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

May 8, 2017

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 05000461/2017001

Dear Mr. Hanson:

On March 31, 2017, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Clinton Power Station. On April 20, 2017, the NRC inspectors discussed the results of this inspection with Mr. B. Kapellas and other members of your staff. The results of this inspection are documented in the enclosed report.

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

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

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

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

Sincerely,

/RA/

Karla Stoedter, Chief
Branch 1
Division of Reactor Projects

Docket No. 50–461
License No. NPF–62

Enclosure:
Inspection Report 05000461/2017001

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Letter to Bryan C. Hanson from Karla Stoedter dated May 8, 2017

SUBJECT: CLINTON POWER STATION—NRC INTEGRATED INSPECTION REPORT 05000461/2017001

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50–461
License No:	NPF–62
Report No:	05000461/2017001
Licensee:	Exelon Generation Company, LLC
Facility:	Clinton Power Station
Location:	Clinton, IL
Dates:	January 1 through March 31, 2017
Inspectors:	W. Schaup, Senior Resident Inspector E. Sanchez Santiago, Resident Inspector G. Edwards, Health Physicist J. Wojewoda, Reactor Inspector S. Mischke, Resident Inspector Illinois Emergency Management Agency
Approved by:	K. Stoedter, Chief Branch 1 Division of Reactor Projects

Enclosure

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SUMMARY

Inspection Report 05000461/2017001, Clinton Power Station; Operability Determinations and Functional Assessments and Follow-up of Events and Notice of Enforcement Discretion.

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Four 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 associated non-cited violation of Technical Specification 5.4.1, "Procedures," for the licensee's failure to develop and review a worker tag out in accordance with station procedure OP–AA–109–10, "Clearance and Tagging," Revision 12. Specifically, the licensee failed to identify the effect of a worker tag out on the in-service steam jet air ejector suction valve, which caused condenser vacuum to degrade resulting in the operators entering the off normal procedure for loss of condenser vacuum. The licensee entered this issue into their corrective action program as action request (AR) 03980495. As corrective actions, the operations department issued a standing order to require worker tag outs to be challenged by a second senior reactor operator.

  The performance deficiency was determined to be more than minor because it impacted the Initiating Events cornerstone attribute of configuration control and adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during power operations. Specifically, the failure to properly develop the worker tag out caused the condenser vacuum to degrade, challenging the operators to quickly diagnose the issue and take action to avoid a turbine trip. The finding was screened against the Initiating Events cornerstone and determined to be of very low safety significance because it did not cause a reactor trip or the loss of mitigation equipment relied upon to transition the plant from the onset of a trip to a stable shutdown condition. The inspectors determined that this finding affected the cross-cutting area of human performance in the aspect of avoid complacency, where individuals recognize and plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes. Individuals implement appropriate error reductions tools. Specifically, the operations department failed to implement appropriate error reduction tools such as questioning attitude and thorough work product reviews to ensure the worker tag out considered all potential effects to other plant equipment. [H.12] (Section 4OA3.1)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance and an associated non-cited violation of Title 10 of the *Code of Federal Regulations* (CFR) Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the failure to implement the plant barrier control program for an impacted flood barrier. Specifically, the plant barrier impairment (PBI) permit, PBI–2017–02–003, for work on watertight door 1SD1–24, failed to identify the door as a flood barrier and that appropriate compensatory measures for 1SD1–24 being open for an extended period were identified or implemented in accordance with station procedure CC–AA–201, “Plant Barrier Control Program,” Revision 11. The licensee entered this issue into their corrective action program as AR 03980495. The corrective actions in response to this violation were to identify appropriate compensatory measures for impairment of 1SD1–24 and incorporate them into the PBI log.

  The performance deficiency was determined to be more than minor because it impacted the Mitigating Systems cornerstone attribute of protection against external events and adversely affected the cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. With the flood barrier nonfunctional and without compensatory actions in place the residual heat removal (RHR) ‘B’ and RHR ‘C’ pumps were inoperable. The finding was screened against the Mitigating Systems cornerstone and the inspectors determined that the finding involved the loss or degradation of equipment or function specifically designated to mitigate a seismic, flooding or severe weather initiating event. The inspectors determined that the loss of this equipment or function by itself during the external initiating event would degrade one or more trains of a system that supports a risk significant system or function and would require a detailed risk evaluation. The senior reactor analyst (SRA) performed the detailed risk evaluation and concluded the finding was of very low safety significance. The inspectors determined that 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. Proposed actions are determined to be safe in order to proceed, rather than unsafe in order to stop. Specifically, during preparation of the PBI permit, the station PBI log was reviewed and actions for previous work associated with the watertight door were deemed acceptable even though the work on the door in those instances was different than the work being performed this time.
  [H.14] (Section 1R15.1)

- Green. The inspectors identified a finding of very low safety significance and an associated non-cited violation of 10 CFR 50.36(c)(2)(i), “Limiting conditions for operation,” for failing to meet/follow the required actions for limiting condition for operation 3.9.9 and 3.4.10. Specifically, the operators failed to verify a credited alternate decay heat removal method that would satisfy the required action for the limiting condition for operation. The licensee entered this issue into their corrective action program as AR 03987440. The corrective actions in response to this violation were to identify appropriate alternate methods of decay heat removal and incorporate them into the shutdown safety management program utilized during plant outages.

  The performance deficiency was determined to be more than minor because it impacted the Mitigating Systems cornerstone attribute of equipment performance and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, with the operators failing to identify a credited alternate method of decay heat removal and taking credit for the inoperable but in service RHR shutdown cooling train, the actual available methods that could have been credited were not verified to ensure their availability to provide the required function. The finding was screened against the Mitigating Systems Screening questions and determined to be of very low safety significance because the answer to all of the applicable screening questions was "No." The inspectors determined that 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. Proposed actions are determined to be safe in order to proceed, rather than unsafe in order to stop. Specifically, the senior reactor operators at the station had historically credited inoperable RHR shutdown cooling subsystems as their own alternate decay heat remove method because they believed it was allowable without determining that it was safe in order to proceed. [H.14] (Section 1R15.2)

- Green. The inspectors documented a self-revealed finding of very low safety significance and associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," for the licensee's failure to perform maintenance on a safety related breaker in accordance with station procedure Clinton Power Station (CPS) 8410.12C001, "Westinghouse DHP Circuit Breaker Checklist," Revision 7. Specifically, the licensee failed to ensure the remaining travel on the latch check switch for the RHR 'C' pump breaker was within the acceptable range resulting in the RHR 'C' pump failing to start. The licensee entered this issue into their corrective action program as AR 03949655. The corrective actions taken by the licensee included providing coaching to the involved individuals as well as changing the procedure to include a block to record the latch check switch over travel.

  The performance deficiency was determined to be more than minor because it impacted the Mitigating Systems cornerstone attribute of equipment performance and adversely affected the cornerstone objective of ensuring the availability, capability, and reliability of equipment that responds to initiating events. Specifically, the performance deficiency adversely impacted the operability of the RHR 'C' pump. The inspectors reviewed the Mitigating Systems screening questions and determined a detailed risk evaluation was required because question A.3 was answered yes. The SRA performed the detailed risk evaluation and concluded the finding was of very low safety significance. The inspectors determined that 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, the organization failed to ensure the procedure step included a block for recording the latch check switch over travel value, which led to confusion on whether the value was required to be recorded and ultimately resulted in a failure to perform the step as written. [H.1] (Section 4OA3.2)

REPORT DETAILS

Summary of Plant Status

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

- On February 19, 2017, power was reduced to approximately 90 percent to perform a rod sequence exchange and to perform scram time testing for control rod recovery. The unit was returned to full power the same day.
- On March 10, 2017, power was reduced to approximately 78 percent to perform turbine driven reactor feed pump testing, turbine valve testing, a rod sequence exchange and channel bowing monitoring. The unit was returned to full power the same day.
- On March 19, 2017, power was reduced to approximately 86 percent to perform the final rod sequence exchange for all rods out. The unit was returned to full power the same day.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness for Impending Adverse Weather Condition—Severe Thunderstorm Watch

a. Inspection Scope

Since thunderstorms with potential tornados and high winds were forecast in the vicinity of the facility for February 28, 2017, the inspectors reviewed the licensee’s overall preparations/protection for the expected weather conditions. On February 27, 2017, the inspectors walked down the switchyard, main power transformers, reserve auxiliary transformers, emergency reserve auxiliary transformer, exterior portions of the containment building, auxiliary building and fuel building because their safety-related functions could be affected or required as a result of high winds or tornado-generated missiles or the loss of offsite power. The inspectors evaluated the licensee staff’s preparations against the site’s procedures and determined that the staff’s actions were adequate. During the inspection, the inspectors focused on plant-specific design features and the licensee’s procedures used to respond to specified adverse weather conditions. The inspectors also toured the plant grounds to look for any loose debris that could become missiles during a tornado. The inspector’s evaluated operator staffing and accessibility of controls and indications for those systems required to control the plant. Additionally, the inspectors reviewed the Updated Safety Analysis Report (USAR) and performance requirements for systems selected for inspection, and verified that operator actions were appropriate as specified by plant specific procedures. The inspectors also reviewed a sample of corrective action program (CAP) items to verify that the licensee identified adverse weather issues at an appropriate threshold and dispositioned them through the CAP in accordance with station corrective action procedures.

This inspection constituted one readiness for impending adverse weather condition 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:

- diesel fuel oil transfer pumps ‘A' and 'B' during testing of the 'C' pump;
- division 3 emergency diesel generator (EDG) after surveillance testing;
- standby gas treatment system 'A' during maintenance on standby gas treatment system 'B'; and
- low pressure core spray system during maintenance on high pressure core spray system.

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

These activities constituted four 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 March 23, 2017, the inspectors completed a complete system alignment inspection of the division 1, division 2, and division 3 EDG systems 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 work orders was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the CAP database to ensure that system equipment alignment problems were being identified and appropriately resolved.

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

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

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

- Fire Zone A–2(b), RHR [residual heat removal] ‘A’ Equipment Room—elevation 707';
- Fire Zone CB–1(i), Air Handling Equipment Area—elevation 825';
- Fire Zone CB–2, Division 2 Cable Spreading Rooms—elevation 781';
- Fire Zone D–5 and D–5(a), Division 1 Diesel Generator Room—elevation 737'; and
- Fire Zone CB–6, Main Control Room Complex—elevation 800'.

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.

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

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

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

a. Inspection Scope

On March 15, 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 February 19, 2017, the inspectors observed the control room operators perform a down power to support a rod sequence exchange and scram time response testing. This was an activity that required heightened awareness and was related to increased risk. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;

- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of procedures;
- control board manipulations; and
- oversight and direction from supervisors.

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

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

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations

a. Inspection Scope

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

- main control room recorders.

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

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

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the CAP with the appropriate significance characterization.

This inspection constituted one quarterly maintenance effectiveness sample as defined in IP 71111.12–05.

b. Findings

No findings were identified.

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

.1 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

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

- green due to planned maintenance on fuel pool cooling train ‘A’;
- yellow due to planned maintenance on standby gas treatment train ‘B’;
- yellow due to planned maintenance on RHR train ‘B’;
- yellow for adverse weather—severe thunderstorm warning;
- yellow due to planned surveillance testing of the division 3 EDG; and
- yellow due to emergent work to change out oil in the in the ‘B’ standby liquid control pump.

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) 03980495 Inadequate plant barrier impairment [PBI] 2017–02–003;

- AR 03984708 NRC Question on Low Pressure Core Spray Test Valve Lineup Station Procedure CPS [Clinton Power Station] 3313.01V002;
- AR 03987440 Inoperable RHR Loop Cannot be Credited for SDC [Shutdown Cooling] Actions;
- AR 03966642 Request RR [Reactor Recirculation] 'A' FCV [Flow Control Valve] Lockout be Evaluated for Work-Around; and
- AR 03968392 Fuse Blown During Maintenance on Standby Gas Treatment Differential Pressure Switch.

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. <u>Findings</u>

.1 <u>Plant Barrier Control Program Failed to Compensate for an Impacted Flood Barrier</u>

<u>Introduction</u>: The inspectors identified a finding of very low safety significance and an associated non-cited violation (NCV) of Title 10 of the *Code of Federal Regulations* (CFR) Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to implement the plant barrier control program for an impacted flood barrier. Specifically, the PBI permit, PBI–2017–02–003, for work on watertight door 1SD1–24, failed to identify the door as a flood barrier and that appropriate compensatory measures for 1SD1–24, being open for an extended period were not identified or implemented in accordance with station procedure CC–AA–201, "Plant Barrier Control Program," Revision 11.

<u>Description</u>: On February 9, 2017, the licensee performed work on watertight door 1SD1–24 located between the RHR 'B' and RHR 'C' pump rooms that included replacement of the door seal. The work made the door inoperable and took approximately four hours to complete. The licensee entered the TS for the RHR 'B' heat exchanger room leak detection instrumentation becoming inoperable since the volume of the room increases with the door open. The work was completed, the watertight door was restored to operable, and the TS was exited.

The following day, while reviewing the operations logs, the inspectors noted the entries pertaining to the work on the watertight door. The inspectors noted that the licensee only entered the TS for the RHR 'B' heat exchanger room leak detection instrumentation becoming inoperable. The inspectors asked the operations staff if the RHR 'B' and RHR ‘C’ pumps were operable during the time period the door seal was removed and what mitigating or compensatory actions were in place with the flood barrier removed to allow the pump to remain operable. The licensee documented the issues in the CAP as AR 03972544.

The inspectors were informed that a PBI permit had been issued in accordance with station procedure CC–AA–201, “Plant Barrier Control Program” and would provide the compensatory actions that had been put in place during the work. Section 6.1, Step 2 of CC–AA–201, “Plant Barrier Control Program,” Revision 11, stated, “Use site specific documents to determine which of the barrier types outlined in Attachment 1, Section II of the PBI permit are applicable.” Additionally, Section 6.4, Step 2, stated, “Review the PBI permit and ensure the compensatory actions and operability impact is acceptable.” The inspectors reviewed PBI–2017–02–003 and determined that no compensatory actions had been established to mitigate the removal of the seal on the watertight door. This issue was documented by the licensee in the CAP as AR 03980495.

The inspectors’ review of the station TS and plant barrier control program determined the licensee failed to declare the RHR ‘B’ and RHR ‘C’ pumps inoperable during the maintenance when the required flood barrier became inoperable and enter the applicable required actions to satisfy the limiting conditions for operations. Since the allowed outage time for the inoperable pumps was not exceeded, the inspectors determined that the violation was of minor significance.

Additionally, the licensee’s review of the issue determined that the PBI permit was inadequate. The PBI did not identify the door as a flood barrier and therefore appropriate compensatory measures for 1SD1–24 being open for an extended period were not identified or implemented as required by station procedure CC–AA–201.

The corrective actions in response to this issue were to identify appropriate compensatory measures for impairment of 1SD1–24 and incorporate them into the PBI log.

<u>Analysis</u>: The inspectors determined the licensee’s failure to implement the plant barrier control program for an impacted flood barrier was a performance deficiency. Plant barrier impairment permit, PBI–2017–02–003, for work on watertight door 1SD1–24, failed to identify the door as a flood barrier, therefore, the appropriate compensatory measures for 1SD1–24 being open for an extended period were not identified or implemented in accordance with station procedure CC–AA–201, “Plant Barrier Control Program,” Revision 11. 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 Mitigating Systems cornerstone attribute of protection against external events and adversely affected the cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. With the flood barrier nonfunctional and without compensatory actions in place, the RHR ‘B’ and RHR ‘C’ pumps would become inoperable during a flooding scenario. Using IMC 0609, Attachment 4, “Initial Characterization of Findings,” and Appendix A, “The Significance Determination Process (SDP) for Findings at Power,” issued June 19, 2012, the finding was screened against the Mitigating Systems cornerstone and the inspectors determined that the finding involved the loss or degradation of equipment or function specifically designated to

mitigate a seismic, flooding or severe weather initiating event and proceeded to Exhibit 4. The inspectors evaluated the criteria of the first question, “If the equipment or safety function is assumed to be completely failed or unavailable, are any of the following three statements true?” The inspectors determined that the loss of this equipment or function by itself during the external initiating event would degrade one or more trains of a system that supports a risk significant system or function and would require a detailed risk evaluation. The senior reactor analyst determined the finding was of very low safety significance (Green) because of the very limited time that the flood barrier was non-functional. The NRC Risk Assessment of Operational Events Handbook for External Events was reviewed for internal flood frequencies, specifically the boiling water reactor safety injection and recirculation piping failure rates. Considering the very low pipe failure rates and flood frequencies and the limited exposure time, the risk impact of the condition was judged to be much less than 1E–6/year.

The inspectors determined that 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. Proposed actions are determined to be safe in order to proceed rather than unsafe in order to stop. Specifically, during preparation of the PBI permit, the station PBI log was reviewed and actions for previous work associated with the watertight door were deemed acceptable even though the work on the door in those instances was different than the work being performed this time. [H.14]

Enforcement: Title 10 CFR 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 6.1, Step 2 of CC–AA–201, “Plant Barrier Control Program,” Revision 11, stated, “Use site specific documents to determine which of the barrier types outlined in attachment 1, Section II of the PBI permit are applicable.” Additionally, Section 6.4, Step 2, stated, “Review the PBI permit and ensure the compensatory actions and operability impact is acceptable.”

Contrary to the above, on February 6, 2017, the licensee failed to accomplish work on watertight door 1SD1–24, an activity affecting quality, in accordance with procedure CC–AA–201. Specifically, PBI permit, PBI–2017–02–003, for work on watertight door 1SD1–24, failed to identify the door as a flood barrier and ensure the compensatory measures and the operability impact for 1SD1–24, being open for an extended period were acceptable.

The corrective actions in response to this violation were to identify appropriate compensatory measures for impairment of 1SD1–24 and incorporate into the PBI log. Because this finding was of very low safety significance and was entered into the licensee’s CAP as AR 03980495, this violation is being treated as an NCV, in accordance with Section 2.3.2 of the NRC Enforcement Policy.
(NCV 5000461/2017001–01: Plant Barrier Control Program Failed to Compensate for an Impacted Flood Barrier)

.2 Failed to Verify an Appropriate Alternate Method of Decay Heat Removal

Introduction: The inspectors identified a finding of very low safety significance and an associated NCV of 10 CFR 50.36(c)(2)(i), "Limiting conditions for operation," for failing to meet/follow the required actions for limiting conditions for operation (LCO) 3.9.9, "Residual Heat Removal—Low Water Level," and 3.4.10, "Residual Heat Removal Shutdown Cooling System—Cold Shutdown." Specifically, the operators failed to verify a credited alternate decay heat removal method that would satisfy the required action for the limiting condition for operation.

Description: On May 26, 2016, the inspectors were reviewing the operation logs and noted the following log entry: "Restarted RHR–B in shutdown cooling operation. Residual heat removal 'B' remains inoperable due to division 2 switchgear room cooler being inoperable. Entered the following LCO [limiting condition for operation] action (previously at 00:30 on 5/25/2017): LCO 3.4.10, Action A.1—Verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem. This action is met by having RHR 'B' available for decay heat removal via the shutdown cooling pathway. This initial 1 hour action is met."

At the time of the inspectors' review (on the following day) the licensee had restored operability of the division 2 switchgear room cooler and the RHR 'B' shutdown cooling subsystem and were in compliance with their TS. However, the inspectors questioned the appropriateness of crediting the inoperable RHR shutdown cooling subsystem as its own alternate method of decay heat removal because it failed to meet the required actions for the LCO. This was discussed with the control room operators and operations management. The inspectors requested a review of this issue by the Office of Nuclear Reactor Regulation. The results of this review concluded that "a system which has been declared inoperable cannot be designated its own alternate system for the purposes of complying with the technical specification… Clearly, an alternate 'method' of decay heat removal was not provided for shutdown cooling since the 'method' of providing decay heat removal did not change when the inoperable system was declared to be its own alternate." This issue was documented in the licensee's CAP as AR 03987440.

The inspectors reviewed the operations logs for the entire 2016 outage and determined that the operations staff had verified the inoperable RHR–B shutdown cooling subsystem as its own alternate method of decay heat removal on numerous occasions, in Mode 4 and Mode 5, to satisfy the applicable TS. The inspectors' discussions with operations staff revealed that the senior reactor operators believed it was acceptable to have the inoperable subsystem be its own method of alternate decay heat removal since it was available and had been done this way historically.

The licensee entered this issue into their CAP as AR 03987440. The corrective actions in response to this violation were to identify appropriate alternate methods of decay heat removal and incorporate them into the shutdown safety management program utilized during plant outages.

Analysis: The inspectors determined that failing to meet/follow the required actions for LCOs 3.9.9 and 3.4.10 was a performance deficiency. Specifically, the operators failed to verify a credited alternate decay heat removal method that would satisfy the required action for the LCO. 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 Mitigating Systems cornerstone attribute of equipment performance and adversely affected the cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, with the operators failing to identify a credited alternate method of decay heat removal and taking credit for the inoperable but in service RHR shutdown cooling train, the actual available methods that could have been credited were not verified to ensure their availability to provide the required function.

Using IMC 0609, Attachment 4, "Initial Characterization of Findings," Table 2, the finding was screened against the Mitigating Systems cornerstone and using Table 3 "SDP Appendix Router" the inspectors determined the answer to question A, "does the finding pertain to operations, an event, or a degraded condition while the plant was shut down?" was "Yes" and proceeded to IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," dated May 9, 2014. The finding was screened against the Exhibit 3—Mitigating Systems Screening questions and determined to be of very low safety significance (Green) because the answer to all of the applicable screening questions was "No."

The inspectors determined that this finding affected the cross-cutting aspect 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. Proposed actions are determined to be safe in order to proceed, rather than unsafe in order to stop. Specifically, the senior reactor operators at the station had historically credited inoperable RHR shutdown cooling subsystems as their own alternate decay heat remove method because they believed it was allowable. [H.14]

<u>Enforcement</u>: Title 10 CFR 50.36(c)(2)(i) states, in part, "When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met."

Technical Specification 3.9.9 stated, in Mode 5 with irradiated fuel in the reactor vessel and the water level less than 22 feet 8 inches above the top of the reactor vessel flange, "Two RHR shutdown cooling subsystems shall be operable and one RHR shutdown cooling subsystem shall be in operation." Condition A.1 "One or two RHR shutdown cooling systems inoperable" has a required action to "Verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem." The completion time is within 1 hour and every 24 hours thereafter.

Technical Specification 3.4.10 stated, in Mode 4, "Two RHR shutdown cooling subsystems shall be operable, and, with no circulation pump in operation, at least one RHR shutdown cooling subsystem shall be in operation". Condition A.1 "One or two RHR shutdown cooling systems inoperable" has a required action to "Verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem." The completion time is within 1 hour and every 24 hours thereafter.

Contrary to the above:

- From May 24, 2016, at 4:00 a.m. through May 25, 2016, at 12:30 a.m., after declaring the RHR 'B' shutdown cooling subsystem inoperable with the reactor vessel water level drained down to below 22 feet 8 inches, the licensed operators failed to identify an alternate method to remove decay heat to satisfy the required action for TS 3.9.9, Condition A “with one RHR shutdown cooling subsystem inoperable, verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem.”

- From May 25, 2016, at 12:30 a.m. through May 26, 2016, at 3:40 p.m., after declaring the RHR ‘B’ shutdown cooling subsystem inoperable, the licensed operators failed to identify an alternate method to remove decay heat to satisfy the required action for TS 3.4.10, Condition A “with one RHR shutdown cooling subsystem inoperable, verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem.”

Specifically, in each TS entry above, the licensed operators incorrectly verified the inoperable RHR shutdown cooling subsystem as its own alternate method of decay heat removal.

The corrective actions in response to this violation were to identify appropriate alternate methods of decay heat removal and incorporate them into the shutdown safety management program utilized during plant outages. Because this finding was of very low safety significance and was entered into the CAP as AR 03987440, this violation is being treated as an NCV, in accordance with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000461/2017001–02: Failed to Verify an Appropriate Alternate Method of Decay Heat Removal)**

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

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

- testing of RHR 'C' pump breaker;
- testing of the emergency reserve auxiliary transformer;
- testing of the reactor pressure vessel pressure transmitter;
- testing of the reactor protection system reactor water level transmitter;
- testing of the EDG ventilation load shed relay; and
- testing of the ‘B’ standby liquid control pump.

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 technical specifications (TS), 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 PM 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 six PM testing samples as defined in IP 71111.19–05.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

a. Inspection Scope

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

- Clinton Power Station (CPS) 9080.26, "DG [diesel generator] 1C Test Mode Override and Load Reject Operability" (routine test);
- CPS 9431.01, "RPS [Reactor Protection System] Drywell Pressure C71–N050C Channel Calibration" (routine test);
- CPS 9080.26, "DG 1C Test Mode Override and Load Reject Operability" (routine test);
- CPS 9080.20, "DG 1C Differential Overcurrent Trip test and Trip Bypass Operability" (routine test); and
- CPS 9015.01, "Standby Liquid Control 'A' Pump Operability" (inservice test).

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

- did preconditioning occur;
- the effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis;
- plant equipment calibration was correct, accurate, and properly documented;
- as-left setpoints were within required ranges; and the calibration frequency was in accordance with TSs, the USAR, procedures, and applicable commitments;
- measuring and test equipment calibration was current;
- test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied;
- test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other

applicable procedures; jumpers and lifted leads were controlled and restored where used;
- test data and results were accurate, complete, within limits, and valid;
- test equipment was removed after testing;
- where applicable for inservice testing activities, testing was performed in accordance with the applicable version of Section XI, American Society of Mechanical Engineers code, and reference values were consistent with the system design basis;
- where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure;
- where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished;
- prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;
- equipment was returned to a position or status required to support the performance of its safety functions; and
- all problems identified during the testing were appropriately documented and dispositioned in the CAP.

This inspection constituted four 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.

2. RADIATION SAFETY

2RS5 Radiation Monitoring Instrumentation (71124.05)

.1 Walkdowns and Observations (02.02)

a. Inspection Scope

The inspectors assessed select portable survey instruments that were available for use for current calibration, source check stickers, instrument material condition, and operability.

The inspectors observed licensee staff demonstrate performance checks of various types of portable survey instruments. The inspectors assessed whether high-range instruments responded to radiation on all appropriate scales.

The inspectors walked down area radiation monitors and continuous air monitors to determine whether they were appropriately positioned relative to the radiation sources or areas they were intended to monitor. The inspectors compared monitor response with actual area conditions for selected monitors.

The inspectors assessed the functional checks for select personnel contamination monitors, portal monitors, and small article monitors to verify they were performed in accordance with the manufacturer's recommendations and licensee procedures.

These inspection activities constituted one complete sample as defined in IP 71124.05–05.

b. Findings

No findings were identified.

.2 Calibration and Testing Program (02.03)

a. Inspection Scope

The inspectors assessed laboratory analytical instruments used for radiological analyses to determine whether daily performance checks and calibration data indicated that the frequency of the calibrations was adequate and there were no indications of degraded instrument performance. The inspectors assessed whether appropriate corrective actions were implemented in response to indications of degraded instrument performance.

The inspectors reviewed the methods and sources used to perform whole body count functional checks before daily use and assessed whether check sources were appropriate and aligned with the plant's isotopic mix. The inspectors reviewed whole body count calibration records since the last inspection and evaluated whether calibration sources were representative of the plant source term and that appropriate calibration phantoms were used. The inspectors looked for anomalous results or other indications of instrument performance problems.

Inspectors reviewed select containment high-range monitor calibration and assessed whether an electronic calibration was completed for all range decades, with at least one decade at or below 10 rem/hour calibrated using an appropriate radiation source, and calibration acceptance criteria was reasonable.

The inspectors reviewed select monitors used to survey personnel and equipment for unrestricted release to assess whether the alarm set-points were reasonable under the circumstances to ensure that licensed material was not released from the site. The inspectors reviewed the calibration documentation for each instrument selected and discussed the calibration methods with the licensee to determine consistency with the manufacturer's recommendations.

The inspectors reviewed calibration documentation for select portable survey instruments, area radiation monitors, and air samplers. The inspectors reviewed detector measurement geometry and calibration methods for portable survey instruments and area radiation monitors calibrated on-site and observed the licensee demonstrate use of the instrument calibrator. The inspectors assessed whether appropriate corrective actions were taken for instruments that failed performance checks or were found significantly out of calibration, and that the licensee had evaluated the possible consequences of instrument use since the last successful calibration or performance check.

The inspectors reviewed the current output values for instrument calibrators. The inspectors assessed whether the licensee periodically measured calibrator output over the range of the instruments used with measuring devices that have been calibrated by a facility using National Institute of Standards and Technology traceable sources and corrective factors for these measuring devices were properly applied in its output verification.

The inspectors reviewed the licensee's 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," source term to assess whether calibration sources used were representative of the types and energies of radiation encountered in the plant.

These inspection activities constituted one complete sample as defined in IP 71124.05–05.

b. Findings

No findings were identified.

.3 Problem Identification and Resolution (02.04)

a. Inspection Scope

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

These inspection activities constituted one complete sample as defined in IP 71124.05–05.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

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

4OA1 Performance Indicator Verification (71151)

.1 Unplanned Scrams per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams per 7000 Critical Hours performance indicator (PI) for the period from the first quarter of 2016 through the fourth quarter of 2016. In order 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, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports and NRC integrated inspection reports for the period of January 1, 2016, through December 31, 2016, to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator, and none were identified.

This inspection constituted one unplanned scrams per 7000 critical hours sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.2 Unplanned Scrams with Complications

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams with Complications PI for the period from the first quarter of 2016 through the fourth quarter of 2016. In order 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, event reports and NRC integrated inspection reports for the period of January 1, 2016, through December 31, 2016, to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator, and none were identified.

This inspection constituted one unplanned scrams with complications sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.3 Unplanned Power Changes per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Transients per 7000 Critical Hours PI for the period from the first quarter of 2016 through the fourth quarter of 2016. In order 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, maintenance rule records, event reports and NRC integrated inspection reports for the period of January 1, 2016, through December 31, 2016, to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator, and none were identified.

This inspection constituted one unplanned power changes per 7000 critical hours sample as defined in IP 71151–05.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems

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

a. Inspection Scope

As discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify they were being entered into the licensee's CAP at an appropriate threshold, adequate attention was being given to timely corrective actions, and adverse trends were identified and addressed. Some minor issues were entered into the licensee's corrective action program as a result of the inspectors' observations; however, they are not discussed in this report.

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

b. Findings

No findings were identified.

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

.1 Entry into Loss of Vacuum Off Normal Procedure Due to Degrading Condenser Vacuum

a. Inspection Scope

The inspectors reviewed the operators' response degrading condenser vacuum. This event occurred on January 19, 2017. Main control room annunciators alarmed indicating off gas flow was lowering and condenser vacuum was degrading. The operators entered the loss of vacuum off normal procedure and determined the transient was a result of the in service steam jet air ejector (SJAE) suction valve closing. The operators determined the cause to be due to a worker tag out an equipment operator was in the process of hanging. The equipment operator removed the worker tag out and the condenser vacuum returned to normal.

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

b. Findings

Failure to Develop and Review a Worker Tag Out

Introduction: The inspectors documented a self-revealed finding of very low safety significance and associated NCV of TS 5.4.1, "Procedures," for the licensee's failure to develop and review a worker tag out in accordance with station procedure OP–AA–109–10, “Clearance and Tagging,” Revision 12. Specifically, the licensee failed to identify the effect of a worker tag out on the in-service SJAE suction valve, which caused condenser vacuum to degrade resulting in the operators entering the off normal procedure for loss of condenser vacuum.

Description: On January 19, 2017, main control room annunciators alarmed indicating off gas flow was lowering and condenser vacuum was degrading. The operators entered station procedure CPS 4004.02, “Loss of Vacuum,” Revision 6c, and determined the transient was a result of the in-service SJAE suction valve closing. The operators determined the cause to be a loss of configuration control due to a worker tag out an equipment operator was in the process of hanging. The purpose of the tag out was to establish the electrical isolation necessary to replace a local hand switch for a condenser air removal valve. The operators immediately requested the equipment operator to remove the tag out and restore the electrical equipment. Upon restoration of the electrical equipment, the SJAE suction valve re-opened and the condenser vacuum recovered. The loss of vacuum procedure would have directed the operators to enter a rapid plant shutdown if they could not maintain vacuum below -24” Hg; the automatic main turbine trip set point was -21.6” Hg. The lowest condenser vacuum recorded during the transient was -26.58” Hg.

The inspectors’ independent review included interviewing the individuals involved with the development, review and execution of the worker tag out and station procedure OP–AA–109–101, “Clearance and Tagging,” Revision 12, for performing worker tag outs. Section 13.7.4, for worker tag outs, stated in part, “The same standards apply for technical rigor in development, authorizations and field application that apply to the clearance and tagging process.” Section 7.1.4, for clearance development standards, stated, “Clearance order impacts must be evaluated to ensure that effects on systems and components outside the clearance order boundary are identified and are acceptable, or properly dispositioned.” The equipment operator that developed the worker tag out did not identify that opening the control power breaker to the SJAE suction valve would cause the valve to close while reviewing the electrical drawings. Per the station procedure a senior reactor operator performed an independent review of the worker tag out prior to authorizing the work which also failed to identify the effect of the tag out on other equipment. Both reviews required looking at multiple complex electrical diagrams with a high level of attention to detail and understanding to determine the control power breaker being opened would cause the suction valve to close.

The licensee entered this issue into their CAP as AR 03964405 and performed a CAP evaluation. As an interim corrective action, site senior leadership implemented a temporary face to face screening of all work in the plant in order to reinforce the human performance attributes discovered in the corrective action program evaluation. Additionally the operations department issued a standing order to require worker tag outs to be challenged by a second senior reactor operator.

Analysis: The inspectors determined that the licensee’s failure to appropriately develop and review a worker tag out in accordance with procedure OP–AA–109–10, “Clearance and Tagging,” was a performance deficiency. Specifically, the licensee failed to identify the effect of the worker tag out on the in-service SJAE suction valve, which caused condenser vacuum to degrade resulting in entry into the off normal procedure for loss of condenser vacuum. 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 Initiating Events cornerstone attribute of configuration control and adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during power operations. Specifically, the failure to properly develop the worker tag out caused the condenser vacuum to degrade, challenging the operators to quickly diagnose the issue and take action to avoid a turbine trip.

Using IMC 0609, Attachment 4, “Initial Characterization of Findings,” issued October 7, 2016, 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 it did not cause a reactor trip or the loss of mitigation equipment relied upon to transition the plant from the onset of a trip to a stable shutdown condition.

The inspectors determined that this finding affected the cross-cutting area of human performance in the aspect of avoid complacency, where individuals recognize and plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes. Individuals implement appropriate error reductions tools. Specifically, the operations department failed to implement appropriate error reduction tools such as questioning attitude and thorough work product reviews to ensure the worker tag out considered all potential effects to other plant equipment. [H.12]

<u>Enforcement</u>: Technical Specification 5.4.1, “Procedures,” states, “Written procedures shall be established, implemented and maintained covering the applicable procedures recommended in Regulatory Guide (RG) 1.33, Revision 2, Appendix A.” Section 1.c of RG 1.33 requires administrative procedures for equipment control (locking and tagging).

Procedure OP–AA–109–101, “Clearance and tagging,” Section 7.1.4 stated, “Clearance Order impacts must be evaluated to ensure that effects on systems and components outside of the clearance order boundary are identified and are acceptable, or properly dispositioned.”

Contrary to the above the licensee did not implement the applicable procedures recommended in RG 1.33. Specifically, the licensee did implement the procedure for equipment control when they failed to identify, evaluate or properly disposition the impacts of the clearance order on the in-service SJAE suction valve, which caused condenser vacuum to degrade and resulted in entry into the loss of vacuum off-normal procedure.

As corrective actions, the operations department issued a standing order to require worker tag outs to be challenged by a second senior reactor operator. Because the violation was of very low safety significance and was entered into the licensee's corrective action program as AR 03964405, this violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement policy. **(NCV 05000461/2017001–03: Failure to Develop and Review a Worker Tag Out)**

.2 (Closed) Licensee Event Report 05000461/2016–012–00: Residual Heat Removal ‘C’ Pump Failure during Surveillance Testing as a Result of Breaker Latch Check Switch Adjustments Results in Condition Prohibited by Technical Specifications

This event occurred on December 5, 2016. During a surveillance test of the residual heat removal (RHR) ‘C’ pump, the breaker that supplies power to the pump failed to close as expected resulting in the pump failing to start. The pump was declared inoperable and TS LCO 3.5.1, “Emergency Core Cooling System—Operating,” required action A.1 was entered requiring the pump to be restored to an operable status in seven days. An investigation determined that the breaker latch check switch contacts were not closed and the latch check switch was not set in accordance with station procedures. The licensee’s corrective actions were to properly set the latch check switch, reinstall the breaker and performance of the surveillance to demonstrate operability. The licensee planned to change the station procedure to ensure proper latch switch operation. This licensee event report is closed.

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

a. Findings

Failure to Perform Maintenance on the Residual Heat Removal Pump ‘C’ Breaker in Accordance with Procedures

Introduction: The inspectors documented a self-revealed finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures and Drawings,” for the licensee’s failure to perform maintenance on a safety related breaker in accordance with station procedure CPS 8410.12C001, “Westinghouse DHP Circuit Breaker Checklist,” Revision 7. Specifically, the licensee failed to ensure the remaining travel on the latch check switch for the RHR ‘C’ pump breaker was within the acceptable range resulting in ‘C’ RHR pump failing to start.

Description: On December 5, 2016, while performing a surveillance test on the RHR ‘C’ pump, the pump failed to start. The licensee performed troubleshooting and determined the breaker latch check switch contacts were open. The purpose of the latch check switch was to provide a permissive for the breaker to close, after ensuring the breaker was fully reset and the breaker closing springs were fully charged.

When the licensee inspected the breaker, the closing springs were still fully charged after the attempt to close the breaker. With the springs fully charged, if the latch check switch was adjusted properly the latch switch contacts would have been closed. The licensee measured the distance past the contact point for over travel and identified there was no over travel past the contact closure point as required by procedure.

The breaker was installed on March 11, 2016, in accordance with station procedure CPS 8410.12C001. Step 8.13.4 would adjust the latch check switch over travel past the point that the contacts are closed from 1/8 inches to 3/16 inches. This step was marked complete by the licensee and a value of 3/16 inches was recorded in the procedure. This step was performed in the maintenance shop prior to transferring the breaker to the cubicle where it would be installed. Since the setting could be disturbed during transport of the breaker from the shop to the breaker cubicle, Step 8.31.1.4 of the procedure required checking and recording the setting again before the breaker was installed. This step was marked complete but did not record the actual measurement. Additionally, the supervisory review performed on the work order did not identify the missing data.

The surveillance test was successfully performed two times between the breaker installation and the surveillance test run when the pump failed to start. Since the latch check switch was found to be right at the point of contact, it was possible that during the previous surveillance tests it made contact just enough to complete the circuit. The specific purpose of the over travel distance was to ensure full contact and prevent this scenario.

The licensee's causal investigation concluded that the apparent cause for the RHR 'C' pump failure to start was an inadequate latch check switch verification just prior to the installation of the breaker into the cubicle. This was based upon no data recorded for the setting check just prior to installation of the breaker and the highly unlikely occurrence that the switch adjustment changed during the physical installation or operation of the breaker. Based on this information the licensee concluded that switch contacts failed to close following the last successful surveillance test on September 6, 2016. The licensee also identified a contributing cause in the area of procedure adequacy because though Step 8.31.1.4 stated to record the latch check switch over travel it did not include a block to write down the number. This is different than all the other steps in the procedure that requiring a value be recorded.

The corrective actions taken by the licensee included providing coaching to the involved individuals as well as changing the procedure to include a block to record the latch check switch over travel value for Step 8.31.1.4.

<u>Analysis</u>: The inspectors determined that the licensee's failure to perform maintenance on a safety-related breaker in accordance with station procedure CPS 8410.12C001, "Westinghouse DHP Circuit Breaker Checklist," was a performance deficiency. Specifically, the licensee failed to ensure the remaining travel on the latch check switch for the RHR 'C' pump breaker was within the acceptable range resulting in RHR 'C' pump failing to start. 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 Mitigating Systems cornerstone attribute of equipment performance and adversely affected the cornerstone objective of ensuring the availability, capability and reliability of equipment that responds to initiating events. Specifically, the performance deficiency adversely impacted the operability of the RHR 'C' pump.

The inspectors evaluated the finding using the significance determination process in accordance with IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power," Exhibit 2, dated June 19, 2012. The inspectors reviewed the Mitigating Systems screening questions in Exhibit 2 and answered "No" to question A.1, "If the finding is deficiency affecting the design or qualification of a mitigating SSC, does the SSC maintain its operability or functionality," and A.2, "Does the finding represent a loss of system and/or function." The inspectors answered "Yes" to question A.3, "Does the finding represent an actual loss of function of at least a single Train for > its Tech Spec Allowed Outage Time or two separate safety systems out-of-service for > its Tech Spec Allowed Outage Time." Therefore, a detailed risk evaluation was required.

A RIII senior reactor analyst (SRA) performed a detailed risk evaluation of the finding using the NRC's Standardized Plant Analysis Risk (SPAR) model, revision 8.50. The

SRA modeled the degraded condition as a failure to start of RHR pump ‘C’ for an exposure period of 90 days. The exposure period was the time from the last successful surveillance test until the breaker for the pump failed to operate on December 5, 2016. The change in core damage frequency estimated using the SPAR model for internal events was less than 1E–6/year but greater than 1E–7/year. The dominant core damage sequence was a loss of condenser heat sink initiating event followed by the failure of all RHR pumps, the failure to vent containment to remove decay heat, and the failure of injection after containment failure. After considering the potential contribution from external events using both the SPAR model and the licensee’s interim fire PRA model, the SRA concluded that the total change in core damage frequency as less than 1E–6/year, which is a finding of very low safety significance (Green). The SRA also considered the change in large early release frequency using IMC 0609 Appendix H, “Containment Integrity Significance Determination Process” and concluded that the change was less than 1E–7/year.

The inspectors determined that 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, the organization failed to ensure procedure Step 8.31.1.4 included a block for recording the latch check switch over-travel value, which led to confusion on whether the value was required to be recorded and ultimately resulted in a failure to perform the step as written. [H.1]

<u>Enforcement</u>: Title 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that activities affecting quality be prescribed by documented procedures of a type appropriate to the circumstances and be accomplished in accordance with these procedures. The licensee established CPS 8410.12C001, “Westinghouse DHP Circuit Breaker Checklist,” Revision 7 as the implementing procedure for performing preventive and corrective maintenance on Westinghouse switchgear cubicles, an activity affecting quality.

Procedure CPS 8410.12C001, Step 8.31.1.4 stated, “Record remaining travel of tripping trigger from where latch check switch contacts close. Acceptable measurement is 1/8 inches to 3/16 inches.”

Contrary to the above, on March 11, 2016, the licensee failed to follow Step 8.31.1.4 of procedure CPS 8410.12C001. Specifically, the licensee failed to record the remaining travel of the of the tripping trigger from where latch check switch contacts close, and ensure the measurement was acceptable.

As corrective actions, the licensee revised the procedure to include a block to record the latch check switch over travel value in the breaker installation section. Because the violation was of very low safety significance and was entered into the licensee's corrective action program as AR 03949655, this violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement policy. **(NCV 05000461/2017001–04: Failure to Perform Maintenance on Residual Heat Removal Pump ‘C’ Breaker in Accordance with Procedures)**

4OA5 Other Activities

.1 (Closed) NRC Temporary Instruction 2515/192, "Inspection of the Licensee's Interim Compensatory Measures Associated with the Open Phase Condition Design Vulnerabilities in Electric Power Systems"

a. Inspection Scope

The objective of this performance based temporary instruction is to verify implementation of interim compensatory measures associated with an open phase condition (OPC) design vulnerability in electric power system for operating reactors. The inspectors conducted an inspection to determine if the licensee had implemented the following interim compensatory measures. These compensatory measures are to remain in place until permanent automatic detection and protection schemes are installed and declared operable for OPC design vulnerability. The inspectors verified the following:

- The licensee had identified and discussed with plant staff the lessons-learned from the OPC events at the US operating plants including the Byron station OPC event and its consequences. This includes conducting operator training for promptly diagnosing, recognizing consequences, and responding to an OPC event.
- The licensee had updated plant operating procedures to help operators promptly diagnose and respond to OPC events on off-site power sources credited for safe shutdown of the plant.
- The licensee had established and continue to implement periodic walkdown activities to inspect switchyard equipment such as insulators, disconnect switches, and transmission line and transformer connections associated with the offsite power circuits to detect a visible OPC.
- The licensee had ensured that routine maintenance and testing activities on switchyard components have been implemented and maintained. As part of the maintenance and testing activities, the licensee assessed and managed plant risk in accordance with 10 CFR 50.65(a) (4) requirements.

b. Findings and Observations

No findings of significance were identified. The inspectors verified the criteria were met.

4OA6 Meetings

.1 Exit Meeting Summary

On April 20, 2017, the inspectors presented the inspection results to Mr. B. Kapellas, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered propriety.

.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 March 2, 2017.

The inspectors confirmed that none of the potential report input discussed was considered proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

T. Stoner, Site Vice President
B. Kapellas, Plant Manager
D. Avery, Regulatory Assurance
R. Bair, Work Management Director
J. Cunningham, Maintenance Director
C. Dunn, Operations Director
M. Friedmann, Emergency Preparedness Manager
S. Gackstetter, Engineering Director
M. Heger, Senior Manager Plant Engineering
N. Hightower, Radiation Protection Manager
N. Keen, Design Engineering
T. Krawyck, Senior Manager Plant Engineering
B. Marchese, ODCM Specialist
W. Marsh, Security Manager
S. Minya, Operations Training Manager
J. Pfabe, Licensing Consultant
K. Pointer, Regulatory Assurance
D. Shelton, Regulatory Assurance Manager
S. Strickland, Shift Operations Superintendent
J. Ward, Chemistry Manager

U.S. Nuclear Regulatory Commission

K. Stoedter, Chief, Reactor Projects Branch 1
W. Schaup, Clinton Senior Resident Inspector
E. Sanchez Santiago, Resident Inspecto

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000461/2017001–01	NCV	Plant Barrier Control Program Failed to Compensate for an Impacted Flood Barrier (Section 1R15.1)
05000461/2017001–02	NCV	Failed to Verify an Appropriate Alternate Method of Decay Heat Removal (Section 1R15.2)
05000461/2017001–03	NCV	Failure to Develop and Review a Worker Tag Out (Section 4OA3.1)
05000461/2017001–04	NCV	Failure to Perform Maintenance on Residual Heat Removal Pump ‘C’ Breaker in Accordance with Procedures (Section 4OA3.2)

Closed

05000461/2016–012–00	LER	Residual Heat Removal RHR ‘C’ Pump Failure during Surveillance Testing as a Result of Breaker Latch Check Switch Adjustments Results in Condition Prohibited by Technical Specifications (Section 4OA3.02)
2515/192	TI	Inspection of the Licensee’s Interim Compensatory Measures Associated with the Open Phase Condition Design Vulnerabilities in Electric Power Systems (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather, that selected sections of 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

- MA–AA–716–026; “Station Housekeeping/Material Condition Program,” Revision 13
- CPS 4302.01; “Tornado High Winds,” Revision 21F
- OP–AA–108–111–1001; “Severe Weather and Natural Disaster Guidelines,” Revision 12
- SY–AA–101–146; “Severe Weather Preparation and Response,” Revision 1

1R04 Equipment Alignment

- OP–AA–108–103; Locked Equipment Program; Revision 2
- CPS 3506.01E001; Diesel Generator and Support Systems Electrical Lineup; Revision 18c
- CPS 3506.01V002; Diesel Generator and Support Systems Instrument Valve Lineup; Revision 11b
- CPS 3506.01V001; Diesel Generator and Support Systems Valve Lineup Division 2; Revision 13a
- CPS 8801.06C001; H22 Panel Mounted Instrument Valve Operation Checklist; Revision 33c
- CPS 9433.20; ECCS LPCS Pump Discharge Pressure ADS E21–N052 Channel Calibration; Revision 34f
- CPS 3313.01V001; Low Pressure Core Spray Valve Lineup; Revision 13b
- CPS 3313.01V002; Low Pressure Core Spray Instrument Valve Lineup; Revision 8a
- CPS 3313.01E001; Low Pressure Core Spray Electrical Lineup; Revision 11b
- CPS 3319.01V002; Standby Gas Treatment Instrument Valve Lineup; Revision 6a
- CPS 3319.01E001; Standby Gas Treatment Electrical Lineup; Revision 11
- CPS 3319.01V002; Standby Gas Treatment Instrumentation Valve Lineup; Revision 6a
- M05–1035; Diesel Gen Aux System (DG) Exhaust Starting Air Exhaust & Combustion System; Revision AB
- M05–1036; P&ID Diesel Generator Fuel Oil System (DO); Revision S
- M05–1073; P&ID Low Pressure Core Spray (LPCS); Revision AH
- WO 01684074; 9433.20A20 CC ECCS LPCS Discharge ADS E21–N052 CC
- AR 03984708; Question on LPCS Test Valve Lineup 3313.01V002
- AR 02715746; 1DG01KC Diesel Generator 1C Has Minor Oil Seepage from Kiene
- AR 03955951; FME Concern Division 2 Diesel Generator Control Panel

1R05 Fire Protection

- OP–AA–201–003; Fire Drill Performance; Revision 16
- OP–AA–201–005; Fire Brigade Qualification; Revision 9
- OP–AA–201–003; Attachment 1; Fire Drill Record (U2016–15); September 6, 2016
- OP–AA–201–003; Attachment 3; Fire Drill Scenario (U2016–15); August 3, 2016
- CPS 1893.01; Fire Protection Impairment Reporting; Revision 20d
- CPS 1893.04M720; 762 Turbine: Turbine Auxiliaries Prefire Plan; Revision 6c
- CPS 1893.01M001; Fire Door Compensatory Measures; Revision 5f
- CPS 1893.04; Fire Fighting; Revision 18
- CPS 1893.04M001; Prefire Plan Cross Index; Revision 3c

- CPS 1893.04M002; Prefire Plan/Fire Zone Cross Index; Revision 3b
- CPS 1893.04M003; Prefire Plan Legend; Revision 1
- CPS 3822.13C002; Visual Inspection of Portable Fire Extinguishers Checklist—NON- Essential Areas within the Protected Area; Revision 2c
- CPS 3822.13C001; Visual Inspection of Portable Fire Extinguishers Checklist—Power Block and Essential Areas within the Protected Area; Revision 2c
- CPS 3822.13; Visual Inspection of Portable Fire Extinguishers; Revision 4e
- CPS 1893.06; Fire Protection Maintenance and Testing Program; Revision 12d
- CPS 1893.04M400; 712 Fuel: Basement Prefire Plan; Revision 5
- CPS 1893.04M102; 707–781 Auxiliary: RHR ‘A’ Pump and Heat Exchanger Room Prefire Plan; Revision 5
- CPS 1893.04M350; 781 Control: Div 2 Cable Spreading Room Prefire Plan; Revision 5c
- CPS 1893.04M370; 825 Control: Control Room HVAC Prefire Plan; Revision 7a
- CPS 1893.04M364; 800 Control: Main Control Room Prefire Plan; Revision 3a
- CPS 1893.04M511; 737 Diesel Generator: Div 1 Diesel Generator & Day Tank Room Prefire Plan; Revision 6a
- AR 03969335; Procedure Enhancement CPS 3822.13

1R11 Licensed Operator Requalification Program

- OP–AA–101–111–1001; Operations Standards and Expectations; Revision 17
- OP–AA–300; Reactivity Management; Revision 9
- OP–CL–108–101–1003; Operations Department Standards and Expectation; Revision 35
- TQ–AA–150; Operator Training Programs; Revision 12
- TQ–AA–155; Conduct of Simulator Training and Evaluation; Revision 5
- CPS 3005.01; Unit Power Changes; Revision 42c
- Reactivity Maneuver Guidance Sheet # C17–007
- Reactor Engineer’s Evolution Plan # C17–007
- WO 01923734; WO for Cycle 17 Down Power Activities

1R12 Maintenance Effectiveness

- ER–AA–310; Implementation of Maintenance Rule; Revision 9
- ER–AA–310–1001; Maintenance Rule Scoping; Revision 4
- ER–AA–310–1002; Maintenance Rule Functions—Safety Significance Classification; Revision 3
- ER–AA–310–1003; Maintenance Rule—Performance Criteria Selection; Revision 4
- ER–AA–310–1004; Maintenance Rule—Performance Monitoring; Revision 13
- ER–AA–310–1005; Maintenance Rule—Dispositioning Between (a)(1 and (a)(2); Revision 7
- ER–AA–310–1006; Maintenance Rule—Expert Panel Roles and Responsibilities; Revision 5
- AR 03970055; 98 Recorders Maintenance Rule Requires (A)(1) Determination
- AR 02455894; Recorder Screen Has Gone Dark
- AR 02590230; 1TR–CM017 Div 1 Suppression Pool Temperature Recorder Printer Failed
- AR 02602955; 1C51–R603D Div 4 NMS Recorder Failed
- AR 02620629; 1LLRCM240 Recorder has Failed
- AR 02639615; Recorder 1PR–CM257 Display is Blank/Dark
- AR 0269766; 1PR–CM257 MCR Recorder Failed
- AR 03942990; 1C51R603B Display Screen Blank
- AR 03959970; 1LR–CM240 Suppression Pool Level and Temp Screen Blank
- AR 02718948; 98 Recorders System Maintenance Rule (A)(2) At Risk
- (A)(1) Determination Template for “98” Recorders

1R13 Maintenance Risk Assessments and Emergent Work Control

- AD–AA–3000; Nuclear Risk Management Process; Revision 1
- AD–AA–3000; Nuclear Risk Management Process; Revision 1
- ER–AA–600; Risk Management; Revision 7
- ER–AA–600–1011; Risk Management Program; Revision 14
- ER–AA–600–1012; Risk Management Documentation; Revision 12
- ER–AA–600–1014; Risk Management Configuration Control; Revision 7
- ER–AA–600–1042; On-line Risk Management; Revision 9
- OP–AA–108–117; Protected Equipment Program; Revision 4
- OP–AA–108–117; Protected Equipment Program; Revision 4
- WC–AA–101; On-Line Work Control Process; Revision 26
- WC–AA–104; Integrated Risk Management; Revision 23
- WC–AA–101; On-Line Work Control Process; Revision 26
- WC–AA–101–1006; On-Line Risk Management and Assessment; Revision 2
- CPS 4302.01; Tornado High Winds; Revision 21F
- AR 03988367; Results of the Standby Liquid Control A and B Oil Sample
- OP–AA–108–117–1001; Spent Fuel Storage Pool Heat-up rate with Loss of Normal Cooling; Revision 0

1R15 Operability Evaluations

- OP–AA–108–115; Operability Determinations; Revision 19
- OP–AA–108–115-1002; Supplemental Consideration On-shift Immediate Operability Determinations; Revision 3
- OP–AA–102–103; Operator Work-Around Program; Revision 4
- OP–AA–102–103–1001; Operator Burden and Plant Significant Decisions Impact Assessment Program; Revision 7
- CC–AA–201; Plant Barrier Control Program; Revision 11
- LS–AA–1400; Event Reportability Guidelines 10 CFR 50.72 and 50.73; Revision 3
- CPS 1401.09; Control of System and Equipment Status; Revision 9b
- CPS 1052.01; Conduct of System Line-ups; Revision 9
- AR 03968392; Fuse Blown During Maintenance on 0PDSVG136
- WO 01188518; Perform Calibration, Replace Cover O-ring 0PDSVG136
- HURB; Fuse Blown During Maintenance on 0PDSVG136
- HU–AA–1211; Pre-job Briefings; Revision 11
- C1R16 Shutdown Safety Management Program
- EC 404691; Decay Heat C1R16; Revision 0
- EC 379554; Clarification of the Safety Function of RHR Shutdown Cooling; Revision 0
- CPS 3312.02; Alternate Shutdown Cooling (A-SDC) Methods; Revision 9a
- CPS 4006.01; Loss of Shutdown Cooling; Revision 5a
- CPS 3312.03; RHR-Shutdown Cooling (SDC) and Fuel Pool Cooling and Assist (FPC&A); Revision 10e
- CPS 3312.01; Residual Heat Removal (RHR); Revision 45d
- OU–CL–104; Shutdown Safety Management Program Clinton Power Station; Revision 15
- AR 03987440; Inoperable RHR Loop Cannot be Credited for SDC Actions
- WO 01567286; 1SD–24 Replace Door Shaft and Lock Shaft O-rings
- CPS 8250.02C001; Modified Watertight Door Maintenance Checklist; Revision 2a
- Plant Barrier Impairment Permit; PBI–2017–02–003; 1SD1–24
- AR 03972544; Senior Resident Inspector Operability Question
- AR 03980495; Inadequate PBI 2017–02–003

- AR 03966642; Request RR ‘A’FCV Lockout be Evaluated for Work-Around
- AR 0396141; 0HSCA030B Failed to Spring Return Back to Normal
- AR 03985892; 1SX003C Indicates Intermediate When Valve Locally Shut
- AR 03974371; Refuel Bridge Procedural Work-Around
- AR 03957200; RR FCV ‘A’ Position Change With No Operator Demand

1R19 Post-Maintenance Testing

- MA–AA–716–012; Post Maintenance Testing; Revision 20
- CPS 9381.01C002; MOV Thermal Overload Bypass Post Maintenance Verification Checklist; Revision 29
- CPS 8451.06; Corrective Maintenance for Limitorque SMB–0 through SMB–4 and SB–0, SB–1 and SB–3 Operators; Revision 9
- CPS 9015.01; Standby Liquid Control System Operability; Revision 41d
- CPS 9015.01D001; SLC Pump and Valve Data Sheet; Revision 38b
- CPS 8801.06C001; H22 Panel Mounted Instrument Valve Operation Checklist; Revision 33c
- CPS 9431.04; RPS Reactor Water Level B21–N080 Channel Calibration; Revision 35f
- CPS 9431.04D04; RPS Reactor Water Level B21–N080D Channel Calibration Data Sheet; Revision 36b
- CPS 9431.03; RPS Reactor Pressure B21–N078D Channel Calibration; Revision 37e
- CPS 9431.03D002; RPS Reactor Pressure B21–N078D Channel Calibration Data Sheet; Revision 33b
- CPS 8801.24; Rosemount Series 1153/1154 Pressure Transmitter Replacement; Revision 1d
- CPS 8801.05D002; Rosemount D/P Transmitter Zero Shift Determination Data Sheet; Revision 2b
- CPS 9053.07; RHR B/C Pumps and RHR B/C Water Leg Pump Operability; Revision 47d
- CPS 9053.07D001; RHR B/C & RHR B/C Water Leg Pump Operability Data Sheet; Revision 45c
- CPS 8410.21C001; Westinghouse DHP Circuit Breaker Checklist; Revision 7
- WO 01812696; Perform MOV Thrust Verification and Clean and Inspect 1E12F048
- WO 04606492; Ops PMT Run SLC Pump B
- WO 04583682; 9015.01B23 SLB Pump B Operability
- WO 00918049; EQ-CL021-03 Replace Rosemount 1153
- WO 00918047; EQ-CL021-03 Replace Rosemount 1153
- WO 04573693; RHR Pump Failed to Start
- WO 04582195; EM Swap Breaker/Latch Check
- WO 04582195; Breaker 1AP09EF Latch Check Switch
- WO 04609243–03; EM PMT New 1AP11E/3A–427X3–41A
- AR 03982969; Adverse Trend in 1C41-C001A Reduced Margin to Alert Range

1R22 Surveillance Testing

- OP–AA–108–103; Locked Equipment Program; Revision 3
- CPS 9080.03; Diesel Generator 1C Operability—Manual and Quick Start Operability; Revision 35
- CPS 9080.20; DG IC Differential Overcurrent Trip Test and Trip Bypass Operability; Revision 3a
- CPS 9080.26; DG 1C Test Mode Override and Load Reject Operability; Revision 5e
- CPS 9080.26D001; DG 1C Test Mode Override and Load Reject Operability Data Sheet; Revision 1
- CPS 3506.01C005; Diesel Generator Start Log; Revision 1b

- CPS 3506.01C003; Diesel Generator 1C Pre-start Checklist; Revision 6b
- CPS 9080.03D001; Diesel Generator 1C Operability—Manual and Quick Start Data Sheet; Revision 23a
- CPS 3506.01E001; Diesel Generator and Support Systems Electrical Lineup; Revision 18C
- 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 3810.01; Diesel Generator Fuel Oil Transfer Pump Operability; Revision 1f
- CPS 9431.01; RPS Drywell Pressure C71-N050C Channel Calibration; Revision 36b
- CPS 9431.01D003; RPS Drywell Pressure C71-N050C Channel Calibration Data Sheet; Revision 33b
- CPS 8801.12C001; Local Mounted Instrument Valve Operation Checklist; Revision 15b
- CPS 9015.01; Standby Liquid Control System Operability; Revision 41d
- CPS 9015.01D001; SLC Pump and Valve Data Sheet; Revision 38b
- WO 01864259; 9080.20 DG 1C Differential Overcurrent Trip Test and Trip
- WO 04609142; 9080.03A23 OP DG 1C Operability Monthly Test
- WO 01898675; 9080.12 DG Fuel Oil Transfer Pump Operability 1C
- WO 01923844; 9080.12 DG Fuel Oil Transfer Pump Operability 1C
- WO 01946494; 9080.12 DG Fuel Oil Transfer Pump Operability 1C
- WO 04569381; 9080.12 DG Fuel Oil Transfer Pump Operability 1C
- WO 04569394; DO Check Valve IST Closure Testing 1DO001C
- WO 04572298; 9015.01 SLC Valve Operability
- WO 04583681; 9015.01A23 OP SLC Pump A Operability
- WO 01863494; 9431.01C20 RPS Drywell Pressure C71-N050C Calibration
- WO 01956172; 9080.03R20 Op DG 1C Operability—Semi-Annual Quick Start Test
- AR 02642429; Identified Degraded Div 3 1PIDG189 Gauge not Marked
- AR 02642048; Div 3 DG Fuel Priming Pump not Developing Proper Pressure
- AR 03971981; NRC ID: Div 3 DO Transfer Pump Tested to Old Data
- AR 02738547; 1DO01PC Div 3 FO Transfer Pump Flow Higher than Acceptable

2RS5   Radiation Monitoring Instrumentation

- PI–AA–126–1005–F–01; Check in Self-Assessment; Radiation Protection Instrumentation; Dated December 22, 2016
- RP–AA–700–1235; Operation and Calibration of the PM-12 Gamma Portal Monitor; Revision 3
- RP–AA–229; Fastscan ABACOS Plus Whole Body Counter (WBC) Calibration; Revision 3
- RP–AA–700; Controls for Radiation Protection Instrumentation; Revision 4
- RP–AA–230; Operation of Canberra FASTSCAN Whole Body Counter (WBC) Using ABACOS Plus; Revision 3
- RP–AA–700–1240; Operation and Calibration of the Canberra ARGOS–5 Personnel Contamination
- RP–AA–700–1246; Operation of Air Samplers; Revision 4
- RP–AA–700–1306; Attachment 2, MDH/RADCAL 2026C Correction Factor Worksheet; Revision 0; Dated January 4, 2017
- Work Order 01840626–01; IM 9437.65C20 CC CNMT/DW High Range Gamma Mon. (CM061) CC; Dated January 20, 2017
- Work Order 01689514–01; IM 9437.65 1RIXCM062 CNMT/DW High Range Gamma Monitor; Dated June 1, 2015
- Certificate of Calibration #0010949054; Asset/Equipment # 0012717, Ludlum 3030P, Serial # 275361; Dated March 31, 2016
- AR 02584812; RP ID: Safety Issue with SAM 12 #150; Dated November 10, 2015

- AR 02601517; RP ID: Instruments Cal Due Dates; Dated December 17, 2015
- AR 02623016; RP ID: ARGOS Source Jig Needs Improvement

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

- CC–AA–20; Configuration Management; Revision 2
- OP–AA–109–101; Clearance and Tagging; Revision 12
- ACE 3949655; Failure of Breaker 1AP09AF for Residual Heat Removal Pump 'C' to Close When Requested During Surveillance Testing
- CPS 5130.02; Alarm Panel 5130 Annunciators—Row 2; Revision 29c
- CPS 5130.01; Alarm Panel 5130 Annunciators—Row 1; Revision 26e
- CPS 5130.03; Alarm Panel 5130 Annunciators—Row 3; Revision 27e
- CPS 9053.07; RHR B/C Pumps and RHR B/C Water Leg Pump Operability; Revision 47d
- CPS 9053.07D001; RHR B/C & RHR B/C Water Leg Pump Operability Data Sheet; Revision 45c
- CPS 8410.21; Westinghouse DHP 6900, 4160 Volt Power Circuit Breaker; Revision 9
- CPS 8410.21C001; Westinghouse DHP Circuit Breaker Checklist; Revision 7
- WO 01588956; Swap Breakers for 1E12C002C Cubicle Checks, Relay Functional Test
- WO 04573693; RHR Pump Failed to Start
- WO 01952669; 9053.07C21 OP RHR Pump Operability Test
- AR 03964405; Unplanned Entry into Loss of Vacuum 4004.02 ON

4OA5 Other Activities

- EC 387664; Potential Design Vulnerability Single Open Phase; Revision 2
- EC 387664; Potential Design Vulnerability Single Open Phase; Revision 1
- EC 387664; Potential Design Vulnerability Single Open Phase; Revision 0
- CPS 3501.01; High Voltage Power System; Revision 28d
- AR 01323827; Potential Design Vulnerability Single Open Phase
- AR 02739265; Verify Compliance with NRC Temporary Instruction 2515/192

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
AR	Action Request
CAP	Corrective Action Program
CPS	Clinton Power Station
CFR	*Code of Federal Regulations*
DG	Diesel Generator
EDG	Emergency Diesel Generator
FCV	Flow Control Valve
IMC	Inspection Manual Chapter
IP	Inspection Procedure
LCO	Limiting Condition for Operation
LER	Licensee Event Report
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
OPC	Open Phase Condition
PBI	Plant Barrier Impairment
PI	Performance Indicator
PM	Post-Maintenance
RG	Regulatory Guide
RHR	Residual Heat Removal
RPS	Reactor Protection System
RR	Reactor Recirculation
SDC	Shutdown Cooling
SDP	Significance Determination Process
SJAE	Steam Jet Air Ejector
SPAR	Standardized Plant Analysis Risk
SRA	Senior Reactor Analyst
SSC	System, Structure, and Component
TS	Technical Specification
USAR	Updated Safety Analysis Report