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

August 11, 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 05000461/2017002 AND NOTICE OF VIOLATION

Dear Mr. Hanson:

On June 30, 2017, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Clinton Power Station. On July 13, 2017, the NRC inspectors discussed the results of this inspection with Mr. T. Stoner and other members of your staff. The results of this inspection are documented in the enclosed report.

Based on the results of this inspection, the NRC identified one finding of very-low safety significance (Green). The finding was determined to involve a violation of NRC requirements. The violation is cited in the enclosed Notice of Violation (Notice) and the circumstances surrounding the violation are described in detail in the enclosed report. Although determined to be of very-low safety significance (Green), in accordance with Section 2.3.2 of the NRC's Enforcement Policy, this violation is being cited because, after identification of the initial violation in the NRC Inspection Report 05000461/2016010(DNMS)/0721046/2016001(DNMS), your completed corrective actions failed to restore compliance with the design basis. The NRC Enforcement Policy is included on the NRC website at (https://www.nrc.gov/about-nrc/regulatory/enforcement/enforce-pol.html).

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. If you have additional information that you believe the NRC should consider, you may provide it in your response to the Notice. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

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

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:
1. Notice of Violation
2. Inspection Report 05000461/2017002

cc: Distribution via LISTSERV®

Letter to Bryan C. Hanson from Karla Stoedter dated August 11, 2017

SUBJECT: CLINTON POWER STATION—NRC INTEGRATED INSPECTION REPORT 05000461/2017002 AND NOTICE OF VIOLATION

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NOTICE OF VIOLATION

Exelon Generation Company, LLC
Clinton Power Station

Docket No. 50–461
License No. NPF–62

During a U.S. Nuclear Regulatory Commission (NRC) inspection conducted on July 21, 2016, through April 21, 2017, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> Title 10 of the *Code of Federal Regulations* (CFR), Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that the design control measures provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculation methods, or by the performance of a suitable testing program. The fuel building crane and the supporting fuel building steel structure are Seismic Category I components/structures subject to the 10 CFR Part 50, Appendix B, quality assurance requirements, and the applicable structural steel design specification is the American Institute of Steel Construction (AISC), 7th or 8th Edition.
>
> Contrary to the above,
>
> - On May 27, 2016, in Calculation No. SDQ15–24–DG09, Revision 12, and in the associated responses dated July 11, 2016, and July 30, 2016, the licensee failed to verify or check adequacy of the design for the fuel building crane and crane support structure elements. Specifically, the licensee verification process failed to identify that the load distribution used in the calculations was not justified, and that the use of the plate girder bearing stiffener provisions of the AISC Section 1.10.5.1 and the assumption of flexible supports at specific bolted connections in determination of wheel load distribution were not consistent with the field configuration based on design drawings.
>
> - On August 11, 2016, in Calculation No. SDQ15–24–DG09, Revision 13, the licensee failed to verify or check adequacy of the design for the fuel building crane and crane support structure elements. Specifically, the licensee verification process failed to identify that the unexpectedly high design margins for the bolt shear as indicated by the analysis were not justified because the evaluation incorrectly credited friction forces for reduction in the bolt stresses. Additionally, the bolt tensions calculated in the analysis were incorrect because the lever arm between the force couple used for determining bolt tension was incorrect and the AISC method used for determination of prying force was not applicable to the configuration being evaluated.
>
> - On October 11, 2016, in Calculation No. SDQ15–24–DG09, Revision 14, and in the associated response dated January 5, 2017, the licensee failed to verify or check adequacy of the design for the fuel building crane and crane support structure elements. Specifically, the licensee verification process failed to identify that the unexpectedly high design margins indicated by the analysis in Revision 14 of the calculation were not justified because the evaluation incorrectly credited frictional and bolt pre-tension forces in reduction of bolt shear and tension stresses. In the response dated January 5, 2017, the licensee did not correctly combine the vertical and horizontal seismic responses while applying the 100–40–40 percent rule for combining effects of spatial components of earthquake as described in Regulatory Guide 1.92.

This violation is associated with a Green Significance Determination Process finding.

Pursuant to the provisions of 10 CFR 2.201, Exelon Generation Company, LLC is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555–0001 with a copy to the Regional Administrator, Region III, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a, "Reply to a Notice of Violation; IR 05000461/2017002–02," and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level; (2) the corrective steps that have been taken and the results achieved; (3) the corrective steps that will be taken; and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555–0001.

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

In accordance with 10 CFR 19.11, you may be required to post this Notice within two working days of receipt.

Dated this 11th day of August 2017

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50–461
License No:	NPF–62
Report No:	05000461/2017002
Licensee:	Exelon Generation Company, LLC
Facility:	Clinton Power Station
Location:	Clinton, IL
Dates:	April 1 through June 30, 2017
Inspectors:	W. Schaup, Senior Resident Inspector E. Sanchez Santiago, Resident Inspector M. Doyle, Acting Resident Inspector S. Bell, Health Physicist G. Edwards, Health Physicist V. Meghani, Reactor Inspector L. Torres, ASME Inspector Illinois Emergency Management Agency S. Mischke, Resident Inspector Illinois Emergency Management Agency
Approved by:	K. Stoedter, Chief Branch 1 Division of Reactor Projects

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SUMMARY

Inspection Report 05000461/2017002; 04/01/2017–06/30/2017; Clinton Power Station; Operability Determinations and Functional Assessments, Plant Modifications, Outage Activities, Surveillance Testing, Radioactive Solid Waste Processing and Radioactive Material Handling, Storage, and Transportation, and Identification and Resolution of Problems.

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Seven Green findings were identified by the team. Five of these findings were considered Non-Cited Violations of U.S. Nuclear Regulatory Commission (NRC) regulations while one of these findings was considered a Notice of Violation of NRC regulations. 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.

NRC-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

- Green. The inspectors documented a self-revealed 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 licensee's failure to follow steps in Work Order (WO) 04640788 while performing troubleshooting on blown power transformer fuses in the division 3 emergency diesel start circuitry. Specifically, the electricians opened test switches in the wrong electrical cubicle resulting in the unexpected start of the division 3 emergency diesel generator and a loss of power to the 1C1 bus from an offsite source. The licensee entered this issue into their corrective action program (CAP) as Action Request (AR) 04012393. As corrective actions, the licensee performed a human performance review to identify the reasons the procedure was not followed and restored power to the 1C1 safety bus.

  The performance deficiency was determined to be more than minor because it impacted the Initiating Events cornerstone attribute of human performance and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the failure of the electrical maintenance technicians to follow their procedures resulted in a loss of power to the 1C1 electrical bus. The finding was screened against the Initiating Events cornerstone and determined to be of very low safety significance because the loss of power to the 1C1 bus occurred while Clinton was in a refueling outage when the high pressure core spray system was removed from service and not being relied upon for shutdown safety defense in depth. The loss of the 1C1 bus did not affect decay heat removal from the core, did not affect reactor coolant inventory, and the event occurred while the refuel cavity was flooded up for refueling operations. The inspectors determined that this finding affected the cross-cutting area of human performance in the aspect of avoid complacency where individuals implement

appropriate error reduction tools. Specifically, as documented in the licensee's human performance review, the electricians performing the work did not utilize any human performance tools to flag the equipment to be operated and improperly performed the concurrent verification of the component to be manipulated. [H.12] (Section 1R20.1)

- Green. The inspectors identified a finding of very low safety significance and an associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion II, "Quality Assurance Program," for the failure to implement a quality assurance program procedure. Specifically, the licensee failed to document a root cause and develop a corrective action to preclude repetition for the 1A bus transformer failure in accordance with quality assurance procedure PI–AA–125–1001, "Root Cause Analysis Manual." The licensee entered this issue into their CAP as AR 01594407. The corrective actions in response to this issue were to revise the root cause report with a root cause of insulation degradation of the phase windings over time and develop a corrective action to prevent recurrence by using Doble testing to ensure indication of transformer insulation degradation was discovered prior to failure.

  The performance deficiency was determined to be more than minor because if left uncorrected the performance deficiency had the potential to lead to a more significant safety concern. Specifically, the root cause and corrective actions to prevent recurrence were not identified until the licensee was prompted by the inspectors. As a result, additional transformer failures could have occurred. The finding was screened against the Initiating Events cornerstone and determined to be of very low safety significance because the finding did not involve the complete or partial loss of a support system that contributes to the likelihood of or cause an initiating event nor did it affect mitigation equipment. The inspectors determined this finding affected the cross-cutting area of human performance, in the aspect of resources, where leaders ensure that personnel, equipment, procedures and other resources are available and adequate to support nuclear safety. Specifically, the licensee's station procedure did not provide guidance on when a corrective action to preclude repetition is required, regardless of whether a risk assessment was performed. [H.1] (Section 4OA2.4)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance for the licensee's failure to perform maintenance on a safety-related motor control center cubicle. Specifically, the licensee failed to perform thermography on the division 1 shutdown service water pump room cooler breaker cubicle in accordance with the maintenance strategy/template without providing justification for differing from the template as required by MA–AA–716–210, "Performance Centered Maintenance Process," Revision 3. This resulted in the division 1 shutdown service water pump room cooler fan failing because of a high resistance connection that went undetected. The licensee entered this issue into their CAP as AR 02667822. As corrective actions, the licensee replaced the thermal overload relays and created a preventative maintenance action to perform thermography on this equipment on a periodic basis.

  This 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 room cooler fan failure directly impacted the operability of the division 1 shutdown service water pump and the

division 1 emergency diesel generator which are safety-related, risk significant systems. The finding was screened against the Mitigating Systems cornerstone and determined to be of very low safety significance because the inspectors were able to answer all of the associated screening questions "No." The inspectors determined that this finding is not indicative of current plant performance and therefore did not assign a cross-cutting aspect. (Section 4OA2.3)

- Green. The inspectors identified a finding of very low safety significance and an associated non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to assure that applicable regulatory requirements and the design basis was correctly translated into specifications, drawings, procedures, and instructions and that design control measures provided for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Specifically, the licensee failed to assure/validate operators were able to complete the standby liquid control time critical action for an anticipated transient without a scram specified in their licensing documents. The licensee entered this issue into their CAP as AR 03980202. As corrective actions, the licensee determined the scram choreography required to complete the time critical action in the specified time, initiated a standing order to inform the operating crews, processed a procedure change for the anticipated transient without scram choreography and performed an evaluation to determine the impact of initiating the standby liquid control system at 172 seconds.

  The performance deficiency was determined to be more than minor because the finding was associated with the procedure quality attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, with the operators initiating standby liquid control at 172 seconds instead of 120 seconds, the accident analysis calculations were required to be re-performed to assure the accident analysis requirements were met. The finding was screened against the Mitigating Systems cornerstone and determined to be of very low safety significance because the inspectors were able to answer all of the associated screening questions "No." The inspectors determined that this finding is not indicative of current performance and therefore did not assign a cross-cutting aspect. (Section 1R15.1)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a finding of very-low safety significance and an associated cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to properly verify the adequacy of design of the fuel building crane and crane support structure elements. Specifically, calculations involving the crane rail clips and clip bolts had multiple technical errors and failed to adequately demonstrate that the design met the design basis requirements. The licensee initiated corrective actions by documenting the deficiency in AR 4001089 and performed an evaluation demonstrating that the functionality of the crane was maintained.

  The finding was determined to be more-than-minor because it was associated with the design control attribute of the Barrier Integrity cornerstone and adversely affected the cornerstone objective of maintaining the functionality of the spent fuel pool (SFP) cooling system. Specifically, crane rail clip bolts were required to ensure structural integrity of structures, systems, and components described in the Updated Safety Analysis Report,

when subjected to design loads as part of safe load handling of heavy loads near the SFP and to ensure integrity of the spent fuel cask. In accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Initial Characterization of Findings," Table 2, the inspectors determined the finding affected the Barrier Integrity cornerstone because it was associated with SFP/fuel handling activities. Based on answering "No" to questions A through F in Table 3, the inspectors determined the finding could be evaluated using Appendix A, "The Significance Determination Process for Findings At-Power," Exhibit 3, for the Barrier Integrity cornerstone screening questions. Based on the crane remaining functional, the inspectors answered "No" to Questions D.1 through D.4 because the finding did not adversely affect decay heat removal capabilities, did not result from fuel handling errors, did not result in loss of SFP inventory, and did not affect the SFP neutron absorber or fuel bundle misplacement; therefore, the finding screened as having very-low safety significance. The finding was cross-cutting in the resolution aspect of the problem identification and resolution area because the licensee failed to take effective corrective actions in a timely manner to address issues identified earlier in the rail clip evaluations. [P.3] (Section 1R18.1)

- Green. The inspectors documented a self-revealed finding of very low safety significance and an associated non-cited violation of 10 of CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure of the licensee to provide sufficient work instructions for performing maintenance on the control room ventilation charcoal filter bed. Specifically, the work order used to change out the charcoal filter bed (Work Order 01494189) contained only the minimum required amount of charcoal to place in the bed. Sometime after filling the bed April 6, 2015, the charcoal settled, resulting in the 'B' control room ventilation system being declared inoperable after failing a surveillance test. The licensee entered this issue into their CAP as AR 03995612. As corrective actions, the licensee is revising the WO instructions and Clinton Power Station Procedure 9866.03 to require that charcoal be filled completely to the bottom of the deluge piping to allow for settling.

  The performance deficiency was determined to be more than minor because it impacted the Barrier Integrity cornerstone attribute of procedure quality and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to provide sufficient guidance in the work order regarding the quantity of charcoal to be installed resulted in the 'B' control room ventilation system failing a surveillance test and being declared inoperable. The finding was screened against the Barrier Integrity cornerstone and determined to be of very low safety significance because the finding only represents a degradation of a radiological barrier function provided for the control room. The inspectors determined that this finding affected the cross-cutting area of human performance in the aspect of design margins, where the organization operates and maintains equipment within design margins. Special attention is placed on maintaining fission product barriers, defense in depth, and safety-related equipment. Specifically, when performing maintenance on the charcoal bed, the licensee failed to recognize that filling the charcoal to the minimum bed level provided no margin if settling occurred. [H.6] (Section 1R22.1)

Cornerstone: Public Radiation Safety

- Green. A finding of very low safety significance and an associated non-cited violation of Title 10 of CFR 71.5(a) and 49 CFR 173.421(b) was self-revealed when the licensee

failed to properly classify a shipment per Department of Transportation (DOT) regulations. The failure to properly classify the shipment per DOT regulations allowed the shipment to proceed in transit with dose rates that were greater than what was stated on the shipping manifest. When the discrepancy in dose rates was noticed by the receiving entity, the shipment was immediately isolated and the licensee was contacted about the survey results. The licensee then dispatched two radiation protection technicians to perform confirmatory surveys. The survey data was confirmed, and the licensee was able to determine that the misclassification of the shipment was caused by dust and debris contained inside of a dust collector shifting during transportation, which created the elevated dose rate. The site implemented immediate corrective actions which included all shipments classified as limited quantity to be approved by a senior manager in the Radiation Protection Department prior to shipping. Another immediate corrective action required that the first 4 shipments conducted by the site shipper after this event be under the direct observation of a fleet independent shipper and a senior manager in the Radiation Protection Department. The licensee entered this event into their CAP as AR 03961544.

The inspectors determined that the performance deficiency was more than minor because the finding impacted the program and process attribute of the Public Radiation Safety cornerstone and adversely effected the cornerstone objective of ensuring adequate protection to public health and safety from exposure to radiation from routine civilian nuclear operations. Specifically, the misclassification of the shipment per DOT regulations could have led to individuals in the public domain being exposed to radiation dose that was greater than anticipated if conditions had been slightly altered. The finding was screened against the Public Radiation Safety cornerstone and determined to be of very low safety significance because: (1) the finding did not involve a certificate of compliance issue; (2) the failure to make emergency Notifications; (3) a low-level burial issue; or (4) a breach of the transportation package occurring during transit. The finding did involve a radioactive shipment above radiation limits. However, the shipment contained less than a Type A quantity of material (LSA I shipment), and dose rates were <2 millirem per hour on contact. The inspectors determined that this finding affected the cross-cutting area of human performance in the aspect of challenging the unknown, where individuals stop when faced with uncertain conditions. Risks are evaluated and managed before proceeding. Specifically, the risk associated with the content of the dust-collector shifting during transportation and creating an area that would lead to elevated dose rates was not evaluated by Clinton Power Station radiation protection staff. [H.11] (Section 2RS8.5)

REPORT DETAILS

Summary of Plant Status

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

- On May 8, 2017, the unit was shut down to perform scheduled Refueling Outage (RFO) C1R17.
- On May 28, 2017, reactor startup commenced and the unit was returned to approximately 28 percent reactor power on May 30, 2017, to perform control rod scram time testing.
- On May 30, 2017, while performing control rod scram time testing at approximately 28 percent reactor power, the unit experienced an automatic reactor scram. Forced Outage C1F59 was commenced to determine and take corrective actions for the cause of the automatic scram.
- On June 2, 2017, reactor startup commenced and the unit was returned to full power on June 5, 2017.
- On June 10, 2017, a manual scram was initiated due to a loss of feedwater heating. Forced Outage C1F60 was commenced to determine and take corrective actions for the cause of the loss of feedwater heating.
- On June 11, 2017, reactor startup commenced and the unit was returned to full power on June 14, 2017.
- On June 16, 2017, power was reduced to approximately 85 percent to perform rod pattern adjustments. The unit returned to full power on the same day.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness of Offsite and Alternate Alternating Current Power Systems

a. Inspection Scope

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

- coordination between the TSO and the plant during off-normal or emergency events;

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

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

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

The inspectors also reviewed corrective action program (CAP) items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their CAP in accordance with station corrective action procedures.

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

b. Findings

No findings were identified.

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:

- reactor core isolation cooling system with the high pressure core spray system inoperable;
- division 2 emergency diesel generator (EDG) during maintenance on the division 1 EDG;
- residual heat removal (RHR) train ‘B’ in shutdown cooling mode during maintenance on RHR train ‘A’; and
- division 1 shutdown service water (SX) system during emergent repairs to division 3 shutdown service water 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, Updated Safety Analysis Report (USAR), Technical Specification (TS) requirements, outstanding work orders (WOs), condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the CAP with the appropriate significance characterization.

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

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

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

- Fire Zone D–6(a)(b), Diesel Generator Building Division 2 Diesel Generator and Day Tank Room—Elevation 737’;
- Fire Zone M–2(a), Screen House Division 2 and 3 SX Pump Room and Tunnel—Elevations 657’ and 699’;
- Fire Zone D–10, Diesel Generator Building Ventilation Equipment Area and Unit 2 Rooms—Elevation 762’; and
- Fire Zone CB–3(f)(g), Control Building Auxiliary Electrical Equipment, Inverter and Battery Rooms—Elevation 781’.

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 four quarterly fire protection inspection samples as defined in IP 71111.05–05.

b. Findings

No findings were identified.

1R06 Flooding (71111.06)

.1 Underground Vaults

a. Inspection Scope

The inspectors selected underground bunkers/manholes subject to flooding that contained cables whose failure could disable risk-significant equipment. The inspectors determined that the cables were not submerged, that splices were intact, and that appropriate cable support structures were in place. In those areas where dewatering devices were used, such as a sump pump, the device was operable and level alarm circuits were set appropriately to ensure that the cables would not be submerged. In those areas without dewatering devices, the inspectors verified that drainage of the area was available, or that the cables were qualified for submergence conditions. The inspectors also reviewed the licensee's corrective action documents with respect to past submerged cable issues identified in the corrective action program to verify the adequacy of the corrective actions. The inspectors performed a walkdown of the following underground bunkers/manholes subject to flooding:

- OSHA 1a, 1b, and 1c cable vaults; and
- OSHB 1a, 1b, and 1c cable vaults.

This inspection constituted one underground vaults sample as defined in IP 71111.06--05.

b. Findings

No findings were identified.

1R07 Annual Heat Sink Performance (71111.07)

.1 Heat Sink Performance

a. Inspection Scope

The inspectors reviewed the licensee's inspections of the RHR 'B' pump room cooler to verify that potential deficiencies did not mask the licensee's ability to detect degraded performance, to identify any common cause issues that had the potential to increase

risk, and to ensure that the licensee was adequately addressing problems that could result in initiating events that would cause an increase in risk. The inspectors reviewed the licensee's observations as compared against acceptance criteria, the correlation of scheduled inspection and the frequency of inspection, and the impact of instrument inaccuracies on inspection results. Inspectors also verified that inspection acceptance criteria considered differences between inspections and previously identified issues.

This annual heat sink performance inspection constituted one sample as defined in IP 71111.07–05.

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08)

From May 15, 2017, through May 19, 2017, the inspectors conducted a review of the implementation of the licensee's inservice inspection (ISI) program for monitoring degradation of the reactor coolant system, emergency feedwater systems, risk-significant piping and components and containment systems.

The inspections described in Sections 1R08.1 and 1R08.5 below constituted one ISI sample as defined in IP 71111.08–05.

.1 Piping Systems Inservice Inspection

a. Inspection Scope

The inspectors observed the following non-destructive examinations (NDE) mandated by the American Society of Mechanical Engineers (ASME) Section XI Code to evaluate compliance with the ASME Code Section XI and Section V requirements and if any indications and defects were detected, to determine if these were dispositioned in accordance with the ASME Code or an U.S. Nuclear Regulatory Commission (NRC) approved alternative requirement:

- magnetic particle examination of RHR system heat exchanger shell to nozzle weld HEA–4;
- dye penetrant examination of reactor recirculation system welded attachment 1–RR–B–4PR–1–WA;
- ultrasonic examination (UT) of feedwater system pipe to elbow weld 1–FW–1–6–3;
- UT of RHR elbow to pipe weld 1–RH–14–23; and
- UT of reactor core isolation cooling valve to pipe weld 1–RI–13–8.

The inspectors observed the following NDE conducted as part of the licensee's Industry Initiative Inspection Program for inter-granular stress corrosion cracking to determine if the examination was conducted in accordance with the licensee's Augmented Inspection Program and associated licensee examination procedures. If any indications and defects were detected, the inspectors determined if these indications and defects were dispositioned in accordance with approved procedures and NRC requirements:

- phased array UT of reactor water cleanup system pipe to pipe dissimilar metal weld 1–RT–36–1C;
- UT of RHR pipe to elbow weld 1–RH–20–4–2;
- UT of RHR elbow to pipe weld 1–RH–20–7; and
- UT of RHR reducing tee to pipe weld 1–RH–20–7.

The licensee had not identified any recordable indications during surface and volumetric examinations performed since the beginning of the previous refueling outage. Therefore, no NRC review was completed for this inspection procedure attribute.

The inspectors reviewed records for the following pressure boundary welds completed for risk-significant systems during the outage to determine if the licensee applied the pre-service NDE and acceptance criteria required by the construction Code. Additionally, the inspectors reviewed the welding procedure specification and supporting weld procedure qualification records to determine if the weld procedure were qualified in accordance with the requirements of the Construction Code and the ASME Code Section IX:

- code class 1, main steam line flexible hose replacements (WO 04613137–01); and
- code class 2, reactor recirculation system valve leak-off line connection (WO 1508953–01).

b. Findings

No findings were identified.

.2 Reactor Pressure Vessel Upper Head Penetration Inspection Activities—Not Applicable

.3 Boric Acid Corrosion Control—Not Applicable

.4 Steam Generator Tube Inspection Activities—Not Applicable

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI related problems entered into the licensee's CAP and conducted interviews with licensee staff to determine if:

- the licensee had established an appropriate threshold for identifying ISI-related problems;
- the licensee had performed a root cause (if applicable) and taken appropriate corrective actions; and
- the licensee had evaluated operating experience and industry generic issues related to ISI and pressure boundary integrity.

The inspectors performed these reviews to evaluate compliance with Title 10 of the *Code of Federal Regulations* (CFR), Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the Attachment to this report.

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 June 7, 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 May 28, 2017, and June 2, 2017, the inspectors observed the control room operations staff perform reactor plant startups. This was an activity that required heightened awareness or was related to increased risk. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms (if applicable);
- correct use and implementation of procedures;
- control board (or equipment) manipulations; 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 ventilation system charcoal beds.

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 samples as defined in IP 71111.12–05.

b. Findings

No findings were identified.

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

.1 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

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

- yellow due to planned maintenance on standby gas treatment train 'A';
- yellow due to emergent work on the division 2 shutdown service water pump;
- yellow for adverse weather—severe thunderstorm warning; and
- green due to emergent work to the control room cooling coil flow control valve 1SX019B.

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 four samples as defined in IP 71111.13–05.

b. Findings

No findings were identified.

1R15 Operability Determinations and Functional Assessments (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- Action Request (AR) 03980202, OP–CL–102–106–1001 Times Impacted;
- AR 04002714, NRC Questions on Inverter Room Cooling;
- AR 03975643, FME Found in NSPS Division 2 Cabinet Bay C;
- AR 03999982, Instrument Tubing on VX Condensing Unit Touching Angle Iron;
- AR 04018770 Pinhole Leak at Pipe Weld for 1sx019B; and
- AR 03974026, Potential Water Intrusion in Charcoal Bed.

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 the 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.

These operability inspections constituted six samples as defined in IP 71111.15–05.

b. Findings

Failure of Operators to Meet Time Critical Operator Actions

Introduction: The inspectors identified a finding of very low safety significance and an associated non-cited violation (NCV) of Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to assure that applicable regulatory requirements and the design basis were correctly translated into specifications, drawings, procedures, and instructions and that design control measures provided for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Specifically, the licensee failed to assure/validate operators were able to complete the standby liquid control time critical action for an anticipated transient without a scram (ATWS) specified in (design analysis EPU–T0902) licensing documents.

Description: On November 15, 2016, the NRC component design basis inspection team held a discussion with operations training personnel regarding the validation of time critical operator actions (TCAs). Specifically, the team was concerned that some of the validations were conducted with greater than the minimum main control room staffing. This issue was documented in the licensee's CAP as AR 02741339. The AR stated that TCAs 4, 5, 9, and 10 were validated with greater than minimum control room staffing and recommended evaluating the TCAs against the current procedural guidance in station procedure OP-AA-102-106, "Operator Response Time Program," Revision 3, which contained guidance for more than minimum control room staffing.

On February 25, 2017, the licensee was validating TCA 10. This TCA requires injection of the standby liquid control system (SBLC) within 120 seconds of a main steam line isolation valve closure during an ATWS. Based upon the scram choreography prior to taking actions to initiate SBLC, the time to complete the TCA was 172 seconds. The licensee documented the issue as AR 03980202. Corrective actions taken by the licensee included validating that SBLC could be initiated in less than 120 seconds by changing when the TCA action should be performed during the scram choreography, issuing a daily order to the operation crews to inform crews of when to perform TCA 10, and changing the necessary station procedures.

Additionally, the licensee performed an operability evaluation that included review of design analysis EPU–T0902, "Extended Power Uprate Task Report Anticipated

Transients Without Scram," Revision 0 to evaluate the margin available to safety limits with an extended SBLC injection time of 172 seconds. The licensee concluded there was still sufficient margin available before reaching the safety limits.

Analysis: The inspectors determined that failing to assure that applicable regulatory requirements and the design basis be correctly translated into specifications, drawings, procedures, and instructions and that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program was a performance deficiency.

The performance deficiency was determined to be more than minor in accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated September 7, 2012, because the finding was associated with the procedure quality attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, with the operators initiating SBLC at 172 seconds instead of 120 seconds, the accident analysis calculations were required to be re-performed to assure the accident analysis requirements were met.

Using IMC 0609, Attachment 4, "Initial Characterization of Findings," and Appendix A, "The Significance Determination Process (SDP) for Findings at Power," Exhibit 2, June 19, 2012, the finding was screened against the Mitigating Systems cornerstone and determined to be of very low safety significance (Green) because the inspectors were able to answer all of the questions "No."

The inspectors determined that this finding is not indicative of current performance and therefore did not assign a cross-cutting aspect.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, measures shall be established to assure that applicable regulatory requirements and the design basis be correctly translated into specifications, drawings, procedures, and instructions and that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.

Contrary to the above, on February 25, 2017, the licensee did not establish measures to assure that applicable regulatory requirements and the design basis was correctly translated into specifications, drawings, procedures, and instructions and that design control measures were provided for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Specifically, the licensee failed to assure operators were able to complete the SBLC time critical action for an ATWS specified in (design analysis EPU–T0902) licensing documents.

As corrective actions, the licensee determined the scram choreography required to complete the time critical action in the specified time, initiated a standing order to inform the operating crews, processed a procedure change for the ATWS scram choreography and performed an evaluation to determine the impact of initiating the SBLC system at

172 seconds. Because this finding was of very low safety significance and was entered into the CAP as AR 03980202, this violation is being treated as a NCV, in accordance with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000461/2017002–01: Failure of Operators to Meet Time Critical Operator Actions)**

1R18 Plant Modifications (71111.18)

.1 Plant Modifications

a. Inspection Scope

The inspectors reviewed the following modifications:

- Engineering Change (EC) 618150, Main Steam Line Flexible Hose Material Upgrade; and
- EC 405113, Replace Fuel Building Crane Rail Clips, Engineering Change Annunciator Replacement Project.

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

Specifically for the replacement fuel building crane rail clips, the inspectors reviewed the structural details and calculations associated with the crane rail clip modification to verify resolution of the performance deficiency identified in the NRC Inspection Report 05000461/2016010(DNMS)/0721046/2016001(DNMS).

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

b. Findings

Failure to Perform Adequate Evaluation of Crane Rail Clips

Introduction: A finding of very-low safety significance (Green) and an associated cited violation of Title 10 of the *Code of Federal Regulations* (CFR), Part 50, Appendix B, Criterion III, “Design Control,” was identified by the inspectors for the failure to properly verify the adequacy of design of the fuel building crane and crane support structure elements. Specifically, calculations involving the crane rail clips and clip bolts had multiple technical errors and failed to adequately demonstrate that the design met the design basis requirements.

Description: A finding of very-low safety significance (Green) and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was documented in NRC Inspection Report 05000461/2016010(DNMS)/07201046/2016001(DNMS). The finding included multiple examples. Examples b, c, and d in the report identified deficiencies in the licensee's evaluation of the crane rail clip connections. Some of the deficiencies included not accounting for the prying force, use of non-conservative methods without adequate justification for distribution of lateral loads to the rail clips, and incorrectly crediting bolt pre-load and friction forces in the evaluation. As part of corrective actions, the licensee performed field modifications involving use of stronger rail clips and higher strength bolts and also revised the design calculation. Inspectors noted that the American Institute for Steel Construction (AISC) specification (7th or 8th Edition) was the design basis for structural steel.

Upon review of the revised evaluations in Calculation No. SDQ15–24DG09, Revision 12, and the subsequent revisions/responses, the inspectors identified numerous technical concerns as described below.

Revision 12:

The inspectors noted that the revised Calculation No. SDQ15–24DG09, Revision 12, used the same non-conservative load distribution as before for reducing the load on the clips being evaluated. The calculation did not provide adequate justification for the engineering judgment that one pair of clips would resist only 50 percent of the wheel load when the wheel was centered on the clips. Inspectors also noted that such distribution was not consistent with the technical information provided by the crane rail vendor GANTRAIL indicating insignificant horizontal load distribution to the adjacent pair of clips. Additionally, the calculation used incorrect allowable load for tension in bolts. The licensee used the bolt installation pre-load of 49 kips as the allowable tension instead of the AISC specified value of 32.5 kips.

The inspectors' questions were followed by multiple responses and calculation revisions where the licensee used various finite element programs (STAAD PRO and ANSYS) and models. Inspectors' review of these responses identified additional concerns described below.

Response dated July 11, 2016:

The licensee provided results of a finite element analyses using STAAD computer code to justify using 62.5 percent of the wheel load for evaluation of one pair of clips. The applied load for this analysis was distributed over a length of 44 inches using a provision of the AISC Specification, Section 1.10.5.1. The inspectors noted that the provision was applicable to bearing stiffeners in plate girders which are designed as columns, an application that is non-conservative and completely different from the current one.

Response dated July 30, 2016:

The licensee performed an analysis using a more refined STAAD model. Instead of a rigid support at the bolt locations, the analysis assumed that 1/16 inch movement would be permitted due to the slightly larger bolt holes and correspondingly modeled spring support at specific locations. This analysis demonstrated that the load on one pair of clips to be 49 percent of the wheel load. The inspectors noted that the assumption

regarding the gaps providing flexibility at specific desired locations only was not consistent with the installation drawings and therefore was not justified.

Revision 13:

The revised analysis used a finite element model of a crane rail length including seven pairs of clips using the ANSYS computer code. Bolt shears were calculated by the program while the tensions were hand calculated from the analysis output. Results of this analysis indicated that for the governing Operating Basis Earthquake (OBE) load case the critical pair of clips will carry about 84 percent of the load or approximately 59 kips. However, the inspectors noted that the indicated shear on the bolts was less than 3 kips. The calculation did not explain how the remaining force was accounted for. Consequently, the calculation indicated a large margin for shear on the bolts while if correctly determined, the acceptance criteria would not have been met. In the hand calculation of bolt tensions, incorrect lever arm (distance between the up and down reaction points) was used resulting in under prediction of the bolt tensions. Additionally, for determining effect of prying forces, a formula from the AISC manual was used that did not apply to the subject configuration. This resulted in significant under prediction of the bolt tension loads.

Revision 14:

The licensee further refined the ANSYS model from Revision 13 to include bolt tension calculations also in the computer analysis. The inspectors noted that the design margins indicated by the analysis were unexpectedly high and inconsistent with the previous revisions as well as engineering judgment. The licensee used a finite element analysis using ANSYS software. The analytical model included a rail length including seven pairs of clips and lateral loads were applied at two locations, W1 and W2, to represent the worst case loading. The inspectors identified discrepancies in the tabulation of output for the OBE load case, raising concerns regarding adequacy of the model and consequently results for all loading cases. For the pair of clips under wheel 1 (W1), the total load transmitted was 47 kips. However, the analysis indicated that shear reaction in the bolts was only about 8 kips instead of the expected 47 kips. Similarly, the analysis indicated negligible bolt tensions while, based on the applied 47 kips load, a value of more than 50 kips was to be expected. A closer review revealed that the analysis model incorrectly reduced the bolt shear and tension values by taking credit for the clamping force and the friction developed by the clamping force. This was contrary to the standard engineering practice as well as the AISC specification which is the Clinton Power Station design basis.

Response dated January 5, 2017:

The licensee performed a hand calculation using conventional methods to address the concerns identified in the previous revisions. As expected, this analysis resulted in significantly different results. For the most critical OBE condition identified, the calculation showed that the bolt tension would exceed the allowable by about 30 percent. Since the applied wheel loads was based on an envelope of hundreds of load combinations from the crane analysis, the licensee refined the calculation by reviewing individual load combinations in order to remove unnecessary conservatism. Additionally, the licensee decided to use the 100–40–40 method for combining the horizontal and vertical seismic components in accordance with the Regulatory Guide 1.92, which allows use of Square Root of Sum of Squares or 100–40–40 method for combining spatial components of the

earthquake in lieu of an absolute sum method. The inspectors identified the following concerns:

- Sixty-seven percent of the wheel load was applied to the rail based on the results of Revision 14 discussed above. Considering the concerns with the adequacy of Revision 14 as discussed above, the assumption that 67 percent is a reliable value was not adequately justified.
- The licensee considered OBE load combinations stating that they were more critical. While this may have been the case when enveloping loads were used, the calculation did not provide adequate justification to show that it would also be true when considering numerous individual load conditions.
- In order to apply the 100–40–40 method for combining effects of horizontal and vertical seismic motions, the licensee applied the 40 percent factor to all vertical seismic wheel load. In order to correctly apply the method, the factor would need to be applied to only the portion that is caused by the vertical seismic earthquake motion. A part of the vertical reaction may be caused by the horizontal seismic motion to which no reduction would be applied. The licensee calculation did not provide a breakdown of the vertical load to demonstrate that the method was applied correctly.
- Application of the 100–40–40 method was not permitted in this case because the crane analysis from which the wheel loads were obtained had already used the Square Root of Sum of Squares method for spatial combination of earthquake components.

The licensee initiated corrective actions by documenting the deficiency in AR 04001089 and performed an evaluation demonstrating that the functionality of the crane was maintained.

Analysis: The inspectors determined that the failure to determine accurate loading and performing adequate calculations to demonstrate compliance with the American Institute of Steel Construction (AISC) specification which is the design basis, was a performance deficiency.

The finding was determined to be more-than-minor because it was associated with design control attribute of the Barrier Integrity cornerstone and adversely affected the cornerstone objective of maintaining the functionality of the spent fuel pool (SFP) cooling system. Specifically, crane rail clip bolts were required to ensure structural integrity of structures, systems, and components described in the USAR, when subjected to design loads as part of safe load handling of heavy loads near the SFP, and to ensure integrity of the spent fuel cask. In accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Initial Characterization of Findings,” Table 2, the inspectors determined the finding affected the Barrier Integrity cornerstone because it was associated with SFP/fuel handling activities. Based on answering “No” to questions A through F in Table 3, the inspectors determined the finding could be evaluated using Appendix A, “The Significance Determination Process for Findings At-Power,” Exhibit 3, for the Barrier Integrity cornerstone screening questions. Based on the crane remaining functional, the inspectors answered “No” to Questions D.1 through D.4 because the finding did not adversely affect decay heat removal capabilities, did not result from fuel handling errors, did not result in loss of SFP inventory, and did not affect the SFP neutron absorber or fuel bundle misplacement, and therefore the finding screened as having very-low safety significance (Green).

The inspectors identified a problem identification and resolution, resolution [P.3] cross-cutting aspect associated with this finding. Specifically, the licensee failed to take effective corrective actions in a timely manner to address issues identified earlier in the rail clip evaluations.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that the design control measures provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculation methods, or by the performance of a suitable testing program. The fuel building crane and the supporting fuel building steel structure are Seismic Category I components/structures subject to the 10 CFR Part 50, Appendix B, quality assurance requirements, and the applicable structural steel design specification is the AISC, 7th or 8th Edition.

Contrary to the above,

- On May 27, 2016, in Calculation No. SDQ15–24–DG09, Revision 12, and in the associated responses dated July 11, 2016, and July 30, 2016, the licensee failed to verify or check adequacy of the design for the fuel building crane and crane support structure elements. Specifically, the licensee verification process failed to identify that the load distribution used in the calculations was not justified, and that the use of the plate girder bearing stiffener provisions of the AISC specification, Section 1.10.5.1 and the assumption of flexible supports at specific bolted connections in determination of wheel load distribution were not consistent with the field configuration based on design drawings.
- On August 11, 2016, in Calculation No. SDQ15–24–DG09, Revision 13, the licensee failed to verify or check adequacy of the design for the fuel building crane and crane support structure elements. Specifically, the licensee verification process failed to identify that the unexpectedly high design margins for the bolt shear as indicated by the analysis were not justified because the evaluation incorrectly credited friction forces for reduction in the bolt stresses. Additionally, the bolt tensions calculated in the analysis were incorrect because the lever arm between the force couple used for determining bolt tension was incorrect and the AISC method used for determination of prying force was not applicable to the configuration being evaluated.
- On October 11, 2016, in Calculation No. SDQ15–24–DG09, Revision 14, and in the associated response dated January 5, 2017, the licensee failed to verify or check adequacy of the design for the fuel building crane and crane support structure elements. Specifically, the licensee verification process failed to identify that the unexpectedly high design margins indicated by the analysis in Revision 14 of the calculation were not justified because the evaluation incorrectly credited frictional and bolt pre-tension forces in reduction of bolt shear and tension stresses. In the response dated January 5, 2017, the licensee did not correctly combine the vertical and horizontal seismic responses while applying the 100–40–40 percent rule for combining effects of spatial components of earthquake as described in Regulatory Guide 1.92.

The licensee entered the condition in its CAP to initiate actions to restore compliance (AR 4001089). The licensee also made a determination that the non-compliance did not affect the crane functionality and therefore did not present any immediate safety concerns.

The inspectors determined that after identification of the initial violation (NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," documented in the NRC Inspection Report 05000461/2016010 (DNMS)/0721046/2016001(DNMS)), the licensee's completed corrective actions failed to restore compliance with the design basis. As a result, the condition for considering it as an NCV per Section 2.3.2(a)(2) of the NRC's Enforcement Policy was not met. Therefore, the violation is being cited in the attached Notice of Violation. **(VIO 05000461/2017002–02: Failure to Perform Adequate Evaluation of Crane Rail Clips)**

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 the control room ventilation train A;
- testing of the main steam line isolation valves;
- testing of the division 3 shutdown service water pump;
- testing of the main steam line instrumentation flex hoses;
- testing of the reactor coolant system safety relief valves;
- testing of the control rods;
- testing of reactor core isolation cooling reactor pressure vessel isolation check valve 1E51F066;
- testing of the reactor core isolation cooling pump;
- testing of the standby liquid control train A;
- testing of the control room cooling coil flow control valve 1SX019B;
- testing of the division 1 switchgear heat removal condensing unit; and
- testing of the division 2 inverter room cooling coil 1VX13AB.

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

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

b. Findings

No findings were identified.

1R20 Outage Activities (71111.20)

.1 Refueling Outage Activities

a. Inspection Scope

The inspectors reviewed the Shutdown Safety Management Program and contingency plans for RFO C1R17, conducted May 8, 2017, through May 30, 2017, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the RFO, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below:

- licensee configuration management, including maintenance of defense-in-depth commensurate with the outage safety plan (OSP) for key safety functions and compliance with the applicable TS when taking equipment out of service;
- implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing;
- controls over the status and configuration of electrical systems to ensure that TS and OSP requirements were met, and controls over switchyard activities;
- monitoring of decay heat removal processes, systems, and components;
- controls to ensure that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system;
- reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss;
- controls over activities that could affect reactivity;
- maintenance of secondary containment as required by TS;
- licensee fatigue management, as required by 10 CFR 26, Subpart I;
- refueling activities, including fuel handling and sipping to detect fuel assembly leakage;
- startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the drywell (primary containment) to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing; and
- licensee identification and resolution of problems related to RFO activities.

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

b. Findings

Unexpected Start of the Division 3 Emergency Diesel Generator

Introduction: The inspectors documented a self-revealed finding of very low safety significance and an associated NCV of Title 10 of CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to follow steps in WO 04640788 while performing troubleshooting on blown power transformer (PT) fuses in the division 3 emergency diesel start circuitry. Specifically, the electricians opened test switches in the wrong electrical cubicle resulting in the unexpected start of the division 3 EDG and a loss of power to the 1C1 bus from an offsite source.

Description: On May 18, 2017, the 1C1 safety bus was being powered from offsite power via the emergency reserve auxiliary transformer (ERAT). Electrical maintenance technicians were investigating blown PT fuses associated with the degraded voltage and loss of voltage relay circuitry between the ERAT and the 1C1 safety bus. They were performing troubleshooting in accordance with WO 04640788 to determine the cause of the failed fuses. The technicians performed the first step of the WO, a visual inspection for signs of damage, at cubicle 1E22S004–102. The second step of the procedure instructed the technicians to open the test switches on cubicle 1E22S004–101, a different cubicle from the one they had just performed the inspection for damage on, but on the same safety bus.

The technicians opened test switches in the 1E22S004–102 cubical, the panel they had just completed the inspection on, and this removed power from the degraded voltage and loss of voltage relay circuitry for the 1C1 safety bus. The removal of power to the relay circuitry resulted in the circuitry performing its intended function which is to start the division 3 EDG. Additionally, the removal of power to the degraded voltage relays started a 15 second timer that caused the breaker feeding the 1C1 safety bus from the ERAT to open leaving the bus de-energized. The division 3 EDG output breaker then closed to restore power to the 1C1 bus. The control room operators secured the division 3 diesel generator because they had no indications in the control room of voltage on the 1C1 bus because the test switches that were opened removed indication to the control room. The 1C1 bus remained de-energized for approximately 2 hours until the bus was recovered by aligning it to an offsite source.

The licensee documented this issue as AR 04012393. The immediate corrective actions included securing the electricians work, performing a human performance review, and restoring power to the 1C1 safety bus. The human performance review (HURB) determined that the correct use of flagging and a properly performed concurrent verification of the component to be manipulated would have prevented the event.

Analysis: The inspectors determined that the electricians failing to follow the steps in WO 04640788 while performing troubleshooting on blown PT fuses was contrary to the requirements of 10 CFR Part 50, Appendix B, Criterion V and was a performance deficiency. The performance deficiency was determined to be more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated September 7, 2012, because it impacted the Initiation Events cornerstone attribute of human performance and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the failure of

the electrical maintenance technicians to follow their procedures resulted in a loss of power to the 1C1 electrical bus.

Using IMC 0609, Attachment 4, “Initial Characterization of Findings,” issued October 7, 2016, and Appendix G, “Shutdown Operations Significance Determination Process,” issued May 19, 2014, the finding was screened against the Initiating Events cornerstone and determined to be of very low safety significance (Green) because the loss of power to the 1C1 bus occurred while Clinton Power Station was in an RFO when the high pressure core spray system (the division 3 system) was removed from service and not being relied upon for shutdown safety defense in depth. The loss of the 1C1 bus did not affect decay heat removal from the core, did not affect reactor coolant inventory, and the event occurred while the refuel cavity was flooded up for refueling operations.

The inspectors determined that this finding affected the cross-cutting area of human performance in the aspect of avoid complacency where individuals implement appropriate error reduction tools. Specifically, as documented in the licensee’s HURB, the electricians performing the work did not utilize any human performance tools to flag the equipment to be operated and improperly performed the concurrent verification of the component to be manipulated. [H.12]

Enforcement: 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 WO 04640788, “Inspect 4160 Bus 1C1 Main Feed Breaker PT Drawer” as the implementing procedure for performing troubleshooting on the blown PT fuses, an activity affecting quality.

Step 2 of WO 04640788 instructed the technicians to open the test switches on cubicle 1E22S004–101.

Contrary to the above, on May 18, 2017, the technicians failed to accomplish troubleshooting of blown PT fuses associated with the degraded voltage and loss of voltage relay circuitry between the ERAT and the 1C1 safety bus in accordance with WO 04640788, Step 2. Specifically, the technicians opened the test switches on cubicle 1E22S004–102 rather than the test switches on cubicle 1E22S004–101 resulting in the unexpected start of the division 3 EDG and a loss of power to the 1C1 bus from an offsite source. As corrective actions, the licensee performed a human performance review to identify the reasons the procedure was not followed and restored power to the 1C1 safety bus. Because this finding was of very low safety significance and was entered into the CAP as AR 04012393, this violation is being treated as an NCV, in accordance with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000461/2017002–03: Unexpected Start of the Division 3 Emergency Diesel Generator)**

.2 Forced Outage C1F59

a. Inspection Scope

The inspectors evaluated outage activities for a forced outage that began on May 30, 2017, and continued through June 2, 2017, due to an automatic reactor scram received from the oscillating power range monitors. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule.

The inspectors observed or reviewed the reactor shutdown and cooldown, outage equipment configuration and risk management, electrical lineups, selected clearances, control and monitoring of decay heat removal, control of containment activities, personnel fatigue management, startup and heat up activities, and identification and resolution of the automatic reactor scram received from the oscillating power range monitors.

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

b. Findings

No findings were identified.

.3 Forced Outage C1F60

a. Inspection Scope

The inspectors evaluated outage activities for a forced outage that began on June 10, 2017, and continued through June 11, 2017, due to a manual reactor scram due to a loss of feed water heating. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule.

The inspectors observed or reviewed the reactor shutdown and cooldown, outage equipment configuration and risk management, electrical lineups, selected clearances, control and monitoring of decay heat removal, control of containment activities, personnel fatigue management, startup and heat up activities, and identification and resolution of the manual reactor scram due to a loss of feed water heating.

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

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety

function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- Clinton Power Station (CPS) 9052.01, “Low Pressure Core Spray(LPCS)/Residual Heat Removal (RHR) ‘A’ Pump and LPCS/RHR ‘A’ Water Leg Pump Operability” Section 8.2 (routine test);
- CPS 9052.01, “Low Pressure Core Spray(LPCS)/Residual Heat Removal (RHR) ‘A’ Pump and LPCS/RHR ‘A’ Water Leg Pump Operability,” Section 8.3 (routine test);
- CPS 9861.04, “Main Steam Isolation Valve local Leak Rate Test” (isolation valve);
- CPS 9866.02, “VG/VC Charcoal Absorber Leak Test” (routine test);
- CPS 9069.01, “Shutdown Service Water Operability Test” (inservice test);
- CPS 9333.30, “Division II 4.16kV Degraded Voltage Trip—Functional Test” (routine test);
- CPS 9054.01, “Reactor Containment Isolation Cooling System Operability Check” (inservice test); and
- CPS 9843.01V006, “Leak Rate Testing of RHR Shutdown Suction” (isolation valve).

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

- did preconditioning occur;
- the effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis;
- plant equipment calibration was correct, accurate, and properly documented;
- as-left set points were within required ranges; and the calibration frequency was in accordance with TSs, the USAR, procedures, and applicable commitments;
- measuring and test equipment calibration was current;
- test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied;
- test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used;
- test data and results were accurate, complete, within limits, and valid;
- test equipment was removed after testing;
- where applicable for inservice testing activities, testing was performed in accordance with the applicable version of Section XI, American Society of Mechanical Engineers code, and reference values were consistent with the system design basis;
- where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure;

- where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished;
- prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;
- equipment was returned to a position or status required to support the performance of its safety functions; and
- all problems identified during the testing were appropriately documented and dispositioned in the CAP.

This inspection constituted four routine surveillance testing samples, two in-service test samples and two containment isolation valve samples as defined in IP 71111.22, Sections–02 and–05.

b. Findings

Failure to Provide Sufficient Work Instructions for Performing Maintenance on the Control Room Ventilation System Charcoal Filter

Introduction: The inspectors documented a self-revealed finding of very low safety significance and an associated NCV of Title 10 of CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to provide sufficient work instructions for performing maintenance on the control room ventilation charcoal filter bed. Specifically, WO 01494189 to change out the charcoal in the filter bed contained only the minimum required amount of charcoal to place in the bed. Sometime after filling the bed April 6, 2015, the charcoal settled, resulting in the 'B' control room ventilation system failing a surveillance test requiring the ventilation system to be declared inoperable.

Description: On April 7, 2017, the 0VC07SB charcoal filter bed of the 'B' control room ventilation system failed a bypass leakage test required by TS surveillance requirement 3.7.3.3. A subsequent inspection performed by the licensee identified that the charcoal level in the filter bed was below the minimum fill line which allowed bypass flow around the charcoal filter above the TS limit. The licensee entered the issue into the CAP as AR 03995612 and performed an equipment CAP evaluation Equipment Corrective Action Program Evaluation (ECAPE).

The inspectors reviewed the licensee's ECAPE and found that the guidance contained in WO 01494189 and used in April 2015 instructed the maintenance staff to add charcoal to a depth of 1 inch over the bed screen. The work instructions used to change out the charcoal media had identified the minimum level of charcoal to be added to the filter bed, however, this amount of charcoal did not provide margin to account for potential settling below the minimum level. The lower level of charcoal after settling allowed excess air flow to bypass the filter resulting in a failed surveillance test.

As immediate corrective actions, the licensee added charcoal to the 0VC07SB charcoal filter bunker under WO 01656078 and performed a satisfactory surveillance test.

Analysis: The inspectors determined that the licensee's failure to provide sufficient instructions for performing maintenance on the control room ventilation system charcoal filter bed was a performance deficiency. The performance deficiency was determined to be more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports,"

Appendix B, “Issue Screening,” dated September 7, 2012, because it impacted the Barrier Integrity cornerstone attribute of procedure quality and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to provide sufficient guidance in the WO used to change out the charcoal bed resulted in the ‘B’ control room ventilation system failing a surveillance test requiring the ‘B’ control room ventilation system be declared inoperable. Using IMC 0609, Attachment 4, “Initial Characterization of Findings,” and Appendix A, “The Significance Determination Process for Findings at Power,” Exhibit 2, October 7, 2016, the finding was screened against the Barrier Integrity cornerstone and determined to be of very low safety significance because the finding only represents a degradation of a radiological barrier function provided for the control room.

The inspectors determined that this finding affected the cross-cutting are of human performance in the aspect of design margins, where the organization operates and maintains equipment within design margins. Special attention is placed on maintaining fission product barriers, defense in depth, and safety-related equipment. Specifically, when performing maintenance on the charcoal bed, the licensee failed to recognize that filling the charcoal to the minimum bed level provided no margin if settling occurred to maintain the ‘B’ control room ventilation system operable. [H.6]

Enforcement: 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 WO 01494189, “0VC07SB Charcoal Change Out,” as the implementing procedure for replacing charcoal in the ‘B’ control room ventilation system filter bed, an activity affecting quality.

Contrary to the above, on April 7, 2017, the licensee failed to have instructions within WO 01494189 for changing out the charcoal in the ‘B’ control room ventilation system filter bed which were appropriate to the circumstances. Specifically, WO 01494189 contained only the minimum required amount of charcoal to place in the bed. Due to only installing the minimum amount of charcoal, the charcoal settled after installation resulting in the ‘B’ control room ventilation system failing a surveillance test and becoming inoperable.

As corrective actions, the licensee is revising the WO instructions and an associated procedure to require that charcoal be filled completely to the bottom of the deluge piping to provide for settling. Because the violation was of very low safety significance and was entered into the licensee's CAP as AR 03995612, this violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy.
(NCV 05000461/2017002–04: Failure to Provide Sufficient Work Instructions for Performing Maintenance on the Control Room Ventilation System Charcoal Filter)

2. RADIATION SAFETY

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

.1 Radiological Hazard Assessment (02.02)

a. Inspection Scope

The inspectors assessed the licensee's current and historic isotopic mix, including alpha emitters and other hard-to-detect radionuclides. The inspectors evaluated whether survey protocols were reasonable to identify the magnitude and extent of the radiological hazards.

The inspectors determined if there have been changes to plant operations since the last inspection that may have resulted in a significant new radiological hazard for onsite individuals. The inspectors evaluated whether the licensee assessed the potential impact of these changes and implemented periodic monitoring, as appropriate, to detect and quantify the radiological hazard. The inspectors reviewed the last two radiological surveys from selected plant areas and evaluated whether the thoroughness and frequency of the surveys were appropriate for the given radiological hazard.

The inspectors conducted walk-downs of the facility, including radioactive waste processing, storage, and handling areas to evaluate material conditions and performed independent radiation measurements as needed to verify conditions were consistent with documented radiation surveys.

The inspectors assessed the adequacy of pre-work surveys for select radiologically risk-significant work activities.

The inspectors evaluated the radiological survey program to determine if hazards were properly identified. The inspectors discussed procedures, equipment, and performance of surveys with radiation protection staff and assessed whether technicians were knowledgeable about when and how to survey areas for various types of radiological hazards.

The inspectors observed work in potential airborne areas to assess whether air samples were being taken appropriately for their intended purpose and reviewed various survey records to assess whether the samples were collected and analyzed appropriately. The inspectors also reviewed the licensee's program for monitoring contamination, which has the potential to become airborne.

These inspection activities constituted one complete sample as defined in IP 71124.01–05.

b. Findings

No findings were identified.

.2 Instructions to Workers (02.03)

a. Inspection Scope

The inspectors reviewed select radiation work permits used to access high radiation areas and evaluated the specified work control instructions or control barriers. The inspectors also assessed whether workers were made aware of the work instructions and area dose rates.

The inspectors reviewed electronic alarming dosimeter dose and dose rate alarm set-point methodology. For selected electronic alarming dosimeter occurrences, the inspectors assessed the worker's response to the alarm, the licensee's evaluation of the alarm, and any follow-up investigations.

The inspectors reviewed the licensee's methods for informing workers of changes in plant operations or radiological conditions that could significantly impact their occupational dose.

The inspectors reviewed the labeling of select containers of licensed radioactive material that could cause unplanned or inadvertent exposure to workers.

These inspection activities constituted one complete sample as defined in IP 71124.0–05.

b. Findings

No findings were identified.

.3 Contamination and Radioactive Material Control (02.04)

a. Inspection Scope

The inspectors observed locations where the licensee monitors material leaving the radiologically controlled area and assessed the methods used for control, survey, and release of material from these areas. As available, the inspectors observed health physics personnel surveying and releasing material for unrestricted use.

The inspectors observed workers leaving the radiologically controlled area and assessed their use of tool and personal contamination monitors and reviewed the licensee's criterial for use of the monitors.

The inspectors assessed whether instrumentation was used at its typical sensitivity levels based on appropriate counting parameters or whether the licensee had established a de facto release limit.

The inspectors selected several sealed sources from the licensee's inventory records and assessed whether the sources were accounted for and verified to be intact. The inspectors also evaluated whether any transactions, since the last inspection, involving nationally tracked sources were reported in accordance with Title 10 of the *Code of Federal Regulations*, Part 20.2207.

These inspection activities constituted one complete sample as defined in IP 71124.01–05.

b. Findings

No findings were identified.

.4 Radiological Hazards Control and Work Coverage (02.05)

a. Inspection Scope

The inspectors evaluated ambient radiological conditions during tours of the facility. The inspectors assessed whether the conditions were consistent with applicable posted surveys, radiation work permits, and worker briefings.

The inspectors evaluated the adequacy of radiological controls, such as required surveys, radiation protection job coverage, and contamination controls. The inspectors evaluated the licensee's use of electronic alarming dosimeters in high noise areas as high radiation area monitoring devices.

The inspectors assessed whether radiation monitoring devices were placed on the individual's body consistent with licensee procedures. The inspectors assessed whether the dosimeter was placed in the location of highest expected dose or that the licensee properly employed an NRC-approved method of determining effective dose equivalent.

The inspectors reviewed the application of dosimetry to effectively monitor exposure to personnel in work areas with significant dose rate gradients.

For select airborne area radiation work permits, the inspectors reviewed airborne radioactivity controls and monitoring, the potential for significant airborne levels, containment barrier integrity, and temporary filtered ventilation system operation.

The inspectors examined the licensee's physical and programmatic controls for highly activated or contaminated materials stored within pools and assessed whether appropriate controls were in place to preclude inadvertent removal of these materials from the pool.

These inspection activities constituted one complete sample as defined in IP 71124.01–05.

b. Findings

No findings were identified.

.5 Radiation Worker Performance and Radiation Protection Technician Proficiency (02.07)

a. Inspection Scope

The inspectors observed radiation worker performance and assessed their performance with respect to radiation protection work requirements, the level of radiological hazards present, and radiation work permit controls.

The inspectors assessed worker awareness of electronic alarming dosimeter set points, stay times, or permissible dose for radiologically significant work as well as expected response to alarms.

The inspectors observed radiation protection technician performance and assessed whether the technicians were aware of the radiological conditions and radiation work permit controls and whether their performance was consistent with training and qualifications for the given radiological hazards.

The inspectors observed radiation protection technician performance of radiation surveys and assessed the appropriateness of the instruments being used, including calibration and source checks.

These inspection activities constituted one complete sample as defined in IP 71124.01–05.

b. Findings

No findings were identified.

.6 Problem Identification and Resolution (02.08)

a. Inspection Scope

The inspectors assessed whether problems associated with radiological hazard assessment and exposure controls were being identified at an appropriate threshold and were properly addressed for resolution. For select problems, the inspectors assessed the appropriateness of the corrective actions. The inspectors also assessed the licensee's program for reviewing and incorporating operating experience.

The inspectors reviewed select problems related to human performance errors and assessed whether there was a similar cause and whether corrective actions taken resolve the problems.

The inspectors reviewed select problems related to radiation protection technician error and assessed whether there was a similar cause and whether corrective actions taken resolve the problems.

These inspection activities constituted one complete sample as defined in IP 71124.01–05.

b. Findings

No findings were identified.

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

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

a. Inspection Scope

The inspectors reviewed the assumptions and bases in as-low-as-reasonably-achievable (ALARA) work planning documents for selected activities and verified that the licensee has established measures to track, trend, and if necessary to reduce, occupational doses for ongoing work activities.

The inspectors determined whether a dose threshold criteria was established to prompt additional reviews and/or additional ALARA planning and controls and evaluated the licensee's method of adjusting exposure estimates, or re-planning work, when unexpected changes in scope or emergent work were encountered. The inspectors determined if adjustments to exposure estimates were based on sound radiation protection and ALARA principles or if they are just adjusted to account for failures to control the work. The inspectors evaluated whether there was sufficient station management review and approval of adjustments to exposure estimates and that the reasons for the adjustments were justifiable.

These inspection activities constituted a partial sample as defined in IP 71124.02–05.

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors conducted observations of in-plant work activities and assessed whether the licensee had effectively integrated the planned administrative, operational, and engineering controls into the actual field work to maintain occupational exposure ALARA. The inspectors observed pre-job briefings, and determined if the planned controls were discussed with workers. The inspectors evaluated the placement and use of shielding, contamination controls, airborne controls, radiation work permit controls, and other engineering work controls against the ALARA plans.

The inspectors assessed licensee activities associated with work-in-progress to ensure the licensee was tracking doses, performed timely in-progress reviews, and, when jobs did not trend as expected, appropriately communicated additional methods to be used to reduce dose. The inspectors evaluated whether health physics and ALARA staff were involved with the management of radiological work control when in-field activities deviated from the planned controls. The inspectors assessed whether the Outage Control Center and station management provided sufficient support for ALARA re-planning.

The inspectors assessed the involvement of ALARA staff with emergent work activities during maintenance and when possible, attended in-progress review discussions, outage status meetings, and/or ALARA committee meetings.

These inspection activities constituted a partial sample as defined in IP 71124.02–05.

b. Findings

No findings were identified.

.3 Radiation Worker Performance (02.05)

a. Inspection Scope

The inspectors observed radiation worker and radiation protection technician performance during work activities being performed in radiation areas, airborne radioactivity areas, or high radiation areas to assess whether workers demonstrated the ALARA philosophy in practice and followed procedures. The inspectors observed radiation worker performance to evaluate whether the training and skill level was sufficient with respect to the radiological hazards and the work involved.

The inspectors interviewed individuals from selected work groups to assess their knowledge and awareness of planned and/or implemented radiological and ALARA work controls.

These inspection activities constituted one complete sample as defined in IP 71124.02–05.

b. Findings

No findings were identified.

2RS3 In-Plant Airborne Radioactivity Control and Mitigation (71124.03)

.1 Engineering Controls (02.02)

a. Inspection Scope

The inspectors reviewed procedural guidance for use of ventilation systems and assessed whether the systems were used, to the extent practicable, during high-risk activities to control airborne radioactivity and minimize the use of respiratory protection. The inspectors assessed whether installed ventilation airflow capacity, flow path, and filter/charcoal unit efficiencies for selected systems were consistent with maintaining concentrations of airborne radioactivity in work areas below the concentrations of an airborne area to the extent practicable. The inspectors also evaluated whether selected temporary ventilation systems used to support work in contaminated areas were consistent with licensee procedural guidance and ALARA.

The inspectors reviewed select airborne monitoring protocols to assess whether alarms and set points were sufficient to prompt worker action. The inspectors assessed whether the licensee established trigger points for evaluating levels of airborne beta-emitting and alpha-emitting radionuclides.

These inspection activities constituted one complete sample as defined in IP 71124.03–05.

b. Findings

No findings were identified.

.2 Use of Respiratory Protection Devices (02.03)

a. Inspection Scope

The inspectors assessed whether the licensee provided respiratory protection devices for those situations where it was impractical to employ engineering controls such that occupational doses were ALARA. For select instances where respiratory protection devices were used, the inspectors assessed whether the licensee concluded that further engineering controls were not practical. The inspectors also assessed whether the licensee had established means to verify that the level of protection provided by the respiratory protection devices was at least as good as that assumed in the work controls and dose assessment.

The inspectors assessed whether the respiratory protection devices used to limit the intake of radioactive materials were certified by the National Institute for Occupational Safety and Health/Mine Safety and Health Administration (NIOSH/MSHA) or have been approved by the NRC. The inspectors evaluated whether the devices were used consistent with their NIOSH/MSHA certification or any conditions of their NRC-approval.

The inspectors reviewed records of air testing for supplied-air devices and self-contained breathing apparatus (SCBA) bottles to assess whether the air used met or exceeded Grade D quality. The inspectors evaluated whether plant breathing air supply systems satisfied the minimum pressure and airflow requirements for the devices.

The inspectors evaluated whether selected individuals qualified to use respiratory protection devices had been deemed fit to use the devices by a physician.

The inspectors reviewed training curricula for use of respiratory protection devices to assess whether individuals are adequately trained on donning, doffing, function checks, and how to respond to a malfunction.

The inspectors observed the physical condition of respiratory protection devices ready for issuance and reviewed records of routine inspection for selected devices. The inspectors reviewed records of maintenance on the vital components for selected devices and assessed whether onsite personnel assigned to repair vital components received vendor-provided training.

These inspection activities constituted one complete sample as defined in IP 71124.03–05.

b. Findings

No findings were identified.

.3 Self-Contained Breathing Apparatus for Emergency Use (02.04)

a. Inspection Scope

The inspectors reviewed the status and surveillance records for select SCBAs. The inspectors evaluated the licensee's capability for refilling and transporting SCBA air bottles to and from the control room and operations support center during emergency conditions.

The inspectors assessed whether control room operators and other emergency response and radiation protection personnel were trained and qualified in the use of SCBAs and evaluated whether personnel assigned to refill bottles were trained and qualified for that task.

The inspectors assessed whether appropriate mask sizes and types were available for use. The inspectors evaluated whether on-shift operators had no facial hair that would interfere with the sealing of the mask and that appropriate vision correction was available.

The inspectors reviewed the past 2 years of maintenance records for selected in-service SCBA units used to support operator activities during accident conditions. The inspectors assessed whether maintenance or repairs on an SCBA unit's vital components were performed by an individual certified by the manufacturer of the device to perform the work. The inspectors evaluated the onsite maintenance procedures governing vital component work to determine whether there was any inconsistencies with the SCBA manufacturer's recommended practices.
The inspectors evaluated whether SCBA cylinders satisfied the hydrostatic testing required by the U.S. Department of Transportation.

These inspection activities constituted one complete sample as defined in IP 71124.03–05.

b. Findings

No findings were identified.

.4 Problem Identification and Resolution (02.05)

a. Inspection Scope

The inspectors assessed whether problems associated with the control and mitigation of in-plant airborne radioactivity were being identified by the licensee at an appropriate threshold and were properly addressed for resolution. Additionally, the inspectors evaluated the appropriateness of the corrective actions for selected problems involving airborne radioactivity documented by the licensee.

These inspection activities constituted one complete sample as defined in IP 71124.03–05.

b. Findings

No findings were identified.

2RS4 Occupational Dose Assessment (71124.04)

.1 Source Term Characterization (02.02)

a. Inspection Scope

The inspectors evaluated whether the licensee had characterized the radiation types and energies being monitored and that the characterization included gamma, beta, hard-to-detects, and neutron radiation.

The inspectors assessed whether the licensee had developed scaling factors for including hard-to-detect nuclide activity in internal dose assessments.

These inspection activities constituted one complete sample as defined in IP 71124.04–05.

b. Findings

No findings were identified.

.2 External Dosimetry (02.03)

a. Inspection Scope

The inspectors evaluated whether the licensee's dosimetry vendor was National Voluntary Laboratory Accreditation Program accredited and if the approved irradiation test categories for each type of personnel dosimeter used were consistent with the types and energies of the radiation present and the way the dosimeter was being used.

The inspectors evaluated the onsite storage of dosimeters before their issuance, during use, and before processing/reading. For personal dosimeters stored on-site during the monitoring period, the inspectors evaluated whether they were stored in low dose areas with control dosimeters. For personal dosimeters that are taken off-site during the monitoring period, the inspectors evaluated the guidance provided to individuals with respect to care and storage of the dosimeter.

The inspectors evaluated the calibration of active dosimeters. The inspectors assessed the bias of the active dosimeters compared to passive dosimeters and the correction factor used. The inspectors also assessed the licensee's program for comparing active and passive dosimeter results, investigations for substantial differences, and recording of dose. The inspectors assessed whether there were adverse trends for active dosimeters.

These inspection activities constituted one complete sample as defined in IP 71124.04–05.

b. Findings

No findings were identified.

.3 Internal Dosimetry (02.04)

a. Inspection Scope

The inspectors reviewed procedures used to assess internal dose using whole body counting equipment to evaluate whether the procedures addressed methods for differentiating between internal and external contamination, the release of contaminated individuals, the route of intake and the assignment of dose. The inspectors assessed whether the frequency of measurements was consistent with the biological half-life of the nuclides available for intake. The inspectors reviewed the licensee's evaluation for use of portal radiation monitors as a passive monitoring system to determine if instrument minimum detectable activities were adequate to detect internally deposited radionuclides

sufficient to prompt additional investigation. The inspectors reviewed whole body counts and evaluated the equipment sensitivity, nuclide library, review of results, and incorporation of hard-to-detect radionuclides.

The inspectors reviewed procedures used to determine internal dose using in vitro analysis to assess the adequacy of sample collection, determination of entry route and assignment of dose.

The inspectors reviewed the licensee's program for dose assessment based on air sampling, as applicable, and calculations of derived air concentration. The inspectors determined whether flow rates and collection times for air sampling equipment were adequate to allow lower limits of detection to be obtained. The inspectors also reviewed the adequacy of procedural guidance to assess internal dose if respiratory protection was used. The inspectors assessed select dose assessments based on air sampling for adequacy.

The inspectors reviewed select internal dose assessments and evaluated the monitoring protocols, equipment, and data analysis.

These inspection activities constituted one complete sample as defined in IP 71124.04–05.

b. Findings

No findings were identified.

.4 Special Dosimetric Situations (02.05)

a. Inspection Scope

The inspectors assessed whether the licensee informs workers of the risks of radiation exposure to the embryo/fetus, the regulatory aspects of declaring a pregnancy, and the specific process to be used for declaring a pregnancy. The inspectors evaluated whether the monitoring program for declared pregnant workers was technically adequate to assess the dose to the embryo/fetus. The inspectors assessed results and/or monitoring controls for compliance with regulatory requirements.

The inspectors reviewed the licensee's methodology for monitoring external dose in non-uniform radiation fields or where large dose gradients exist. The inspectors evaluated the licensee's criteria for determining when alternate monitoring was to be implemented. The inspectors reviewed dose assessments performed using multi-badging to evaluate whether the assessment was performed consistently with licensee procedures and dosimetric standards.

The inspectors evaluated the licensee’s methods for calculating shallow dose equivalent from distributed skin contamination or discrete radioactive particles.

The inspectors evaluated the licensee’s program for neutron dosimetry, including dosimeter types and/or survey instrumentation. The inspectors reviewed select neutron exposure situations and assessed whether dosimetry and/or instrumentation was appropriate for the expected neutron spectra, there was sufficient sensitivity, and neutron dosimetry was properly calibrated. The inspectors also assessed whether

interference by gamma radiation had been accounted for in the calibration and whether time and motion evaluations were representative of actual neutron exposure events.

For the special dosimetric situations reviewed in this section, the inspectors assessed how the licensee assigned dose of record. This included an assessment of external and internal monitoring results, supplementary information on individual exposures, and radiation surveys and/or air monitoring results when dosimetry was based on these techniques.

These inspection activities constituted one complete sample as defined in IP 71124.04–05.

b. Findings

No findings were identified.

.5 Problem Identification and Resolution (02.06)

a. Inspection Scope

The inspectors assessed whether problems associated with occupational dose assessment 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 involving occupational dose assessment.

These inspection activities constituted one complete sample as defined in IP 71124.04–05.

b. Findings

No findings were identified

2RS8 Radioactive Solid Waste Processing and Radioactive Material Handling, Storage, and Transportation (71124.08)

.1 Radioactive Material Storage (02.02)

a. Inspection Scope

The inspectors selected areas where containers of radioactive waste are stored, and evaluated whether the containers were labeled in accordance with Title 10 of the *Code of Federal Regulations* (CFR) 20.1904, or controlled in accordance with 10 CFR 20.1905.

The inspectors assessed whether the radioactive material storage areas were controlled and posted in accordance with the requirements of 10 CFR Part 20. For materials stored or used in the controlled or unrestricted areas, the inspectors evaluated whether they were secured against unauthorized removal and controlled in accordance with 10 CFR 20.1801 and 10 CFR 20.1802.

The inspectors evaluated whether the licensee established a process for monitoring the impact of low-level radioactive waste storage that was sufficient to identify potential unmonitored, unplanned releases or nonconformance with waste disposal requirements.

The inspectors evaluated the licensee's program for container inventories and inspections. The inspectors selected containers of stored radioactive material, and assessed for signs of swelling, leakage, and deformation.

These inspection activities constituted one complete sample as defined in IP 71124.08–05.

b. Findings

No findings were identified.

.2 Radioactive Waste System Walk-down (02.03)

a. Inspection Scope

The inspectors walked down accessible portions of select radioactive waste processing systems to assess whether the current system configuration and operation agreed with the descriptions in plant and/or vendor manuals.

The inspectors reviewed administrative and/or physical controls to assess whether equipment, which is not in service or abandoned in place would not contribute to an unmonitored release path and/or affect operating systems or be a source of unnecessary personnel exposure. The inspectors assessed whether the licensee reviewed the safety significance of systems and equipment abandoned in place in accordance with 10 CFR 50.59.

The inspectors reviewed the adequacy of changes made to the radioactive waste processing systems since the last inspection. The inspectors evaluated whether changes from what is described in the Final Safety Analysis Report were reviewed and documented in accordance with 10 CFR 50.59 or that changes to vendor equipment were made in accordance with vendor manuals. The inspectors also assessed the impact of these changes on radiation doses to occupational workers and members of the public.

The inspectors selected processes for transferring radioactive waste resin and/or sludge discharges into shipping/disposal containers and assessed whether the waste stream mixing, sampling, and waste concentration averaging were consistent with the process control program, and provided representative samples of the waste product for the purposes of waste classification.

The inspectors evaluated whether tank recirculation procedures provided sufficient mixing.

The inspectors assessed whether the licensee's process control program correctly described the current methods and procedures for dewatering and waste stabilization.

These inspection activities constituted one complete sample as defined in IP 71124.08–05.

b. Findings

No findings were identified.

.3 Waste Characterization and Classification (02.04)

a. Inspection Scope

For select waste streams, the inspectors assessed whether the licensee's radiochemical sample analysis results were sufficient to support radioactive waste characterization as required by 10 CFR Part 61. The inspectors evaluated whether the licensee's use of scaling factors and calculations to account for difficult-to-measure radionuclides was technically sound and based on current 10 CFR Part 61 analysis.

The inspectors evaluated whether changes to plant operational parameters were taken into account to: (1) maintain the validity of the waste stream composition data between the sample analysis update; and (2) assure that waste shipments continued to meet the requirements of 10 CFR Part 61.

The inspectors evaluated whether the licensee had established and maintained an adequate quality assurance program to ensure compliance with the waste classification and characterization requirements of 10 CFR 61.55 and 10 CFR 61.56.

These inspection activities constituted one complete sample as defined in IP 71124.08–05.

b. Findings

No findings were identified.

.4 Shipment Preparation (02.05)

a. Inspection Scope

The inspectors reviewed the technical instructions presented to workers during routine training. The inspectors assessed whether the licensee's training program provided training to personnel responsible for the conduct of radioactive waste processing and radioactive material shipment preparation activities. The inspectors assessed whether shippers were knowledgeable of the shipping regulations and demonstrated adequate skills to accomplish package preparation requirements. The inspectors evaluated whether the licensee was maintaining shipping procedures in accordance with current regulations. The inspectors assessed whether the licensee was meeting the expectations in NRC Bulletin 79–19, "Packaging of Low-Level Radioactive Waste for Transport and Burial," and 49 CFR Part 172, Subpart H, "Training."

The inspectors evaluated whether the requirements for Type B shipment Certificates of Compliance had been met. The inspectors determined whether the user was a registered package user and had an NRC-approved quality assurance program. The inspectors assessed whether procedures for cask loading and closure were consistent with vendor procedures.

The inspectors assessed whether non-Type B shipments were made in accordance with the package quality documents.

The inspectors assessed whether the receiving licensee was authorized to receive the shipment packages.

These inspection activities constituted one complete sample as defined in IP 71124.08–05.

b. Findings

No findings were identified.

.5 Shipping Records (02.06)

a. Inspection Scope

The inspectors reviewed select shipments to evaluate whether the shipping documents indicated the proper shipper name; emergency response information and a 24-hour contact telephone number; accurate curie content and volume of material; and appropriate waste classification, transport index, and UN number. The inspectors assessed whether the shipment marking, labeling, and placarding was consistent with the information in the shipping documentation.

These inspection activities constituted one complete sample as defined in IP 71124.08-05.

b. Findings

Failure to Properly Classify a Shipment per Department of Transportation Regulations

Introduction: The inspectors documented a self-revealed finding of very low safety significance and an associated NCV of 10 CFR 71.5(a) and 49 CFR 173.421(b) that was identified when a shipment containing turbine tooling arrived at its destination and was discovered by the receiving entity to be misclassified per Department of Transportation (DOT) regulations. The shipment was classified as Limited Quantity based on dose rates from the initial surveys conducted prior to shipment. When the container arrived at the destination, the shipment dose rates were greater than the 49 CFR 173.421(b) limits for Limited Quantity shipments. The shipment should have been transported based on its radiological characteristics that represent Schedule 5 through 11 as described in NUREG–1660, "U.S.-Specific Schedules of Requirements for Transport of Specified Types of Radioactive Material Consignments." The licensee's failure to correctly classify this shipment per DOT regulations was reasonably within the licensee ability to foresee and correct, and should have been prevented, therefore, constituting a performance deficiency. This action resulted in the shipment being transported to another facility with a classification that would indicate dose rates were lower than what they actually were during transportation of the shipment.

Description: On December 21, 2016, a shipment (M16–064) containing turbine tooling was classified as Limited Quantity with dose rates <0.1 mRem/hr on contact and sent to another facility licensed to received radioactive material. Upon arrival, security did not have the ability to verify the seals on the shipping containers. The shipment was consequently sent to another authorized location until the ability to verify the seals was regained. The shipment returned to intended receiver on January 10, 2017. Upon surveying the sea-land containers associated with this shipment, a radiation

protection (RP) technician for the receiving entity identified an area on the container that read 1.49 mRem/hr on contact. The sea-land container in question was then resurveyed with a different instrument and the RP technician obtained a reading of 1.51 mRem/hr on contact. The receiver contacted Clinton Power Station to inform them of the unexpected results from the receipt survey. On January 11, 2017, Clinton Power Station sent two RP technicians to the receiving facility and confirmed the dose rate of 1.51 mRem/hr on contact. After the dose rate was confirmed, the sea-land container was opened and it was revealed that the source of elevated dose rate was a dust-collector that was used for sand-blasting. During transportation, the dust and debris inside the dust-collector shifted causing the dose rate of the sea-land container to be above the Limited Quantity classification limit during transit.

When the shipment arrived at the destination, surveys revealed dose rates of 1.49 mRem/hr and 1.51 mRem/hr on contact, respectively. These dose rates indicated that the package radiation level for Limited Quantity shipments of 0.5 mRem/hr on contact was exceeded.

Analysis: The inspectors determined that the licensee's failure to correctly classify shipment (M16–064) per DOT regulations was reasonably within the licensee ability to foresee and correct, and should have been prevented, therefore, constituting a performance deficiency. Specifically, the shipment was classified as Limited Quantity due to on contact dose rates being <0.1 mRem/hr before shipment, but increased during transit with contact dose readings between 1.49–1.51 mRem/hr. The performance deficiency was determined to be more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," because the performance deficiency impacted the program and process attribute of the Public Radiation Safety cornerstone and adversely effected the cornerstone objective of ensuring adequate protection to public health and safety from exposure to radiation from routine civilian nuclear operations. Specifically, the misclassification of this shipment could have led to members of the public and individuals involved in the shipping of the container to receive radiation dose that was greater than expected. This would be caused by dose rates that were reported on the shipping manifest being lower than the actual dose rates of the container.

The finding was determined to be of very low safety significance (Green) in accordance with IMC 0609, Appendix D, "Public Radiation Safety Significance Determination Process," dated August 19, 2008, because: (1) a certificate of compliance issue; (2) the failure to make emergency notifications; (3) a low-level burial issue; or (4) a breach of the transportation package occurring during transit did not occur. The finding did involve a radioactive shipment above radiation limits. However, the shipment contained less than a Type A quantity of material (LSA I shipment), and dose rates were <2 mRem/hr on contact. Consequently, the inspectors determined that the finding was of very low safety significance (Green).

The inspectors determined that this finding affected the cross-cutting area of human performance in the aspect of challenging the unknown, where individuals stop when faced with uncertain conditions. Risks are evaluated and managed before proceeding. Specifically, the risk associated with the content of the dust-collector shifting during transportation and creating an area that would lead to elevated dose rates was not evaluated by Clinton Power Station radiation protection staff. [H.11]

Enforcement: Title 10 CFR 71.5(a) states, in part, "each licensee who transports licensed material outside the site of usage, as specified in the NRC license, or where transport is on public highways, or who delivers licensed material to a carrier for transport, shall comply with the applicable requirements of the DOT regulations in 49 CFR Parts 107, 171 through 180, and 390 through 397, appropriate to the mode of transport."

Specifically, 49 CFR 173.421(b) states, in part, "A Class 7 (radioactive) material with an activity per package which does not exceed the limited quantity package limits specified in Table 4 in CFR 173.425, and its packaging, are excepted from requirements in this subchapter for specification packaging, marking (except for the UN identification number marking requirement described in CFR 173.422(a), labeling, and if not a hazardous substance or hazardous waste, shipping papers, and the requirements of this subpart (B). The radiation level at any point on the external surface of the package does not exceed 0.005 mSv/h (0.5 mRem/h)."

Contrary to the above, on December 21, 2016, the licensee failed to comply with the applicable requirements of the DOT regulations in 49 CFR Parts 107, 171 through 180, and 390 through 397, appropriate to the mode of transport, when they failed to properly classify a shipment per DOT regulations. The shipment was classified as Limited Quantity after the initial surveys were taken and on contact dose rates were <0.1 mRem/hr. When the shipment arrived at the destination, surveys revealed dose rates of 1.49 mRem/hr and 1.51 mRem/hr on contact, respectively, which exceeded the limited quantity package limits. The site implemented immediate corrective actions which included all shipments classified as limited quantity to be approved by a senior manager in the Radiation Protection Department prior to shipping. Another immediate corrective action required that the first 4 shipments conducted by the site shipper after this event be under the direct observation of a fleet independent shipper and a senior manager in the Radiation Protection Department. Because this violation was of very low safety significance and was entered into the licensee's CAP as AR 03961544, this violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. **(NCV 05000461/2017002-05: Failure to Properly Classify a Shipment per DOT Regulations)**

.6 Identification and Resolution of Problems (02.07)

a. Inspection Scope

The inspectors assessed whether problems associated with radioactive waste processing, handling, storage, and transportation, were being identified by the licensee at an appropriate threshold, were properly characterized, and were properly addressed for resolution. Additionally, the inspectors evaluated whether the corrective actions were appropriate for a selected sample of problems documented by the licensee that involve radioactive waste processing, handling, storage, and transportation.

These inspection activities constituted one complete sample as defined in IP 71124.08–05.

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 Mitigating Systems Performance Index—Emergency AC Power System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index (MSPI)—Emergency AC Power System performance indicator (PI) for the period from the second quarter 2016 through first quarter 2017. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 31, 2013, were used. The inspectors reviewed the licensee's operator narrative logs, MSPI derivation reports, issue reports, event reports and NRC integrated inspection reports for the period of April 1, 2016, through March 31, 2017, to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator, and none were identified.

This inspection constituted one MSPI emergency AC power system sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index—High Pressure Injection Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index—High Pressure Injection Systems PI for the period from the second quarter 2016 through first quarter 2017. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 31, 2013, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, MSPI derivation reports, event reports and NRC integrated inspection reports for the period of April 1, 2016, through March 31, 2017, to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator, and none were identified.

This inspection constituted one MSPI high pressure injection system sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.3 Mitigating Systems Performance Index—Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index—Heat Removal System PI for the period from the second quarter 2016 through first quarter 2017. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 31, 2013, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports, MSPI derivation reports, and NRC integrated inspection reports for the period of April 1, 2016, through March 31, 2017, to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator, and none were identified.

This inspection constituted one MSPI heat removal system sample as defined in IP 71151–05.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems (71152)

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

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.2 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also considered the results of daily inspector CAP item screening discussed in Section 4OA2.1 above, licensee trending efforts, and licensee human performance results. The inspectors' review nominally considered the 6-month period of January 1, 2017, through June, 30, 2017, although some examples expanded beyond those dates where the scope of the trend warranted.

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

Observations

Over the last 6 months, the inspectors identified a negative trend regarding the timeliness of completing operability determinations as outlined in station procedures and NRC IMC 0326. The licensee documented the issue in the CAP as AR 04030777. The inspectors' plan to follow up on the licensee's corrective actions later this year as part of the biennial problem identification and resolution inspections occurring in September 2017.

Additionally, during the review period the inspectors noted that the licensee is self-identifying and placing into the corrective action processes, issues associated with the completeness and accuracy of the USAR. None of the issues identified by the licensee crossed the more than minor threshold and all were documented in an action request. This is a positive trend in this area where previous violations were written. The licensee's sensitivity to the issue is at a low threshold for identifying errors and the errors are being placed into the corrective action processes for resolution.

This review constituted one semi-annual trend review inspection sample as defined in IP 71152.

b. Findings

No findings were identified.

.3 Annual Follow-up of Selected Issues: Division 1 Shutdown Service Water Pump Room Cooling Fan Failure

a. Inspection Scope

The inspectors selected the following Condition Reports for in-depth review:

- AR 02667822, Unexpected alarms 5050—1F/2F and 1VH01CA trip.

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

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

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

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

b. Findings

Failure to Perform Preventive Maintenance on a Safety-Related Breaker Cubicle

Introduction: The inspectors identified a finding of very low safety significance for the licensee's failure to perform maintenance on a safety-related motor control center cubicle. Specifically, the licensee failed to perform thermography on the division 1 SX pump room cooler breaker cubicle in accordance with the maintenance strategy/template without providing justification for differing from the template as required by MA–AA–716–210, "Performance Centered Maintenance Process," Revision 3. This resulted in the division 1 SX room cooler fan failing because of a high resistance connection that went undetected.

Description: On May 10, 2016, the division 1 shutdown SX pump room cooler fan tripped during routine testing and the licensee declared the division 1 SX pump inoperable. This was documented in AR 02667822 in the licensee's CAP. As part of

troubleshooting activities, the licensee performed thermography on the SX fan breaker cubicle and identified a hot spot on the 'C' thermal overload relay connection. Corrective actions were taken by the licensee to replace the divisions 1 and 2 SX pump room cooler fan breaker thermal overload relays. A successful surveillance test was performed on the room cooler fan and the division 1 SX pump was declared operable.

The licensee completed an apparent cause evaluation (ACE) that determined the fan thermal overload relays tripped due to excessive heating of a loose connection between one of the screws that connects the starter with the 'C' thermal overload relay. When the screw was installed, it was cocked, resulting in the two connectors not sitting flush. The misaligned connection left a small gap which created the high resistance across the connection. The connection likely degraded overtime due to thermal cycling each time the fan was started and stopped.

The inspectors reviewed the ACE, the licensee procedures, and maintenance records to determine whether the licensee had previously identified or implemented a method to detect high resistance connections for motor control centers. During their review, the inspectors identified the licensee's preventive maintenance template for motor control center compartments stated that thermography would detect increases in temperature that affect the breaker and its connections. The template additionally stated that thermography was an effective condition monitoring task for failures caused by loose, damaged, or contaminated connections, or contacts that are out of adjustment, pitted or corroded, or which have damaged plating. The template was applicable to the division 1 SX pump room cooler fan breaker cubicle, a level 3 component, which should have established a 12 month periodicity for thermography activities.

Procedure MA–AA–716–210, Step 4.14.5.2 stated, "When new or revised PCM [performance centered maintenance] templates are issued/released, the sites should: 1) Implement templates identified as higher priority within 6 months of the issue/release date; and 2) Implement the remaining templates in a site determined priority, not to exceed 18 months from the issue/release date." Additionally, Step 5.1.2 stated, "Document reasons for applying preventive maintenance different than recommended by the PCM template."

The inspectors' review of the issue determined the licensee had not been performing thermography of the division 1 SX pump room cooler fan breaker cubicle. On January 26, 2004, the licensee issued a revision to the PCM template for circuit breakers that added periodic thermography to the maintenance strategy with the purpose of detecting and monitoring high resistance connections. The licensee failed to add this activity to their planned maintenance for the SX room cooler fan breaker cubicle and was unable to provide any written documentation justifying why thermography was not being performed.

The license completed EC 619008, "Evaluate Survivability of 1SX01PA and Associated Equipment on Loss of SX Room Cooling," to demonstrate whether the division 1 SX pump motor and associated equipment would have performed their required functions for 24 hours after a loss of SX room cooling. The inspectors reviewed this analysis to determine if the licensee had reasonably demonstrated the equipment would have survived the conditions determined to exist after fan failure occurred. The inspectors noted the licensee's evaluation determined the peak temperature in the SX pump room was as high as 182 degrees Fahrenheit (F), 24 hours after a loss of room cooling. The

licensee then evaluated the 1SX01PA pump motor and required support equipment in the room to determine if the equipment would meet their required function at a temperature of 200 degrees F for 24 hours. Inspectors determined that the licensee's evaluations were reasonable in concluding the 1SX01PA pump and required support equipment would have performed their functions as identified in component remaining life predictions and margin analyses.

In addition to replacing the thermal overload relays, the licensee also created an action to perform thermography on this equipment on a periodic basis that was documented in AR 02667822.

Analysis: The inspectors determined that the licensee's failure to perform maintenance on a safety-related motor control center cubicle was a performance deficiency. Specifically, the licensee failed to perform thermography on the division 1 SX pump room cooler breaker cubicle in accordance with the maintenance strategy/template or to provide justification for differing from the template, resulting in the division 1 SX room cooler fan failing due to an undetected high resistance connection. 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 respond to initiating events. Specifically, the room cooler fan failure directly impacted the operability of the division 1 SX pump and the division 1 EDG which are safety-related risk significant systems. Using IMC 0609, Attachment 4, "Initial Characterization of Findings," and Appendix A, "The Significance Determination Process (SDP) for Findings at Power," Exhibit 2, June 19, 2012, the finding was screened against the Mitigating Systems cornerstone and determined to be of very low safety significance (Green) because the inspectors were able to answer all of the questions "No." The inspectors determined that this finding is not indicative of current performance and therefore did not assign a cross-cutting aspect.

Enforcement: The inspectors did not identify a violation of a regulatory requirement associated with this finding since performing thermography is not required by the NRC. The licensee has entered this issue into their CAP as AR 02667822. Because this finding does not involve a violation and is of very low safety significance, it is identified as a Finding. **(FIN 05000451/2017002–06: Failure to Perform Preventive Maintenance on a Safety-Related Breaker Cubicle)**

.4 (Closed) Unresolved Item 05000461/2014008–02: Evaluation of Root Cause Evaluation 1594407 Against Inspection Procedure 95001 Objectives

a. Inspection Scope

In July 2014, as part of the 95001 inspection for the Unplanned Reactor Scrams per 7,000 Critical Hours PI exceeding the Green-to-White threshold, it was determined that the IP 95001 objectives could not be evaluated at the time of the onsite inspection for root cause evaluation (RCE) 1594407, "Automatic Trip of Breaker 1AP07EJ–0AP05E2 Transformer," because a final failure analysis of the failed transformer was required to complete the RCE.

In June 2016, the licensee finalized Root Cause Report (RCR) 01594407, "Automatic Trip of Breaker 1AP07EJ—0AP05E2 Transformer Failure," in accordance with station procedure PI–AA–125–1001, "Root Cause Analysis Manual," Revision 2.

The inspectors completed a review of the RCE using the objectives outlined in IP 95001 of providing assurance that the root causes and contributing causes of risk-significant performance issues were understood; that the extent of condition and extent of cause of risk-significant performance issues were identified; and that the licensee's corrective actions for risk-significant performance issues were sufficient to address the root and contributing causes and prevent recurrence.

This review constituted one in-depth problem identification and resolution sample as defined in IP 71152–05.

b. Findings

Failure to Identify a Root Cause and Develop Corrective Actions to Preclude Repetition

Introduction: The inspectors identified a finding of very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion II, "Quality Assurance Program," for the failure to implement a quality assurance program procedure. Specifically, the licensee failed to identify/document a root cause and develop a corrective action to preclude repetition (CAPR) for the 1A bus transformer failure in accordance with quality assurance procedure PI–AA–125–1001, "Root Cause Analysis Manual."

Description: On December 9, 2013, safety-related transformer 0AP07EJ failed causing a loss of power to all vital loads powered by the division 1 electrical bus. In accordance with their process, the licensee does not document whether an issue is a significant condition adverse to quality; however, they communicated to the inspectors that this failure was considered a significant condition adverse to quality. The licensee performed an RCE in accordance with PI–AA–125–1001 to satisfy the quality assurance program requirements to identify and document significant conditions adverse to quality as well as determine the cause and establish a CAPR.

On June 6, 2016, the licensee finalized RCR 01594407, "Automatic Trip of Breaker 1AP07EJ—0AP05E2 Transformer Failure," in accordance with station procedure PI–AA–125–1001, "Root Cause Analysis Manual," Revision 2. The licensee documented the evaluation using Attachment 13 of PI–AA–125–1001. Section 4.4.2, Step 4 of PI–AA–125–1001, stated in part, "analyze each causal factor to determine the root causes." Step 2.25 of PI–AA–125, Revision 4, "Corrective Action Program," stated, in part, that a root cause was "the fundamental underlying cause(s) that, when corrected, will prevent recurrence of an inappropriate action or equipment failure that results in a consequential event or condition." Section 4.5, Step 1 of PI–AA–125–1001, stated in part, "identify at least one corrective action to prevent recurrence for each root cause."

The licensee had the transformer removed and sent off site for laboratory analysis to determine why the transformer failed. The failure analysis report stated that it was not possible to determine a definitive root cause of the unit's failure. Based on the evidence gathered during the analysis, there were two failures apparent in the windings; a phase 'B' arc and a phase 'A' turn-to-turn short circuit. It was not possible to determine which failure had occurred first because either event could occur first resulting in the other

event. However, the common factor for either event occurring was insulation degradation.

Since the report was not definitive on which event led to the other the licensee chose to assign a most probable causal factor and cause and perform a risk assessment in accordance with PI–AA–125–1001, Section 4.5, rather than identifying a root cause and a CAPR. The procedure also stated that the development of a CAPR could be waived by the Management Review Committee if it was determined that a CAPR was not required or feasible based on the performance of an effective risk assessment. The procedure did not provide additional guidance on when a CAPR was required regardless of the performance of a risk assessment, as in the case of significant conditions adverse to quality.

The inspectors performed an independent review of the RCR and the stations quality assurance program. The inspectors review determined that the licensee had over looked the common factor that for either scenario insulation degradation had occurred and was the root cause of the failure. Additionally, the inspectors determined that the licensee had failed to take measures to assure that the cause of the significant condition adverse to quality was determined and corrective action was taken to preclude repetition. The inspectors discussed these issues with the licensee. The licensee reevaluated and revised the RCR and assigned a CAPR to the root cause as required.

Analysis: The inspectors determined that the failure to document a root cause and develop a CAPR for the 1A bus transformer failure in accordance with quality assurance procedure PI–AA–125–1001 was a performance deficiency. The performance deficiency was determined to be more than minor in accordance with IMC 0612, "Power Reactor inspection Reports," Appendix B, "Issue Screening," dated September 7, 2012, because if left uncorrected the performance deficiency had the potential to lead to a more significant safety concern. Since the transformer failure was a significant condition adverse to quality if the root cause and appropriate corrective actions to prevent recurrence were not identified and implemented additional transformer failures could occur. Using 0609, Attachment 4, "Initial Characterization of Findings at Power," and Appendix A, "The Significance Determination Process for Findings at Power," issued June 19, 2012, the finding was screened against the Initiating Events cornerstone and determined to be of very low safety significance (Green) because the finding did not involve the complete or partial loss of a support system that contributes to the likelihood of, or cause an initiating event and did not affect mitigation equipment.

The inspectors determined this finding affected the cross-cutting area of human performance, in the aspect of resources, where leaders ensure that personnel, equipment, procedures and other resources are available and adequate to support nuclear safety. Specifically, the licensee's station procedure did not provide guidance on when a CAPR is required, regardless of whether a risk assessment was performed. [H.1]

Enforcement: Title 10 CFR 50 Appendix B, Criterion II, "Quality Assurance Program," requires, in part, that "the applicant shall establish at the earliest practicable time, consistent with the schedule for accomplishing the activities, a quality assurance program which complies with the requirements of this appendix. This program shall be documented by written policies, procedures, or instructions and shall be carried out throughout plant life in accordance with those policies, procedures, or instructions."

Section 4.4.2, Step 4 of PI–AA–125–1001, Revision 2, "Root Cause Analysis Manual," states, in part, "analyze each causal factor to determine the root causes." Step 2.25 of PI–AA–125, Revision 4, "Corrective Action Program," states, in part, that a root cause is "the fundamental underlying cause(s) that, when corrected, will prevent recurrence of an inappropriate action or equipment failure that results in a consequential event or condition."

Section 4.5, Step 1 of PI–AA–125–1001, states, in part, "identify at least one corrective action to prevent recurrence for each root cause."

Contrary to the above, on June 6, 2016, the licensee failed to carry out the quality assurance program in accordance with their procedures. Specifically, the licensee failed to follow quality assurance program procedure PI–AA–125–1001, Revision 2, "Root Cause Analysis Manual," Section 4.4.2, Step 4, to "analyze each causal factor to determine the root causes" and Section 4.5, Step 1, "identify at least one corrective action to prevent recurrence for each root cause." Specifically, RCR 01594407, "Automatic Trip of Breaker 1AP07EJ—0AP05E2 Transformer Failure," identified a most probable cause of a turn to turn failure of the high side windings and assigned no corrective action to prevent recurrence for the root cause. The corrective actions in response to this violation were to revise the RCR with a root cause of insulation degradation of the phase windings over time and develop a corrective action to prevent recurrence by using Doble testing to ensure indication of insulation degradation of the transformers were discovered prior to failure. Because this finding was of very low safety significance and was entered into the CAP as AR 01594407, this violation is being treated as an NCV, in accordance with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000461/2017002–07: Root Cause Evaluation Failed to Identify Corrective Action to Preclude Repetition)**

4OA6 Management Meetings

.1 Exit Meeting Summary

On April July, 13, 2017, the inspectors presented the inspection results to Mr. T. Stoner, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meetings

Interim exits were conducted for:

- The inspection results of the Plant Modifications Inspection with Mr. T. Stoner, Site Vice President, and other member of the licensee staff on April 21, 2017.
- The inspection results of the ISI with Mr. T. Stoner, Site Vice President, and other members of the licensee staff on May 19, 2017.
- The inspection results of the Radiation Safety Program review with Mr. T. Stoner, Site Vice President, on May 19, 2017.
- The inspection results for the Radiation Safety Program review with Mr. B. Kapellas, Plant Manager, on June 23, 2017.

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

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
T. Dean, Training Director
C. Dunn, Operations Director
K. Engelhardt, Outage Manager
M. Friedmann, Emergency Preparedness Manager
M. Heger, Senior Manager Plant Engineering
T. Krawyck, Engineering Director
W. Marsh, Organizational Effectiveness Manager
S. Minya, Operations Training Manager
F. Paslaski, Radiation Protection Manager
K. Pointer, Regulatory Assurance
D. Shelton, Regulatory Assurance Manager
S. Strickland, Shift Operations Superintendent
J. Ward, Chemistry Manager
J. Wilson, Senior Manager Plant Engineering

U.S. Nuclear Regulatory Commission

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

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000461/2017002–01	NCV	Failure of Operators to Meet Time Critical Operator Actions (Section 1R15.1)
05000461/2017002–02	VIO	Failure to Perform Adequate Evaluation of Crane Rail Clips (Section 1R18.1)
05000461/2017002–03	NCV	Unexpected Start of the Division 3 Emergency Diesel Generator (Section 1R20.1)
05000461/2017002–04	NCV	Failure to Provide Sufficient Work Instructions for Performing Maintenance on the Control Room Ventilation System Charcoal Filter (Section 1R22.1)
05000461/2017002–05	NCV	Failure to Properly Classify a Shipment per DOT Regulations (Section 2RS8.5)
05000461/2017002–06	FIN	Failure to Perform Preventive Maintenance on a Safety-Related Breaker Cubicle (Section 4OA2.3)
05000461/2017002–07	NCV	Root Cause Evaluation Failed to Identify Corrective Action to Preclude Repetition (Section 4OA2.4)

Closed

05000461/2017002–01	NCV	Failure of Operators to Meet Time Critical Operator Actions (Section 1R15.1)
05000461/2017002–03	NCV	Unexpected Start of the Division 3 Emergency Diesel Generator (Section 1R20.1)
05000461/2017002–04	NCV	Failure to Provide Sufficient Work Instructions for Performing Maintenance on the Control Room Ventilation System Charcoal Filter (Section 1R22.1)
05000461/2017002–05	NCV	Failure to Properly Classify a Shipment per DOT Regulations (Section 2RS8.5)
05000461/2017002–06	FIN	Failure to Perform Preventive Maintenance on a Safety-Related Breaker Cubicle (Section 4OA2.3)
05000461/2017002–07	NCV	Root Cause Evaluation Failed to Identify Corrective Action to Preclude Repetition (Section 4OA2.4)
05000461/2014008-02	URI	Evaluation of RCE 1594407 Against Inspection Procedure 95001 Objectives (Section 4OA2.4)

LIST OF DOCUMENTS REVIEWED

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

1R01 Adverse Weather Protection

- CPS 3501.01, "High Voltage Auxiliary Power System" Revision 28d
- CPS 3505.01C001, "Generator Backfeed Checklist" Revision 10a
- CPS 3501.01E001, "High Voltage Auxiliary Power System Electrical Lineup" Revision 14
- CPS 3502.01, "480 VAC Distribution" Revision 10a
- CPS 3502.01E001, "480 VAC Distribution Electrical Lineup" Revision 14
- CPS 3505.01, "345 & 138 kV Switchyard (SY)" Revision 20b
- CPS 3505.01E001, "Switchyard Electrical Lineup" Revision 11c
- CPS 3505.01V001, "Switchyard Valve Lineup" Revision 8c
- CPS 3506.01, "Diesel Generator and Support Systems (DG)" Revision 37f
- CPS 4200.01, "Loss of AC Power" Revision 24c
- CPS 4201.01, "Loss of DC Power" Revision 8c
- CPS 9082.01, "Offsite Source Power Verification" Revision 40d
- CPS 9082.02, "Electrical Distribution Verification" Revision 36a

1R04 Equipment Alignment

- CPS 3310.01, "Reactor Core Isolation Cooling," Revision 29e
- CPS 3310.01C001, "Restoring RCIC After Maintenance Outage," Revision 4
- CPS 3310.01E001, "Reactor Core Isolation Cooling Electrical Lineup," Revision 16
- CPS 3310.01V001, "Reactor Core Isolation Cooling Valve Lineup," Revision 12e
- CPS 3310.01V002, "Reactor Core Isolation Cooling Instrument Valve Lineup," Revision 9e
- CPS 3312.01, "Residual Heat Removal," Revision 45f
- CPS 3312.01E001, "Residual Heat Removal Electrical Lineup," Revision 17
- CPS 3312.01V001, "Residual Heat Removal Valve Lineup," Revision 17c
- CPS 3312.01V002, "Residual Heat Removal Instrument Valve Lineup," Revision 9a
- CPS 3506.01, "Diesel Generator and Support Systems" Revision 13a
- CPS 3506.01E001, "Diesel Generator and Support Systems Electrical Lineup" Revision 18c
- CPS 3506.01P001, "Division 2 Diesel Generator Operations" Revision 3a
- 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 3211.01, "Shutdown Service Water (SX)" Revision 32
- CPS 3211.01E001, "Shutdown Service Water Electrical Lineup" Revision 18a
- CPS 3211.01V001, "Shutdown Service Water Valve Lineup" Revision 28d
- CPS 3211.01V002, "Shutdown Service Water Instrument Valve Lineup" Revision 9

1R05 Fire Protection

- CPS 1893.01, "Fire Protection Impairment Reporting" Revision 20d
- 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 1893.04M351, “781 Control: Auxiliary Electrical Equipment, Inverter and Battery Rooms Prefire Plan” Revision 7c
- AR 04017957, NRC Identified TCP and Exposed Combustibles
- CPS 1893.04M532, “762 Diesel Generator: Diesel Generator HVAV Equipment Area and Unit 2 Rooms Prefire Plan” Revision 5a
- AR 040225257, Wall Integrity Questioned During NRC Walkdown
- AR 04025235, Latch Sticking on Door 445
- AR 04025263, Damaged and Missing Seal on Fire Door 446
- CPS 1893.04M801, “699 Screen House: Division 2 and 3 SX Pump Rooms and Tunnel Prefire Plan” Revision 6
- LS–AA–104–1001, 50.59 Review Form, Scaffold S-026 Erected More Than 90 Days to Support Piping Installation per WO 01690866
- CPS 1893.04M512, “737 Diesel Generator: Division 2 Diesel Generator and Day Tank Room Prefire Plan” Revision 7a

<u>1R06 Flood Protection Measures</u>

- CPS 8378.01, “Inspection of Cable Vault and Cable Vault Sump Pumps Checklist” Revision 1
- SA–AA–114, “Confined Space Entry” Revision 19

<u>1R07 Heat Sink Performance</u>

- ER–AA–340, “GL 89–13 Program Implementing Procedure,” Revision 8
- ER–AA–340–1001, “GL 89–13 Program Implementation Instruction Guide,” Revision 10
- ER–AA–340–1002, “Service Water Heat Exchanger and Component Inspection Guide,” Revision 6
- CPS 1003.10, “Clinton Power Station Program For NRC Generic Letter 89-13 Service Water Problems Affecting Safety Related Equipment,” Revision 7
- EPRI NP–7552, “Heat Exchanger Performance Monitoring Guidelines”
- EPRI TR–107397, “Service Water Heat Exchanger Testing Guidelines”
- WO 01771103, Inspect, Boroscope, Clean, Eddy Current and Clean 1VY06AA
- WO 01771102, Inspect, Boroscope, Clean, Eddy Current and Clean 1VY06AB

<u>1R08 Inservice Inspection Activities</u>

- ER–AA–335–003; Magnetic Particle Examination; Revision 7
- ER–AA–335–018; Visual Examination of ASME IWE Class MC and Metallic Liners of IWL Class CC Components; Revision 12
- ER–AA–335–002; Liquid Penetrant Examination; Revision 9
- ER–AA–330–1002; Clinton Power Station 2016 Inservice Inspection Program Health Report
- GEH–UT–247; Procedure for Phased Array Ultrasonic Examination of Dissimilar Metal Welds; Revision 3
- GEH–PDI–UT–1; PDI Generic Procedure for the Ultrasonic Examination of Ferritic Pipe Welds; Revision 10

- GEH–PDI–UT–2; PDI Generic Procedure for The Ultrasonic Examination of Austenitic Pipe Welds; Revision 10
- GEH–PDI–UT–10; PDI Generic Procedure for The Ultrasonic Examination of Dissimilar Metal Welds; Revision 8
- EPRI–DMW–PA–1; Procedure for Manual Phased Array Ultrasonic Examination of Dissimilar Metal Welds; Revision 6
- Report 06–074; PT Examination Summary Sheet; February 11, 2006
- Report 06–071; UT Examination Summary Sheet; February 16, 2006
- Report C1–001; MT Examination Summary Sheet; January 12, 2008
- C1R13–044; Ultrasonic Calibration and Examination Record; December 10, 2011
- C1R17–APR–24; UT Examination; May 17, 2017
- C1R17–UT–018; UT Calibration and Examination; May 17, 2017
- C1R17–MT–003; Magnetic Particle Examination; May 17, 2017
- C1R17–PT–002; Liquid Penetrant Examination; May 18, 2017
- WO 01508953; Replace 1B33F067B Valve Disc with Anti-Rotation Disc
- WO 04613137; Replace MS Flex Hoses
- WPS 8–8–GTSM; ASME Welding Procedure Specification; Revision 6
- WPS 1–1–GTSM–PWHT; ASME Welding Procedure Specification; Revision 2
- PQR 1–15A; Welding Procedure Qualification Record; December 28, 1983
- PQR 4–51A; Welding Procedure Qualification Record; September 12, 1986
- PQR A–003; Welding Procedure Qualification Record; February 8, 2000
- PQR A–004; Welding Procedure Qualification Record; February 8, 2000
- PQR A–001; Welding Procedure Qualification Record; October 19, 1998
- PQR A–002; Welding Procedure Qualification Record; March 9, 1999
- PQR 1–50C; Welding Procedure Qualification Record; January 3, 1984
- PQR 1–51A; Welding Procedure Qualification Record; April 6, 1984
- EC 368729; Pipe Coupling Installation for The Replacement of MS and RR Flex Hoses
- AR 02706215; NRC NCV 2016002–05 Lack of Acceptance Criteria for CNMT Exam
- AR 04012181; C1R17: Improper Weld Rod Used on RR67B Leak-off Line Weld
- AR 04012712; Improvement Opportunity in Providing Outage Weld Plan to NRC
- AR 02678790; EWP Should Not Be Used for ASME Class 1, 2, and 3 WO
- AR 03999900; Eng OE: UT Indications Discovered During Planned Inspection
- AR 04012784; Review ER–AA–335–003 Rev 7 for Prod Spacing Measurements
- AR 04011345; PMRQ 178930–27 Requires Due Date Adjustment
- AR 02715763; Procedure Enhancement to 9080.03
- AR 2706215; NRC NCV 2016002–05 Lack of Acceptance Criteria for CNMT Exam
- AR 040009711; OE—Lesson Learned—Limerick RPV Instrument Line Weld Leak
- AR 02173967; NOS ID: Issued Identified in the C1R14 ISI Summary Report
- AR 04012849; C1R17 LL from NRC ISI Inspection

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

- OP–AA–101–111–1001, “Operations Standards and Expectations” Revision 17
- OP–AA–300, “Reactivity Management” Revision 9
- OP–CL–108–101–1003, “Operations Department Standards and Expectation” Revision 35
- TQ–AA–150, “Operator Training Programs” Revision 12
- TQ–AA–155, “Conduct of Simulator Training and Evaluation” Revision 5
- CPS 3001.01, “Preparation for Startup and Approach to Critical,” Revision 28a
- CPS 3001.01C001, “Preparation for Startup Checklist,” Revision 18e
- CPS 3001.01C002, “Mode 2 Checklist,” Revision 17b
- CPS 3002.01, “Heat up and Pressurization,” Revision 32b

- CPS 3002.01C001, “Heat up and Pressurization Checklist,” Revision 10
- CPS 3002.01C002, “Mode 1 Checklist,” Revision 12
- CPS 3002.01F001, “Heat up and Pressurization Flowchart,” Revision 0
- CPS 3004.01, “Turbine Startup and Generator Synchronization,” Revision 33e
- CPS 3005.01, “Unit Power Changes,” Revision 43b

1R12 Maintenance Effectiveness

- AR04020232; NRC–ID M. Rule Failure Classification Approval is Past Due
- CPS 2104.02D001, “HEPA Filter Test Data Sheet”, Revision 7a
- WO 04625004; 0VG08FA Charcoal Inspection EOC from 0VC07SB
- WO 01899000; VQ Train B Ventilation Testing
- WO 01876322; Perform Charcoal Sample, Absorber, Heater, HEPA Filter Tests
- WW-AA-104, “Risk Screening/Mitigation Plan”; Revision 24

1R13 Maintenance Risk Assessments and Emergent Work Control

- AD-AA-3000, “Nuclear Risk Management Process” Revision 1
- ER–AA–600, “Risk Management” Revision 7
- ER–AA–600–1011, “Risk Management Program” Revision 14
- ER–AA–600–1012, “Risk Management Documentation” Revision 12
- ER–AA–600–1014, “Risk Management Configuration Control” Revision 7
- ER–AA–600–1042, “On-line Risk Management” Revision 9
- OP-AA-108-117, “Protected Equipment Program” Revision 4
- WC–AA–104, “Integrated Risk Management” Revision 23
- WC–AA–101, “On-Line Work Control Process” Revision 26
- WC–AA–101-1006, “On-Line Risk Management and Assessment” Revision 2
- AR 04004451, Unplanned Off-Normal Entry 4302.01, Tornado-High Winds
- CPS 4302.01, “Tornado-High Winds’ Revision 22
- OP–AA–108–111–1001, “Severe Weather and Natural Disaster Guidelines” Revision 16
- AR 04022176, “Division 3 SX Pump Tripped During Start Up of 9096.01
- AR 04018770, Pinhole Leak at Pipe Weld for 1SX019B

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
- CC–AA–201; Plant Barrier Control Program; Revision 11
- LS–AA–1400; Event Reportability Guidelines 10 CFR 50.72 and 50.73; Revision 7
- OP–AA–102–106, “Operator Response Time Program,” Revision 4
- OP–AA–102–106, Attachment 1, “Operator Response Time Validation Sheet,” Revision 1
- OP–CL–102–106–1001, “Operator Response Time Program at CPS,” Revision 5a
- OP–CL–102–106–1001, “Operator Response Time Master List at CPS,” Revision 6
- Design Analysis EPU–T0902, Revision 0
- Design Analysis EPU–T0902, Revision 2
- EC 619167, Evaluate Extended SLC Initiation Time, Revision 0
- ANI/ANS 58.8 – 1984, Time Response Design Criteria for Nuclear Safety Related Operator Actions
- AR 02739012, Validation Errors of Operator Response Time Action
- AR 02741339, Past TCAs Conducted with Greater Than minimum Staffing
- AR 03979481, Time Critical Validation of TCA 10 Time Not Met

- AR 03980202, OP–CL–102–106–1001 Times Impacted
- AR 03974026, 0VG01SB, Potential Water Intrusion in Charcoal Bed
- EC 618563, Evaluation of VG B Charcoal Test Results, Revision 0
- AR 04002714, NRC Questions on Invertor Room Cooling
- Design Analysis, VX–01, Switchgear Heat Removal System Cooling Loads
- DWG M05–1001, Standard Symbols Piping and Instrument Diagram, Sheet 2, Revision B
- DWG M05–1115, P&ID Essential Switchgear Heat Removal (VX), Sheet 1, Revision R
- DWG M05–1115, P&ID Essential Switchgear Heat Removal (VX), Sheet 2, Revision M
- AR 03975643, FME Found in NSPS Division 2 Cabinet Bay C
- AR 03999982, Instrument Tubing on VX Condensing Unit Touching Angle Iron
- AR 04018770, Pinhole Leak at Pipe Weld for 1SX019B
- DWG VX–900, Auxiliary Building HVAC Switchgear Heat Removal System Piping, Revision 1
- DWG M10–9115, P&ID/C&I Diagram HVAC Switch Gear Heat Removal System (VX), Revision C

1R18 Plant Modifications

- Calculation SDQ15–24DG09; Mezzanine Floor—Design of crane Girder; Revision 12
- Calculation SDQ15–24DG09; Mezzanine Floor—Design of crane Girder; Revision 13
- Calculation SDQ15–24DG09; Mezzanine Floor—Design of crane Girder; Revision 14
- EC 405113; ISFSI—Replace Fuel Building Crane Rail Clips; Revisions 0, 1
- AR 02690731; NRC Inquiry About Fuel Building Crane Rail Clip Calculation; July 8, 2016
- AR 03959310; Track Resolution of NRC Inquiries; January 4, 2017
- AR 04001089; NRC Cited Violation for Design Control; April 21, 2017
- EC 618150, Main Steam Line (MSL) Flexible Hose Material Upgrade, Revision 0

1R19 Post Maintenance Testing

- MA–AA–716–012; Post Maintenance Testing; Revision 20
- WO 01788731, PMT for 1VX13AB – Verify Proper Operation with No Leaks
- CPS 9381.01C001, “MOV Thermal Overload Bypass Post Maintenance Verification Checklist,” Revision 29
- CPS 8451.04, “Limitorque Operator Removal/Installation,” Revision 16
- MA–AA–723-302, “Installation and Checkout of Quick Stem Sensor (QSS) on Valve Stems,” Revision 5
- WO 01859548, EQ–CL064 Replace Parts After Firing Per 9015.02
- CPS 8120.22C001, “Squib Explosive Valve Checklist,” Revision 12d
- CPS 9015.02, “Standby Liquid Control Injection Operability,” Revision 39
- CPS 9015.02D001, “Standby Liquid Control Injection Operability Data Sheet,” Revision 27
- CPS 9015.06,”Cold Shutdown Standby Liquid Control Pump and Valve Operability Check,” Revision 30a
- CPS 9015.06D001, ”Cold Shutdown Standby Liquid Control Pump and Valve Data Sheet,” Revision 29a
- WO 01872469, Inspect/Clean condenser, Hydro lance tubes
- Magnetic Particle Examination 2017–MT–001 – 1SX22AA and 1SX23CA 3” Flanges
- Magnetic Particle Examination 2017–MT–002 – 1SX22AA and 1SX23CA 3” Flanges (weld Prep)
- CPS 9843.02D001, ”Generic Class 1, 2, and 3 Operational Pressure Test Data Sheet,” Revision 44
- CPS 4003.01C009, ”Division 1 VX Heat Removal Operation,” Revision 1
- AR 04014454, 1VX06CB, Division 2 Shutdown Service Water Condensing Unit End Bells Leak

- WO 01527213, 1B21F047F Replace SRV with a Qualified Spare
- CPS 8216.02, “Safety/Relief Valve Removal and Installation,” Revision 21
- WO 01889917, Leak Rate Test Vessel Pressure Test
- CPS 9059.01, “Reactor Coolant System Leakage Test,” Revision 11a
- CPS 9059.01V001, “Reactor Coolant System Leakage Test Valve Lineup (MCR),” Revision 3
- CPS 3310.01V001, “Reactor Core Isolation Cooling Valve Lineup,” Revision 12e
- CPS 3315.02V001, “Leak Detection Valve Lineup,” Revision 8c
- CPS 3314.01V001, “Standby Liquid Control Valve Lineup,” Revision 10a
- CPS 9059.01V002, “Reactor Coolant System Leakage Test Valve Lineup (Manual Valves – All in the Drywell),” Revision 1
- AR 04015467, Leakage Identified During C1R17 Pressure Test – CRDM 52-29
- AR 04015459, Leakage Identified During C1R17 Pressure Test – CRDM 44-45
- AR 04015456, Leakage Identified During C1R17 Pressure Test – CRDM 36-05
- AR 04015518, Leakage Identified During C1R17 Pressure Test – CRDM 45-25
- EC 619855, Evaluation of CRDM Leakage Identified during Performance of Pressure Test, CPS 9059.01, in C1R17
- AR 04015448, Leakage Identified During C1R17 Pressure Test – 1E21F013
- AR 04015450, Leakage Identified During C1R17 Pressure Test – 1E21F040
- AR 04015455, Leakage Identified During C1R17 Pressure Test – 1B33F052B
- AR 04015466, Leakage Identified During C1R17 Pressure Test – 1B33F052A
- AR 04015468, Leakage Identified During C1R17 Pressure Test – 1B21F357A
- AR 04015517, Leakage Identified During C1R17 Pressure Test – HCU 16-37
- AR 04015486, Leakage Identified During C1R17 Pressure Test – HCU 04-33
- AR 04015526, Leakage Identified During C1R17 Pressure Test – HCU 40-25
- AR 04015457, Leakage Identified During C1R17 Pressure Test – 1C41F006
- AR 04015461, Leakage Identified During C1R17 Pressure Test – 1B21F098D
- AR 04015460, Leakage Identified During C1R17 Pressure Test – 1B21F098B
- AR 04015451, Leakage Identified During C1R17 Pressure Test – 1E51F064
- AR 04015483, Leakage Identified During C1R17 Pressure Test – 1C11F101
- AR 04015484, Leakage Identified During C1R17 Pressure Test – 1C11F1010437
- AR 04015472, Leakage Identified During C1R17 Pressure Test – 1C11HCU4025
- AR 04015482, Leakage Identified During C1R17 Pressure Test – 1C11F321C
- AR 04015494, Leakage Identified During C1R17 Pressure Test – HCU 20-37
- AR 04015498, Leakage Identified During C1R17 Pressure Test – 1C11F126CU
- AR 04015471, Leakage Identified During C1R17 Pressure Test – 1C11F107
- WO 01889967, Operations Control Rod Scram Time Testing
- CPS 9813.01C001, “Control Rod Scram Timing Checklist,” Revision 33
- CPS 9813.01D003, “Scram Time Testing – Containment Data Sheet,” Revision 31
- CPS 9813.01D004, “Scram Time Testing – Main Control Room Data Sheet,” Revision 31b
- CPS 9813.01D005, “Scram Time Testing – Stopwatch,” Revision 31a
- CPS 9813.02D001, “Control Rod Scram Time Option B 20% Insertion Calculation,” Revision 0
- CPS 9813.01D002, “Control Rod Scram Time Option B OLMCPR Calculation,” Revision 31
- WO 04613137, Replace Main Steam Flex Hoses – Extent of Condition
- AR 04008698, White Residue Observed on End Connectors of Flex Hose
- WO 01347633, 1B21F028D Refurbish MSIV – Disassemble Valve – Replace
- WO 01340312, 1B21F022D Contingent Refurb MSIV – Disassemble/Replace/Repair
- CPS 8216.11, “Main Steam Isolation Valve Maintenance,” Revision 27
- CPS 8120.37, “Valve Packing Installation,” Revision 1
- WO 0134630, 1B21F028A Contingent Refurb MSIV – Disassemble/Replace/Repair
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- AR 04011789, VT–3 Indications Identified in the MSIV Body 1B21F028A

- WO 01897599, 1E51F066 Check Valve Inspection per MA–AA–733–1001
- MA–AA–733–1001, “Guidance For Check Valve General Visual Inspection” Revision 10
- CPS 8120.04, “Maintenance of Anchor Darling Tilting Disc Check Valves” Revision 16a
- CPS 8120.04C001, “Maintenance of Anchor Darling Tilting Disc Check Valves Checklist” Revision 13
- CPS 9054.02, “Reactor Core Isolation Cooling Valve Operability Checks’ Revision 42d
- CPS 9054.02D001, “RCIC Valve Operability Data Sheet” Revision 41b
- CPS 9843.01V015, “Leak Rate Testing of RCIC Head Spray (1E51F066)” Revision 25c
- CPS 9843.01D002, “Category A Valve Leak Rate Test Via Flowmeter” Revision 25b
- WO 01858713, EQ–CL050–01 Replace Switch Drain Mechanism and Housing Gasket
- CPS 9054.01C003, “RCIC Low Pressure Operability Checks” Revision 5e
- WO 01862980, RCIC Overspeed Trip Test with Steam
- CPS 9054.01C002, “RCIC High Pressure Operability Checks” Revision 8d
- CPS 9096.01, “Shutdown Service Water Operability Test” Revision 49c
- CPS 2858.01, “Division 3 SX Baseline Testing” Revision 0a
- CPS 2858.01D001, “Division 3 SX Baseline Testing Data Sheet” Revision 0
- CPS 8801.12C001, “Local Mounted Instrument Valve Operation Checklist” Revision 15b
- CPS 9061.09, “MS/FW System Valve Operability (Cold Shutdown)” Revision 36b
- CPS 9061.09D001, “MS/FW System Valve Operability Data Sheet” Revision 32a
- WO 01347633, 1B21F028D Refurbish MSIV
- CPS 9861.04, “MSIV Local leak Rate Test (MC-5,6,7,8)” Revision 27
- CPS 9861.04D001, “MSIV A LLRT Data Sheet (1MC-6)” Revision 27e
- CPS 9861.04D002, “MSIV B LLRT Data Sheet (1MC-8)” Revision 28
- CPS 9861.04D003, “MSIV C LLRT Data Sheet (1MC-5)” Revision 27e
- CPS 9861.04D004, “MSIV D LLRT Data Sheet (1MC-7)” Revision 28
- CPS 3101.01V001, “Main Steam Valve Lineup” Revision 14b
- CPS 3101.01E001, “Main Steam Electrical Lineup” Revision 16
- CPS 3101.01V004, “Main Steam Instrument Valve Lineup” Revision 8
- CPS 3101.01E003, “MSIV Leakage Control (IS) Electrical Lineup” Revision 9b
- CPS 8801.06C001, “H22 Panel Mounted Instrument Valve Operation Checklist” revision 33c
- AR 04018770, Pinhole Leak at Pipe Weld for 1SX019B
- WO 04648253, Pinhole leak at Pipe Weld for 1SX109B
- WO 04624950, Perform Charcoal Leak Test
- CPS 9866.02, “Charcoal Adsorber Leak Test” Revision 33
- CPS 9866.02D001, “Charcoal Adsorber Leak Test Data Sheet” Revision 32
- AR 040004205, Anomaly During VC Charcoal Adsorber Leak Test
- Procedure CPS 9843.02, “Operational Pressure Testing of Class 1, 2 and 3 Systems” Rev. 44
- Procedure CPS 9061.11C017, “Pressure Test of Outboard MSIV 1A Piping” Revision 4a
- Procedure CPS 9061.11C018, “Outboard MSIV IA Check Valve Close Test” Revision 5

<u>1R20 Refueling and Other Outage Activities</u>

- OU–AA–103, “Shutdown Safety Management Program,” Revision 15
- C1R17 Shutdown Safety Management Program
- CPS 9000.06, “Reactor Coolant and Vessel Metal/Pressure/Temperature Limit Logs,” Revision 32
- CPS 9000.06D001, “Heat up/Cooldown, Inservice Leak and Hydrostatic Testing 30 Minute Temperature Log,’ Revision 30b
- CPS 9000.06D003, “Shutdown Cooling Temperature Data Sheet,” Revision 31
- CPS 9000.06D002, “Vessel Head and Shell Flange Temperature Log,” Revision 30
- OP–AA–108–108, “Unit Restart Review (C1R17),” Revision 19

- OP–AA–108–108, “Unit Restart Review (C1F59),” Revision 19
- OP–AA–108–108, “Unit Restart Review (C1F60),” Revision 19
- CPS 3007.01C005, “Operations with a Potential for Draining the Reactor Vessel Checklist,” Revision 3
- LS–AA–119, “Fatigue Management and Work Hour Limits,” Revision 12
- SY–AA–102, “Exelon’s Nuclear Fitness-For-Duty Program,” Revision 19
- SY–AA–102–222, “Handling and Storage of Fitness-For-Duty Records,” Revision 9
- SY–AA–103–513, “Behavioral Observation Program,” Revision 12
- NEI 06–11, “Managing Personnel Fatigue at Nuclear Power Reactor Sites,” Revision 1
- CPS 3001.01, “Preparation for Startup and Approach to Critical,” Revision 28a
- CPS 3001.01C001, “Preparation for Startup Checklist,” Revision 18e
- CPS 3001.01C002, “Mode 2 Checklist,” Revision 17b
- CPS 3002.01, “Heat up and Pressurization,” Revision 32b
- CPS 3002.01C001, “Heat up and Pressurization Checklist,” Revision 10
- CPS 3002.01C002, “Mode 1 Checklist,” Revision 12
- CPS 3002.01F001, “Heat up and Pressurization Flowchart,” Revision 0
- CPS 3004.01, “Turbine Startup and Generator Synchronization,” Revision 33e
- CPS 3005.01, “Unit Power Changes,” Revision 43b
- CPS 3006.01, “Unit Shutdown,” Revision 45a
- CPS 3006.01C001, “Mode 4 Checklist,” Revision 12d
- CPS 3304.04, “Control Rod Exercising Modes 3, 4 or 5,” Revision 2b
- CPS 4005.01, “Loss of Feedwater Heating,” Revision 18d
- CPS 4100.01, “Reactor Scram,” Revision 23a
- OP–AA–108–114, “Post Transient Review,” Revision 12
- OP–AA–300–1540, “Reactivity Management Administration,” Revision 13
- CC–AA–5001, “Post Transient or Scram Walkdown,” Revision 5
- CLP Clinton 1, Clinton Unit 1 Cycle 18 Core Loading Plan, Revision 9
- AR 04020665, Scram Walkdown Results CC-AA-5001
- AR 04012057, 1E22S004102-F01 and F04 1C1 PT Fuses Found Blown
- AR 04012393, Unexpected Division 3 Diesel Generator Start
- WO 4640788, Inspect 4160 Bus 1C1 Main Feed Breaker PT Drawer
- HU–AA–101, “Human Performance Tools and Verification Practices” Revision 9
- HURB, Under-voltage Trip of the Division III Safety Related Bus
- DWG E02–1HP99, High Pressure Core Spray System (HP) HPCS Power Supply System (1E22-1070), Sheet 106
- DWG E02–1HP99, High Pressure Core Spray System (HP) HPCS Power Supply System (1E22–1070), Sheet 107
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- CPS 3006.01F001, “Unit Shutdown Flowchart” Revision 0a
- CPS 3302.02P001, “Reactor Recirculation Hydraulic Power Unit 1A” Revision 0
- OU–CL–104, “Shutdown Management Program Clinton Power Station” Revision 15a
- WO 01889997, 981101 Verify S/D Margin Prior to Core Alts
- Clearance 00138636, C1R17 1SX063A Remove/Install Actuator
- Clearance 00138217, C1R17 RPV Disassembly/Assembly
- AR 04007861, Entered 4008.01 and 4002.01 Due to FCV Runback
- AR 04007832, 1B21-F303B Failed to Close Electrically
- AR 04007827, 1B21F303B Failed to Fully Close from the MCR
- AR 04007862, 1B21F302B Large Packing Leak

1R22 Surveillance Testing

- WO 04634841, EM Channel Functional Division 2 4KV Bus (2nd Level) Under Voltage Relay
- CPS 9333.33, "Division II 4.16KV Degraded Voltage Trip – Functional Test" Revision 6
- CPS 9333.31, "Division II 4.16KV Bus Under Voltage Relay (Loss of Voltage) – Functional Test" Revision 2
- CPS 9333.32, "Division II 4.16KV Main Feed Under Voltage Relay (Loss of Voltage) – Functional Test" Revision 2
- CPS 9333.35, "Division II 4.16KV Bus Under Voltage Relay Calibration with Doble F2000 Test Equipment" Revision 4
- CPS 9333.36, "Division II 4.16KV Main Feed Under Voltage Relay (Loss of Voltage) Calibration with Doble F2000 Test Equipment" Revision 2
- CPS 9333.37, "Division II 4.16KV Reserve Feed Under Voltage Relay (Loss of Voltage) – Functional Test" Revision 2
- WO 01773118, Motor Operated Valve 1SX063A Seating Torque Increase
- CPS 9381.01C001, "MOV Thermal Overload Bypass Post Maintenance Verification Checklist" Revision 29
- CPS 9054.01, "RCIC System Operability Check" Revision 44a
- CPS 9054.01C003, "RCIC Low Pressure Operability Checks" Revision 5e
- CPS 9054.01D003, "RCIC Low Pressure Operability Checks Checklist" Revision 2e
- CPS 9054.01C002, "RCIC High Pressure Operability Checks" Revision 8d
- CPS 9054.01D002, "RCIC High Pressure Operability Checks Checklist" Revision 26b
- CPS 9054.01C001, "RCIC Water leg Operability Test and 1E50-F040 Closure Test and 1SX037 Stroke Timing" Revision 9a
- CPS 9054.01D001, "RCIC Water leg Operability Test and 1E50-F040 Closure Test and 1SX037 Stroke Timing Data Sheet" Revision 48
- CPS 9000.05D001, "Suppression Pool Temperature Log Data Sheet" Revision 27a
- CPS 9052.01, "Low Pressure Core Spray(LPCS)/Residual Heat Removal (RHR) 'A' Pump and LPCS/RHR 'A' Water Leg Pump Operability" Revision 49
- AR 03995612, 0VC07SB Unacceptable Charcoal Test
- CPS 9866.02, "VG/VC Charcoal Absorber Leak Test" Revision
- CPS 9096.01, "Shutdown Service Water Operability Test" Revision 49c
- CPS 9861.04, "MSIV Local leak Rate Test (MC-5,6,7,8)" Revision 27
- CPS 9861.04D001, "MSIV A LLRT Data Sheet (1MC-6)" Revision 27e
- CPS 9861.04D002, "MSIV B LLRT Data Sheet (1MC-8)" Revision 28
- CPS 9861.04D003, "MSIV C LLRT Data Sheet (1MC-5)" Revision 27e
- CPS 9861.04D004, "MSIV D LLRT Data Sheet (1MC-7)" Revision 28
- CPS 3101.01V001, "Main Steam Valve Lineup" Revision 14b
- CPS 3101.01E001, "Main Steam Electrical Lineup" Revision 16
- CPS 3101.01V004, "Main Steam Instrument Valve Lineup" Revision 8
- CPS 3101.01E003, "MSIV Leakage Control (IS) Electrical Lineup" Revision 9b
- CPS 8801.06C001, "H22 Panel Mounted Instrument Valve Operation Checklist" Revision 33c
- WO 01858821, LRT Category A Valve LRT (1E12-F009) RHR SDC Suction
- WO 01858816, LRT Category A Valve LRT (1E12-F008) RHR SDC Suction
- CPS 9843.01D002, "Category A Valve Leak Rate Test Via Flowmeter" Revision 25b
- CPS 9843.01V006, "Leak Rate Testing of RHR Shutdown Suction" Revision 23d
- CPS 1305.01F001, "Type B Local Leak Rate Summary Sheet" Revision 4
- CPS 1305.01F002, "Type C Local Leak Rate Summary Sheet" Revision 7
- CPS 1305.01F003, "Bypass Leakage Summary Sheet" Revision 3
- CPS 1305.01F004, "Summary Report of Local Leak Rate Test Results" Revision 5
- CPS 1305.01F005, "Appendix J Procedure Frequency Update" Revision 1

2RS1 Radiological Hazard Assessment and Exposure Controls

- PI–AA–126–1005–F–01; Pre-NRC Inspection, 71124.01 Radiological Hazard Assessment and Exposure Control; March 8, 2017
- RP–AA–16; ALARA Program; Revision 0
- RP–AA–18; Radiological Posting and Labeling Program Description; Revision 1
- RP–AA–19; High Radiation Area Program Description; Revision 2
- RP–AA–376; Radiological Postings, Labeling and Markings; Revision 9
- RP–AA–376–1001; Radiological Posting, Labeling and Marking Standard; Revision 14
- RP–AA–551; Cobalt Reduction Program; Revision 2
- RP–AA–551–1002; Evaluation and Estimation of Cobalt Introduction into System by Valves; Revision 4
- RP–AA–551–1003; Cobalt Reduction Program Work Process; Revision 7
- RP–AA–870–1001; Set-Up and Operation of Portable Air Filtration Equipment; Revision 7
- RP–AA–870–1002; Use of Vacuum Cleaners in Radiologically Controlled Areas
- RWP CL–1–17–00403; C1R17 ABST FW/MS Activities; Revision 1
- RWP CL–1–17–00413; C1R17 ABST–CTST RT System Activities; Revision 3
- RWP CL–1–17–00501; C1R17 DW Operations Department Activities; Revision 3
- RWP CL–1–17–00506; C1R17 DW Scaffolding Activities; Revision 1
- RWP CL–1–17-00509; C1R17 DW MSIV Activities; Revision 1
- RP–AA–800; Attachment 1, Source Maintenance Record, Radioactive Source Inventory Report, Revision 7; December 21, 2016
- U.S. NRC NSTS Annual Reconciliation Confirmation; Confirmation of Annual Inventory Reconciliation; January 11, 2017
- 2017 DAW 10 CFR 61 Database Analysis
- 2016 Spent Resin 10 CFR 61 Database Analysis
- 2015 Waste Sludge Resin 10 CFR 61 Database Analysis
- Annual Fastscan Calibration; February 2, 2017
- AR 04011815 Report; NRC ID: Boundary Observation; May 17, 2017
- AR 03993493 Report; RP ID: Universal Waste Not Identified or Stored Correctly; April 3, 2017
- AR 03996572 Report; Need More Qualified Techs for Vacuum Cleaner HEPA Filter; April 10, 2017
- AR 04008352 Report; Door Closed but Not Latched; May 9, 2017

2RS2 Occupational As-Low-As-Is-Reasonably-Achievable Planning and Controls

- PI–AA–126–1005–F–01; Pre NRC Inspection IP 71124.02 Occupational ALARA Planning & Control; March 15, 2017
- RP–AA–400; ALARA Program; Revision 14
- RP–AA–400–1006; Outage Exposure Estimating and Tracking; Revision 7
- RP–AA–400–1007; Elevated Dose Rate Response Planning; Revision 2
- RP–AA–400–401; Guidelines for Unconditional Release of Laundered Material; Revision 22
- RP–AA–401–1001; Dose Reporting Guidance; Revision 7
- C1R17 05–05 SAC Committee; Dose Reduction through Ownership and Accountability; May 15, 2017
- Work in Progress Reviews; C1R17; Various Records
- RP–AA–407; Attachment 2, Combined ALARA Plan/Micro-ALARA Plan; RWP Number: CL–1–17–00403; ALARA Plan Number: 17–00403–0; C1R17 Outage ABST FW MS Work Activities
- RWP CL–1–7–00505; C1R17 DW Shielding Activities; Revision 03

2RS3 In-Plant Airborne Radioactivity Control and Mitigation

- RP–AA–441; Evaluation and Selection Process for Radiological Respirator Use; Revision 6
- RP–AA–825; Maintenance, Care, and Inspection of Respiratory Protective Equipment; Revision 8
- RP–AA–870–1001; Set-Up and Operation of Portable Air Filtration Equipment; Revision 7
- RP–CL–825–101; CPS Maintenance and Care of Respiratory Protective Equipment; Revision 22
- SCBA Annual Flow Test Reports; 2016–2017 Data
- SCBA Monthly Inspection Records; May 2017
- SCBA Qualification Matrix; Radiation Protection Department; Undated
- Grade D Air Tests; 2016–2017
- Lesson Plan; Respiratory Protection Equipment—MSA Firehawk SCBA; Revision 2
- Respiratory Protection Inventory/Inspection Records; May 2017
- MSA SCBA Maintenance Training Records; June 22, 2016
- Radiation Protection, Chemistry and Operations Departments SCBA Training Records; Various Records
- HEPA/Vacuum Inspections; May 15, 2017
- AR 04006241; Rad Protection Plans When VQ/VRis Secured During C1R17
- AR 04011975; RP ID: NRC Questioned Personal Air Sampler Left in the Field; May 17, 2017
- AR 04012001; NRC Questions about Licensed Operator Respirator Quals; May 17, 2017

2RS4 Occupational Dose Assessment

- Self-assessment; NRC Inspection Procedures 71124.03 and 71124.04; March 24, 2017
- RP–AA–210; Dosimetry Issue, Usage and Control; Revision 27
- RP–AA–220; Bioassay Program; Revision 12
- RP–AA–222; Methods for Estimating Internal Exposure from in Vivo and in Vitro Bioassay Data; Revision 5
- RP–AA–230; Operation of the Canberra Fastscan Whole-Body Counter (WBC) using Abacos Plus; Revision 3
- RP–AA–270; Prenatal Radiation Exposure; Revision 8
- Multiple Dosimetry EDE Evaluation; 752’ and 777’ C1R17 Drywell Bioshield ISI
- Radiological Technical Evaluation; TLD/ED Evaluation—Period 2017; April 18, 2017
- Fastscan Whole Body Counter Calibration; January 31, 2017
- Declared Pregnant Worker Records; Various Records
- Intake Investigation Forms; Various Records
- Personnel Exposure Investigation Forms; Various Records
- Exelon DLR Performance Validation Data; First through Third Quarters of 2016
- ED Dose Alarm; May 12, 2017
- AR 03955753 Report; RP ID: Dosimetry Check-In Deficiency Identified; December 21, 2016
- AR 02607362 Report; RP ID: SYS Health CRE IR for Emergent Dose; January 4, 2016
- AR 02596923 Report; RP ID: Duplicate Dose Transferred to Sentinel; December 7, 2015

2RS8 Radioactive Solid Waste Processing and Radioactive Material Handling, Storage, and Transportation

- PI–AA–126–1005–F–01; NRC Inspection 71124.08 Radioactive Solid Waste Processing and Radioactive Material Handling, Storage and Transportation; Dated June 5, 2017
- RP–AA–22; Radioactive Material/Waste Transportation/Disposal Program Description; Revision 0
- RP–AA–600; Radioactive Material/Waste Shipments; Revision 16
- RP–AA–600–1001; Exclusive Use and Emergency Response Information; Revision 9
- RP–AA–600–1002; Highway Route Controlled Quantity/Advance Notification for Radioactive/Waste Shipments; Revision 5
- RP–AA–600–1003; Radioactive Waste Shipments to Barnwell and The Defense Consolidation Facility (DCF); Revision 9
- RP–AA–600–1004; Radioactive Waste Shipments to Energy Solutions Clive Utah Disposal Site Containerized Waste Facility; Revision 12
- RP–AA–600–1005; Radioactive Material and Non Disposal Site Waste Shipments; Revision 18
- RP–AA–600–1006; Shipment of Category 1 Quantities of Radioactive Material or Waste (Category 1 RAMQC); Revision 11
- RP–AA–600–1009; Shipment of Category 2 Quantities of Radioactive Material or Waste (Category 2 RAMQC); Revision 2
- 2015 Concentrate Waste 10 CFR 61 Database Analysis
- 2015 Waste Sludge Resin 10 CFR 61 Database Analysis
- 2016 Spent Resin 10 CFR 61 Database Analysis
- 2017 Dry Activated Waste (DAW) Database Analysis
- RP–AA–600–1005; Attachment 6, Radioactive Waste Processor Checklist; Radioactive Shipment Number W16-001; Dated January 6, 2016
- RP–AA–600–1007; Attachment 1, Energy Solutions Bulk Waste Facility Shipment Checklist; Radioactive Waste Shipment Number: W16-016; Dated November 14, 2016
- RP–AA–600–1005; Attachment 6, Radioactive Waste Processor Checklist; Radioactive Shipment Number M16-064; Dated December 19, 2016
- RP–AA–600–1005; Attachment 6, Radioactive Waste Processor Checklist; Radioactive Shipment Number: W17-008; Dated May 18, 2017
- RP–AA–600–1005; Attachment 6, Radioactive Waste Processor Checklist; Radioactive Shipment Number: M17-064; Dated June 22, 2017
- PI–AA–125–1003; Corrective Action Program Evaluation Report; Radioactive Shipment Exceeded Classification Limits; Dated February 24, 2017
- AR 02538433; Enhancements to Sludge Transfers to High Integrity Container; Dated August 6, 2015
- AR 02632615; Housekeeping in RW Shipping Bay; Dated February 27, 2016
- AR 02635440; Enhancements to Pendant Crane Qualifications to Support Rad Shipping; Dated March 3, 2016
- AR 03961544; RP ID: Radioactive Shipment Exceeds Classification Limits; Dated January 10, 2017
- AR 04018643; RP ID: Vendor Paperwork Request Change; Dated June 5, 2017

4OA1 Performance Indicator Verification

- MSPI Derivation Report; MSPI Emergency AC Power System
- MSPI Derivation Report; MSPI High Pressure Injection System
- MSPI Derivation Report; MSPI Heat Removal System

4OA2 Problem Identification and Resolution

- PI–AA–125, “Corrective Action Program (CAP) Procedure” Revision 5
- PI–AA–120, “Issue identification and Screening Process” Revision 7
- PI–AA–125–1001, “Root Cause Analysis Manual” Revision 3
- PI–AA–125–1003, “Corrective Action Program Evaluation Manual” Revision 2
- PI–AA–125–1004, “Effectiveness Review Manual’ Revision 2
- AR 04030777, Adverse Trend Regarding the Timeliness of Operability Determinations
- AR 04008869, Equipment Stored Within 3 Feet of RHR B Ventilation Panel
- AR 04009570, Question Regarding Shift Review of AR 04008869
- AR 03983161, SRI Questions Length of Time on Operability Determination
- AR 03980202, OP-CL-102-106-1001 times Impacted
- AR 03989866, USAR Update for VC Smoke Mode Functional Test
- AR 04008240, Clarify USAR 5.4.7.1.1.3 Description
- AR04016719, Remove Airline Specific Information from USAR
- AR 04021077, Evaluate USAR 15.1 Description of Loss of Feed Water Heating
- AR 04026209, USAR Change package 2005-010 Incorporated in Error
- AR 01604880, 0AP05E Questions Regarding Maintenance Electrical Testing
- CPS 8440.01, “Insulation Testing” Revision 13
- CPS 8440.01, “Insulation Testing” Revision 14
- MA–AA–716–210–1001, “Performance Centered Maintenance” Revision 5
- MA–AA–716–210, “Performance Centered Maintenance (PCM) Process” Revision 0 – 8
- MA–AA–716–230, “Predictive Maintenance Program” Revision 3 – 11
- MA–AA–716–010, “Maintenance Planning” Revision 24
- MA–AA–716–230–1003, “Thermography Program Guide” Revision 4
- ER–AA–200, “Preventive Maintenance Program” Revision 2
- ER–AA–310–1004, “Maintenance Rule-Performance Monitoring” Revision 13
- ER–AA–200–1001, “Equipment Classification” Revision 1 – 2
- LS–AA–110–1003, “Processing of Level 3 OPEX Evaluations” Revision 2 - 3
- WC–AA–120, “Preventive Maintenance Database Revision Requirements” Revision 2
- PA–AA–125–1006, “Investigation Techniques Manual” Revision 2
- DWG E02–1AP70, Shutdown Service Water MCC 1A (1AP29E), Revision N
- CPS 3405.01, “Screenhouse HVAC System” Revision 9b
- CPS 8410.03, “Motor Overload Relay Testing” Revision 11a, 12
- CPS 8410.04, “Molded Case Circuit Breaker/ Bucket Component Functional Testing and Maintenance” Revision 34
- CPS 3211.01, “Shutdown Service Water” Revision 32
- CPS 3211.01C001, “Division 1 SX System Flush Checklist” Revision 12
- EC 619008, Evaluate Survivability of 1SX01PA and Associated Equipment on Loss of SX Room Cooling, Revision 000
- VH–01, Shutdown Service Water Pump Room Cooling Load, Revision 0
- NAI–2300–001, Room Heat up for Clinton Shutdown Service Water Pump Room, Revision 0
- Motor Control Center PCM Template, Revision 1
- Motor Control Center PCM Template, Revision 2
- 1VH02CA Supply Fan Maintenance Strategy
- TR–106857, Preventive Maintenance Basis, Revision 1
- WO 01822399, Perform Thermography on Division 1 Breakers
- WO 00554147, Molded Case Breaker/Bucket Test 1AP29E–3S/1VH01CA
- WO 01923407-01, EM Inspect/ Rework 1AP29E3C, Megger/ Monitor 1VH01CA
- WO 01913189, Flush Division 1 SX System per 3211.01C001
- WO 01885026, Flush Division 1 SX System per 3211.01C001

- WO 01908962, Flush Division 1 SX System per 3211.01C001
- WO 01901295, Flush Division 1 SX System per 3211.01C001
- WO 01897661, Flush Division 1 SX System per 3211.01C001
- WO 01894080, Flush Division 1 SX System per 3211.01C001
- WO 01889530, Flush Division 1 SX System per 3211.01C001
- WO 01890991, Flush Division 1 SX System per 3211.01C001
- WO 01917395, Flush Division 1 SX System per 3211.01C001
- AR 02668083, Incorrect Thermal Heaters Installed
- AR 02667822, Unexpected Alarms 5050–1F/2F and 1VH01CA Trip
- AR 02670965, Test Thermal Overload Relays Removed From 1AP29E–3C
- AR 02668849, NRC Question About Thermography Requirements on 1VH01CA PMT
- AR 01594407, Automatic Trip of Breaker 1AP07EJ

LIST OF ACRONYMS USED

AC	Alternating Current
ACE	Apparent Cause Evaluation
ADAMS	Agencywide Document Access Management System
AICS	American Institute of Steel Construction
ALARA	As-Low-As-Reasonably-Achievable
AR	Action Request
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient without Scram
CAP	Corrective Action Program
CAPR	Corrective Action to Preclude Repetition
CFR	*Code of Federal Regulations*
CPS	Clinton Power Station
DOT	Department of Transportation
EC	Engineering Change
ECAPE	Equipment Corrective Action Program Evaluation
EDG	Emergency Diesel Generator
ERAT	Emergency Reserve Auxiliary Transformer
F	Fahrenheit
HURB	Human Performance Review
IMC	Inspection Manual Chapter
IP	Inspection Procedure
ISI	Inservice Inspection
LPCS	Low Pressure Core Spray
mRem	Millirem
MSHA	Mine Safety and Health Administration
MSPI	Mitigating Systems Performance Index
NCV	Non-Cited Violation
NDE	Non-Destruction Examination
NEI	Nuclear Energy Institute
NIOSH	National Institute of Safety & Health
NRC	U.S. Nuclear Regulatory Commission
OBE	Operating Basis Earthquake
OSP	Outage Safety Plan
PARS	Publicly Available Records System
PI	Performance Indicator
PM	Post Maintenance
PT	Power Transformer
RCE	Root Cause Evaluation
RCR	Root Cause Report
RFO	Refueling Outage
RHR	Residual Heat Removal
RP	Radiation Protection
SBLC	Standby Liquid Control
SCBA	Self-Contained Breathing Apparatus
SDP	Significance Determination Process
SFP	Spent Fuel Pool
SSC	Structure, System, and Component
SX	Service Water
TCA	Time Critical Actions

TS	Technical Specification
TSO	Transmission System Operator
UFSAR	Updated Final Safety Analysis Report
USAR	Updated Safety Analysis Report
UT	Ultrasonic Examination
WO	Work Order