, the inspectors answered “No” to Questions 1 and 2 of Exhibit 4. 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. Specifically, the licensee failed to establish a systematic approach to evaluating snubbers that did not meet the acceptance criteria to ensure all required aspects were addressed. [H.13] (Section 4OA2.2)

Cornerstone: Barrier Integrity

- Green. The inspectors documented a self-revealed finding of very low safety significance and an associated NCV of TS limiting condition for operation (LCO) 3.6.1.3, for the failure to follow station procedure ER–AA–200, “Preventative Maintenance Program,” Revision 3. Specifically, the licensee utilized a condition-based maintenance approach on the main steam isolation valves (MSIVs) that failed to monitor and trend equipment performance so that planned maintenance could be performed prior to the MSIVs exceeding the TS leakage limits. The licensee entered this issue into their CAP as AR 04009845. As corrective actions, the licensee repaired and tested the valves prior to returning the unit to the modes of applicability.

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 impacted the Barrier Integrity cornerstone attribute of configuration control and adversely affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the monitoring and trending of local leak rate tests on the MSIVs did not provide performance data that would allow planned maintenance to the valves prior to the valves failing resulting in exceeding TS leakage requirements for the MSIVs. Using IMC 0609, Attachment 4, “Initial Characterization of Findings,” and Appendix A, “The Significance Determination Process for Findings at Power,” Exhibit 2, October 7, 2016, the finding was screened against the Barrier Integrity cornerstone Reactor Containment and did represent an actual open pathway in the physical integrity of reactor containment. The inspectors proceeded to Appendix H, “Containment Integrity Significance Determination Process,” and determined that it was a Type B finding that was related to a degraded condition that has potentially important implications for the integrity of the containment, without affecting the likelihood of core damage. The inspectors used Figure 6.1, Road Map for LERF based Risk Significance for Evaluation of Type-B Findings at Full Power and determined this finding is of very low safety significance (Green). The inspectors determined that this finding affected the cross-cutting area of human performance in the aspect of design margins, where the organization operates and maintains equipment within design margins. Special attention is placed on maintaining fission product barriers, defense-in-depth and safety related equipment. Specifically, the procedure for testing the MSIVs utilized an administrative limit that provided no margin to correct performance prior the valves becoming inoperable. [H.6] (Section 1R15)

- Green. The inspectors documented a self-revealed finding of very low safety significance and an associated NCV of TS LCO 3.0.4, for the failure to follow station procedure CC–AA–201, “Plant Barrier Control Program,” Revision 11. Specifically, the licensee entered MODE 2 from MODE 4 without meeting the requirements of LCO 3.0.4 for entering a mode when an applicable LCO is not met. The licensee had not met LCO 3.6.4.1 because the doors to the ‘B’ reactor water cleanup room were both opened instead of being closed to make secondary containment operable as required in MODE 2. The licensee entered this issue into their CAP as AR 04017613. As corrective actions, the licensee planned to conduct training for site personnel.

  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 impacted the Barrier Integrity cornerstone attribute of configuration control and adversely affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to follow the station procedure by not identifying that the open doors required a plant barrier impairment (PBI) permit that would have identified the doors as a constraint to entering MODE 2 resulted in the unit transitioning to MODE 2 with the secondary containment inoperable. Using IMC 0609, Attachment 4, “Initial Characterization of Findings,” and Appendix A, “The Significance Determination Process for Findings at Power,” Exhibit 2, October 7, 2016, the finding was screened against the Barrier Integrity cornerstone and determined to be of very low safety significance because the finding only represented a degradation of a radiological barrier function provided for auxiliary building. The inspectors determined that this finding affected the cross-cutting are of human performance in the aspect of training, where the organization provides training and ensures knowledge transfer to maintain a knowledgeable, technically competent work force and instill nuclear safety values. Specifically, station personnel did not know the process for routing a PBI permit and did not know when a PBI permit was required. [H.9] (Section 4OA3.2)

Cornerstone: Occupational Radiation Safety

- Green. A finding of very low safety significance and an associated NCV of TS 5.4.1 was self-revealed when individuals failed to adequately control access in locked high radiation areas (LHRAs). Specifically, the failure to meet all of the requirements of Procedure RP–AA–460, Attachment 5, represented a failure to comply with Radiation Work Permit CL–1–7–00518, "C1R17 [Drywell] DW Bioshield Inservice Inspection Activities." This resulted in four individuals entering a LHRA that they had not been specifically authorized to enter. These individuals entered the incorrect location and were inside the area for approximately 2-3 minutes before they noticed that they were in the incorrect area. The individuals knew that they were in the incorrect location when they could not find the nozzles that they planned on inspecting. The individuals exited the area and were simultaneously told to exit the area by the radiation protection technician (RPT) providing remote coverage which demonstrated that the four workers were not in the authorized work area. Immediate corrective actions taken by the licensee included immediately suspending the work that was scheduled to take place within the bioshield associated with this job. Electronic dosimeters and dosimeters were immediately collected from the individuals that entered the area so the dose that was received could be known. The licensee also interviewed all the individuals that were involved in this bioshield entry, and the RPT that performed the brief. These interviews were conducted to understand which parts of the process associated with entry into LHRAs failed and led to this event transpiring. The licensee entered this event into their CAP as AR 04012075. As corrective actions the licensee planned to observe high radiation area and locked high radiation area briefs, for both in house and traveling RPTs. The licensee also planned to modify the bioshield as-low-as-reasonably-achievable (ALARA) plan template to label all accessible bioshield doors with elevation and azimuth.

  The inspectors determined that the performance deficiency was more-than-minor in accordance with IMC 0612, Appendix B, because the finding impacted the program and process attribute of the Occupational Radiation Safety Cornerstone, and adversely affected the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation, in that, the workers entered an area that required the radiation dosimeter to be relocated to the workers knee, and the workers were wearing them on the head for the intended work location. The finding was determined to be of very-low safety significance (Green) in accordance with IMC 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," dated August 19, 2008, because: (1) it did not involve as-low-as-reasonably-achievable planning or work controls; (2) there was no overexposure; (3) there was no substantial potential for an overexposure; and (4) the ability to assess dose was not compromised. The inspectors determined this finding affected the cross-cutting area of human performance in the aspect of resources, where leaders ensure that personnel, equipment, procedures and other resources are available and adequate to support nuclear safety. Specifically, radiation protection leadership failed to ensure that the RPT was capable of meeting the expectations for performing the LHRA briefing in accordance with station procedure RP–AA–460, Attachment 5. [H.1] (Section 2RS1)

REPORT DETAILS

Summary of Plant Status

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

- On July 8, 2017, power was lowered to approximately 5 percent to perform a drywell entry to investigate the lowering oil level on reactor recirculation pump ‘B’ motor lower bearing. The source of the oil leak was identified, repaired, and the Unit returned to full power on July 11, 2017.

- On September 10, 2017, power was lowered to approximately 80 percent to perform control rod pattern adjustments, control rod timing, and testing of the main steam isolation valves, the turbine stop valves/combined intermediate valves and the turbine control valves. The Unit returned to full power on September 11, 2017.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 External Flooding

a. Inspection Scope

The inspectors evaluated the design, material condition, and procedures for coping with the design basis probable maximum flood. The evaluation included a review to check for deviations from the descriptions provided in the Updated Safety Analysis Report (USAR) for features intended to mitigate the potential for flooding from external factors. As part of this evaluation, the inspectors checked for obstructions that could prevent draining, checked that the roofs did not contain obvious loose items that could clog drains in the event of heavy precipitation, and determined that barriers required to mitigate the flood were in place and operable. Additionally, the inspectors performed a walkdown of the protected area to identify any modification to the site which would inhibit site drainage during a probable maximum precipitation event or allow water ingress past a barrier. The inspectors also reviewed the abnormal operating procedure for mitigating the design basis flood to ensure it could be implemented as written.

This inspection constituted one external flooding 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:

- high pressure core spray while performing maintenance on reactor core isolation cooling (RCIC);
- division 1 AC/DC power distribution while performing maintenance on the ‘B’ residual heat removal (RHR) system; and
- emergency reserve auxiliary transformer (ERAT) and division 1 emergency diesel generator (EDG) while performing maintenance on the reserve auxiliary transformer.

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 corrective action program (CAP) with the appropriate significance characterization.

These activities constituted three 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 August 9, 2017, the inspectors performed a complete system alignment inspection of the RCIC system to verify the functional capability of the system. This system was selected because it was considered both safety significant and risk significant in the licensee’s probabilistic risk assessment. The inspectors walked down the system to review mechanical and electrical equipment lineups; electrical power availability; system pressure and temperature indications, as appropriate; component labeling; component lubrication; component and equipment cooling; hangers and supports; operability of support systems; and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding WOs was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the CAP database to ensure that system equipment alignment problems were being identified and appropriately resolved. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

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

- Fire Zone D–2, Division 1 EDG Fuel Tank Room—Elevation 712'-0", 719'-0";
- Fire Zone A–3f, Division 2 Switchgear Room—Elevation 781'-0";
- Fire Zone A–4, Division 1 Battery Room—Elevation 781'-0";
- Fire Zone A–2c, LPCS Pump Room—Elevation 707'-6", 712'-0";
- Fire Zone D–6 and D–6a, Division 2 DG Room—Elevation 737'-0"; and
- Fire Zone D–6b, Division 2 DG Day Tank Room—Elevation 737'-0".

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

These activities constituted six 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 15, 2017, the inspectors observed an unannounced fire brigade activation for a fire on the elevation 762’ in the radwaste relay repair shack. 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
- 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 CAP 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:

- Division 1 EDG fuel oil storage tank room.

Documents reviewed during this inspection are listed in the Attachment to this report. This inspection constituted one internal flooding sample as defined in IP 71111.06–05.

b. Findings

No findings were identified.

1R07 Heat Sink Performance (71111.07T)

.1 Triennial Review of Heat Sink Performance

a. Inspection Scope

The inspectors reviewed completed surveillances, vendor manual information, associated calculations, performance test results and inspection results associated with the 1E12B001B—'B' RHR heat exchanger (HX) and the 1VX06CA—division 1 switchgear heat removal condensing unit HX. These HXs were chosen based on their risk significance in the licensee's probabilistic safety analysis and their important safety-related mitigating system support functions.

The inspectors reviewed the testing, inspection, maintenance, and monitoring of biotic fouling and macro fouling programs to assess the heat transfer capability of the HXs. For both the 'B' RHR HX and the division 1 switchgear heat removal condensing unit HX, the inspectors reviewed whether: (1) the methods used to inspect and clean the HXs were consistent with as-found conditions identified, expected degradation trends, and industry standards; (2) the licensee's inspection and cleaning activities had established acceptance criteria consistent with industry standards; and (3) the as-found results were recorded, evaluated, and dispositioned such that the as-left condition was consistent with the established criteria.

For the 'B' RHR HX the inspectors also reviewed whether: (1) the test method used was consistent with accepted industry practices; (2) the test conditions were consistent with the selected methodology; (3) the test acceptance criteria were consistent with the design basis values; and (4) the results of the HX performance tests met established acceptance criteria. The inspectors also reviewed whether: (1) the test results considered differences between testing conditions and design conditions; (2) the frequency for testing considered previous test result trends; and (3) test results considered test instrument inaccuracies and differences.

In addition, the inspectors reviewed the condition and operation of the two HXs to determine consistency with design assumptions in heat transfer calculations and as described in the USAR. This included an assessment of the number of plugged tubes compared to pre-established limits based on capacity and heat transfer assumptions. The inspectors also reviewed whether the licensee established adequate controls and operational limits to prevent HX degradation due to excessive flow-induced vibration during operation. In addition, eddy current test reports and visual inspection records were reviewed to determine the structural integrity of the HXs.

The inspectors reviewed the performance of the ultimate heat sink (UHS) and safety-related service water system and their subcomponents such as piping, intake screens, pumps, valves, etc., by tests or other equivalent methods to ensure availability and accessibility to the in-plant cooling water systems. Specifically, the inspectors reviewed the UHS in accordance with U.S. Nuclear Regulatory Commission Inspection Procedure 71111.07, “Heat Sink Performance,” Section 02.02, Sub-Section d.2 and Sub-Section d.4.

The inspectors reviewed the results of the licensee’s inspection of the UHS weirs or excavations. The inspectors also reviewed whether identified settlement or movement indicating loss of structural integrity and/or capacity was appropriately evaluated and dispositioned by the licensee. In addition, the inspectors assessed the licensee’s trending and removing debris or sediment buildup in the UHS to ensure sufficient reservoir capacity.

The inspectors reviewed the licensee’s operation of the service water system and UHS. This included a review of procedures for a loss of the service water system or UHS, and a review of the availability and functionality of instrumentation which is relied upon for decision making. In addition, the inspectors assessed whether macro fouling was adequately monitored, trended, and controlled by the licensee to prevent clogging. The inspectors reviewed whether the licensee’s biocide treatments for biotic control were adequately conducted and the results monitored, trended, and evaluated. The inspectors reviewed the service water system susceptibility to strong pump-weak pump interaction, and the licensee’s controls in place for susceptible systems. In addition, the inspectors reviewed design changes to the service water system and the UHS to verify they were not adversely impacted by the changes.

In addition, the inspectors reviewed corrective action documents related to HXs and heat sink performance issues to verify that the licensee had an appropriate threshold for identifying issues and to evaluate the effectiveness of their corrective actions.

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

b. Findings

No findings of significance 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 16, 2017, 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 that training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- licensed operator performance;
- crew’s clarity and formality of communications;

- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- 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 July 8, 2017, the inspectors observed a reactor shutdown to repair the 'B' reactor recirculation pump oil leak. This was an activity that required heightened awareness or was related to increased risk. The inspectors evaluated the following areas:

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

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

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

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations

a. Inspection Scope

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

- low pressure core spray pump room coolers;
- Clinton Power Station (CPS) periodic (a)(3) assessment;
- division 3 shutdown service water; and
- main steam isolation valves (MSIVs).

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 Title 10 C*ode of Federal Regulations* (CFR), Part 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/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 four quarterly maintenance effectiveness samples as defined in IP 71111.12–05.

b. Findings

No findings were identified.

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

.1 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

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

- yellow risk due to planned maintenance on the Division 1 EDG;
- yellow risk due to planned maintenance on the reactor containment isolation cooling system;
- yellow risk due to emergent work on the ‘B’ RHR pump room supply fan 1VY06C;
- yellow risk due to planned replacement of the Division 3 shutdown service water pump;
- yellow risk due to planned maintenance on the Division 1 EDG; and
- yellow risk due to planned maintenance on the ERAT.

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

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

b. Findings

No findings were identified.

1R15 Operability Determinations and Functional Assessments (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- Action Request (AR) 4034246, 1SX01PA: Haze in Division 1 SX Pump Room After Pump Run EC [engineering change] 620644;
- AR 04028568, MCR [Main Control Room] Received Alarm 5062–6C [Division 3 diesel generator control panel];
- AR 04031197, NRC Question on 1RIXCM060TS LCO 3.3.1 During Mode Changes;
- AR 04014871, Non-Safety Related Parts Installed on Actuators; and
- AR 04009845 C1R17 MSIV LLRT [local leak rate test] TS limit exceeded.

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

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

b. Findings

Main Steam Isolation Valve Technical Specification Leakage Limits Exceeded Due to Condition Based Maintenance Approach

Introduction: The inspectors documented a self-revealed finding of very low safety significance and an associated non-cited violation (NCV) of TS limiting condition for operation (LCO) 3.6.1.3, for the failure to follow station procedure ER-AA-200, "Preventative Maintenance Program," Revision 3. Specifically, the licensee utilized a condition based maintenance approach on the MSIVs that failed to monitor and trend equipment performance so that planned maintenance could be performed prior to the MSIVs exceeding the TS leakage limits.

Description: On May 12, 2017, the licensee completed LLRT on the MSIVs in accordance with CPS TS and 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Option B. Based on the results the licensee determined that the 'D' main steam line leakage pathway and the total combined main steam line leakage had exceeded TS leakage limits of LCO 3.6.1.3 for operability. The issue was documented in the CAP as AR 04009845.

The licensee utilized station procedure ER–AA–200, "Preventative Maintenance Program," Revision 3, to implement the condition based maintenance approach for the MSIVs. Station procedure ER–AA–200 defined condition based maintenance as "a maintenance strategy that performs maintenance on components as a result of performing predictive maintenance or performance monitoring and trending. Performance monitoring and trending with condition monitoring techniques are used to detect equipment degradation over time." The procedure defined condition monitoring as "continuous or periodic monitoring, trending and diagnosis of equipment and components using techniques such as leak rates to forecast equipment failures. Condition monitoring results are used to monitor and trend equipment performance so that planned maintenance can be performed prior to equipment failure."

The licensee performed repairs on the affected valves and completed testing to demonstrate that the leakage through the main steam line leakage pathways were within the TS limits. The licensee then performed a CAP evaluation to determine why the valves had exceeded the TS limits. The licensee determined that the cause was expected wear of the MSIV internals. The basis was that all internal maintenance for the MSIVs were based off as-found LLRT test and diagnostic tests (Condition Based Maintenance) and because of this there were no preventative maintenance tasks that were performed on a given frequency. Additionally, the licensee noted that the LLRT results were not completely trendable, nor were future results fully predictable.

The inspectors independently reviewed the issue, applicable station procedures and programs, and questioned whether the administrative limit for the MSIVs being the same as the TS limits met the intent of ER–AA–200 to be specified such that they are an indicator of potential valve or penetration degradation. Once the TS limit is exceeded, the valve is degraded and the pathway must be declared inoperable. The program engineer documented the issue in the CAP as AR 04059351.

The inspectors determined the 'D' main steam line leakage pathway and the total combined main steam line leakage pathways had exceeded TS leakage limits of LCO 3.6.1.3 for operability during the previous operating cycle based on the licensee's determination that the cause was expected wear of the MSIV internals during power operation. Additionally, the inspectors determined the licensee's condition based maintenance approach failed to monitor and trend equipment performance so that planned maintenance could be performed prior to equipment failure.

The licensee modified station procedures to require and document a risk assessment if one or more MSIV leakage rates exceed 50 percent of the allowable value.

<u>Analysis</u>: The inspectors determined that the licensee's failure to adequately use a condition-based maintenance approach on the MSIVs, in accordance with ER–AA–200, was a performance deficiency. The performance deficiency was determined to be more than minor in accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated September 7, 2012, because it impacted the Barrier Integrity cornerstone attribute of configuration control and adversely affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the monitoring and trending of LLRT on the MSIVs did not provide performance data that would allow planned maintenance to the valves prior to the valves failing, resulting in exceeding TS leakage requirements for the MSIVs. Using IMC 0609, Attachment 4, "Initial Characterization of Findings," and Appendix A, "The Significance Determination Process for Findings at Power," Exhibit 2, October 7, 2016, the finding was screened against the Barrier Integrity cornerstone Reactor Containment and did represent an actual open pathway in the physical integrity of reactor containment. The inspectors proceeded to Appendix H, "Containment Integrity Significance Determination Process," and determined that it was a Type B finding that was related to a degraded condition that has potentially important implications for the integrity of the containment, without affecting the likelihood of core damage. The inspectors used Figure 6.1, Road Map for LERF based Risk Significance for Evaluation Type-B Findings at Full Power and determined this finding is of very low safety significance (Green).

The inspectors also determined that this finding affected the cross-cutting area of human performance in the aspect of design margins, where the organization operates and maintains equipment within design margins. Special attention is placed on maintaining fission product barriers, defense-in-depth and safety related equipment. Specifically, the procedure for testing the MSIVs utilized an administrative limit that provided no margin to correct performance prior the valves becoming inoperable. [H.6]

Enforcement: Clinton Power Station TS LCO 3.6.1.3 requires each primary containment isolation valve to be OPERABLE in MODES 1, 2 and 3.

Limiting condition for operation 3.6.1.3, Condition C, states with one or more penetration flow paths with leakage rate not within limit, except for purge valve leakage restore leakage rate to within limit in four hours.

Contrary to the above, between May 2015 and May 12, 2017, primary containment isolation valves became inoperable in applicable modes with one or more penetration flow paths with leakage rates not restored within limit in four hours. Specifically, testing performed by the licensee demonstrated that the leakage rate through the 'D' main steam leakage path was in excess of the TS limit of 100 scfh prior to May 12, 2017, and after the last successful LLRT in May 2015; therefore, the valves were inoperable during the modes of applicability.

As corrective actions the licensee made repairs to the valves and tested the valves prior to returning the unit to the modes of applicability. Because this finding was of very low safety significance and was entered into the CAP as AR 04009845, this violation is being treated as an NCV, in accordance with Section 2.3.2 of the Nuclear Regulatory Commission (NRC) Enforcement Policy. **(NCV 05000461/2017003–01: MSIV TS Leakage Limits Exceeded Due to Condition Based Maintenance Approach)**

1R18 Plant Modifications (71111.18)

.1 Plant Modifications

a. Inspection Scope

The inspectors reviewed the following modification:

- Division 3 shutdown service water pump motor replacement.

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

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

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

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

- testing of the plant service water to shutdown service water 1B header isolation valve 1SX014B;
- testing of the Division 1 cross tie valve SX011A;
- testing of the Division 1 batteries; and
- testing of the Division 1 EDG.

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

This inspection constituted four post-maintenance testing sample 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 9080.02, “Diesel Generator 1B Operability-Manual and Quick Start Operability” (Routine);
- CPS 9080.03, “Diesel Generator 1C Operability-Manual and Quick Start Operability” (Routine); and
- CPS 9069.01, “Shutdown Service Water Operability Test” (in-service 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 two routine surveillance testing samples, and one in-service test sample, 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 29, 2017, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the control room simulator and technical support center to determine whether the event classification, notifications, and protective action recommendations were performed in accordance with procedures. The inspectors also attended the licensee drill critique to compare any inspector-observed weakness with those identified by the licensee staff in order to evaluate the critique and to verify whether the licensee staff was properly identifying weaknesses and entering them into the CAP. As part of the inspection, the inspectors reviewed the drill package and other documents listed in the Attachment to this report.

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

b. Findings

No findings were identified.

2. RADIATION SAFETY

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

.1 High Radiation Area and Very High Radiation Area Controls (02.06)

a. Inspection Scope

The inspectors observed posting and physical controls for high radiation areas and very high radiation areas to assess adequacy.

The inspectors conducted a selective inspection of posting and physical controls for high radiation areas and very high radiation areas to assess conformance with performance indicators.

The inspectors reviewed procedural changes to assess the adequacy of access controls for high and very high radiation areas to determine whether procedural changes substantially reduced the effectiveness and level of worker protection.

The inspectors assessed the controls for high radiation areas greater than 1 rem/hour and areas with the potential to become high radiation areas greater than 1 rem/hour for compliance with TS and procedures.

The inspectors assessed the controls for very high radiation areas and areas with the potential to become very high radiation areas. The inspectors also assessed whether individuals were unable to gain unauthorized access to these areas.

These inspection activities supplemented those documented in the NRC Integrated Inspection Report 05000461/2017002 and constituted one complete sample as defined in IP 71124.01–05.

b. Findings

Introduction: A self-revealed finding of very low safety significance and an associated NCV of TS 5.4.1 was identified when the licensee failed to adequately control access (i.e., exert measures to preclude inadvertent entry) within a Locked High Radiation Area (LHRA).

Description: On May 17, 2017, four individuals began preparing for in-service inspection, nozzle inspection work that was to be conducted on the 752' elevation of the bio shield in Unit 1 containment. These individuals began by receiving an as-low-as-reasonably-achievable (ALARA) briefing for the work that they would be performing on the 752' elevation. The ALARA briefing covered topics such as electronic dosimetry placement based on dose gradient (on the head), the type of inspection work that would be performed and on contact general area dose rates that were present at the nozzles to be inspected. The ALARA brief also covered general area dose rates showing the area the work would be performed met the definition for a LHRA, where dose rates were >1000 mrem in one hour at 30 centimeters. After the ALARA brief, the individuals were to receive a LHRA brief that specifically covered the conditions on the 752' elevation of the bioshield.

The individuals arrived at the LHRA brief to receive their specific area briefing for the 752' elevation as required by the Radiation Work Permit (RWP) CL–1–7–00518, "C1R17 [Drywell] DW Bioshield Inservice Inspection [ISI] Activities." The RWP was an authorization by the licensee's management to perform a specific procedure involving radiation exposure of personnel in a particular area. The RWP contained detailed procedures for every aspect of the work to be done. The RWP CL–1–7–00518, "C1R17 DW Bioshield ISI Activities," required a LHRA briefing. The required content for the LHRA briefing was detailed in Procedure RP–AA–460, "Controls for High and LHRAs," Revision 29. Specifically, Section 1.1 stated, "*The purpose of this procedure is to provide administrative and physical controls and instructions for access to High Radiation Areas and Locked High Radiation Areas"*. Section 4.4.3.3 of Procedure RP–AA–460, Revision 29 stated, in part, *"RP Personnel CONDUCTS and DOCUMENTS LHRA briefing for each non-Radiation Protection individual making LHRA entry per Attachment 5 at the following frequencies…"* During the LHRA briefing for this task, the Radiation Protection Technician (RPT) failed to perform the briefing per Attachment 5 as required by Procedure RP–AA–460. The technician failed to check each box on the attachment as covered, utilize survey or location maps as appropriate to accurately determine work locations, identify sources of radiation and low dose waiting area information in the work area, etc. All of the conditions of Attachment 5 were required by Procedure RP–AA–460 to meet the requirements of a specific area brief for LHRAs and the RWP associated with this task, which consequently ensured compliance with TS 5.4.1. The RPT failing to perform the briefing per Attachment 5, consequently caused the requirements of TS 5.4.1 to not be satisfied.

During the ineffective LHRA brief, the RPT also instructed the individuals to take a travel path that could not get the four workers to the intended and authorized work area. The travel path led to bioshield 777' elevation not 752' elevation, which was contrary to the area conditions that were covered during the LHRA briefing. Area conditions for the 752' elevation were covered during the LHRA briefing not for the 777' elevation. The individuals reverse briefed the area conditions associated with 752' elevation to the RPT and placed the electronic dosimetry in a position on the whole body that was for conditions that were present on 752' elevation not the 777'elevation. The conditions on the 752' elevation led the RPT to make the decision to place electronic dosimetry on the head of the individuals that were performing work in this area due to the present dose gradient. This was different for workers authorized to enter at the 777' elevation where the dosimetry was placed at the knee due to the change in relative location of the radiation source.

After the brief, the individuals proceeded to enter to the bio shield as instructed by the RPT that gave the LHRA brief. The individuals followed the travel path that was given to them by RPT. This travel path led to the 777' elevation, a fact that was not known to the inspection group. Before the individuals entered the bioshield on the 777' elevation, permission was requested via headset by the individuals to enter the bioshield. Permission was inappropriately granted by the Radiation Protection Staff, and the individuals entered the area. This represented a missed opportunity to adequately control access (i.e. exert measures to preclude inadvertent entry) to areas outside of the authorized work area and four individuals entering a LHRA for which they had not been specifically authorized by the RWP.

The individuals then began to search for the nozzle they intended to inspect but realized after 2–3 minutes that they had entered the wrong elevation when the nozzle could not be located in the area. The individuals exited the area and were simultaneously told to exit the area by the RPT providing remote coverage demonstrating the four workers were not in the authorized work area. The job was immediately suspended by the licensee, and all electronic dosimeters and multi-pack dosimeters were collected from the individuals that entered the area. The licensee also interviewed all the individuals that were involved in this bioshield entry, and the RPT that performed the brief.

The licensee entered this event into their CAP as AR 04012075. As corrective actions, the licensee planned to observe high radiation area and locked high radiation area briefs, for both in house and traveling RPTs. The licensee also planned to modify the bioshield ALARA plan template to label all accessible bioshield doors with elevation and azimuth.

Analysis: The inspectors determined that the failure to adequately control access (i.e., exert measures to preclude inadvertent entry) to areas outside of the authorized work area when four individuals entered a LHRA they had not been specifically authorized to enter by the RWP was reasonably within the licensee's ability to foresee and correct and was a performance deficiency. Specifically, the failure to meet all of the requirements of Procedure RP–AA–460, Attachment 5, and the failure to place the electronic dosimetry in the appropriate location for the conditions that the individuals were briefed to enter represents a failure to comply with RWP CL–1–7–00518, "C1R17 DW Bioshield ISI Activities." Consequently the requirements of TS 5.4.1 were not satisfied. The performance deficiency was determined to be more-than-minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because the performance deficiency impacted the program and process attribute of the Occupational Radiation Safety Cornerstone, and adversely affected the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation, in that, the workers entered an area that required the radiation dosimeter to be relocated to the workers knee, and the workers were wearing them on the head for the intended work location.

The finding was determined to be of very low safety significance (Green) in accordance with IMC 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," dated August 19, 2008, because: (1) it did not involve ALARA planning or work controls; (2) there was no overexposure; (3) there was no substantial potential for an overexposure; and (4) the ability to assess dose was not compromised. Immediate corrective actions taken by the licensee included suspending the work that was scheduled to take place within the bioshield associated with this job. Electronic dosimeters and dosimeters were immediately collected from the individuals that entered the area so the dose that was received could be known. The licensee also interviewed all the individuals that were involved in this bioshield entry, and the RPT that performed the brief. These interviews were conducted to understand which parts of the process associated with entry into LHRAs failed and led to this event transpiring. The licensee entered this event into their CAP as AR 04012075.

The inspectors determined this finding affected the cross-cutting area of human performance in the aspect of resources, where leaders ensure that personnel, equipment, procedures and other resources are available and adequate to support nuclear safety. Specifically, radiation protection leadership failed to ensure that the RPT was capable of meeting the expectations for performing the LHRA briefing in accordance with station procedure RP–AA–460, Attachment 5. [H.1]

Enforcement: Technical Specification 5.4.1 requires that written procedures shall be established, implemented and maintained covering activities contained in Regulatory Guide (RG) 1.33, Revision 2, Appendix A, dated February 1978. Procedures specified in Regulatory Guide 1.33 include access control to radiation areas including radiation work permits.

Contrary to the above, on May 17, 2017, the licensee failed to meet all the requirements of Procedure RP-AA-460 "Controls for High and Locked High Radiation Areas" Revision 29. This failure resulted with four individuals entering a LHRA that they had not been specifically briefed to enter. Because this violation was of very-low safety significance and was entered into the licensee's Corrective Action Program as Action Request 04012075 this violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. **(NCV 05000461/2017003–02: Failure to Adequately Control Access in Locked High Radiation Area)**

2RS2 Occupational As-Low-As-Reasonably-Achievable Planning and Controls (71124.02)

.1 Radiological Work Planning (02.02)

a. Inspection Scope

The inspectors selected three to five work activities of the highest exposure significance or involve work in high dose rate areas.

The inspectors reviewed the radiological work planning ALARA evaluations, initial and revised exposure estimates, and exposure mitigation requirements. The inspectors determined if the licensee had reasonably grouped the radiological work into work activities.

The inspectors assessed whether the licensee's planning identified appropriate dose reduction techniques; appropriately considered alternate reduction features; and defined reasonable dose goals. The inspectors evaluated whether the licensee's ALARA assessment had taken into account decreased worker efficiency from use of respiratory protective devices and/or heat stress mitigation equipment. The inspectors determined if the licensee's work planning considered the use of remote technologies and dose reduction insights from industry and plant-specific operating experience. The inspectors assessed whether these ALARA requirements were integrated into work procedure and/or RWP documents.

The inspectors compared the results achieved with the intended dose established in the ALARA planning. The inspectors compared the person-hour estimates provided by work groups to the radiation protection group with the actual work activity time results, and evaluated the accuracy of these time estimates. The inspectors evaluated the reasons for any inconsistencies between intended and actual work activity doses.

The inspectors evaluated whether post-job reviews were conducted to identify lessons learned and entered into the licensee's CAP.

These inspection activities supplemented those documented in IR 05000461/2017002 and constituted one complete sample as defined in IP 71124.02–05.

b. Findings

No findings were identified.

.2 Verification of Dose Estimates and Exposure Tracking Systems (02.03)

a. Inspection Scope

The inspectors assessed whether the assumptions and basis for the current annual collective exposure estimate were reasonably accurate. The inspectors assessed source term reduction effectiveness and reviewed applicable procedures for estimating exposures from specific work activities.

The inspectors reviewed selected occasions with inconsistent or incongruent results from the licensee's intended radiological outcomes to determine whether the cause was attributed to a failure to adequately plan work activities, or failure to provide sufficient management oversight of in-plant work activities, or failure to conduct the work activity without significant rework, or failure to implement radiological controls as planned.

These inspection activities supplemented those documented in IR 05000461/2017002 and constituted one complete sample as defined in IP 71124.02–05.

b. Findings

No findings were identified.

.3 Implementation of As-Low-As-Reasonably-Achievable and Radiological Work Controls (02.04)

a. Inspection Scope

The inspectors reviewed the radiological administrative, operational, and engineering controls planned for selected radiologically significant work activities and evaluated the integration of these controls and ALARA requirements into work packages, work procedures and/or RWPs.

The inspectors compared the radiological results achieved with the intended radiological outcomes and verified that the licensee captured lessons learned for use in the next outage.

These inspection activities supplemented those documented in IR 05000461/2017002 and constituted one complete sample as defined in IP 71124.02–05.

b. Findings

No findings were identified.

.4 Problem Identification and Resolution (02.06)

a. Inspection Scope

The inspectors reviewed self-assessments and/or audits performed of the ALARA program and determined if these reviews identified problems or areas for improvement.

The inspectors assessed whether problems associated with ALARA planning and controls were being identified by the licensee at an appropriate threshold and properly addressed for resolution.

These inspection activities supplemented those documented in IR 05000461/2017002 and constituted one complete sample as defined in IP 71124.02–05.

b. Findings

No findings were identified.

2RS6 Radioactive Gaseous and Liquid Effluent Treatment (71124.06)

.1 Walkdowns and Observations (02.02)

a. Inspection Scope

The inspectors walked down select effluent radiation monitoring systems to evaluate whether the monitor configurations aligned with Offsite Dose Calculation Manual (ODCM) descriptions and to observe the materiel condition of the systems.

The inspectors walked down selected components of the gaseous and liquid discharge systems to evaluate whether equipment configuration and flow paths align with plant documentation and to assess equipment materiel condition. The inspectors also assessed whether there were potential unmonitored release points, building alterations which could impact effluent controls, and ventilation system leakage that communicated directly with the environment.

For equipment or areas associated with the systems selected for review that were not readily accessible, the inspectors reviewed the licensee's materiel condition surveillance records.

The inspectors walked down filtered ventilation systems to assess for conditions such as degraded high-efficiency particulate air/charcoal banks, improper alignment, or system installation issues that would impact the performance or the effluent monitoring capability of the effluent system.

As available, the inspectors observed selected portions of the routine processing and discharge of radioactive gaseous effluent to evaluate whether appropriate treatment equipment was used and the processing activities aligned with discharge permits.

The inspectors determined if the licensee has made significant changes to their effluent release points.

As available, the inspectors observed selected portions of the routine processing and discharging of liquid waste to determine if appropriate effluent treatment equipment was being used and that radioactive liquid waste was being processed and discharged in accordance with procedure requirements and aligned with discharge permits.

These inspection activities constituted one complete sample as defined in IP 71124.06–05.

b. Findings

No findings were identified.

.2 Calibration and Testing Program (02.03)

a. Inspection Scope

The inspectors reviewed calibration and functional tests for select effluent monitors to evaluate whether they were performed consistent with the ODCM. The inspectors assessed whether National Institute of Standards and Technology traceable sources were used, primary calibration represented the plant nuclide mix, secondary calibrations verified the primary calibration, and calibration encompassed the alarm set points.

The inspectors assessed whether effluent monitor alarm set points were established as provided in the ODCM and procedures.

The inspectors evaluated the basis for changes to effluent monitor alarm set points.

These inspection activities constituted one complete sample as defined in IP 71124.06–05.

b. Findings

No findings were identified.

.3 Sampling and Analyses (02.04)

a. Inspection Scope

The inspectors reviewed select effluent sampling activities and assessed whether adequate controls had been implemented to ensure representative samples were obtained.

The inspectors reviewed select effluent discharges made with inoperable effluent radiation monitors and assessed whether controls were in place to ensure compensatory sampling was performed consistent with the ODCM and that those controls were adequate to prevent the release of unmonitored effluents.

The inspectors determined whether the facility was routinely relying on the use of compensatory sampling in lieu of adequate system maintenance.

The inspectors reviewed the results of the Inter-Laboratory Comparison Program to evaluate the quality of the radioactive effluent sample analyses and assessed whether the Inter-Laboratory Comparison Program included hard-to-detect isotopes as appropriate.

These inspection activities constituted one complete sample as defined in IP 71124.06–05.

b. Findings

No findings were identified.

.4 Instrumentation and Equipment (02.05)

a. Inspection Scope

The inspectors reviewed the methodology used to determine the effluent stack and vent flow rates to determine if the flow rates were consistent with plant documentation, and that differences between assumed and actual stack and vent flow rates did not affect the results of the projected public doses.

The inspectors assessed whether surveillance test results for TS required ventilation effluent discharge systems met TS acceptance criteria.

The inspectors assessed calibration and availability for select effluent monitors used for triggering emergency action levels or for determining protective action recommendations.

These inspection activities constituted one complete sample as defined in IP 71124.06–05.

b. Findings

No findings were identified.

.5 Dose Calculations (02.06)

a. Inspection Scope

The inspectors reviewed significant changes in reported dose values compared to the previous radiological effluent release report to evaluate the factors which may have resulted in the change.

The inspectors reviewed radioactive liquid and gaseous waste discharge permits to assess whether the projected doses to members of the public were accurate.

Inspectors evaluated the isotopes that are included in the source term to assess whether analysis methods were sufficient to satisfy detectability standards. The review included the current Part 61 analyses to ensure hard-to-detect radionuclides are included in the source term.

The inspectors reviewed changes in the licensee's offsite dose calculations to evaluate whether changes were consistent with the ODCM and Regulatory Guide (RG) 1.109. Inspectors reviewed meteorological dispersion and deposition factors used in the ODCM and effluent dose calculations to evaluate whether appropriate factors were being used for public dose calculations.

The inspectors reviewed the latest Land Use Census to assess whether changes have been factored into the dose calculations.

For select radioactive waste discharges, the inspectors evaluated whether the calculated doses where within 10 CFR Part 50, Appendix I, and TS dose criteria.

The inspectors reviewed select records of abnormal radioactive waste discharges to ensure the discharge was monitored by the discharge point effluent monitor. Discharges made with inoperable effluent radiation monitors, or unmonitored leakages were reviewed to ensure that an evaluation was made to account for the source term and projected doses to the public.

These inspection activities constituted one complete sample as defined in IP 71124.06–05.

b. Findings

No findings were identified.

.6 Problem Identification and Resolution (02.07)

a. Inspection Scope

Inspectors assessed whether problems associated with the effluent monitoring and control program were being identified by the licensee at an appropriate threshold and were properly addressed for resolution. In addition, they evaluated the appropriateness of the corrective actions for a selected sample of problems documented by the licensee involving radiation monitoring and exposure controls.

These inspection activities constituted one complete sample as defined in IP 71124.06–05.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

Cornerstones: Mitigating Systems, Barrier Integrity, Occupational and Public Radiation Safety

.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 third quarter 2016 through the second quarter 2017. 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 WOs, issue reports, event reports and NRC integrated inspection reports for the period of July 1, 2016, through June 30, 2017, to validate the accuracy of the submittals. The inspectors also reviewed the licensee’s issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator, and none were identified.

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

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index—Residual Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index (MSPI)—Residual Heat Removal (RHR) System PI third quarter 2016 through the second quarter 2017. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99–02, “Regulatory Assessment Performance Indicator Guideline,” Revision 7, dated August 31, 2013, were used. The inspectors reviewed the licensee’s operator narrative logs, issue reports, MSPI derivation reports, event reports and NRC integrated inspection reports for the period of July 1, 2016, through June 30, 2017, to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. 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 MSPI RHR system sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.3 Mitigating Systems Performance Index—Cooling Water Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index—Cooling Water Systems PI third quarter 2016 through the second quarter 2017. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 31, 2013, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, MSPI derivation reports, event reports and NRC integrated inspection reports for the period of July 1, 2016, through June 30, 2017, to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. 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 MSPI cooling water system sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.4 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors sampled licensee submittals for the Occupational Exposure Control Effectiveness PI for the period from the second quarter 2016 through the first quarter 2017. The inspectors used PI definitions and guidance contained in the 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 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 and accumulated dose alarms and 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.

.5 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 first quarter 2016 through the second quarter 2017. The inspectors used PI definitions and guidance contained in the 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. 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.

.6 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/ODCM radiological effluent occurrences PI for the period from the first quarter 2016 through the second quarter 2017. The inspectors used PI definitions and guidance contained in the 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/ODCM 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 CAP at an appropriate threshold, adequate attention was being given to timely corrective actions, and adverse trends were identified and addressed. Some minor issues were entered into the licensee's CAP 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: PSU-C1R17 Mechanical Snubber 1RT01003S Failed Function Test

a. Inspection Scope

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

- AR 04011410; PSU–C1R17 Mechanical Snubber 1RT01003S Failed Function Test

As appropriate, the inspectors verified the following attributes during their review of the licensee's corrective actions for the above condition reports and other related condition reports:

- 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;

- identification of the root and contributing causes of the problem; and
- identification of corrective actions, which were appropriately focused to correct the problem; and
- completion of corrective actions in a timely manner commensurate with the safety significance of the issue.

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

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

b. Findings

Failure to Perform Evaluation to Determine Cause of Snubber Failure

Introduction: The inspectors identified a finding of very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the licensee’s failure to demonstrate compliance with the requirement as prescribed in procedure ER–CL–330, “CPS Snubber Program,” Revisions 1 and 2. Specifically, the licensee failed to perform engineering evaluations to determine the cause of failure of snubbers that did not satisfy their functional testing acceptance criteria.

Description: A snubber is a mechanical device designed to protect the piping system from excess dynamic load caused by seismic disturbances or other transients. The snubber allows for pipe thermal displacements in both tension and compression during normal plant operation, and becomes activated in a seismic event or transient to provide restraint to the piping system and transfers the load to the supporting structure.

The CPS Snubber Program was governed by procedure ER–CL–330, “CPS Snubber Program,” which established the requirements for activities including in-service testing and data evaluation. In each fuel cycle, snubbers within the scope of the program were tested for operational readiness during normal system operation or during system/plant outages in accordance with a sampling plan. The operational readiness tests were performed to ensure that certain parameters were within the established design range. In the event a snubber was found not satisfying the acceptance criteria of the test, the CPS Snubber Program dictated that appropriate corrective actions were to be taken to address the snubber failure and its effects to the components supported by the failed snubber.

Among the corrective actions, one was described in Section 7.1.2 of ER–CL–330, which stated, in part, “[…] IF a snubber fails to satisfy the acceptance criteria of an operational readiness test, THEN an engineering evaluation of the failed snubber shall be performed within 72 hours to determine the cause of the failure […].” Subsections 7.1.2.1 and 7.1.2.2 described the requirements of subsequent actions that were to be taken, which were dependent upon the result of the engineering evaluation. Subsection 7.1.2.1 stated, “The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the operability of other snubbers, irrespective of type, which may be subject to the same failure mode.” Subsection 7.1.2.2 stated, “IF the snubber fails to lock up or fails to move AND the cause of the failure is determined to be a manufacturing or design deficiency, THEN all snubbers of the same type, subject to the same defect, shall be functionally tested. This testing requirement shall be independent of the requirements stated for snubbers not meeting the functional test acceptance criteria."

The inspectors identified the following three representative examples in which the licensee failed to perform an engineering evaluation to determine the failure mode for snubbers that did not pass the functional test acceptance criteria:

- On October 20, 2013, the licensee performed the functional testing of Snubber 1RH17002S during C1R14 and identified that it did not satisfy the functional testing acceptance criteria. The mechanical type PSA–1/2 size snubber, which provided seismic restraint to the 4-inch RHR piping in the RHR 'B' heat exchanger room, failed to meet the drag test acceptance criteria in the tension direction when it locked up at approximately 1.25-inch position. However, the inspectors were not able to identify the existence of an engineering evaluation that determined the cause of failure. Specifically, the inspectors were able to identify the vendor failure analysis containing details of the as-found conditions of the failed snubber, but were not able to identify an engineering evaluation utilizing that information and other pertinent information to determine the cause of failure. The inspectors noted that this particular snubber was later permanently deleted under EC 397737. Nevertheless, the inspectors also noted the requirement from American Society of Mechanical Engineers OM Code, 2004 Ed, Subsection ISTD–1700, which stated that "snubbers may be deleted from the plant based on analysis of the affected piping system. When an unacceptable snubber is deleted, the deleted snubber shall nevertheless be considered in its respective examination population, examination category, or failure mode group (FMG) for determining the corrective action."
- On May 7, 2015, the licensee performed the functional testing of Snubber 1RT06014S during C1R15 and identified that it did not satisfy the functional testing acceptance criteria. The mechanical type PSA–1/2 size snubber, which provided seismic vertical restraint to the safety-related 3" reactor water clean-up (RWCU) piping upstream of 'A' RWCU pump, failed the as-found activation portion of the functional test. However, the inspectors were not able to identify the existence of an engineering evaluation that determined the cause of failure. Specifically, the inspectors were able to identify the vendor failure analysis containing details of the as-found conditions of the failed snubber, but were not able to identify an engineering evaluation utilizing that information and other pertinent information to determine the cause of failure.
- On May 16, 2017, the licensee performed the functional testing of Snubber 1RT01003S during C1R17 and identified that it did not satisfy the functional testing acceptance criteria. The mechanical type PSA–1 size snubber, which provided seismic lateral restraint to the safety-related 4-inch RWCU piping that branched off the 20-inch reactor recirculation loop 'B' piping, failed the as-found drag test portion of the functional test, and was found to be locked up. However, the inspectors were not able to identify the existence of an engineering evaluation that determined the cause of failure. Specifically, the inspectors were able to identify the vendor failure analysis containing details of the as-found conditions of the failed snubber, but were not able to identify an engineering evaluation utilizing that information and other pertinent information to determine the cause of failure.

The inspectors determined that no operability concern existed because, as part of their corrective actions, the licensee evaluated the components affected by the failed snubber for adverse impact in each instance and determined that no operability issues existed. Further, the licensee replaced the failed snubbers with fully functional ones to restore the affected piping systems to the original design configuration. The licensee documented the inspectors' concerns into their CAP as AR 04015342 and AR 04041302 dated May 26, 2017, and August 7, 2017, respectively.

Analysis: The inspectors determined that the licensee's failure to perform engineering evaluations in accordance with station procedure ER–CL–330, to determine the cause of failure of snubbers that did not satisfy their functional testing acceptance criteria 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," because it was associated with the Mitigating Systems cornerstone attribute of Protection against External Factors and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability for mitigating systems to respond to initiating events. Specifically, compliance with ER–CL–330 would ensure the failed snubber was evaluated for the cause of failure, and would direct the licensee to identify other snubbers that may have been vulnerable to the same type of deficiency. This would ensure that any potential undesired loading on the piping system could be avoided and the affected safety-related RHR and reactor water cleanup (RWCU) piping systems could continue to perform their design function of maintaining the pressure boundary and structural integrity following a postulated design basis seismic event.

The inspectors determined the finding could be evaluated using the Significance Determination Process (SDP) in accordance with IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," for the Mitigating Systems cornerstone and then Exhibit 4, "External Events Screening Question." The finding screened as having very low safety significance (Green) because the inspectors answered "No" to Questions 1 and 2 of Exhibit 4.

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. Specifically, the licensee failed to establish a systematic approach to evaluating snubbers that did not meet the acceptance criteria to ensure all required aspects were addressed. [H.13]

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

Section 7.1.2 of Procedure ER–CL–330, stated, in part, "[…] IF a snubber fails to satisfy the acceptance criteria of an operational readiness test, THEN an engineering evaluation of the failed snubber shall be performed within 72 hours to determine the cause of the failure […]."

Contrary to the above, as of September 30, 2017, the licensee failed to perform an engineering evaluation to determine the cause of failure of snubbers tested on

October 20, 2013, May 7, 2015, and May 16, 2017, that did not satisfy its functional testing acceptance criteria.

The licensee entered this issue into their CAP as ARs 04015242 and 04041302. As corrective actions the licensee evaluated the components affected by the failed snubber and determined that no operability issues existed. This violation is being treated as an NCV, in accordance with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000461/2017003–03: Failure to Perform Engineering Evaluation to Determine the Cause of Failure of Snubbers)**

.3 Annual Follow-up of Selected Issues: Corrective Action for IR 3949655 not Implemented as Intended

a. Inspection Scope

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

- AR 04013626: Corrective Action for AR 3949655 not Implemented as Intended.

The inspectors noted this AR during routine review of condition reports and determined to follow up because the corrective action documented in the issue report was associated with a previous NRC NCV and work had been completed with the corrective action not implemented as intended.

As appropriate, the inspectors verified the following attributes during their review of the licensee's corrective actions for the above condition reports and other related condition reports:

- 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;
- identification of corrective actions, which were appropriately focused to correct the problem; and
- completion of corrective actions in a timely manner commensurate with the safety significance of the issue.

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

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

b. Observations and Assessments

The licensee identified the issue as part of the CAP follow up for AR 03949655, residual heat removal pump ‘C’ failed to start. The action was to revise station procedure CPS 8410.21C001, “Westinghouse DHP Circuit Breaker Checklist.” The reviewer noted that the procedure revision was not implemented as intended.

The licensee immediately documented the issue as AR 04013626 and reviewed previous work to see if work had been performed per the procedure revision. The licensee determined it had been worked one time for breaker B33C1A2, reactor recirculation pump 1A breaker.

The licensee reviewed the associated work order and procedure, and determined the work had been done correctly. The licensee revised the procedure to reflect the intent of the corrective action. The inspectors did not identify any discrepancies with the licensee's disposition of this issue in accordance with their corrective action program.

c. Findings

No findings were identified.

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

.1 Unexpected Reactor Recirculation Pump Runback During Unit Down Power

a. Inspection Scope

On May 7, 2017, the licensee was shutting down the reactor to support the C1R17 refueling outage. When the operators were in the process of shifting from a turbine driven feedwater pump (TDFWP) to the motor driven feedwater pump (MDFWP) the water level in the reactor vessel unexpectedly increased and reached the level 4 setpoint. As expected the reactor recirculation pump ran back resulting in a 9.2 percent power reduction. The inspectors observed the operator actions associated with this event. They also performed a review of the corrective action documents, causal analysis and completed procedures associated with this issue.

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

b. Findings

Flow Control Valves Not Locked Out Results in Reactor Recirculation Pump Runback

Introduction: The inspectors documented a self-revealed finding of very low safety significance and an associated NCV of TS 5.4.1, "Procedures," for the licensee's failure to establish sufficient instructions in station procedure CPS 3103.01, "Feedwater (FW)," Revision 31e, for changing modes of operation for the nuclear steam supply system. Specifically, the station procedure did not provide instructions requiring the locking out of the flow control valves (FCVs) to prevent a reactor recirculation (RR) FCV runback while changing the feedwater pump lineup, resulting in an unexpected plant transient and 9.2 percent change in reactor power.

Description: On May 7, 2017, Clinton Power Station was shutting down the reactor to support refueling outage C1R17. The control room operating crew was performing station procedure CPS 3103.01, "Feedwater (FW)," to shift the feedwater pump alignment from two turbine driven feedwater pumps (TDRFP) to one TDRFP and a motor driven reactor feedwater pump. During the realignment, water level in the reactor pressure vessel (RPV) lowered to Level 4 and caused a RR FCV runback. This RR FCV runback unexpectedly lowered reactor power by 9.2 percent and closed the RR FCVs.

Station procedure CPS 3103.01, "Feedwater (FW)," Section 8.1, "TDRFP shutdown," contained a caution that stated "A Level 4 runback will occur at 30.8" (refer to Appendix B) if less than two TDRFPs are in operation," to inform the crew of the potential of an RR FCV runback.

Additionally, the first step of Section 8.1 for shutting down a TDRFP stated in part, "Discuss with the control room supervisor the need to lockout the RR FCVs during the TDRFP S/D [shutdown]. Lockout the RR FCVs per station procedure CPS 3302.02, 'Reactor Recirculation Flow Control Hydraulic System,' if deemed necessary." The operations crew discussed the current plant status and decided to not lockout the RR FCVs. Locking out the FCVs prior to the reactor feedwater pump alignment shift would have prevented the RR FCV runback.

The control room operating crew responded to the RR FCV runback transient by entering CPS procedures 4800.01, "Abnormal Reactor Coolant Flow" and, CPS 4002.01, "Abnormal RPV Level/Loss of FW at Power." The crew stabilized the plant, reset the RR FCV runback, and exited the abnormal procedures about 20 minutes later.

The licensee subsequently revised CPS 3103.01 to clearly state that lockout of the RR FCVs is required prior to commencing the operation.

<u>Analysis</u>: The inspectors determined the licensee's failure to establish sufficient instructions in station procedure CPS 3103.01 for changing modes of operation for the nuclear steam supply system, as required by RG 1.33, was a performance deficiency. Specifically, the licensee failed to establish sufficient procedural guidance for preventing an RR FCV runback while shifting feedwater pumps during a plant shutdown, resulting in an unexpected plant transient and 9.2 percent change in reactor power. The performance deficiency was more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated September 7, 2012, because the finding was associated with the procedure quality attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the failure to have adequate procedures for shifting feedwater pumps during a plant shutdown on May 7, 2017, resulted in an unexpected recirculation pump runback and a 9.2 percent change in reactor power. 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 (Green) because the event did not cause a reactor scram.

The inspectors determined this finding affected the cross-cutting area of human performance in the aspect of conservative bias, where individuals use decision making practices that emphasize prudent choices over those that are simply allowable and a proposed action is determined to be safe in order to proceed, rather than unsafe in order to stop. Specifically, the procedure provided for the option to lockout the reactor recirculation flow control valves if deemed necessary during a shift of the reactor feedwater pumps and the operations crew did not make the prudent choice of locking out the valves before determining that it was safe to proceed. [H.14]

Enforcement: Technical Specification 5.4.1, "Procedures," states, in part, "Written procedures shall be established, implemented and maintained covering the applicable procedures recommended in RG 1.33, Revision 2, Appendix A."

Section 4 of RG 1.33 requires, in part, instructions for energizing, filling, venting, draining, startup, and changing modes of operation should be prepared, as appropriate for the Nuclear Steam Supply System (Vessel and Recirculation System).

Contrary to the above, on May 7, 2017, the licensee failed to establish written procedures covering the applicable procedures recommended in RG 1.33, Appendix A, Section 4. Specifically, the licensee failed to establish instructions, in station procedure CPS 3103.01, "Feedwater (FW)" which is used to change modes of operation as part of the Nuclear Stream Supply System. The station procedure did not provide instructions requiring locking out the FCVs to prevent an RR FCV runback during the feedwater pump alignment resulting in an unexpected plant transient and 9.2 percent change in reactor power.

As corrective actions, the licensee revised CPS 3103.01 to require that the FCVs be locked out prior to shifting reactor feedwater pumps. Because the violation was of very low safety significance and was entered into the licensee's CAP as AR 04007861, this violation is being treated as an Non-Cited Violation (NCV), consistent with Section 2.3.2 of the Enforcement policy. **(NCV 05000461/2017003–04: Flow Control Valves Not Locked Out Results in Reactor Recirculation Pump Runback)**

.2 (Closed) Licensee Event Report 05000461/2017–006–00: Secondary Containment Inoperable During Mode Change Due to Doors Propped Open.

a. Inspection Scope

On June 2, 2017, while the plant was in MODE 4, maintenance personnel were performing welding activities associated with the 'B' RWCU pump. In support of these activities operations provided authorization to prop open both RWCU 'B' pump doors, which was a secondary containment boundary. The licensee did not use the plant barrier integrity (PBI) program which would have provided guidance on actions necessary to evaluate and compensate for the impaired barrier. As the plant prepared for startup and the transition to operating MODE 2, the operators made a plant announcement requiring primary and secondary containment be established. The announcement was not heard by the personnel working on the 'B' RWCU pump, therefore they did not close the pump room doors. The doors were found propped open shortly after the plant entered MODE 2 by operations shift personnel traversing through the area. Technical Specification 3.6.4.1, "Secondary Containment," actions were then entered to restore secondary containment. Both RWCU doors were later closed and secondary containment was restored. The inspectors reviewed the licensee's response to this issue, including a review of the corrective actions, causal products, operator logs, and licensee procedures. The inspectors concluded there was a violation associated with this issue.

This licensee event report (LER) is closed.

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

b. Findings

Failure to Establish Secondary Containment Prior to Entering MODE 2

Introduction: The inspectors documented a self-revealed finding of very low safety significance and associated NCV of TS LCO 3.0.4, for the failure to follow station procedure CC–AA–201, "Plant Barrier Control Program," Revision 11. Specifically, the licensee entered Mode 2 from Mode 4 without meeting the requirements of LCO 3.0.4 for entering a Mode when an applicable LCO is not met. The licensee had not met LCO 3.6.4.1 because the doors to the 'B' RWCU pump room were both open instead of being closed to make secondary containment operable as required in MODE 2.

Description: On June 2, 2017, at 0241, Clinton Power Station Unit 1 declared the unit had entered MODE 2. At approximately 0300 hours, a senior rector operator was transiting through the control building on Elevation 737'. While going past the 'B' RWCU pump room he noted the outer pump room door was open. Upon looking into the pump room access through the outer door he noted that the inner door was also open with welding cables routed into the room. The work control supervisor and the control room supervisor were notified, and the secondary containment was declared inoperable and the applicable TS LCO was entered. The senior reactor operator monitored the completion of the work and the restoration of the inner and outer doors to the closed position. Secondary containment was declared operable at 0324 hours. The licensee documented the issue in the CAP as AR 04017613.

The licensee performed a CAP evaluation to determine why the welding cables were run through the 'B' RWCU pump room doors making secondary containment inoperable. The evaluation determined that the work control supervisor gave authorization to open the doors to support welding in the 'B' reactor water cleanup pump room while the unit was in MODE 4 without using the PBI process in accordance with station procedure CC–AA–201. Since the procedure was not implemented, a plant barrier impairment permit was not initiated and the restraints database for MODE 2 was not updated. Additionally, the policy for recognizing the need for a PBI permit and routing the permits to the necessary departments had been changed with regards to the maintenance personnel developing and routing the PBI form prior to going to the work control supervisor for approval. The maintenance personnel also failed to recognize the need for a PBI permit.

The evaluation stated that a knowledge gap existed with the use of the PBI process. Station personnel did not know the process for routing a PBI permit and did not know when a PBI permit was required with the exception of fire barriers.

The inspectors reviewed the AR and CAP evaluation and did not identify any issues with the conclusions.

The licensee planned to conduct training for site personnel as corrective actions.

Analysis: The inspectors determined that the licensee's failure to follow station procedure CC–AA–201 and initiate a PBI permit for the 'B' RWCU pump room doors 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 impacted the Barrier Integrity cornerstone attribute of configuration control and adversely affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to follow the station procedure by not identifying that the open doors required a PBI permit that would have identified the doors as a constraint to entering MODE 2 resulted in the unit transitioning to MODE 2 with the secondary containment inoperable. Using IMC 0609, Attachment 4, "Initial Characterization of Findings," and Appendix A, "The Significance Determination Process for Findings at Power," Exhibit 2, October 7, 2016, the finding was screened against the Barrier Integrity cornerstone and determined to be of very low safety significance (Green) because the finding only represented a degradation of a radiological barrier function provided for the auxiliary building.

The inspectors determined that this finding affected the cross-cutting are of human performance in the aspect of training, where the organization provides training and ensures knowledge transfer to maintain a knowledgeable, technically competent work force and instill nuclear safety values. Specifically, station personnel did not know the process for routing a PBI permit and did not know when a PBI permit was required. [H.9]

<u>Enforcement</u>: Clinton Power Station TS LCO 3.6.4.1 requires the secondary containment to be OPERABLE in MODE 2.

Clinton Power Station Technical Specification LCO 3.0.4, states, in part, when an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made: (1) When the associated Actions to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time; (2) After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate, exceptions to this specification are stated in the individual specifications; or (3) When an allowance is stated in the individual value, parameter, or other specification.

Contrary to the above, on June 2, 2017, the licensee transitioned the unit into MODE 2 without secondary containment operable as required by LCO 3.6.4.1. Specifically, secondary containment was inoperable due to the 'B' reactor water cleanup pump room doors being open and TS LCO 3.0.4 conditions were not met as the associated actions did not allow continued operation for an unlimited period of time, the licensee did not perform a risk assessment and an allowance was not stated for this LCO.

As corrective actions the licensee immediately entered the applicable LCO for secondary containment for MODE 2. The licensee then established secondary containment and exited the LCO. Additionally, the licensee planned to perform site training on the PBI process. Because this finding was of very low safety significance and was entered into the CAP as AR 04017613, this violation is being treated as an NCV, in accordance with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000461/2017003–05: Failure to Establish Secondary Containment Prior to Entering MODE 2)**

.3 <u>(Closed) Licensee Event Report 05000461/2017–003–00: Implementation of Enforcement Guidance Memorandum 11–003, Revision 3</u>

a. <u>Inspection Scope</u>

From May 9, 2017, through May 28, 2017, CPS performed operations with the Potential to Drain the Reactor Vessel (OPDRV) while in Mode 5 without an operable secondary containment. An OPDRV is an activity that could result in the draining or siphoning of the reactor pressure vessel water level below the top of fuel, without crediting the use of mitigating measures to terminate the uncovering of fuel. Secondary containment is required by TS 3.6.4.1 to be operable during OPDRV. The required action for this specification is to suspend OPDRV operations. Therefore, entering the OPDRV without establishing secondary containment integrity was considered a condition prohibited by TS as defined by 10 CFR 50.73(a)(2)(i)(B).

The NRC issued Enforcement Guidance Memorandum (EGM) 11–003, Revision 3, on January 15, 2016, to provide guidance on how to disposition boiling water reactor licensee noncompliance with TS containment requirements during OPDRV operations. The NRC considers enforcement discretion related to secondary containment operability during MODE 5 OPDRV activities appropriate because the associated interim actions necessary to receive the discretion ensure an adequate level of safety by requiring licensees' immediate actions to: (1) adhere to the NRC plain language meaning of OPDRV activities; (2) meet the requirements which specify the minimum makeup flow rate and water inventory based on OPDRV activities with long drain down times; (3) ensure that adequate defense in depth is maintained to minimize the potential for the release of fission products with secondary containment not operable by (a) monitoring RPV level to identify the onset of a loss of inventory event, (b) maintaining the capability to isolate the potential leakage paths, (c) prohibiting MODE 4 (cold shutdown) OPDRV activities, and (d) prohibiting movement of irradiated fuel with the spent fuel storage pool gates removed in MODE 5; and (4) ensure that licensees follow all other MODE 5 TS requirements for OPDRV activities.

The inspectors reviewed LER 2017–003–00 for potential performance deficiencies and/or violations of regulatory requirements. The inspectors also reviewed the stations implementation of the EGM during OPDRVs:

1. The inspectors observed that the OPDRV activities were logged in the control room narrative logs, that the log entry appropriately recorded the standby source of makeup water designated for the evolutions, and defense in-depth criteria were in place.

2. The inspectors noted that the reactor vessel water level was maintained at least 22 feet and 8 inches over the top of the reactor pressure vessel flange as required by TS 3.9.6. The inspectors also verified that at least one safety-related pump was the standby source of makeup designated in the control room narrative logs for the evolutions. The inspectors confirmed that the worst case estimated time to drain the reactor cavity to the reactor pressure vessel flange was greater than 24 hours.

3. The inspectors reviewed Engineering Change documents which calculated the time to drain down during these activities and the feasibility of pre-planned

actions the station would take to isolate potential leakage paths during these periods of time. The inspectors verified that the OPDRVs were not conducted in Mode 4 and that the licensee did not move irradiated fuel during the OPDRVs. The inspectors noted that CPS had in place a contingency plan for isolating the potential leakage path and verified that two independent means of measuring reactor pressure vessel water level were available for identifying the onset of loss of inventory events.

4. The inspectors verified that all other Technical Specifications requirements were met during the May 9, 2017, through May 28, 2017, OPDRVs with secondary containment inoperable.

Technical Specification 3.6.4.1 requires, in part, that secondary containment shall be operable during OPDRVs. Technical Specification 3.6.4.1, Condition C, requires the licensee to initiate action to suspend OPDRVs immediately when secondary containment is inoperable. Contrary to the above, from May 9, 2017, through May 28, 2017, CPS performed eight OPDRV activities while in Mode 5 without an operable secondary containment. Specifically, the station performed the following OPDRV activities without an operable secondary containment:

- Shifting RHR 'B' shutdown cooling to the upper pools;
- Isolating, draining and filling reactor recirculation Loops A and B;
- Intermediate range monitor dry tube replacement;
- Repairing jet pump flow transmitters;
- Reactor water cleanup system pump startup and letdown;
- Starting and aligning RHR 'A' for shutdown cooling to the feedwater line 'A' and 'A' upper pool return line;
- Performing maintenance on control rod hydraulic control units; and
- Shift from RHR a to RHR 'B' shutdown cooling.

Because the violation occurred during the discretion period described in EGM 11–003, Revision 3, the NRC is exercising enforcement discretion in accordance with Section 3.5, "Violations Involving Special Circumstances," of the NRC Enforcement Policy and, therefore, will not issue enforcement action for this violation.

In accordance with EGM 11–003, Revision 3, each licensee that receives discretion must submit a license amendment request within 4 months of the NRC staff's publication in the Federal Register of the notice of availability for a generic change to the standard TS to provide more clarity to the term OPDRV. The licensee submitted a license amendment request on May 1, 2017, which follows the guidance in Technical Specification Task Force traveler TSTF–542, which is the generic resolution to this issue.

This LER is now closed.

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

.4 <u>(Closed) Licensee Event Report 05000461/2017–001–00: Failure of the 138 kV Offsite Power Source Results in a Loss of Secondary Containment Vacuum</u>

a. <u>Inspection Scope</u>

On February 24, 2017, the station experienced a loss of the 138 kilovolt (kV) line, resulting in a loss of the emergency Reserve Auxiliary Transformer (ERAT) and associated static VAR compensator. The Division 1 safety bus was being fed from the ERAT. When the 138 kV line lost power, the Division 1 bus transferred to the Reserve Auxiliary Transformer (RAT). This transfer caused a trip of the Division 1 fuel building ventilation (VF). As a result of the VF trip, secondary containment differential pressure (dP) degraded to less than the TS required 0.25” vacuum water column, causing the licensee to declare secondary containment inoperable and enter TS 3.6.4.1, “Secondary Containment.” The licensee manually actuated standby gas treatment and restored secondary containment to within the TS allowable value within several minutes.

The licensee performed an investigation and determined the inoperability of secondary containment was caused by the fact that the circuit design of VF was not adequately robust to withstand loss of the 138 kV feed. The licensee plans to install a modification to the VF circuitry that includes a time delay feature to allow for the transfer of the VF system to the alternate power supply in an under voltage event.

The inspectors reviewed the CAP evaluation report as well as the preliminary proposed corrective actions to address this issue. The inspectors also reviewed procedures, USAR, previous loss of secondary containment events, and previous CAP documents related to this system. Based on this review, the inspectors did not identify any performance deficiencies related to this event.

This LER is closed.

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

b. <u>Findings</u>

No findings were identified.

.5 <u>(Closed) Licensee Event Report 05000461/2016–011–00: Incorrect Calculation Method Used to Demonstrate Control Room Habitability</u>

a. <u>Inspection Scope</u>

During the 2016 Component Design Basis Inspection the NRC identified that licensee calculation C–020, “Reanalysis of Loss of Coolant Accident (LOCA) Using Alternate Source Terms,” incorrectly took credit for the dual control room ventilation (VC) supply inlets being single failure proof (SFP). This allowed the calculation to reduce one of the dose terms by a factor of four. However, the inlets to VC at Clinton are not SFP. This error, when corrected, resulted in a calculated dose of greater than five rem total effective dose equivalent to occupants of the main control room. This was above the allowable dose in the accident analysis. The licensee received a Green finding and an associated NCV for this issue, documented in NRC Component Design Bases Inspection inspection report (IR) 05000461/2016009 (ML17013A253).

The licensee performed an investigation to determine the apparent and contributing causes, and concluded the cause of the incorrect calculation was the VC USAR sections did not discuss whether the intake structures were single failure proof. The licensee also defined the contributing cause to be a lack of technical rigor on the part of the preparer. As corrective actions, the licensee revised C–020 with the correct inputs and is planning on reviewing additional calculations to verify inputs and assumptions.

The inspectors reviewed the apparent cause evaluation report as well as the preliminary proposed corrective actions and extent of condition to address this issue. The inspectors also reviewed procedures, the USAR, the calculation, the previously documented violation in IR 2016009, and previous CAP documents related to this issue. Based on this review, the inspectors did not identify any performance deficiencies related to this event.

This LER is closed.

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

b. Findings

No findings were identified.

.6 (Closed) Licensee Event Report 05000461/2017–004–00: Main Steam Isolation Valve Local Leak Rate Test Limit Exceeded During Refueling Outage

a. Inspection Scope

During the CPS refueling outage (C1R17) on May 12, 2017, CPS tested its MSIVs and discovered the as-found leakage for main steam line 'D' exceeded the TS 3.6.1.3, "Primary Containment Isolation Valves," surveillance requirements. An investigation determined the as-found condition of the MSIVs did not reveal any damage, only what the licensee classified as normal wear indications. The MSIVs were repaired so the as-left leakage past all the effected MSIVs complied with limits established by TS surveillance requirements. The licensee reported this event due to principle plant safety barriers being seriously degraded under the provisions of 10 CFR 50.73(a)(2)(ii)(A). The inspectors reviewed the licensee's response to this issue, including a review of the corrective actions, causal products, previous evaluations, and relevant licensee procedures. The inspectors also reviewed the reportability criteria associated with this event and identified this issue should have also been reported as a condition prohibited by TS under the provisions of 10 CFR 50.73(a)(2)(i)(B). The inspectors communicated this to the licensee and they entered it into their corrective action program as AR 04061443. The failure to comply with 10 CFR 50.73(a)(2)(i)(B) constitutes a minor violation that is not subject to enforcement action in accordance with the NRC's enforcement policy.

This LER is closed.

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

b. Findings

The inspectors identified a finding associated with this issue as documented in Section 1R15 of this report.

4OA6 Management Meetings

.1 Exit Meeting Summary

On October 19, 2017, 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 July 13, 2017.
- The inspection results for the Triennial Review of Heat Sink Performance were discussed with Mr. T. Stoner, Site Vice President, on August 18, 2017.
- The inspection results for the Radiation Safety Program review with Mr. B. Kapellas, Plant Manager, on September 28, 2017.

The inspectors confirmed that none of the potential report input discussed was considered proprietary. Proprietary material received during the inspection was returned to the licensee.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

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

U.S. Nuclear Regulatory Commission

L. Kozak, Acting Chief, Reactor Projects Branch 1

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000461/2017003–01	NCV	MSIV TS Leakage Limits Exceeded Due to Condition Based Maintenance Approach (Section 1R15)
05000461/2017003–02	NCV	Failure to Adequately Control Access in Locked High Radiation Area (Section 2RS1)
05000461/2017003–03	NCV	Failure to Perform Engineering Evaluation to Determine the Cause of Failure of Snubbers (Section 4OA2.2)
05000461/2017003–04	NCV	Flow Control Valves Not Locked Out Results in Reactor Recirculation Pump Runback (Section 4OA3.1)
05000461/2017003–05	NCV	Failure to Establish Secondary Containment Prior to Entering MODE 2 (Section 4OA3.2)

Closed

05000461/2017003–01	NCV	MSIV TS Leakage Limits Exceeded Due to Condition Based Maintenance Approach (Section 1R15)
05000461/2017003–02	NCV	Failure to Adequately Control Access in Locked High Radiation Area (Section 2RS1)
05000461/2017003–03	NCV	Failure to Perform Engineering Evaluation to Determine the Cause of Failure of Snubbers (Section 4OA2.2)
05000461/2017003–04	NCV	Flow Control Valves Not Locked Out Results in Reactor Recirculation Pump Runback (Section 4OA3.1)
05000461/2017003–05	NCV	Failure to Establish Secondary Containment Prior to Entering MODE 2 (Section 4OA3.2)
05000461/2017–006–00	LER	Secondary Containment Inoperable During Mode Change Due to Doors Propped Open (Section 4OA3.2)
05000461/2017–003–00	LER	Implementation of Enforcement Guidance Memorandum 11–003, Revision 3 (Section 4OA3.3)
05000461/2017–001–00	LER	Failure of the 138kV Offsite Power Source Results in a Loss of Secondary Containment Vacuum (Section 4OA3.4)
05000461/2016–011–00	LER	Incorrect Calculation Method Used to Demonstrate Control Room Habitability (Section 4OA3.5)
05000461/2017–004–00	LER	Main Steam Isolation Valve Local Leak Rate Test Limit Exceeded During Refueling Outage (Section 4OA3.6

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

- CPS 4303.02; "Abnormal Lake Level," Revision 12b
- ER–CL–450–2000; "Lake Monitoring," Revision 2

1R04 Equipment Alignment

- AR 03977720; USAR and Tech Spec Bases Discrepancy, RCIC Not an ESF System
- AR 0397661; H03G—Instrumentation Wires on RCIC Turbine Skid
- AR 04042374; NRC Request Information on RCIC System and Procedure
- CPS 3506.01V002; Diesel Generator and Support Systems Instrument Valve Lineup; Revision 11b
- CPS 3506.01P001; Completing the Diesel Generator Controls Lineup; Revision 5
- CPS 3501.01E001; High Voltage Auxiliary Power System Electrical Lineup; Revision 14
- CPS 3503.01E001; Battery and DC Distribution Electrical Lineup; Revision 14a
- CPS 3310.01V002; RCIC Instrument Valve Lineup; Revision 9e
- CPS 3310.01V001; Reactor Core Isolation Cooling Valve Lineup; Revision 12e
- CPS 3310.01E001; Reactor Core Isolation Cooling Electrical Lineup; Revision 16
- CPS 3506.01V001; Diesel Generator and Support Systems Valve Lineup; Revision 13a
- CPS 3506.01P001; Completing the Diesel Generator Controls Lineup; Revision 5
- AR 4042400; RR System Exceeds Maintenance Rule Reliability Criteria
- AR 2702814; 1E51–F019 Was Not Tested During Performance of 9054.02
- AR 2705353; THU: Follow Up to IR 2702814
- AR 2732407; GETARS Channel 30 out of Calibration
- AR 2732443; Wires Found not Attached to Terminal Block—CCP
- AR 2732593; 1E51–F077 Indicates Mid Position After Trying to Stroke
- AR 2733000; Enhancement to 9058.02D001
- AR 2733105; During 9054.05 RCIC Pump Outboard Bearing Oil Level Went Low
- AR 2733106; Air Seen in RCIC Oil Sightglass
- AR 2735283; RCIC SOW WW 1644 LL
- AR 3966525; Received 5063–1D During RCIC Operability Run
- AR 3970029; Work at E–8 Pulled Into WW 1705
- AR 4003048; Assign Project Coordinator For NDE RCIC Tank Bottom—NER
- AR 4003625; OOT On 9030.01C006 on B21–N692A and B21–N692E
- AR 4010156; PSU: 9861.02D017 Test Set a High Flow
- AR 4015451; PSU: Leakage Identified During C1R17 Pressure Test—1E51F064
- AR 4015453; PSU: 1E51–F064 has 60 DPM Packing Leak ID'd During RX Press
- AR 4015564; 1E51–F046 will not Operate from MCR
- AR 4015792; Unexpected Indications During 9054.03
- AR 4016143; Cancelled WO Lead to Failure to Fix Broken Gauge
- AR 4016274; PMT Task for WO 1862980–07 not Completed
- AR 4016303; C1R17 LL: CPS 9432.06 Procedure Enhancement
- AR 4030932; Replace 1E51F387 EIN Tag and Drain Line Needs Screen—CCP

- AR 4034181; Point 8/9 On 1TRCM326 (RCIC) Approaching Alarm Set Point
- AR 4035641; CPS 9030.001C041 ATM OOT Adjustment
- AR 4035919; Add NDE Task to MO WO for 9054.02
- AR 4035927; CPS 3310.01 RCIC Procedure Enhancement
- AR 4035945; Process Computer Digital Point E51DC001 did not Change State
- AR 4037339; IR Enhancement—Procedure Change Needed 9054.01C002—RCIC
- ATI 1151603–07; Review OE 35733. Farley TDAFW Pump OE
- CPS 3309.01; High Pressure Core Spray (HPCS); Revision 17a
- CPS 3309.01E001; High Pressure Core Spray Electrical Lineup; Revision 8a
- CPS 3309.01V001; High Pressure Core Spray Valve Lineup; Revision 11b
- CPS 3309.01V002; High Pressure Core Spray Instrument Valve Lineup; Revision 9
- CPS 3506.01; Diesel Generator and Support Systems (DG); Revision 37f
- CPS 3506.01V001; Diesel Generator and Support Systems Valve Lineup; Revision 13a
- CPS 3506.01V002; Diesel Generator and Support Systems Instrument Valve Lineup; Revision 11b
- CPS 3506.01E001; Diesel Generator and Support Systems Electrical Lineup; Revision 18c
- CPS 3505.03; RAT & ERAT Static VAR Compensator (SVC); Revision 9

1R05 Fire Protection

- CPS 1893.04M633; 762 Radwaste: Ventilation Equipment Room Prefire Plan; Revision 4a
- OP–AA–201–003, Attachment 3; Fire Drill Scenario Information; Revision 16
- AR 04042526; NRC ID Incomplete Annunciator Response for FP Device 11–04
- AR 04033195; NRC Resident Insp Questions on Receipt of FP Device 11–04
- AR 04032995; FP Alarm 11–04 Recirc Pump Motors Received
- CPA 5121; Alarm Panel 5121 Annunciators; Revision 27b
- AR 04042526; NRC ID Incomplete Annunciator Response for FP Device 11–04
- OP–AA–201–003; Fire Drill Record; Revision 16
- OP–AA–201–003; Fire Drill Scenario Information; Revision 16
- CPS 1893.04M633; 762’ Radwaste Building Ventilation Equipment Room; Revision 4a
- AR 04042843; FM ID Delta’s During Fire Drill U2017–15
- IP–M–0177–AB; NSED/M; Revision 9
- CPS 1893.04; Fire Area A–2 (Fire Zones A–2a through A–2o); Revision 18a
- IP–0177, Fireload Calculation; Revision 11
- BTP APCSB 9.5–1, Appendix A; Plants Under Construction and Operating Plants; Revision 19
- CPS 1893.04M101; 707 to 712 Auxiliary: LPCS Pump Room Prefire Plan; Revision 5
- CPS 1893.04M512; 737 Diesel Generator: Div 2 Diesel Generator & Day Tank Room Prefire Plan; Revision 7a
- CPS 1893.04M130; 781–790 Auxiliary: Div 2 Switchgear Prefire Plan; Revision 5
- CPS 1893.04M134; 781 Auxiliary (East): Div 1 Battery Room Prefire Plan; Revision 5
- CPS 1893.04M003; Clinton Power Station Pre-fire Plan Legend; Revision 1
- AR 04011665; EOID—1FP04S Did Not Actuate Alarm
- AR 02590289; While Performing 3822.01C005 1FS–FP031 Failed to Actuate
- AR 04011664; EOID—1FP05S Did Not Actuate Alarm
- AR 04036920; 1FS–FP321 Did Not Actuate During Flow Testing
- AR 04057193; NRC ID: Discrepancy Identified in Pre-fire Plan 1893.04M130
- CC–AA–211; Fire Protection Program; Revision 8
- OP–AA–201–009; Control of Transient Combustible Material; Revision 19
- Temporary Shielding Package 1999–20; June 1999
- Temporary Shielding Package 2001–105; October 2001

- RP–AA–460–003; Access to HRAs/LHRAs and Contaminated Areas in Response to a Potential or Actual Emergency; Revision 8

1R06 Flood Protection Measures

- IP–M–0471, “Clinton Power Station Post Fire Safe Shutdown Criteria” Revision 4
- AR 04051565, “Error in Calc 3C10-0485-001 Associated with Flood Level”
- 3C10–0485–001, “Internal Flooding Calculations” Revision 9
- CPS 4304.01, “Flooding” Revision 6c
- CPS 3217.02, Diesel Generator Building Floor Drain System (TF)” Revision 8a

1R07 Heat Sink Performance

- AR 02544921; Defer RHR ‘B’ Heat Exchanger Performance Test
- AR 02582904; Clinton Power Station Pre-NRC (EN) Triennial Heat Sink and GL 89–13 Program Check-In Assessment
- AR 02603699; 1SX01PA Vibration Data Point VM1 in Alert Range
- AR 02634735; PC.02 for Circulating Water PI Out of Goal for February
- AR 03970173; Evaluation of Two Additional Severity Level IV Violations in Preparation for NRC IP 92723 Inspection
- AR 03970923; Excessive Residue Noted Around Valve 1CL030A
- AR 04016859; GL 89–13 Check-In ID: ECS Used as Basis for SR not Approved
- AR 04017349; GL 89–13 Self-Assessment ID: CPS 1003.10 Enhancements
- AR 04021222; Incorrect Test Data Taken During Heater Performance Test
- AR 04035181; NRC GL 89–13/UHS Inspection ID: CPS 1003.10 Typo in Appdx. D
- AR 04035331; (SEC–ID) NRC Inspector Identified Two Spent Training Rounds
- AR 04035421; NRC GL 89–13/UHS Inspection ID: No Step Off Pad Outside RHR
- AR 04036143; NRC GL 89–13/UHS Inspect ID: CPS 8130.01, App. A Tube Plug
- AR 04036207; NRC GL 89–13 UHS—Tech Spec 3.7.1 Inventory Question
- AR 04036551; NRC GL 89–13/UHS Inspection—RHR B HX Test Frequency
- AR 04036572; NRC GL 89–13/UHS Inspection: Questions about the Documentation
- AR 04038546; NRC UHS Inspection: Question on USAR Section 9.2.5
- AR 04043562; NRC GL89–13/Assessment of EPU on UHS
- AR 04041000; NRC GL 89–13/UHS Inspection—Minor USAR Discrepancies
- EPU–T0608; Extended Power Uprate Ultimate Heat Sink; Revision 0–A
- MAD 85–364; UHS Minimum Cooling Capacity; Revision 0–A
- MAD 86–0337; UHS Sensitivity Analysis; Revision 01
- MAD 75–403; Ultimate Heat Sink; Revision 0–B
- VX–47; Switchgear Room Div. 1 & 2 Condenser Tube Plugging; Revision 0
- CPS 1003.10; Clinton Power Station Program for NRC Generic Letter 89–13 (Service Water Problems Affecting Safety-Related Equipment); Revision 7
- CPS 2602.01; Heat Exchanger Performance of Shutdown Service Water Coolers Covered By NRC Generic Letter 89–13; Revision 17
- CPS 2700.20; RHR A(B) Heat Exchanger, 1E12B001A(B) Thermal Performance Test Covered by NRC Generic Letter 89–13; Revision 6
- CPS 3211.01; Shutdown Service Water; Revision 32
- CPS 3312.03; RHR—Shutdown Cooling (SDC) & Fuel Pool Cooling and Assist (FPC & A); Revision 11a
- CPS 3412.01; Essential Switchgear Heat Removal (VX); Revision 15e
- CPS 4303.02; Abnormal Lake Level; Revision 12d
- CPS 4306.01P002; Extended Loss of AC Power/Loss of Ultimate Heat Sink; Revision 1

- CPS 5050.01; Alarm Panel 5050 Annunciators—Row 1; Revision 31b
- CPS 8130.01; Heat Exchanger Maintenance/Repairs; Revision 5
- CPS 9801–01; UHS Monitoring Report Review; Revision 24
- EN–CL–402–0005; Extreme Heat Implementation Plan; Revision 9
- ER–AA–340–1001; GL 89–13 Program Implementation Instructional Guide; Revision 10
- ER–AA–340–1002; Service Water Heat Exchanger Inspection Guide; Revision 6
- ER–CL–450–2002; Ultimate Heat Sink Monitoring; Revision 1
- EC 400284; RHR B(Div. II) HX Test Dated 3/16/12 Evaluation; Revision 0
- EC 402993; Defer RHR B Heat Exchanger Performance Test; Revision 0
- EC 404025; Div. 3 SX Pump Shaft Sleeve and Suction Bell Design Change; Revision 0
- EC 405368; Credit RHR B Heat Exchanger Test with 2016 RHR B Heat Exchanger Inspection; Revision 0
- WO 01244266; Test HX Performance 1E12B001B
- WO 01397301; Perform Div. II SX System Testing IAW 2700.13
- WO 01609464; Perform Div. I SX System Testing IAW 2700.12
- WO 01675226; 1E12B001B HX Inspection (GL 89–13 Program) Credited by 15839
- WO 01701930; 1VX06CA Inspect/Clean Condenser, Clean Tubes
- WO 01765922; Divers Inspect/Clean Screenhouse Structure/GL 89–13 Program
- WO 01872469; Inspect/Clean Condenser, Hydrolance Tubes
- WO 01900631; (CPS) 9801.01 UHS Monitoring Report Review
- Clinton Power Station Full Raw Water Report; June 2017
- GE–NE–0000–0010–6516–01; RHR Heat Exchanger Tube Plugging; Revision 1
- K2801–0125; Heat Exchanger Data Sheet
- UTC 2601685; Certificate of Calibration—Ultrasonic Flowmeter; January 24, 2008
- UTC 2896008; Certificate of Calibration—Ultrasonic Flowmeter; January 25, 2012
- U–601532; Response to NRC Bulletin No. 88–04; January 5, 1990

<u>1R11 Licensed Operator Requalification Program</u>

- 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 3001.01, “Preparation for Startup and Approach to Critical,” Revision 28a
- CPS 3001.01C001, “Preparation for Startup Checklist,” Revision 18e
- CPS 3001.01C002, “Mode 2 Checklist,” Revision 17b
- CPS 3002.01, “Heat up and Pressurization,” Revision 32b
- CPS 3002.01C001, “Heat up and Pressurization Checklist,” Revision 10
- CPS 3002.01C002, “Mode 1 Checklist,” Revision 12
- CPS 3002.01F001, “Heat up and Pressurization Flowchart,” Revision 0
- CPS 3004.01, “Turbine Startup and Generator Synchronization,” Revision 33e
- CPS 3005.01, “Unit Power Changes,” Revision 43b
- AR 04050485, “Level Transient on MSDT 1A During Downpower”
- AR 04050498, “Received MCR Annunciator 5006–3G RC&IS Inop”
- AR 04050500, “Unexpected Alarm 5130–4E SJAE Final Stage Process Flow High”
- AR 04050565, “All MCR PPC Screens Locked Up”

1R12 Maintenance Effectiveness

- AR 02716350; Unusual Sound While Starting 1SX01PC
- AR 02618004; Inconsistencies Between Passport CA and EACE Action
- AR 04029986; Division 3 SX Unavailability Exceeds Criteria for June
- AR 02620350; Unnecessary Div. 3 SX Pump Run
- AR 02621409; Div. 3 SX Projected to Exceed Unavailability Criteria
- AR 02626144; EOID: 1SX01PC Oil Level Found 1/16” Below “Oil Level” Line
- AR 02640346; 1SX013F HBC No Grease Inspection Ports
- MA–AA–723–301; Periodic Inspection of Limitorque Model SMB/SB/SBD–000 Through 5 Motor Operated Valves; Revision 13
- AR 02640908; 1SX024C Stem Disc Separation
- AR 02679645; Elevated Connection Temperature on Thermal Overload 1AP30E
- AR 02682433; 1SX01PC (SX ‘C’) Unusual Sound Upon Start
- AR 02717581; 1SX013D HBC No Grease Inspection Ports
- AR 02717895; WK1638 LL For SX SOW
- AR 03946890; EC 367399 Rev. 1 Created Torque Requirement for 1SX019A—CCP
- CPS 8452.04; Hydramotor Actuator Maintenance; Revision 15
- AR 03982090; 1SX017B HBC Does Not Have Inspection Ports
- AR 03989180; Cannabilize Unit 2 Safety-Related Limitorque Operators
- AR 04022176; Div. 3 SX Pump Tripped During Start Up of 9069.01
- AR 04022481; Second Unsuccessful Run of 1SX01PC
- AR 04026746; Follow-up to IR 4025941: Storage Rack in SX Pipe Tunnel
- AR 04029986; Division 3 SX Unavailability Exceeds Criteria for June
- AR 04034246; 1SX01PA: Haze in Div. 1 SX Pump Room After Pump Run
- AR 04034973; Oil Leak from 1SX01PA
- AR 04041378; IR Needed to Record SX A1 Deter. Needing >30 Days to Finish
- AR 04043724; EOID: Multiple Through Wall Leaks on Weld for 1SX019B
- AR 04051284; Condition of the Div. 3 SX Stuffing Box
- EC 404025; Div. 3 SX Pump Shaft Sleeve and Suction Bell Design Change; Revision 0
- ECR 429909; Need Engineering Support for Replacement Motor Design
- ECR 429867; Evaluate the Installation of the New 1SX01PC Motor with New Breaker Settings per EC 618700 Prior to Pump Installation and Instrument per SX Monitoring Plan for Performing Uncoupled Run
- CPS Evaluation No.: 89256
- NSED–S–MS–08.00; Lubrication Level of Rotating Equipment; Revision 32
- 10 CFR 50.65 (a)(3) Periodic Assessment of Maintenance Rule Program
- AR 04042740; Periodic MRule (a)(3) Assessment Report
- AR 04042804; Auto Reactor Scram Plant Level Event Needs MRUL Sys/Function

1R13 Maintenance Risk Assessments and Emergent Work Control

- AD–AA–3000, “Nuclear Risk Management Process” Revision 1
- 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
- WC–AA–104, “Integrated Risk Management” Revision 23

- WC–AA–101, “On-Line Work Control Process” Revision 26
- WC–AA–101–1006, “On-Line Risk Management and Assessment” Revision 2

1R15 Operability Evaluations

- AR 04031197; NRC Question on 1RIXCM060 TS LCO 3.3.3.1 During Mode Changes
- AR 04014871; Non-Safety Parts Installed on Actuators
- AR 04034246; 1SX01PA: Haze in Div. 1 SX Pump Room After Pump Run
- OP–AA–108–115; Operability Evaluation; Revision 19
- Drawing E02–1HP99; Schematic Drawing High Pressure Core Spray Sys (HP) Div. 3 Diesel Gen. CT/PT Cubicle (1E22–S001C); Revision K
- ER–AA–200, “Preventive Maintenance Program,” Revision 3
- ER–AA–310, “Implementation of the Maintenance Rule,” Revision 10
- CPS 1305.01, “Primary Containment Leakage Rate Testing Program,” Revision 12
- Maintenance Rule System Basis Document, “Containment Isolation/Integrity and Reactor Coolant Pressure Boundary Monitoring”
- Maintenance Rule System Basis Document, “Reactor Vessel Injection/Containment Flooding and Emergency Reactor Venting Sources”
- Maintenance Rule System Basis Document, “Main Steam”
- NUMARC 93–01, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants”
- ER–AA–200–1001, “Equipment Classification,” Revision 3
- AR 04009845, “PSU: C1R17 MSIV LLRT TS 3.6.1.3 Limit Exceeded”
- AR 04059351, “C1R17 MSIV Failure”
- AR 0736646–02, “C1R11 LRT Failures for Maintenance Rule System 97 Need Evaluation”
- EACE 0818718, “EACE to Investigate Cause for MSIV Seat Cracking”
- AR 04028568; MCR Received Alarm 5062–6C
- AR 04014871; Non-Safety Related Parts Installed on Actuators
- AR 04015121; Replacement Limit Switch Model # Doesn’t Match
- AR 04013133; 1C11F180/181—Filter Regulator Parts Not Like For Like
- AR 04013220; 1C11F180 Bench Set Low, Friction High
- AR 04013420; 1C11F010 As Left Packing Friction Low
- AR 04014231; 1C11F010 Could Not Be Opened With Handwheel
- AR 04014235; PSU: 1C11F180 Always Indicates Open
- AR 04014716; PSU: SDV Drain Vlvs (F011 & F181) Failing 9012.01 Stroke Surv
- AR 04015737; Retest—SDV Valve Stroked Slow For F180 & F181
- AR 04015151; PSU: 9031.15 Reactor Mode Switch Functional Test
- AR 04015715; 1C11F010 Parts and Repair
- Evaluation No.: M–94–0300–01; Parts Classification Manual
- ER–CL–330; CPS Snubber Program; Revision 1
- ER–CL–330; CPS Snubber Program; Revision 2
- EC 395766; Evaluation of Piping Subsystem 1RH17 for Failed Snubber 1RH17002S; Revision 0
- EC 402079; C1R15 Piping Evaluation of Failed Snubbers per ORM 3.4.1; Revision 0
- EC 619714; C1R17 Snubber (1RT01003S) Failure Evaluation for Snubber Inspection Program; Revision 0
- EC 619866; C1R17 Snubber (1RT01003S) Failure Evaluation; Revision 0
- AR 01574369; Snubber Failure—1RH17002S
- AR 02496866; C1R15 Mechanical Snubber Failure—1RT06014S PSU
- AR 04011410; PSU—C1R17 Mechanical Snubber 1RT01003S Failed Function Test
- AR 04015342; NRC Identified Two Issues Regarding Snubber Testing

- AR 04041302; NRC Request for Info for Snubbers 1R06014S & 1RH17002S
- AR 04009845; C1R17 MSIV LLRT Tech Spec 3.6.1.3 Limit Exceeded

1R18 Plant Modifications

- EC 404025; Div. 3 SX Pump Shaft Sleeve and Suction Bell Design Change; Revision 0
- AR 04024207; NRC Document Request: Documentation for New Div. III SX Motor
- EC 618700; NSED—Design: Review Electrical Calcs, Breaker Settings, and Affected Documents for 1SX01PC Replacement Motor (Critical Spare) in Support of PE 89256 (Item Equivalency Evaluation); Revision 0

1R19 Post Maintenance Testing

- CPS 9080.01; Diesel Generator 1A Operability—Manual and Quick Start Operability; Revision 55f
- CPS 9080.01D001; Diesel Generator 1A Operability—Manual and Quick Start Data Sheet; Revision 45
- CPS 3506.01C005; Diesel Generator Start Log; Revision 1b
- CPS 3506.01D001; Diesel Generator 1A Operating Logs; Revision 5a
- AR 04026322; WW 1726 Missed Impacts
- AR 04015715; 1C11F010 Parts and Repair
- WO 01837075; Replace K19 Relay, With New Pre-Tested Relay
- WO 01822658; Check and Burnish as Required Relay
- WO 01759065; 8410.04 Molded Case Circuit Breaker/Bucket
- WO 01820907; Clean and Inspect Div. I Generator and Exciter
- WO 04662236; 9080.01A22 OP DG 1A Oper—Monthly Test
- WO 01820907; Clean and Inspect Div. I Generator and Exciter
- CPS 3506.01C005; Diesel Generator Start Log; Revision 1b
- CPS 3506.01D001; Diesel Generator 1A Operating Logs; Revision 5a
- CPS 9080.01D001; Diesel Generator 1A Operability—Manual and Quick Start Data Sheet; Revision 45
- WC–AA–111; Initial Sheet; Revision 5
- WO 1829802; Excessive Drain Down Time for Div. 2 SX
- CPS 8451.04; Limitorque Operator Removal/Installation; Revision 16
- CPS 9381.01C002; MOV Thermal Overload Bypass Post Maintenance Verification Checklist; Revision 29
- CPS 9861.09D002; Leakage Test on Valve 1SX014B; Revision 2
- CPS 9843.02D001; Generic Class 1, 2 and 3 Operational Pressure Test Data Sheet; Revision 44
- CPS 9069.01; Shutdown Service Water Operability Test; Revision 49b
- AR 04012610; 1SX014B Failed Leak Test, 150GPM
- WO 04638343; 1SX011A Breaker needs repaired
- WO 01829803; 1SX011A Did not stroke shut with test prep switch in “test”
- AR 04010439; Unexpected open signal on 1SX011A
- CPS 8440.01D001, Insulation Testing Data Sheet, Revision 11
- CPS 8451.04; Limitorque Operator Removal/Installation; Revision 16
- CPS 8410.03D001; ITE Gould Motor Overload Relay Data Sheet; Revision 10
- CPS 8410.04D001; Molded Case Circuit Breaker/Bucket Component Functional Testing and Maintenance Data Sheet; Revision 27
- CPS 9381.01C002; MOV Thermal Overload Bypass Post Maintenance Verification Checklist; Revision 29

- CPS 9861.09D003; Leak Rate Testing for SX Valve 1SX011A; Revision 2b
- AR 04009649, 1DC01E Battery Division 1 Test Data Unsat
- CPS 9382.12; Division 1 125VDC Battery Service Test; Revision 31
- CPS 9382.12; Division 1 125VDC Battery Service Test; Revision 32
- AR 04011221; Potentially Degraded Battery Posts on 1DC01E Battery
- AR 04022351; 125VDC Battery-System No. 1 High Electrolyte Level
- CPS 9382.02; 125VDC Battery ICV and Battery Charger Checks; Revision 36
- CPS 9382.02D001; Battery Control Log; Revision 32
- CPS 9382.12; Division 1 125VDC Battery Service Test; Revision 32
- CPS 9382.12D001; Division 1 125VDC Battery Service Test Data Sheet; Revision 26a
- CPS 9382.01; 125VDC Battery Pilot Cell Check; Revision 3

<u>1R22 Surveillance Testing</u>

- CPS 9080.03; Diesel Generator 1C Operability—Manual and Quick Start Operability; Revision 35
- CPS 9080.02; Diesel Generator 1B Operability—Manual and Quick Start Operability; Revision 53a
- CPS 3506.01C002; Diesel Generator 1B Pre-Start Checklist; Revision 12a
- CPS 9080.02D001; Diesel Generator 1B Operability—Manual and Quick Start Data Sheet; Revision 43
- CPS 9069.01; Shutdown Service Water Operability Test; Revision 49c
- WO 04655224; 9080.02B22 OP DG 1B Oper—Monthly Test
- WO 04669650; OP 9080.03 DG 1C Oper—Monthly Test
- WO 04626032; 9069.01B20 OP SX Pump Oper Test (SX Pump B)
- AR 04025941; Resident Inspector Asked Questions on Scaffolding
- EC 620260; Storage of Scaffold Material in the Screenhouse (Elevation 657’-6”); Revision 0

<u>2RS1 Radiological Hazard Assessment and Exposure Controls</u>

- RP–AA–210–1001; Attachment 13 Multiple Dosimetry EDE Evaluation Sheet; Job Description: 752’ and 777’ C1R17 Drywell Bioshield ISI (RP coverage, Shielding, and ISI Work); Revision 10
- RP–AA–203–1001; Attachment 1: “Sample” Personnel Exposure Investigation; Revision 9; 05/17/2017
- RP–AA–460; Controls for High and Locked High Radiations Areas; Revision 29
- RP–AA–460; Attachment 5: High Radiation Area (HRA) and Locked High Radiation Area (LHRA) Briefing (CM-2); HRA/LHRA to be Entered: 725’ Bioshield N; 05/17/2017
- RP–AA–460; Attachment 5: High Radiation Area (HRA) and Locked High Radiation Area (LHRA) Briefing (CM-2); HRA/LHRA to be Entered: 752’ Bioshield N; 05/17/2017 1515
- RP–AA–460; Attachment 5: High Radiation Area (HRA) and Locked High Radiation Area (LHRA) Briefing (CM-2); HRA/LHRA to be Entered: 777’ Bioshield N; 05/17/2017 1518
- RP–AA–460–002; Additional High Radiation Exposure Control; Revision 3
- RP–AA–401–1002; Radiological Risk Management; Revision 10
- RP–AA–403; Administration of the Radiation Work Permit Program; Revision 9
- RP–AA–400–1009; Remote Monitoring System; Revision 2
- CL–1–17–00518; Radiation Work Permit; C1R17 DW Bioshield ISI Activities; Task 1-3; Revision 4
- PI–AA–125–1003; Corrective Action Program Evaluation Report Template; Workers Enter Incorrect Elevation of the Drywell Bioshield; 06/22/2017

- RP–1316–04; CPS Radiological Survey Sheet; Drywell - 777’ EL.; 777’ EL. Bioshield; 05/12/2017
- RP–1316–04; CPS Radiological Survey Sheet; Drywell - 777’ EL.; 777’ EL. Bioshield; 05/12/2017 0043
- RP–1316–04; CPS Radiological Survey Sheet; Drywell - 777’ EL.; 777’ EL. Bioshield; 05/12/2017 1230
- RP–1316–04; CPS Radiological Survey Sheet; Drywell - 777’ EL.; 777’ EL. Bioshield; 05/15/2017
- RP–1316–04; CPS Radiological Survey Sheet; Drywell - 777’ EL.; 777’ EL. Bioshield; 05/15/2017 2100
- RP–1316–04; CPS Radiological Survey Sheet; Drywell - 777’ EL.; 777’ EL. Bioshield; 05/17/2017 0345
- RP–1316–04; CPS Radiological Survey Sheet; Drywell - 752’ EL.; 752’ EL. Bioshield; 05/12/2017
- RP–1316–04; CPS Radiological Survey Sheet; Drywell - 752’ EL.; 752’ EL. Bioshield; 05/12/2017 0130
- RP–1316–04; CPS Radiological Survey Sheet; Drywell - 752’ EL.; 752’ EL. Bioshield; 05/12/2017 1440
- RP–1316–04; CPS Radiological Survey Sheet; Drywell - 752’ EL.; 752’ EL. Bioshield; 05/13/2017 0830
- AR 04012075; Incorrect Travel Path Discussed At Briefing; 05/17/2017

2RS2 Occupational ALARA Planning and Controls

- RP–AA–16; ALARA Program Description; Revision 0
- RP–AA–400; ALARA Program; Revision 14
- RP–AA–400–1001; Establishing Collective Radiation Exposure Estimates and Goals; Revision 4
- RP–AA–400–1004; Emergent Dose Control and Authorization; Revision9
- RP–AA–400–1006; Outage Exposure Estimating and Tracking; Revision 7
- RP–AA–400–1007; Elevated Dose Rate Response Training; Revision 2
- RP–AA–4002; Radiation Protection Refuel Outage Readiness; Revision 11
- RP–AA–4003; Guidelines for Daily Radiation Protection Outage Report; Revision 8
- RP–AA–401; Operational ALARA Planning and Controls; Revision 22
- RP–AA–401; Attachment 1: ALARA Pre-Planning Tool; Task Description: C1R17 RFF Rx Disassembly Reassembly (Cavity Work); RWP: CL–1–17–00901; Revision 22
- RP–AA–401; Attachment 2: Combined ALARA Plan/Micro-ALARA Plan RFF Rx Disassembly Reassembly; 05/06/2017
- RP–AA–401; Attachment 6: ALARA Work-In-Progress Review: C1R17 RX Disassembly; 05/16/2017
- RP–AA–401; Attachment 1: ALARA Pre-Planning Tool; Task Description: C1R17 RFF Activities (No Cavity); RWP: CL–1–17–00902; Revision 22
- RP–AA–401; Attachment 2: Combined ALARA Plan/Micro-ALARA Plan RFF Activities (No Cavity); 05/04/2017
- RP–AA–401; Attachment 6: ALARA Work-In-Progress Review: RFF Activities (No Cavity); 05/05/2017
- RP–AA–401; Attachment 1: ALARA Pre-Planning Tool; Task Description: C1R17 Outage Drywell Scaffold (17–506); Revision 22
- RP–AA–401; Attachment 2: Combined ALARA Plan/Micro-ALARA Plan C1R17 Outage Drywell Scaffold (17–506); 05/04/2017

- RP–AA–401; Attachment 6: ALARA Work-In-Progress Review: C1R17 Outage Drywell Scaffold (17–506); 05/05/2017
- Clinton Power Station C1R17 Post Outage Report; C1R17/ALARA Refuel Outage Report
- AR 02664478; RP ID: Refuel Team Accrues Unapproved Dose; 05/03/2016
- AR 02668841; Reactor Services May On-Line Dose Estimate Insufficient; 05/12/2016
- AR 04009849; C1R17 Not Practicing ALARA per USAR; 05/12/2017
- AR 04013083; RP ID: ED Dose Alarm; 05/20/2017
- AR 04013629; RP ID: ED Dose Alarm; 05/22/2017
- AR 04015609; RP ID: ED Dose Alarm; 05/27/2017
- AR 04020055; RP ID: Emerging Trend in Electronic Dosimeter Dose Alarms; 06/06/2017

2RS6 Radioactive Gaseous and Liquid Effluent Treatment

- CY–170–200; SGTS Effluent – Noble Gas and Tritium; Revision 23
- CY–170–201; SGTS Stack Effluent – Iodine and Particulates; Revision 28
- CY–CL–6901–01; Off-Gas Post-Treatment Contingency Sampling and Analysis; Revision 0
- CY–CL–6949–01; Off-Gas Pre-Treatment Sampling and Analysis; Revision 5
- CY–CL–6954–01; HVAC Stack Effluent Noble Gas and Tritium; Revision 3
- CY–CL–6954–02; HVAC Stack Effluent – Iodine and Particulates; Revision 3
- CPS 2104.01; HEPA Filter Testing; Revision 12b
- CPS 9911.03; 1RIX – PR035/41 Filter Change-Out; Revision 4d
- CPS 9911.51; Liquid Radioactive Surveillance – Monthly; Revision 28d
- CPS 9911.59; Gaseous Radioactive Effluent Surveillance – Monthly; Revision 29c
- CPS 9911.60; Gaseous Radioactive Effluent Surveillance; Revision 30d
- CPS 9947.01; Gaseous Effluent Monthly Composite Analysis; Revision 32a
- Work Order 01871695 01; 9437.40A20 CC HVAC System Exhaust PRM CC (PR001); 04/19/2017
- Work Order 01783646 01; 9437.41C20 CC SGTS Exhaust PRM CC (PR003); 02/15/2016
- Work Order 01867540 10; Perform CPS 2104.01, HEPA Filter Test; 03/30/2017
- Work Order 04674305 01; 9911.59R20; REC Gaseous Rad Effluent Surv – Monthly; 09/07/2017
- WC–AA–11–F–01; 9911.60 Rec Gaseous Rad Effluent Surv – Trigger; 09/11/2017
- WC–AA–11–F–01; 9911.60 Rec Gaseous Rad Effluent Surv – Trigger; 08/17/2017
- Offsite Dose Calculation Manual (ODCM); Revision 25
- Land Use Census; 2016
- Clinton Power Station Radioactive Effluent Release Report; 01/01/2016–12/31/2016
- Clinton Power Station Radioactive Effluent Release Report; 01/01/2015–12/31/2015
- AR 04016990; 0RIX–PR001 Channel 3 Failed Source Check; 05/31/2017
- AR 03992449; 0RIX–PR001 Channel 3 Failed Source Check; 03/31/2017
- AR 02657934; 0RIX–PR002 Channel 3 Iodine Channel Spiked; 04/19/2016
- AR 03968864; 0RIX–PR003 Channel 4, Iodine Failed Source Check; 01/31/2017
- AR 04037876; ERRATA Data Submittal to Clinton 2017 AREOR; 08/01/2017

4OA1 Performance Indicator Verification

- MSPI Derivation Report; MSPI Residual Heat Removal System
- MSPI Derivation Report; MSPI Cooling Water System
- PI Summary; Occupational Exposure Control Effectiveness; Second Quarter 2016 through First Quarter 2017
- PI Summary; RETS/ODCM Radiological Effluent Data; First Quarter 2016 through Second Quarter 2017

- PI Summary; Reactor Coolant System (RCS) Specific Activity Data; First Quarter 2016 through Second Quarter 2017

4OA2 Problem Identification and Resolution

- AR 04040004; 0.11 Consequential CCF Clock Reset in Variance for July
- AR 04016024; Drywell Closeout Issues Identified by NRC
- AR 04029024; EOID: RR B Motor Lower Oil Level Trending Down 1LR–RR001
- AR 04013626; Corrective Action for IR 3949655 Not Implemented As Intended
- WO 1859216; Swap Breaker at 1RR01ED/1B33C001A
- CPS 8410.21C001; Westinghouse DHP Circuit Breaker Checklist; Revision 10

4OA3 Event Followup

- AR 04007861; Entered 4008.01 and 4002.01 Due to FCV Runback
- AR 04012150; Runback 4.0 Critique
- AR 04009845, “PSU: C1R17 MSIV LLRT TS 3.6.1.3 Limit Exceeded”
- AR 04059351, “C1R17 MSIV Failure”
- OP–AA–101–113–1004; CPS Human Performance Issue Verbal Report Format; Revision 37
- CPS 3103.01; Feedwater (FW); Revision 31e
- PI–AA–120; Issue Identification and Screening Process; Revision 7
- PI–AA–125–1003; Corrective Action Program Procedure; Revision 4
- CPS 3005.01; Unit Power Changes; Revision 43b
- NF–161.03, “Checklist for Review of Calculation/Analysis” Revision 0
- PI–AA–12–1006, “Investigation Techniques Manual” Revision 3
- PI–AA–125, “Corrective Action Program” Revision 6
- ACE 2742442, “CDBI: Inappropriate Calculation Method for CR Habitability”
- C–015, “Calculation of Alternative Source Term (AST) Onsite and Offsite X/Q Values”
- OpEval 2742442–02, Operability of the Main Control Room Ventilation System
- AR 02519380, “Trip of the ERAT SVC 0AP103E and 0AP104E”
- AR 02649780, “RAT 4538 Failure: Numerous Unexpected Alarms 5010–1A, 1C, 8A”
- LER 2016–005–00, “Insulator Failure on the Reserve Auxiliary Transformer Results in a Loss of Secondary Containment Vacuum”
- LER 2016–004–00, “Trip of Emergency Reserve Auxiliary Transformer Static VAR Compensator Causes Positive Secondary Containment Pressure Following Lightning Strike on 138 kV Offsite Source”
- PI–AA–125–1004, “Effectiveness Review Manual” Revision 2
- CAPE 3978324, “Lost 138 kV Feed Causing ERAT Transient”
- CPS–17–0065, “EC 405760 Prevent VF Tripping”
- AR 04047373, “NRC Senior Resident had a Question on Secondary Containment and MDE Shift”
- AR 04017613, “Welding Cables Through Both RT Pump Room ‘B’ Doors”
- CAPE 4017613, “Secondary Containment ITS Violation”
- CC–AA–201, “Plant Barrier Control Program,” Revision 11

LIST OF ACRONYMS USED

ALARA	As-Low-As-Reasonably-Achievable
AR	Action Request
CAP	Corrective Action Program
CFR	*Code of Federal Regulations*
CPS	Clinton Power Station
DW	Drywell
EC	Engineering Change
EDG	Emergency Diesel Generator
EGM	Enforcement Guidance Memorandum
ERAT	Emergency Reserve Auxiliary Transformer
FCV	Flow Control Valves
FW	Feedwater
HX	Heat Exchanger
IMC	Inspection Manual Chapter
IP	Inspection Procedure
kV	Kilo Volt
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LHRA	Locked High Radiation Area
LLRT	Local Leak Rate Testing
LPCS	Low Pressure Core Spray
MSIV	Main Steam Isolation Valve
MSPI	Mitigating System Performance Index
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual
OPDRV	Operations with the Potential to Drain the Reactor Vessel
PBI	Plant Barrier Impairment
PI	Performance Indicator
RAT	Reserve Auxiliary Transformer
RCIC	Reactor Core Isolation Cooling
RG	Regulatory Guide
RHR	Residual Heat Removal
RPT	Radiation Protection Technician
RPV	Reactor Pressure Vessel
RR	Reactor Recirculation
RWCU	Reactor Water Cleanup
RWP	Radiation Work Permit
SFP	Single Failure Proof
TDRFP	Turbine Driven Feedwater Pump
TS	Technical Specification
USAR	Updated Safety Analysis Report
UHS	Ultimate Heat Sink
VC	Control Room Ventilation
VF	Fuel Building Ventilation
WO	Work Order