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

February 13, 2017

Mr. Joel P. Gebbie
Senior VP and Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

SUBJECT: DONALD C. COOK NUCLEAR POWER PLANT, UNITS 1 AND 2—NRC INTEGRATED INSPECTION REPORT 05000315/2016004; 05000316/2016004; 05000315/2016501; 05000316/2016501

Dear Mr. Gebbie:

On December 31, 2016, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Donald C. Cook Nuclear Power Plant, Units 1 and 2. On January 19, 2017, NRC inspectors discussed the results of this inspection with yourself and other members of your staff. The enclosed report represents the results of this inspection. The NRC also completed its annual inspection of the Emergency Preparedness Program. This inspection began on January 1, 2016, and issuance of this letter closes Inspection Report Number 2016501.

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

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

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

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, (if any), will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal, privacy or proprietary information so that it can be made available to the Public without redaction.

Sincerely,

/RA/

Kenneth Riemer, Chief
Branch 2
Division of Reactor Projects

Docket Nos. 50–315; 50–316
License Nos. DPR–58; DPR–74

Enclosure:
IR 05000315/2016004; 05000316/2016004;
05000315/2016501; 05000316/2016501

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	05000315; 05000316
License Nos:	DPR–58; DPR–74
Report Nos:	05000315/2016004; 05000316/2016004; 05000315/2016501; 05000316/2016501
Licensee:	Indiana Michigan Power Company
Facility:	Donald C. Cook Nuclear Power Plant, Units 1 and 2
Location:	Bridgman, MI
Dates:	October 1 through December 31, 2016
Inspectors:	J. Ellegood, Senior Resident Inspector T. Taylor, Resident Inspector G. Edwards, Health Physicist E. Fernandez, Reactor Inspector M. Garza, Emergency Preparedness Inspector T. Go, Health Physicist V. Meghani, Reactor Inspector J. Neurauter, Senior Reactor Inspector P. Smagacz, Resident Inspector, Fermi
Approved by:	Kenneth Riemer, Chief Branch 2 Division of Reactor Projects

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SUMMARY

Inspection Report (IR) 05000315/2016004, 05000316/2016004; 05000315/2016501; 05000316/2016501; 10/01/2016 – 12/31/2016; Donald C. Cook Nuclear Power Plant, Units 1 and 2; Outage Activities; Identification and Resolution of Problems

This report covers a 3–month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. One Green finding was identified by the inspectors and one Green finding was self-revealed. One finding involved a Non-Cited Violation (NCV) of the U.S. Nuclear Regulatory Commission (NRC) requirements. The significance of inspection findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," dated April 29, 2015. Cross-cutting aspects are determined using IMC 0310, "Aspects Within the Cross-Cutting Areas," dated December 4, 2014. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated November 1, 2016. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG–1649, "Reactor Oversight Process," dated July 2016.

Cornerstone: Initiating Events

- Green. A self-revealed finding of very low safety significance (Green), occurred on July 6, 2016, when a portion of the Unit 2 Right Moisture Separator Reheater (MSR) 'B' bellows assembly ruptured, causing a steam leak which damaged the adjacent turbine building wall. There were no associated violations of regulatory requirements since the piping was non-safety-related. Reacting to the rupture, operators tripped the reactor and isolated the leak by shutting the Main Steam Isolation Valves. While addressing a number of issues with the MSR's that occurred following a re-design of the internals in 2010, the licensee changed the design of the rods that hold the bellows assembly on each MSR pipe together. The design change called for tack welds to only be used on the end nuts of the rod. Contrary to the design change (EC–51875), tack welds were placed on other nuts as well. The tack welds were determined to have changed the material properties of the rod in the vicinity of the welds, which caused cracking to initiate during operation. Eventually, the cracks grew to a point where two rods completely severed, causing the bellows to tear and rupture. Following the safe shutdown, the licensee repaired the bellows, inspected other rods, and restarted the plant. The issue was entered into their Corrective Action Program (CAP) as Action Request (AR)–2016–7865.

  The issue was more than minor because it adversely affected the Design Control Attribute of the Initiating Events cornerstone because it resulted in a reactor trip and Unusual Event. Per the Significance Determination Process, a detailed risk evaluation was required because during the rupture operators had to close the Main Steam Isolation Valves, which isolated the main condenser (the preferred post-trip decay heat removal path). An NRC Regional Senior Reactor Analyst performed the evaluation and concluded the finding was of very low risk significance (Green). The inspectors determined the finding had an associated cross-cutting aspect in the Human Performance Area, specifically, H.12, Avoid Complacency. Specifically, site personnel did not plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes. (Section 4OA2)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a finding and associated NCV of Technical Specification (TS) 5.4.1 for failing to station a designated individual at the airlocks. Licensee procedure 2–OHP–4030–227–041, Revision 34 required that a designated person be available at the airlock at all times during fuel handling if both air lock doors are open. TS 5.4.1, Procedures, requires, in part, that the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, be established, implemented, and maintained. Regulatory Guide 1.33 states, in part, that general plant operating procedures for refueling and core alterations should be covered by written procedures. Contrary to this requirement, on October 18, 2016, the licensee failed to implement procedure 2–OHP–4030–227–041, "Refueling Integrity." In response to the inspectors concern, the licensee stationed the designated individual. The licensee entered the issue into their CAP as AR–2106–11898.

  The issue screened as more than minor because it adversely affected the Human performance attribute of the barrier integrity cornerstone. The inspectors concluded the issue was of very low safety significance using IMC 0609 Appendix G, Attachment 1 dated May 9, 2014 because the issue did not increase Core Damage Frequency or Large Early Release Frequency. The finding included a cross-cutting aspect of H.9, training, because operations staff had an incorrect understanding of the procedural requirements. (Section 1R20)

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at or near 100 percent power for the entire inspection period.

Unit 2 started a downpower for a refueling outage (RFO) on October 2, 2016.
On October 5, the licensee shutdown Unit 2. Unit 2 remained shutdown for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Winter Seasonal Readiness Preparations

a. Inspection Scope

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

- Fire pumps; and
- Unit 1 and Unit 2 Refueling Water Storage Tanks

This inspection constituted one winter seasonal readiness preparations sample as defined in Inspection Procedure (IP) 71111.01–05.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk-significant systems:

- Unit 1 essential service water system;
- 1 CD emergency diesel generator (EDG) with 1 AB EDG out of service; and
- main fire header with Hydrant 13 tagged out of service.

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, UFSAR, Technical Specification (TS) requirements, outstanding work orders (WOs), condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

.2 Semi-Annual Complete System Walkdown

a. Inspection Scope

During the fourth quarter of 2016, the inspectors performed a complete system alignment inspection of the Unit 2 safety injection system to verify the functional capability of the system. This system was selected because it was considered both safety significant and risk significant in the licensee's probabilistic risk assessment. The inspectors walked down the system to review mechanical and electrical equipment lineups; electrical power availability; system pressure and temperature indications, as appropriate; component labeling; component lubrication; component and equipment cooling; hangers and supports; operability of support systems; and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding WOs was performed to determine whether any deficiencies significantly affected the system function.

In addition, the inspectors reviewed the CAP database to ensure that system equipment alignment problems were being identified and appropriately resolved. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

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

- Unit 1 switch yard gear room;
- Unit 2 switch yard gear room;
- Unit 1 safety-related direct current battery room; and
- Unit 2 safety-related direct current battery room.

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

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

b. Findings

No findings were identified.

.2 Annual Fire Protection Drill Observation (71111.05A)

a. Inspection Scope

On November 29, 2016, the inspectors observed the control room response during a fire drill involving a fire in the fire pump house. Inspectors also attended the critique which also evaluated fire brigade performance. Direct observation of fire brigade performance was accomplished during the third quarter. Based on this observation, the inspectors evaluated the readiness of the plant fire brigade and control room to fight fires. The inspectors verified that the licensee staff identified deficiencies openly discussed them in a self-critical manner at the drill debrief, and took appropriate corrective actions. Specific attributes evaluated were:

- effectiveness of fire brigade leader communications, command, and control;
- utilization of pre-planned strategies; and
- adherence to the pre-planned drill scenario and drill objectives.

Additional attributes were evaluated as part of the third quarter inspection activities. Documents reviewed are listed in the Attachment to this report.

These activities constituted one annual fire protection inspection sample as defined in IP 71111.05–05.

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08)

From October 11, 2016, through December 22, 2016, the inspectors conducted a review of the implementation of the licensee's Inservice Inspection (ISI) Program for monitoring degradation of the Unit 2 reactor coolant system (RCS), steam generator (SG) tubes, emergency feedwater systems, and risk-significant piping and components.

The reviews described in Sections 1R08.1, 1R08.2, R08.3, IR08.4 and 1R08.5 below constituted one inspection sample as described by IP 71111.08–05.

.1 Piping Systems Inservice Inspection

a. Inspection Scope

The inspectors observed and reviewed records for the following non-destructive examinations (NDE) mandated by the American Society of Mechanical Engineers (ASME) Code Section XI (or approved U.S. Nuclear Regulatory Commission (NRC) alternative) 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 NRC requirements.

- Ultrasonic examination (UT) of RCS welds: Pressurizer 2–OME–4; 2–RC–22–SH2; welds 20A (pipe to elbow), 20B (elbow to elbow), and 20C (elbow to elbow) (WO 55471193–04);

- Liquid penetrant examination (PT) of essential service water (ESW) system replacement pipe welds: WO 55416451–01 Weld Map, OW–1 (stainless steel pipe to stainless steel pipe) and OW–2 (stainless steel pipe to carbon steel pipe);
- PT of ESW system replacement flange weld: WO 55463106–27 Weld Map, OW–1 (2–WMO–738 discharge flange); and
- Magnetic particle examination (MT) of #21 steam generator feedwater nozzle weld (WO 55440842–01).

The inspectors reviewed records of the following risk-significant pressure boundary ASME Code Section XI welds fabricated since the beginning of the last refueling outage to determine if the licensee: followed the welding procedure; applied appropriate weld filler material; and implemented the applicable Section XI or construction Code non-destructive examinations and acceptance criteria. Additionally, the inspectors reviewed the welding procedure specification and supporting weld procedure qualification records to determine if the weld procedure was qualified in accordance with the requirements of Construction Code and the ASME Code Section IX.

- ISI Code Class 2: Fabricate and Install new valve and associated piping per EC–53179 – Weld No. OW1, OW2, OW3, OW4, OW5, OW6 and OW7 (WO 5544758–02 and WO 5544758–05);
- ISI Code Class 3: Replace first six feet of piping downstream of pipe elbow downstream of valve 2–WMO–738 in the ESW system – Weld Nos. OW–1 and OW–2 (WO 55416451–01); and
- ISI Code Class 3: Install valve 2–WMO–738 discharge flange in the ESW system – Weld No. OW–1 (WO 55463106–27).

Periodic examination of reactor internal components is a License Renewal commitment. Based on licensee and industry operating experience that has identified baffle-former bolt degradation, the licensee performed UT examination for all baffle-former bolts and visual examination (VT)–3 examinations for all baffle-edge bolts during this Unit 2 RFO. The inspectors observed the NDE for baffle-bolts, components of the reactor coolant support structures, to evaluate compliance with the Materials Reliability Program (MRP)–227–A, “Pressurized Water Reactor Internals Inspection and Evaluation Guidelines,” and MRP–228, “Inspection Standard for PWR Internals,” requirements, and if any degradation and defects were detected, to determine if these were dispositioned within the licensee’s corrective action program.

- UT examination of all 832 baffle-former bolts identified 170 bolts with degradation (9 additional bolts were non-testable, and 2 bolts were previously removed and not replaced in 2010). Six of the bolts with UT identified degradation had been previously replaced in 2010. This result was entered into the licensee’s corrective action program as Action Request (AR) 2016–12216; and
- VT–3 examination of the 1,232 baffle-edge bolts visible from the core side of the baffle plates identified five bolts with degradation. This result was entered into the licensee’s corrective action program as AR 2016–12286.

b. Findings

No findings were identified.

.2 Reactor Pressure Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

For the Unit 2 vessel head, a bare metal visual (C) examination was required this outage pursuant to Title 10 of the *Code of Federal Regulations* (CFR) 50.55a(g)(6)(ii)(D).

The inspectors observed portions of the examination and reviewed the final record for the BMV examination conducted on the Unit 2 reactor vessel head to determine if the activities were conducted in accordance with the requirements of ASME Code Case N–729–1 and 10 CFR 50.55a(g)(6)(ii)(D). In particular, the inspectors confirmed for a sample of penetration locations that:

- the required visual examination scope/coverage was achieved and limitations (if applicable) were recorded in accordance with the licensee procedures;
- the licensee criteria for visual examination quality and instructions for resolving interference and masking issues were adequate; and
- if indications of potential through-wall leakage were identified, the licensee entered the condition into the corrective action system and implemented appropriate corrective actions.

Based upon the licensee's examination, no new relevant indications were accepted for continued service. Therefore, no NRC review was completed for these inspection procedure attributes.

The licensee did not conduct UT on the Unit 2 reactor vessel head penetrations in accordance with the requirements of ASME Code Case N–729–1 and 10 CFR 50.55a(g)(6)(ii)(D). The inspectors verified the NRC approved the licensee's request for use of alternative inservice inspection request 04–02 associated with reactor vessel closure head volumetric/surface examination frequency requirements for the inservice inspection program. The proposed alternative would allow deferral of the volumetric/ surface examinations of each unit's reactor vessel closure head for two refuel cycles beyond the nominal 10–year inservice inspection ISI interval. Specifically, by letter dated June 11, 2015, the NRC staff authorized the one-time use of alternative inservice inspection request 04–02 at Cook Nuclear Plant for the duration up to and including the twenty-fifth RFO for Unit 2 that is scheduled to occur in 2019 during the fourth 10–year ISI interval.

b. Findings

No findings were identified.

.3 Boric Acid Corrosion Control

a. Inspection Scope

The inspectors independently walked down the reactor coolant system loop piping, including the reactor coolant pumps, valves, pressurizer, and emergency core cooling systems within containment to identify boric acid leakage. The inspectors then reviewed the walkdown performed by the licensee to ensure that components with boric acid deposits were identified and entered into the corrective action program.

The inspectors observed these examinations to determine whether the licensee focused on locations where boric acid leaks can cause degradation of safety significant components.

The inspectors reviewed the following licensee evaluations of components with boric acid deposits to determine if the affected components were documented and properly evaluated in the corrective action system. Specifically, the inspectors evaluated the licensee's corrective actions to determine if degraded components met the component Construction Code and/or the ASME Section XI Code.

- AR 2015–04039; Dry Boric Acid on Compression Fitting Near 2–RC–136;
- AR 2015–04045; Dry Boric Acid on 2–RC–101–L3; and
- AR 2016–14247; Boric Acid on Bottom of Unit 2 Reactor Vessel (from Refueling Cavity Leakage).

The inspectors reviewed the following corrective actions related to evidence of boric acid leakage to determine whether the corrective actions completed were consistent with the requirements of the ASME Code Section XI, and 10 CFR Part 50, Appendix B, Criterion XVI:

- AR 2015–05174; Valve 2–QVR–451 Exhibited Packing Leak with Wet Boric Acid;
- AR 2015–05182; Golf Ball Boric Acid Piece Found on Pipe Cap Near 2–SV–100–2;
- AR 2016–14603; Inactive Boric Acid Leak on 2–NFP–221–IH, Associated with Unit 2 Reactor Coolant Transmitter 2–NFP–221; and
- AR 2016–12847; 2–CS–356, Reactor Coolant Pump Seal Water Return Filter Bypass Valve Body to Bonnet Leakage.

b. Findings

No findings were identified.

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

The NRC inspectors observed acquisition and analysis of Eddy Current testing (ET) data, interviewed ET data analysts, and reviewed documentation related to the SG ISI Program to determine if:

- in-situ SG tube pressure testing screening criteria used were consistent with those identified in the Electric Power Research Institute (EPRI) TR–1025132, "SG In-Situ Pressure Test Guidelines," and that these criteria were properly applied to screen degraded SG tubes for in-situ pressure testing;
- the numbers and sizes of SG tube flaws/degradation identified was bounded by the licensee's previous outage operational assessment predictions;
- the SG tube ET examination scope and expansion criteria were sufficient to meet the TS's, and the EPRI 1013706, "Pressurized Water Reactor Steam Generator Examination Guidelines," Revision 7;

- the SG tube ET examination scope included potential areas of tube degradation identified in prior outage SG tube inspections and/or as identified in NRC generic industry operating experience applicable to these SG tubes;
- the licensee identified new tube degradation mechanisms and implemented adequate extent of condition inspection scope and repairs for the new tube degradation mechanism;
- the licensee implemented repair methods which were consistent with the repair processes allowed in the plant TS requirements and to determine if qualified depth sizing methods were applied to degraded tubes accepted for continued service;
- the licensee implemented an inappropriate "plug on detection" tube repair threshold (e.g., no attempt at sizing of flaws to confirm tube integrity);
- the licensee primary-to-secondary leakage (e.g., SG tube leakage) was below 3 gallons per day or the detection threshold during the previous operating cycle;
- the ET probes and equipment configurations used to acquire data from the SG tubes were qualified to detect the known/expected types of SG tube degradation in accordance with Appendix H, "Performance Demonstration for ET Examination," of EPRI 1013706, "Pressurized Water Reactor Steam Generator Examination Guidelines," Revision 7; and
- the licensee performed secondary side SG inspections for location and removal of foreign materials.

The licensee did not perform in-situ pressure testing of SG tubes. Therefore, no NRC review was completed for this inspection attribute.

b. Findings

No findings were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI/SG–related problems entered into the licensee's corrective action program, and conducted interviews with licensee staff to determine if:

- The licensee had established an appropriate threshold for identifying ISI/SG–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 10 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.

In addition, the inspectors performed a review of licensee actions taken to address degraded baffle bolts noted in Section 1R08.1a of this report and entered into the licensee's corrective action program. Specifically, the inspectors observed a sample of baffle-former bolt removals and installation of replacement bolts and evaluated additional licensee actions focused on the: (1) six replacement baffle-former bolts installed in 2010 and identified with degradation in 2016; and (2) the five degraded baffle-edge bolts which the licensee decided to leave in service. In particular, the inspectors observed a portion of the laboratory metallurgical testing performed at a vendor facility, reviewed the results of that testing, reviewed various licensee technical analyses, evaluated licensee compensatory measures, and reviewed the licensee's operability determination for baffle bolt degradation (AR 2016–12216–19).

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

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

a. Inspection Scope

On December 16, 2016, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification training. The inspectors verified that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and 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. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly licensed operator requalification program simulator sample as defined in IP 71111.11–Findings

No findings were identified.

.2 Resident Inspector Quarterly Observation during Periods of Heightened Activity or Risk (71111.11Q)

a. Inspection Scope

On December 29 and 30, 2016, the inspectors observed operators conduct plant heat up and pressurization on Unit 2. This was an activity that required heightened awareness or was related to increased risk. The inspectors evaluated the following areas:

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

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

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

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations

a. Inspection Scope

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

- engineered safety feature ventilation; and
- procurement dedication and annual assessment of maintenance effectiveness.

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

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

The inspector performed a quality review for procurement dedication, as discussed in IP 71111.12, Section 02.02.

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

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

.1 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

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

- emergent repair of reserve feed;
- emergent repair of Unit 1 AB emergency diesel generator (EDG); and
- dual train ESW outage effects on Unit 1.

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

Documents reviewed during this inspection are listed in the Attachment to this report. These maintenance risk assessments and emergent work control activities constituted three 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:

- unsecured argon bottles;
- Unit 2 aggregate operability evaluation; and
- Unit 1 CD EDG fuel pump seizure.

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 Updated Final Safety Analysis Report (UFSAR) to the licensee's evaluations to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

.2 Annual Sample: Review of Operator Workarounds

a. Inspection Scope

The inspectors evaluated the licensee's implementation of their process used to identify, document, track, and resolve operational challenges. Inspection activities included, but were not limited to, a review of the cumulative effects of operator workarounds on system availability and the potential for improper operation of the system, for potential impacts on multiple systems, and on the ability of operators to respond to plant transients or accidents.

The inspectors performed a review of the cumulative effects of operator workarounds. The documents listed in the Attachment were reviewed to accomplish the objectives of the inspection procedure. The inspectors reviewed both current and historical operational challenge records to determine whether the licensee was identifying operator challenges at an appropriate threshold, had entered them into their CAP and proposed or implemented appropriate and timely corrective actions which addressed each issue. Reviews were conducted to determine if any operator challenge could increase the possibility of an Initiating Event, if the challenge was contrary to training, required a change from long-standing operational practices, or created the potential for inappropriate compensatory actions. Additionally, all temporary modifications were reviewed to identify any potential effect on the functionality of Mitigating Systems, impaired access to equipment, or required equipment uses for which the equipment was not designed. Daily plant and equipment status logs, degraded instrument logs, and operator aids or tools being used to compensate for material deficiencies were also assessed to identify any potential sources of unidentified operator workarounds.

This review constituted one operator workaround annual inspection sample as defined in IP 71111.15–02.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

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

- 2AB battery replacement;
- reserve feed cables after crushed by truck (Tan–D and visual results);
- Unit 2 east component cooling water heat exchanger outage restoration; and
- 2–ICM–111 outage work/restoration.

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 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 four 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 continued their evaluation of the Unit 2 refueling outage (RFO) which commenced in October 2016. The outage continued into the first quarter of 2017. During the RFO, the inspectors observed portions of the heatup and pressurization 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;
- installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error;
- 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;
- refueling activities, including fuel handling and sipping to detect fuel assembly leakage;
- walkdown of the 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.

Documents reviewed are listed in the Attachment to this report.

This inspection did not constitute an RFO sample as defined in IP 71111.20–05. The outage continued into the first quarter of 2017.

b. Findings

Introduction: The inspectors identified a finding and associated violation of TS 5.4.1 for failing to follow procedures with respect to stationing personnel at airlocks. Specifically, 2–OHP–4030–227–041, Revision 34 requires that a designated person be available at the air at all times during fuel handling if both air lock doors are open. Contrary to this requirement, the licensee did not have personnel stationed at the air locks.

Discussion: On October 18, 2016, the inspectors assessed the licensee's containment closure capabilities during refueling activities. As part of the inspection, the inspectors asked personnel at both the upper and lower airlocks who the designated person was to close the airlocks. At both locations, personnel at the airlocks were not aware who had been designated to close the airlocks. The inspectors then inquired of on shift operators whom had been designated to close the airlocks. Operators responded that one of the auxiliary operators had been designated but was not required to be at the airlock. The operators also informed the inspectors that the designated individual could be anywhere on site except inside containment. The operators stated that the cavity was flooded, thus providing a long time to boil. The inspectors pointed out that the accident of concern was a fuel handling accident with the release starting immediately. The Licensee concurred with the inspector. Operations reviewed the applicable procedure and concurred with the inspectors that it required the designated person to be at the airlock.

Subsequently, the licensee stated that they met the requirements of the TS in that the technical specifications include a note that states:

> *"Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere via the auxiliary building vent may be unisolated under administrative controls."*

The administrative controls described in the bases state that when both doors are open "a designated individual shall be available at all times during movement of irradiated fuel to close an air lock if required." In addition, the general discussion on containment penetrations states that "specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident." The inspectors concurred with the licensee that neither the TS nor its bases stipulated that a person needed to be at the airlock. However, the inspectors concluded that the site's practice of not restricting the location of the designated individual could lead to the site failing to meet the administrative controls outlined in the TS bases.

Analysis: The inspectors determined the licensee's failure to meet the procedural requirements of 2–OHP–4030–227–041 was a performance deficiency that warranted a significance review. The issue was more than minor because it adversely affected the Human Performance attribute of the Barrier Integrity Cornerstone, whose objective is to provide reasonable assurance that physical design barriers (containment) protect the public from radionuclide releases caused by accidents or events. Specifically, the human performance error of failing to have a designated individual at the airlock created a condition where closure of following a containment evacuation might not be readily achieved. Using Inspection Manual Chapter (IMC) 0609 Appendix G, Attachment 1, dated May 9, 2014, the inspectors answered yes to question B. 6 of Exhibit 4 and determined IMC 0609 Appendix H applied. The finding screened as Green, or very low safety significance, because the finding neither increased CDF nor Increased LERF. Therefore, per Figure 4.1 of IMC 0609, Appendix H, dated May 6, 2004, the finding screens as Green. The inspectors answered 'no' to the Exhibit 2, Section 'A' questions in IMC 0609 Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 1, 2012.

The inspectors determined the finding had an associated cross-cutting aspect in the Human Performance area, specifically, H.9., Training. The licensee determined that Operations staff interpreted the procedure to mean that a designated individual is available and designated to close the airlock.

Enforcement: TS 5.4.1, Procedures, requires, in part, that the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, be established, implemented, and maintained. Regulatory Guide 1.33 states, in part, that general plant operating procedures for refueling and core alterations should be covered by written procedures. Contrary to this requirement, on October 18, 2016, the licensee failed to implement procedure 2–OHP–4030–227–041, "Refueling Integrity." As a result, the licensee compromised containment closure capability since a designated individual was not at the personnel airlock. The licensee documented the issue in action request (AR)–2016–11898. This violation is being treated as a Non-Cited Violation (NCV), consistent with Section 2.3.2.a of the Enforcement Policy.
(NCV 05000316/2016004–01, Designated Individual not at Airlock)

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:

- 2–OHP–4030–219–022E, Unit 2 East essential service water test, (In-Service Test);
- 2–ICM–111 valve timing (In-Service Test);
- 2–EHP–4030–234–203, Unit 2 local leak rate testing, ICM–260, (Containment Isolation Valve);
- 2–OHP–4030–232–217B, DG2AB load sequencing & ESF testing (Routine); and
- Unit 2 Ice Condenser basket weighing (Ice Condenser).

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

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

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

Documents reviewed are listed in the Attachment to this report.

This inspection constituted one routine surveillance testing sample, two in-service test samples, one containment isolation valve sample, and one ice condenser sample as defined in IP 71111.22, Sections–02 and–05. In addition, the inspectors did not identify any performance degradation in the reactor coolant system leakage for the entire cycle. The reactor coolant system leak detection inspection sample was not performed as defined in IP 71111.22, Section–02.

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

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

a. Inspection Scope

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

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

This Emergency Action Level and Emergency Plan Change inspection constituted one sample as defined in IP 71114.04–06.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2RS6 Radioactive Gaseous and Liquid Effluent Treatment (71124.06)

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

a. Inspection Scope

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

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

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

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

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

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

b. Findings

No findings were identified.

.2 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 supplemented those documented in Inspection Report (IR) 2016002 and constituted a complete sample as defined in IP 71124.01–05.

b. Findings

No findings were identified.

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

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

a. Inspection Scope

The inspectors reviewed the radiological administrative, operational, and engineering controls planned for selected radiologically significant work activities and evaluated the integration of these controls and as-low-as-reasonably-achievable (ALARA) requirements into work packages, work procedures and/or radiation work permits.

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.

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

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

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—High Pressure Injection Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index (MSPI)—High Pressure Injection Systems performance indicator (PI) for Units 1 and 2 for the period from the third quarter of 2015 through the second quarter of 2016. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99–02, “Regulatory Assessment Performance Indicator Guideline,” Revision 7, dated August 31, 2013, were used. The inspectors reviewed the licensee’s operator narrative logs, issue reports, MSPI derivation reports, event reports and NRC Integrated IRs for the period of the third quarter of 2015 through the second quarter of 2016 to validate the accuracy of the submittals. The inspectors also reviewed the licensee’s issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index—Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI—Heat Removal System PI for Unit 1 and Unit 2 for the period from the third quarter of 2015 through the second quarter of 2016. 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 IRs for the period of the third quarter of 2015 through the second quarter of 2016 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two MSPI heat removal system samples as defined in IP 71151–05.

b. Findings

No findings were identified.

.3 Mitigating Systems Performance Index—Residual Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI—Residual Heat Removal System PI for Unit 1 and Unit 2 for the period from the third quarter of 2015 through the second quarter of 2016. 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 IRs for the period of the third quarter of 2015 through the second quarter of 2016 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two MSPI—Residual Heat Removal system samples as defined in IP 71151–05.

b. Findings

No findings were identified.

.4 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors sampled licensee submittals for the Occupational Exposure Control Effectiveness PI for the period from the first quarter 2015 through the second quarter 2016. The inspectors used PI definitions and guidance contained in the NEI Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 2013, to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee's assessment of the PI for occupational radiation safety to determine if the indicator related data was adequately assessed and reported. To assess the adequacy of the licensee's PI data collection and analyses, the inspectors discussed with radiation protection staff the scope and breadth of its data review and the results of those reviews. The inspectors independently reviewed electronic personal dosimetry dose rate and accumulated dose alarms and dose reports and the dose assignments for any intakes that occurred during the time period reviewed to determine if there were potentially unrecognized occurrences. The inspectors also conducted walkdowns of numerous locked high and very high radiation area entrances to determine the adequacy of the controls in place for these areas. Documents reviewed are listed in the Attachment to this report.

This inspection activities constituted a complete sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.5 Mitigating Systems Performance Index—Emergency AC Power System

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI—Emergency AC Power System PI for Units 1 and 2 for the period from the fourth quarter 2015 through the third quarter 2016. 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, MSPI derivation reports, issue reports, event reports and NRC Integrated Inspection Reports 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. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

.6 Mitigating Systems Performance Index—Cooling Water Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index—Cooling Water Systems PI for Units 1 and 2 for the period from the fourth quarter 2015 through the third quarter 2016. 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 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. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two MSPI cooling water system samples as defined in IP 71151–05.

b. Findings

No findings were identified.

.7 Reactor Coolant System Leakage

Inspection Scope

The inspectors sampled licensee submittals for the reactor coolant system (RCS) Leakage PI Donald C. Cook Units 1 and 2 for the period from the fourth quarter of 2015 through the third quarter of 2016. 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 logs, RCS leakage tracking data, issue reports, event reports and NRC Integrated Inspection Reports for the period of October 2015 through September of 2016, to validate the accuracy of the submittals. The inspectors also reviewed the licensee’s issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two reactor coolant system leakage samples as defined in IP 71151–05.

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 corrective action program 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 April 1, 2016, through September 30, 2016, although some examples expanded beyond those dates where the scope of the trend warranted.

The review also included issues documented outside the corrective action program 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 corrective action program trending reports. Corrective actions associated with a sample of the issues identified in the licensee's trending reports were reviewed for adequacy.

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: Rupture of Bellows on Unit 2 Right Moisture Separator Reheater

a. Inspection Scope

The inspectors completed the review of the Unit 2 Moisture Separator Reheater (MSR) bellows rupture which began in the third quarter. Observations were provided in the third quarter report (IR 05000316/2016003). Upon completion of their inspection activities, the inspectors identified a finding as discussed below.

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

b. Findings

Introduction: A finding of very low safety significance was self-revealed when an expansion bellows on the Unit 2 Right MSR ruptured, causing a steam leak which resulted in a manual trip of the reactor.

Description: On July 6, 2016, the balance bellows on piping that connects the Unit 2 Right MSR to the ‘B’ low pressure turbine suddenly failed. Unit 2 has two MSR’s which function to reheat some of the steam used to drive the main turbine to improve efficiency. They also help improve the lifetime and reliability of the components in the turbine. The MSR’s each have three outlets (‘A’, ‘B’, and ‘C’…with ‘B’ in the middle) which go to the ‘A’, ‘B’, and ‘C’ low pressure turbines, respectfully. A ‘bellows assembly,’ consisting of multiple, accordion-like segments, is provided on each line to allow some relative motion between the turbine and MSR and to account for the forces on the piping due to the steam flow through the system. Part of the bellows assembly extends out from an elbow in the pipe and is held in place by four tie rods. On July 6, 2016, two of the tie rods broke, which allowed the bellows to tear and release steam from the system into the turbine building. The steam blew a hole in the turbine building wall nearby. Reacting to the rupture, the operators manually tripped the reactor and isolated the steam leak by shutting the Main Steam Isolation Valves. An Unusual Event was declared, which was terminated shortly thereafter when the plant was verified safely shutdown with the leak isolated. The plant remained in a safe shutdown condition until the repairs were completed and the unit restarted.

The inspectors reviewed the root cause performed by the licensee. In October 2010, the internals of the Unit 2 MSR’s were replaced with a different design. In February 2012, a tie rod on the ‘B’ Right MSR failed which resulted in a tear in the bellows with a resultant steam leak. This leak did not require a shutdown and was managed until the April 2012 outage. Of note, the tie rods had a different design in 2012. The licensee took action to modify the tie rod design, going from a pipe-like rod to a threaded rod. This was installed on failed ‘B’ Right line during repairs. The other lines had the modified rod placed inside the old pipe design, effectively resulting in both the old and new design rod being in place at the same time (the new design being a ‘backup’ to the old). In October 2012, the ‘B’ Left MSR line developed a steam leak which also did not require a plant shutdown. Despite the ‘backup’ threaded rod, the old design rod failed and caused enough movement to still tear part of the bellows.

Both issues were attributed to poor original welds from initial construction. In October 2013, the 'B' Left MSR was repaired during the outage, and by then the old design rods had been completely removed and replaced solely with the threaded rods throughout the system (except for the 'A' Right MSR bellows assembly, whose scheduled one-time replacement had not occurred yet).

In February 2014, plant personnel observed that the 'B' lines on the MSR's appeared to be vibrating excessively as compared to the 'A' and 'C' lines. In June 2014, a leak occurred on a different part of the bellows assembly than had been experienced in the past on the 'B' Right line. The station managed this leakage until the spring 2015 outage. During this time, the licensee started to focus attention on the line-vibrations and ways the vibrations could be reduced. In the spring 2015 outage, a support was added to the 'B' MSR lines, which underwent a further modification after vibrations were still thought to be too high. The site continued to gather vibration data but the results were not formally documented. After the 2015 outage, the MSR's were removed from the plant's "Top 10" list, which is a list used to focus resources on enhancing a listed system's reliability.

Failure analysis of the 2016 rupture by the licensee concluded two tie rods cracked, allowing the forces within the pipe to rip the outermost, or balance bellows, apart. When the licensee went to the threaded rod design, which was documented in Engineering Change (EC) 51875, the design called for tack welds to be applied to the rod to hold the outermost bolts on. Based on interviews conducted by the inspectors, personnel involved with the EC development knew that the tack welds should only be applied to the outermost nuts because that portion of the rod was not under stress from the piping and vibrations. The threaded rod was made of a material not conducive to welding. Welding could alter the mechanical properties of the rod, making it more hard/brittle. However, the weld evaluation that was used to put the rods together indicated that tack welds could be placed at any of the nuts along the rod. Contrary to the EC, this is how the threaded rods were constructed. The failure analysis report concluded the cracking of the tie rods initiated from locations where tack welds had been inappropriately placed. The inspectors determined that while the vibrations likely played a part in the various issues experienced by the MSR's since the MSR internals design change in 2010 (along with a loss of focus on the pursuit of an ultimate solution to the vibration issues), the installation of tack welds exacerbated the situation and created a vulnerability which directly led to the failure of two of the rods, causing the rupture on July 6, 2016.

Analysis: The failure to follow approved engineering change documents (EC–51875) by installing tack welds in the wrong locations was a performance deficiency. The issue was more than minor because it adversely affected the Design Control Attribute of the Initiating Events cornerstone because it resulted in a reactor trip and Unusual Event. Per Inspection Manual Chapter (IMC) 0609 Appendix A, "The Significance Determination Process for Findings At-Power," Exhibit 1, question A.2., a detailed risk evaluation was required because the transient required operators to shut the Main Steam Isolation Valves, which isolated the Main Condenser. The Senior Reactor Analyst (SRA) used the Donald C. Cook Standardized Plant Analysis Risk (SPAR) Model, revision 8.22 to complete the evaluation. The impact to the plant as a result of the performance deficiency was a loss of condenser heat sink initiating event. The event was set to "True" and the conditional core damage probability (CCDP) was calculated. The result was a CCDP of 1E–6. The SRA reviewed a similar calculation the licensee performed using the DC Cook plant-specific PRA model. The SRA determined that there were some differences between the results of the two models for the evaluation of this finding, primarily due to the probable risk analysis modeling of recently installed reactor coolant pump shutdown seals. The NRC has not yet updated SPAR models with the shutdown seals, as the NRC continues to review industry information before endorsing the proposed shutdown seal reliability model. However, the SRA determined that the seals would provide additional risk benefit not currently in the SPAR model and the finding is best characterized as having a CCDP of less than 1E–6, which represents a finding of very low safety significance (Green). Dominant sequences involved a loss of condenser heat sink event followed by a failure of the reactor protection system resulting in an anticipated transient without scram and a loss of condenser heat sink event followed by the failure of auxiliary feedwater and feed and bleed.

The inspectors determined the finding had an associated cross-cutting aspect in the Human Performance Area, specifically, H.12, Avoid Complacency. Specifically, site personnel did not plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes. In this case, while vibrations were noted to be an issue and some effort had been expended in trying to resolve the issue, eventually the focus was lost by removing the system from the 'Top–10' list and results from vibration monitoring were not formally documented/assessed. Further, the 'worst-case' scenario of a complete rupture was not considered when assessing actions to address the vibration issues.

Enforcement. Since the MSR's are not safety-related, the inspectors did not identify a violation of regulatory requirements associated with this finding.
(FIN 05000316/2016004–02, Moisture Separator Reheater Rupture)

.4 Annual Follow-Up of Selected Issues: Baffle bolt Failures and Corrective Actions.

a. Inspection Scope

The inspectors reviewed the technical aspects of failures of baffle bolts on Unit 2 in 2010 as well as two corrective actions taken by the licensee. In October of 2010, the licensee identified foreign material on the core plate of Unit 2. The licensee removed the material and identified the source as failing baffle bolts. Subsequently, the licensee performed a visual inspection of the baffle bolts and replaced those which had failed; however, two of the failed bolts were not replaced, although the damaged bolts were removed.

In 2016, Indian Point Energy Center and Salem Nuclear Generating Station identified similar failures. As a result of this operating experience, the licensee performed extensive examination of the baffle bolts during the 2016 Unit 2 refueling outage (RFO). Unlike in 2010, this examination included ultrasonic evaluation of the baffle bolts in addition to a visual inspection. The examination revealed 170 failed bolts, including some previously replaced in 2010. The licensee replaced the affected bolts and reexamined the failure mechanism identified in 2010.

In 2010, the licensee performed a root cause analysis that included a technical evaluation performed by Westinghouse (reference, WCAP–17352–P). The evaluation concluded that the bolts failed as a result of Irradiation Assisted Stress Corrosion Cracking (IASCC) coupled with loss of preload. As stated, in 2010 the licensee performed a visual inspection only, due to the lack of a qualified ultrasonic method at the time. During the 2016 inspection, the ultrasonic inspection identified instances of cracking in some bolts which had previously been evaluated as acceptable during the 2010 visual inspection. Specifically, the cracking occurred at the bolt head to shank interface and could not be detected visually until complete failure had occurred. However, the cracking resulted in additional load being placed on the remaining intact bolts (including those which had been replaced) eventually resulting in failure. The licensee concluded that the original determination that the bolt failure was caused by IASCC was valid, but that the additional load on the intact bolts was not recognized in 2010, due to the limitations of the visual inspection, which had not identified all instances of cracking.

As stated, in 2010 the licensee had left two bolts holes empty. This was because replacement bolts and needed equipment were not available to install new bolts. As part of this decision, the licensee contracted with Westinghouse to evaluate the operational impact of the missing bolts. This evaluation concluded that the fuel assemblies would not be subjected to fretting. An evaluation of operating data of fuel stored in these same locations also showed no impingement had occurred during prior operating cycles. In 2016, the licensee identified two failed fuel assemblies during the Unit 2 core offload. The fuel failures occurred in the area with the missing bolts. The licensee concluded that jet impingement through the bolt holes had damaged the fuel. Specifically, the 2010 analysis had not accounted for the failure of the replaced bolts from the additional loading described above. In the 2010 analysis, Westinghouse had assumed that the replaced bolts would remain intact thereby preventing fretting. The inspectors concluded that the decision made in 2010 to not replace the bolts was reasonable given the extent of the bolt failures recognized at the time.

In 2010, the licensee had also performed visual inspections of the Unit 1 baffle bolts following the Unit 2RFO. This examination did not identify any instances of cracked/failed bolts. The licensee planned to perform visual and ultrasonic examinations of the Unit 1 bolts during the 2017RFO. This was consistent with both industry guidance that was developed following the 2010 identification of the failed bolts on Donald C. Cook, Unit 2 and the subsequent identification of failed bolts at Indian Point Energy Center and Salem Nuclear Generating Station.

This was also consistent with the licensee's aging management program that was approved as part of the licensee's license renewal with the NRC. Since Unit 1 had no identified instances of failed bolts and the licensee was following industry and licensing requirements, the inspectors concluded that there was no immediate safety concern with the integrity of the baffle bolts on Unit 1 or the licensee's timeline to perform the associated inspections.

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

.5 Annual Follow-Up of Selected Issues: Follow-Up on NRC inspection findings from Component Design Bases Inspection associated with Emergency Diesel Generators

a. Inspection Scope

The inspectors reviewed the licensee's response to two previous NRC findings associated with emergency diesel generator (EDG) systems.

The findings were identified by the NRC Component Design Bases Inspection (CDBI) team and documented as Non-Cited Violations (NCV's) in IR 2015008. Specifically, issues with the testing requirements of the air start systems and an issue with fuel oil storage capacity were reviewed.

Regarding certain check valves in the air start systems, the CDBI team identified that the allowed leakage values for the inservice testing program did not account for the fact that the air system might be relied upon for an attempted EDG start following a Station Blackout (SBO). During an SBO, both offsite and onsite emergency AC power sources are assumed to be not available for a period of four hours. Following that time, per the Cook licensing basis, the SBO event is assumed to end once AC power (either onsite or offsite) is restored. The licensee's allowed leakage values would have depleted the EDG air start receivers during a four hour SBO to the point that sufficient energy would not be available to start an EDG, if that were the option utilized to exit the SBO. For corrective actions, the licensee instituted a more conservative allowed-leakage value. In their review of the licensee's corrective actions, the inspectors noted that beyond the inservice test (which checks leakage past specific valves), any leakage from the system could jeopardize functionality of the EDGs. Therefore, any time leakage was discovered, the amount would have to be quantified utilizing the more conservative value the licensee used to respond to the specific CDBI issue. The inspectors noted that the licensee had evaluated this concern under action request (AR)–2016–4337. The AR contained a corrective action to look at several alternatives to address this issue. However, pending selection of a suitable alternative, the inspectors questioned how such leakage would be evaluated in the interim, if discovered. The inspectors discussed this with operations staff, who happened to be aware of the issue and stated they would take into account the new, conservative leakage value for any leakage that happened to be found on the system. The inspectors questioned whether it was necessary to formally capture this in an operations guidance document. This issue was briefed at the exit meeting and the licensee indicated they would evaluate the concern.

The inspectors also looked at a finding associated with the EDG Fuel Oil Storage Tanks. Each of the two storage tanks on site supplies two EDGs, one on each unit. In 2012, the NRC approved Task Interface Agreement 2012–11, "Licensing Basis for Donald C. Cook Nuclear Power Plant, Units 1 and 2, During a Steam Generator Tube Rupture Event Coincident with a Loss of Offsite Power (LOOP)." This TIA established that a LOOP was a station, not an individual unit, event. As a result, the CDBI team discovered that licensee calculations on availability of fuel for the required seven day time period were insufficient given two EDGs would be running off each tank instead of the originally-assumed single EDG. The inspectors reviewed corrective actions from this violation. The inspectors determined the current corrective actions were adequate pending final resolution. The licensee completed changes to EDG operational procedures to reflect a need to conserve fuel. Further solutions to the problem are still being explored by the licensee. The inspectors identified that when the TIA was approved, the licensee had an opportunity to perform an extent of condition to ensure other aspects of equipment operation throughout the site weren't impacted by the outcome of the TIA. The inspectors looked to see if the licensee had added actions to address this in light of the CDBI finding and discovered the licensee was in the process of performing an in-depth engineering change to explore other impacted equipment. The inspectors determined the licensee was taking appropriate action moving forward.

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

b. Findings

No findings were identified.

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

.1 (Closed) Licensee Event Report 05000315/2015003–00: Main Feed Pump Technical Specification 3.3.2 Violation

On June 18, 2015, the licensee completed a review of NRC Information Notice 2015–05, "Inoperability of Auxiliary and Emergency Feedwater Auto-Start Circuits on Loss of Main Feedwater." The notice described a circuit design issue at several plants which would preclude an automatic start of auxiliary feedwater on loss of a main feedwater pump during routine startup of the main feedwater system. The condition existed while one main feed pump was feeding generators and the other one was being started (with circuitry in 'reset' during the pump start sequence). The review identified that Unit 1 was susceptible to the issue in the Information Notice. As a result, the licensee discovered numerous times during the previous three years when the appropriate actions stipulated in the Technical Specifications were not followed due to the circuit design issue. The licensee subsequently submitted a license amendment request to correct the issue. The submittal was approved by the NRC.

In their review, the inspectors noted the licensee had reviewed another nuclear plant's response to existing operating experience about the circuitry in 2012 and had entered the issue into their CAP as GT–2012–10495. However, the licensee's review focused only on performance of the operating experience program, not the underlying technical issue. The inspectors concluded the licensee reasonably should have identified the issue during the 2012 review. Utilizing IMC 0612 Appendix B, "Issue Screening," the inspectors determined the failure to identify the circuit design issue per site procedure PMP–7030–OE–001, "Operating Experience Program," was a performance deficiency. The inspectors concluded the issue was minor as the underlying technical specification violation could be dispositioned as an old design issue. Further, the technical specification function was not credited in the safety analyses. Finally, the approved license amendment which corrected the issue resulted in allowed operation of the system similar to that experienced by the licensee before the design issue was discovered.

Documents reviewed are listed in the Attachment to this report. This licensee event report (LER) is closed.

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

4OA5 Other Activities

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

a. Inspection Scope

The NRC performed Inspection Procedure 92723, “Follow UP Inspection for Three or More Severity Level IV Traditional Enforcement Violations in the Same Area in a 12–Month Period,” in accordance with the assessment letter dated August 31, 2016. The licensee received three Severity Level–IV violations in the traditional enforcement area of impeding regulatory process for failure to perform adequate *Code of Federal Regulations* (10 CFR) 50.59 evaluations. The NRC reviewed the licensee’s corrective action documents for each violation and the overall cause analysis for the following items:

- problem identification;
- cause, extent of condition and extent of cause; and
- evaluation of corrective actions.

b. Findings and Observations

The inspection’s results concluded that the licensee did not fully evaluate or understand the adverse impacts of the changes being performed through the 10 CFR 50.59 process. This included failures to follow a rigorous process along with making assumptions without validation or attention to detail. The licensee identified these gaps and have instituted corrective actions, including, but not limited to, procedure changes, emergency preparedness supervisor reviews for changes to the emergency plan, and the establishment of a 50.59 screen and evaluation review board. All inspection items were met by the licensee.

No findings were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

On January 19, 2016, the inspectors presented the inspection results to Mr. J. Gebbie, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meetings

Interim exits were conducted for:

- The inspection results for the Radiation Safety Program review with Mr. J. Ross, Plant Manager, on October 21, 2016;
- The results of the Emergency Preparedness Program inspection with Mr. R. Seiber, Emergency Preparedness Manager, conducted over the phone on November 8, 2016;

- The results of the Inservice Inspection (ISI) were discussed with Mr. J. Gebbie, and other members of the licensee staff on December 22, 2016; and
- The results for the 92723 inspection with Mr. J. Gebbie and other members of the licensee staff on November 18, 2016.

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

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

J. Gebbie, Chief Nuclear Officer
K. Baker, Manager, Design Engineering – Electrical
M. Ellet, Regulatory Affairs - Compliance
H. Ellison, ISI Program Owner
S. Erickson, Emergency Planning Specialist
K. Harper, Regulatory Affairs
M. Hoholek, Regulatory Affairs
E. Hoskin, Senior Design Engineer– Electrical
H. Kish, Regulatory Affairs Supervisor
M. McLean, Radiation Protection General Supervisor Operations
S. Mitchell, Supervisor, Regulatory Affairs - Compliance
P. Monk, Steam Generators Lead
J. Petro, Design Engineering Director
S. Petro, Baffle Bolt Engineer
B. Roger, Instrumentation and ALARA Supervisor
R. Sieber, Emergency Preparedness Manager
K. Simpson, Emergency Preparedness Supervisor
M. Scarpello, Regulatory Affairs Manager
D. Wood, Radiation Protection Manager

U.S. Nuclear Regulatory Commission

K. Riemer, Chief, Reactor Projects Branch 2
L. Kozak, Senior Reactor Analyst
M. Holmberg, Reactor Inspector
A. Dietrich, Project Manager
J. Cassidy, Senior Health Physicist

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000316/2016004–01	NCV	Designated Individual not at Airlock (Section 1R20)
05000316/2016004–02	FIN	Moisture Separator Reheater Rupture (Section 4OA2.3)

Closed

05000316/2016004–01	NCV	Designated Individual not at Airlock (1R20)
05000316/2016004–02	FIN	Moisture Separator Reheater Rupture (4OA2.3)
05000315/2015003–00	LER	Main Feed Pump Technical Specification 3.3.2 Violation (Section 4OA3)

Discussed

None

LIST OF DOCUMENTS REVIEWED

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

1R01 Adverse Weather Protection

- 12-OHP-4022-001-010; Severe Weather; Revision 19
- AR-2016-12437; 2-HV-EH-30 Will Not Turn on; October 28, 2016
- AR-2016-5610; U1 RWST Valve Enclosure Degradation; May 3, 2016
- Cook Seasonal Readiness Affirmation Letter; November 15, 2016
- PMP-5055-001-001, Winterization/Summerization; Revision 26
- WO 55491762; 2-HV-EH-30 Investigate/Repair

1R04 Equipment Alignment

- 12-5261B-5; Yard Piping for Fire Protection Tanks & Puimp House Units 1 & 2; January 9, 2013
- 12-OHP-4021-019-001; Operation of the Essential Service Water System; Revision 61
- 1-OHP-4021-032-008CD; Operative DG1CD Subsystems; Revision 34
- 2-OHP-4021-001-001; Plant Heatup from Cold Shutdown to Hot Standby
- 2-OHP-4021-008-002; Placing Emergency Core Cooling System in Standby Readiness; Revision 30
- 2-OHP-4021-008-002; Placing Emergency Core Cooling System in Standby Readiness; Revision 30
- AR 2015-15545; Unit 2 Safety Injection Header Pressure Response to Accumulator #24 Fill; December 2, 2015
- AR 2015-8680; Discoloration of Unit 1 South Safety Injection Pump Shaft Seal Components; July 3, 2015
- AR 2016-13587; 2-IMO-361 Leaking Between Body and Bonnet; December 12, 2016
- AR 2016-14184; Packing Leak on 2-IMO-325; December 13, 2016
- AR 2016-14188; NRC Walkdown on 2-IMO-315; December 14, 2016
- AR 2016-14608; Loose Pipe Support on 2S Safety Injection Pump; December 22, 2016
- OP-12-5152-15; Flow Diagram Fire Protection – Water Yard Piping Unit 1 & 2; October 15, 2015
- OP-12-5152T-14; Flow Diagram Fire Protection – Water Piping in Pump House Floor Elevation 598’-0” Units 1 & 2; March 19, 2009
- OP-2-5142-53; Flow Diagram Emergency Core Cooling (SIS); November 30, 2016
- OP-2-5143-74; Flow Diagram Emergency Core Cooling (RHR) Unit No. 2; March 11, 2016
- OP-2-5143A-5; Flow Diagram Emergency Core Cooling (RHR) Accumulator Piping Unit No. 2; December 22, 2015
- Open Corrective Work Order List; Unit 2 Safety Injection System
- Plant Status Report; Monday, November 28, 2016
- Unit 2 Operator Burden Report; December 12, 2016
- WO 55425652; 12-NAPL-NAPL, ESY Essential Service Water Pipe Tunnel Re-Inspect Section; October 14, 2015

- WO 55425652; Make Identified Repairs to Essential Service Water Pipe Tunnel – South 576' Unit 2; August 13, 2016
- WO 55425782; Perform Identified Repairs to Essential Service Water PI; February 12, 2014
- WO 5542782; Perform Identified Repairs to Essential Service Water Pipe Tunnel Main 570'; September 3, 2016
- WO 55460773; 12-BLDG-Turbine, MIS Repair Coatings on Pipe in the Essential Service Water Pipe Tunnel; March 12, 2015
- WO 55460773; Paint Beautification of Unit 1 and 2 CCP, SI and Reciprocal Pump Rooms; February 3, 2016

1R05 Fire Protection

- Fire Pre-Plans; Volume 1, Revision 28
- AR-2016-13628; Failure to Meet Fire Drill Performance Criteria to De-Energize Component; December 12, 2016
- FBD-416-001-B; Fire in the Fire Pump House; June 15, 2016

1R08 Inservice Inspection Activities (71111.08)

- AR 2015-04039; Dry Boric Acid on Compression Fitting Near 2-RC-136; March 26, 2015
- AR 2015-04045; Dry Boric Acid on 2-RC-101-L3; March 26, 2015
- AR 2015-05174; Valve 2-QVR-451 Exhibited Packing Leak with Wet Boric Acid; April 11, 2015
- AR 2015-05182; Golf Ball Boric Acid Piece Found on Pipe Cap Near 2-SV-100-2; April 12, 2015
- AR 2016-09073; Potential Baffle-Former Bolt (BFB) Degradation; August 9, 2016
- AR 2016-12116; Discrepancies Between Work Order Weld Blocks and Qualifications; October 20, 2016
- AR 2016-12121; Lockbar Found on 2-OME-1 Lower Core Plate; October 21, 2016
- AR 2016-12216; U2C23 Baffle-Former Bolt UT Inspection Results; October 23, 2016
- AR 2016-12286; Cracked Baffle-Edge Bolts Identified During VT-3; October 25, 2016
- AR-2016-12847; 2-CS-356, Reactor Coolant Pump Seal Water Return Filter Bypass Valve Body to Bonnet Leakage; Dated November 5, 2016
- AR-2016-14247, Boric Acid on Bottom of Unit 2 Reactor Vessel (from Refueling Cavity Leakage); Dated December 15, 2016
- AR-2016-14603, Inactive Boric Acid Leak on 2-NFP-221-IH, Associated with Unit 2 Reactor Coolant Transmitter 2-NFP-221; Dated December 20, 2016
- AREVA Document 51-9263363-000; Engineering Information Record: D. C. Cook U2C23 Steam Generator Condition Monitoring and Operational Assessment; Revision 0; November 11, 2016 [Proprietary]
- NDE Report U2-MT-16-001; Magnetic Particle Examination on 2-STM-21-FWN; October 14, 2016
- NDE Report U2-VE-16-008; Ultrasonic Examination on 2-RC-22-20B; October 18, 2016
- NDE Report U2-VE-16-009; Ultrasonic Examination on 2-RC-22-20C; October 18, 2016
- NDE Report U2-VE-16-019; Ultrasonic Examination on 2-RC-22-20A; October 18, 2016
- Procedure 12-QHP-5050-NDE-002: Magnetic Particle Examination; Revision 7
- Procedure 12-QHP-5050-NDE-027; Visual Examination for Boric Acid and Condition of Component Surface; Revision 4
- Procedure LMT-10-PAUT-02; Manual Phased Array Ultrasonic Examination of Austenitic and Ferritic Piping Welds; Revision 0
- Procedure PMI-5070; Inservice Inspection; Revision 22
- Procedure PMP-3140-CON-003; Oversight of Contractors; Revision 33

- UTC 0001339526; WO Task 55488446-17: VT-1 Examination - Unit 2 Replacement Baffle Bolt; November 7, 2016
- UTC 0001339528; WO Task 55488446-17: VT-1 Examination - Unit 2 Replacement Baffle Bolt; November 7, 2016
- UTC 0001339530; WO Task 55488446-17: VT-1 Examination - Unit 2 Replacement Baffle Bolt; November 7; 2016
- UTC 0001339531; WO Task 55488446-17: VT-1 Examination - Unit 2 Replacement Baffle Bolt; November 7; 2016
- UTC 0001339532; WO Task 55488446-17: VT-1 Examination - Unit 2 Replacement Baffle Bolt; November 7; 2016
- UTC 0001360637; WO Task 55488446-17: VT-1 Examination - Unit 2 Replacement Baffle Bolt; November 7; 2016
- Westinghouse Document MCOE-TR-16-16; Examination and Testing of Replacement Baffle-Former Bolts from D.C. Cook Unit 2; Revision 0 [Proprietary]
- Westinghouse Letter LTR-PL-16-49 Revision 0; Subject: D. C. Cook Unit 1 Engineering Evaluations Supporting Extent of Condition Review; December 2, 2016 [Proprietary]
- Westinghouse Letter LTR-PL-16-52 Revision 0; Subject: Operability Assessment for Postulated Primary Side Loose Parts from Degraded Reactor Internals Baffle Plate Edge Bolts at D. C. Cook Unit 2; December 2, 2016 [Proprietary]
- Westinghouse Letter LTR-PL-16-84 Revision 1; Subject: D. C. Cook Unit 2 Baffle Bolting Replacement Pattern Summary Letter; December 7, 2016 [Proprietary]
- Westinghouse Letter LTR-PL-16-85 Revision 0; Subject: Assessment Summary for Replacement Bolt Failures at D. C. Cook Unit 2; December 2, 2016 [Proprietary]
- Westinghouse Letter LTR-PL-16-88 Revision 0; Subject: D. C. Cook Unit 2 Baffle-Edge Bolt Summary; December 2, 2016 [Proprietary]
- Westinghouse Letter LTR-PL-16-90 Revision 0; Subject: D. C. Cook Unit 2 Baffle-Former Bolt Evaluation for Normal Loads; December 2, 2016 [Proprietary]
- Westinghouse Letter LTR-PL-16-96 Revision 0; Subject: Description of Events Leading to Baffle-Former Bolt Degradation at D. C. Cook Unit 2; December 2, 2016 [Proprietary]
- WO 55313916-08; Perform VT-2 Bare Metal VT of Reactor Vessel Closure Head per Code Case N-729-1; Completion Date March 29, 2012
- WO 55416451-01; Replace Six Feet of Piping Downstream of 2-WMO-738 with Stainless Steel; Completion Date October 28, 2016
- WO 55440758-02; Pre-Fab Assembly for Valve 2-CC-05; Completion Date December 15, 2014
- WO 55440758-05; Install Valve Assembly 2-CS-314; Completion Date April 6, 2015
- WO 55440842-01; Perform MT ISI Inspection, Nozzle to Shell Weld for #21 Steam Generator Feedwater Nozzle; Completion Date October 18, 2016
- WO 55462921-01;Perform Direct Visual Examination of the Lower Vessel Bottom Mounted Instrumentation Penetrations in Accordance with 12-QHP-5050-027; Completion Date October 10, 2016
- WO 55463106-27; Reinstall Discharge (Outlet) Flange Associated with 2-WMO-738; Completion Date October 27, 2016
- WO 55471193-04; EISI: 2-NMO-151, Vendor Examination on 2-RC-22-SH2, 20A, B, C; Completion Date October 14, 2016
- WO 55488281-02; Perform VT-2 Bare Metal VT of Reactor Vessel Closure Head per Code Case N-729-1; Completion Date October 12, 2016

IR11 Licensed Operator Requalification Program

- 12-EHP-4030-002-356; Low Power Physics Testing with Dynamic Rod Worth; Revision 14
- AR 2016-14012; 12-EHP-4030-002-356 Procedure Enhancements; December 9, 2016

1R12 Maintenance Effectiveness

- AR 2015-15099; Maintenance Rule (a)(1) Action Plan; June 22, 2016
- Evaluation 11194; Pitot Tube Sampling Port; Revision 0
- Evaluation 15018; Shaft for 24 Inch AL Fixed Spray Strainer; Revision 0
- Dedication Plan PS-0045; C&K Components Miniature Toggle Switch; Revision 0
- Evaluation 24263; O-Ring for 550-5 Sight Glass; Revision 0
- Dedication Plan HP-0076; Diesel Driven Fuel Oil Transfer Pump Couplings; Revision 7
- Evaluation 13206; Float, Type 304, Stainless Steel; Revision 0
- Evaluation 17983; Kit, Accessory, Koppers Fast Size 3,5 Inch Type BS Coupling; Revision 0
- Evaluation 16904; Valve, 3-Way Spring Returned Normally Closed, Pilot Operated, Viton Seals, Revision 0
- D.C. Cook Periodic Assessment of Maintenance Effectiveness Report; February 16, 2016

1R13 Maintenance Risk Assessments and Emergent Work Control

- 12-OHP-4021-019-001; Operation of the Essential Service Water System; Revision 61
- Fork Lift Recovery Plan
- Plant Status Report; Tuesday, October 18, 2016
- PMP-2291-OLR-001; On-Line Risk Management; Revision 39
- WOER 20016537; Projects Construction is Requesting an Engineering Evaluation to Drive a Second “Big Red” Forklift to Recover the First “Big Red” Forklift

1R15 Operability Determinations and Functional Assessments

- AR 2016-10994; Portable Teletower Found in Unit 2 CD DG Room Not IAW Procedure
- AR 2016-11092; Potential Interaction with Unit 1 SSC; October 6, 2016
- AR 2016-11092; Potential Seismic Interaction with Unit 1 SSC; October 6, 2016
- AR 2016-12154; U2C23 Aggregate ODE; October 22, 2016
- AR 2016-4732; 3R Fuel Pump Froze Up on 1CD Diesel; April 17, 2916
- AR 2016-8003; Unit 1 CD EDG Seized Fuel Injection Pump Failure Analysis; July 8, 2016
- AR 2016-9198; NRC Concerns with Initial Response to AR 2016-8003; August 12, 2016
- AR-2014-1865; Unit 1 Unexpected Alarm Generator Stator High Temp; February 7, 2014
- AR-2016-13344; 2-NTA-251, P’ZR Liquid Temp Failed High; November 19, 2016
- AR-2016-4200; Unit 1 West CTS Heat Exchanger Troubleshooter Results; April 7, 2016
- Commercial Grade Item Dedication Evaluation 00011651; Emergency Diesel Generator (EDG) Diesel Fuel Injection Pumps (DFIP)AR 00125367; 2CD EDG Fuel Injection Pump Seizure; April 19, 2006
- GT-2016-1327; Resolution to MRB 20: Switchgear Room Ventilation; February 3, 2016
- Haynes Corporation Inspection Checklist – As Built Form, Purchase Order Number: 1545241 (CO024948) FDX Series Pump, Part Number: 10-73422-46
- ISO 4406; 1999 Code Chart
- NUREG-0302; Remarks Presented (Questions/Answers Discussed) at Public Regional Meetings to Discuss Regulations (10 CFR Part 21) for Reporting of Defects and Noncompliance, Revision 1
- PMP-4010-OWA-001; Oversight and Control of Operator Burden; Revision 11

1R19 Post-Maintenance Testing

- 12-EHP-8913-001-002; Heat Exchanger Inspection; Revision 11
- 12-IHP-4030-082-003; AB, CD and N-Train Battery Discharge Test and 24 Month Surveillance Requirements; Revision 33
- 12-IHP-5030-EMP-014; MOV Diagnostic Testing; Revision 21
- 12-MHP-5030-016-001; Component Cooling Water Heat Exchanger Inspection, Cleaning and Tube Plugging; Revision 18
- 2-E-N-ELCP-250-007; 250 VDC Battery 2AB System Analysis; Revision 14
- 2-E-N-ELCP-250-007; Battery Voltage Profile; Attachment H, Revision 0
- 2-ICM-111; Journeyman Worksheet – II; Revision A
- AR 2016-11751; 2-BATT-AB Batteries Installed Different than Vendor Drawings; October 15, 2016
- AR 2016-11868; 2-batt-ab Discharge Failed; October 19, 2016
- AR 2016-12568; 2-HE-15E Tube Plugging Margin Exceeded; October 31, 2016
- AR 2016-12845; 2-HE-15E Tubesheet Degradation, November 8; 2016
- AR 2016-13238; Found Leaking Valves on Post-Maintenance Test for 2-HE-15E; November 17, 2016
- AR-2016-11843; Obtain and Document SQ Report QR2-28441-01; October 17, 2016
- DIT-B-03537-00; Leak Testing the Shell Side of the Component Cooling Water Heat Exchanger Following Tube Plugging; April 8, 2013
- M-8655; Battery Arrangement 2-Step EP3 (28) LC-25 Cells; Revision 0K-7051, Battery Arrangement 2 Step EP III (116) LC-25 Cells; June 6, 1984
- MWO-55266421-22; Perform Static Pressure Test/Leak Check
- WO 55335376; 1-IMO-255 Bit Inlet Leakby
- WO 55436121; MTM, 2-ICM-111, Repack Valve
- WO 55436121-01; Repack Valve
- WO 55436121-14; 2-ICM-111, (PM) Inspect W/RHR Inservice Before Reload, December 10, 2016
- WO 55463592-45; Remove Degraded Areas and Restore via Weld Buildup
- WO 55463592-45; Section XI Repair/Replacement Plan

1R20 Outage Activities

- 02-OHL-5030-SOM-007; Unit 2 Tours – Unit 2 Auxiliary Tour, Revision 32
- 12-EHP-4030-002-356; Low Power Physics Tests with Dynamic Rod Worth Measurement, Revision 14
- 12-EHP-4050-FHP-301; Core Reload
- 12-IHP-6030-IMP-155; Mid-Loop Monitoring System Setup and Calibration; Revision 16
- 12-MHP-4050-FHP-005; Core Unload/Reload and Incore Shuffle; Revision 2
- 12-OHP-4021-001-004; Plant Cooldown from Hot Standby to Cold Shutdown; Revision 68, Data Sheet 1, LTOP Tracking
- 12-OHP-4021-018-002; Placing in Service and Operating the Spent Fuel Pit Cooling and Cleanup System; Revision 30
- 12-OHP-4022-018-001; Loss of Spent Fuel Pit Cooling; Revision 22
- 12-OHP-4050-FHP-001; Refueling Procedure Guidelines; Revision 32
- 12-OHP-4050-FHP-016; Nozzle Inspection Cover and Nuclear Instrumentation Cover Flood up Preparation; Revision 1
- 2-OHP-4021-001-001; Plant Heatup from Cold Shutdown to Hot Standby; Revision 86
- 2-OHP-4021-001-002; Reactor Start-Up; Revision 60
- 2-OHP-4021—001-004; Plant Cooldown from Hot Standby to Cold Shutdown; Revision 67

- 2-OHP-4021-002-001; Filling and Venting the Reactor Coolant System; Revision 33
- 2-OHP-4021-017-001; Operation of the Residual Heat Removal System; Revision 24
- 2-OHP-4030-001-002; Containment Inspection Tours; Revision 36
- 2-OHP-4030-214-030; Daily and Shiftly Surveillance Checks; Revision 30, Data Sheet 20, LTOP Verification
- 2-OHP-4030-214-030; Daily and Shiftly Surveillance Checks; Revision 30, Data Sheet 20A, LTOP Verification LCO 3.4.12.A
- 2-OHP-4030-214-030; Daily and Shiftly Surveillance Checks; Revision 30
- 2-OHP-4030-227-041; Refueling Integrity; Revision 34
- 2-OHP-4030-277-04; Refueling Integrity; Revision 35
- 2-TM-16-48-R0; Remove Fuses to Allow Manual Reset of Rod Control Counters; December 31, 2016
- 2-WD-824; Containment Unit 2; Revision 1
- 2-WD-825; Containment Unit 2
- AR 2016-10795; Spent Fuel Pit Filter Needs Changed; September 27, 2016
- AR 2016-12847; Body to Bonnet Leak 2-CS-356; November 5, 2016
- AR 2016-14490; Unit 2 AB RAT Load Tap Changer Malfunction; December 18, 2016
- AR 2016-14502; Broken Bolt in Vertical Missile Block Hole; December 19,2016
- AR 2016-1898; Discrepancy Between NRC SE re: Airlock Closure and Procedure; October 18, 2016
- AR 2017-0334; Working Hour Violation; January 10, 2017
- Clearance: R-CCW-LTCN-0875(004) 2-CRV-470; December 16, 2016
- CL-R-NEWS-41-0807; Clearance on Containment Penetration 2-CPN-84; October 10, 2016
- Containment Access Logs December 20 Through December 28, 2016
- Donald C. Cook Nuclear Plant Unit 2 Cycle 23 Core Operating Limits Report; Revision 0
- February 26, 2016
- GT 2013-19441; Biodiesel 5 Percent Maybe Unavoidable in the Long Term; December 12, 2013
- GT 2016-11441; Enable Acceptance of Biodiesel Blended Fuel for EDG's; October 11, 2016
- MHI-5097; Medium Voltage Cable Testing, Revision 6
- OP-12-5136-27; Flow Diagram Spent Fuel Pit Cooling & Cleanup Unit 1 & 2,
- OP-2-5114A-38; Flow Diagram, Non-Essential Service Water; May 31, 2016
- OP-2-5120S-12; Control Air System Auxiliary Building Tapoffs Unit 2; March 20, 2013
- OP-2-5128-30; Flow Diagram Reactor Coolant Unit No 2 Sheet 1 of 2; September 14, 2013
- OP-2-5135C-; Flow Diagram Component Cooling Water Miscellaneous Services Auxiliary Building; June 22, 2014
- PMP-4100-SDR-002; Outage Risk Assessment and Management; Revision 8
- U2C23 Shutdown Safety Plan Report
- WO 55480211; 2-OME-25, Disassemble, NDE, & Reassemble (R2P)
- IPTE Brief; Low Power Physics Testing; September 12, 2016
- IPTE Brief; Unit 2 Cycle 23 Rod Drop Time Testing; September 12, 2016

1R22   Surveillance Testing

- 12-EHP-4030-010-262; Ice Condenser Surveillance and Operability Evaluation; Revision 21
- 12-MHP-4030-010-001; Ice Condenser Basket Weighing Surveillance; Revision 21
- 2-EHP-4030-234-203; Unit 2 Local Leak Rate Testing, Revision 24
- 2-OHP-4030-119-022W; East Essential Service Water System Test; Revision 30
- 2-OHP-4030-217-054V; RHR Valve Operability Test, Revision 1
- 2-OHP-4030-232-217A; DG2CD Load Sequencing & ESF Testing; Revision 48
- 2-OHP-4030-232-217B; DG2AB Load Sequencing and ESF Testing; Revision 50

- 50.59 Evaluation; 2014-0454-00; Revise Unit 1 Ice Basket Weight Acceptance Criteria for Unit 1 Cycle 26
- 50.59 Screen; 2014-0454-00; Revise Unit 1 Ice Basket Weight Acceptance Criteria for Unit 1 Cycle 26
- Amendment No. 60 to Facility Operating License No. DPR-74; November 28, 1983
- EHI-5071; Inservice Testing program Implementation; Revision 18
- Ice Condenser Technical Specification Statistical Analysis, DC Cook Unit 2, Cycle 23
- PMP-4030-EXE-001; Conduct of Surveillance Testing; Revision 23
- Revision 3 to the Ice Condenser Utility Group Topical Report No. UCUG-001: Application of the Active Ice Mass Management Concept to the Ice Condenser Ice Mass Technical Specification; October 23, 2003
- WO 55436121-05; OPS: 2-ICM-111 Perform IST Time Stroke; November 14, 2016

1EP4 Emergency Action Level and Emergency Plan Changes

- 10CFR 50.54(q) Effectiveness Evaluation Form 15-22; Revision 1; March 17, 2016
- 10CFR 50.54(q) Screening Form 16-04; February 26, 2016
- 10CFR 50.54(q) Screening Form 16-09, Revision 1; April 14, 2016
- D. C. Cook Nuclear Plant Emergency Action Levels; Revisions 18 and 19
- D. C. Cook Nuclear Plant Emergency Plan; Revisions 35, 36, 37

2RS1 Radiological Hazard Assessment and Exposure Controls

- 12-THP-6010-RPP-104; Personnel Dosimetry Use Varying Radiation Fields; Revision No. 16
- 12-THP-6010-RPP-400; Radiological Protection Job Coverage; Revision 22
- 12-THP-6010-RPP-420; Radiological Controls for Radiography; Revision 7
- 12-THP-6010-RPP-420; Radiological Data Sheet on Radiography Activity in U-2 Auxiliary Building 612' Elevation; October 19, 2016
- 12-THP-6010-RPP-703 Data Sheet 1; Personnel Contamination Log PCL#29 at U-2 Upper Containment; October 6, 2016
- 12-THP-6010-RPP-703 Data Sheet 1; Personnel Contamination Log PCL#30 at U-2 Upper Containment; October 10, 2016
- 12-THP-6010-RPP-703 Data Sheet 1; Personnel Contamination Log PCL#31 at U-2 SI Pump, October 12, 2016
- 12-THP-6010-RPP-703 Data Sheet 1; Personnel Contamination Log PCL#32 at Main Steam Regulating Valve Area; October 16, 2016
- 12-THP-6010-RPP-703 Data Sheet 1; Personnel Contamination Log PCL#33 at 609' Aux Building U-2 CCW HX Area; October 18, 2016
- AR-2016-10301; Elevated Tritium Level in Unit-1 Annulus; September 12, 2016
- AR-2016-10379, Received Un-Briefed to Dose Rate Alarm, August 24, 2016
- AR-2016-10414; Radiological Posting of Areas that had Tritium Result Greater than 0.3 DAC Value; September 12, 2016
- AR-2016-10634; Unanticipated Dose Rate Alarm for RP Technician Performing LHRA Boundary Watch; September 22, 2016
- AR-2016-10639; RP Air Sample was Logged in the Air Sampler Program as Greater than 0.3 DAC but was Not Counted Using Gamma Spec Per IAW12-THP-6010-RPP-405; September 15, 2016
- AR-2016-12204; Potential Violation of PMP-6010-RPP-003 LHRA and Technical Specification 5.7.2, October 23, 2016
- AR-2016-5756; 12 Containers were Not Tagged in Accordance with the Procedure 12-THP-6010-RPP-301; May 6, 2016

- AR-2016-8483; Rad Workers Exiting the RCA Not Meeting Requirements; July 21, 2016
- AR-2016-9059; Rad Workers Not Surveying Personal Items Prior to Exiting RCA; August 8, 2016
- AR-2016-9497; During the Execution of RP HRA /LHRA and VHRA, the VHRA Key was Inadvertently Stored in an LHRA Locked Cabinet; August 19, 2016
- D.C. Cook; Radiation Protection Fundamentals Self-Assessment GT-2015-16529
- PMP-6010-RPP-003; High, Locked High, and Very High Radiation Area Access; Revision 27
- RWP-162100; Unit-2 Refuel Cavity Decontamination Activities that Included Support Work; Revision 0
- RWP-162101; Unit-2 Refuel Prep Activities and Disassembly; Revision 2
- RWP-162102; Unit-2 Refuel Restoration Activities; Revision 0
- RWP-162105; Unit-2 C23 Reactor Baffle Bolting Inspection; Revision 0
- RWP-162130; Unit-2 Perform Radiography in Auxiliary and Turbine Building and Planned Radiologically Controlled Area, Revision 0
- RWP-162148; Unit-2 Steam Generator Primary Platform Activities Including Support Work; Revision 0
- Supplemental Guidance in Response to North Access Building Radon Inversion Event; October 15, 2016
- VSDS Standard Map Survey Report; SW_VSDS-M-20160819 of 617' Demineralizers; August 19, 2016

2RS2 Occupational ALARA Planning and Controls

- ALARA Committee Meeting-A16-43F; October 19, 2016
- ALARA Plan for RWP-162100; Unit-2 Refuel Cavity Decontamination Activities that Included Support Work; Revision 0
- ALARA Plan for RWP-162101; Unit-2 Refuel Prep Activities and Disassembly; Revision 0
- ALARA Plan for RWP-162102; Unit-2 Refuel Restoration Activities; Revision 0
- ALARA Plan for RWP-162105; Unit-2 C23 Reactor Baffle Bolting Inspection; Revision 0
- ALARA Plan for RWP-162130; Unit-2 Perform Radiography in Aux and Turbine Building and Planned Radiologically Controlled Area; Revision 0
- ALARA Plan for RWP-162148; Unit-2 Steam Generator Primary–Platform Activities Including for Support Work; Revision 0
- PMP-6010-ALA-001; ALARA Program Review of Plant Work Activities; Revision 31

4OA1 Performance Indicator Verification

- 12-EHP-7110-ROP-001; Engineering ROP Performance Indicators; Revision 4
- AR-2015-16408; CNAQ-Condition Not Adverse to Quality; December 23, 2015
- PMP-7110-PIP-001; Reactor Oversight Performance Indicators and Monthly Operating Report Data; Reviewed from the Fourth Quarter 2015 Through the Third Quarter 2015
- PMP-7110-PIP-001; Reactor Oversight Program Performance Indicators and Monthly Operating Report Data; Revision 15
- PMP-7110-PIP-001; Occupational Exposure Effectiveness Data Sheet 15 "Source Data Pertinent in Determining the Values"; Reviewed from the First Quarter 2015 Through the Second Quarter 2016
- PRA-MSPI-Basis; Revision 12
- Reactor Coolant System Leakage Rate Database; December 28, 2016
- Two-Year Unavailability Report for the Aux Building Vent System; December 9, 2016
- Two-Year Unavailability Report for the Aux Building Vent System; December 9, 2016
- WO 55227019-01; 2-ITR-311 PM to Perform End of Life Replacement; June 30, 2014
- WO 55227019-01; 2-ITR-311 PM to Perform End of Life Replacement; June 30, 2014

- WO 55227021-01; 2-ITR-321 PM to Perform End of Life Replacement; May 16, 2016
- WO 55227021-01; 2-ITR-321 PM to Perform End of Life Replacement; May 16, 2016
- WO 55465008-01; MTI, 2-IHP-4030-202-002A, RVLIS Train ‘A’ Wide Range Pressure Calibration; May 5, 2016
- WO 55465008-01; MTI, 2-IHP-4030-202-002A, RVLIS Train ‘A’ Wide Range Pressure Calibration; May 5, 2016

4OA2 Identification and Resolution of Problems

- 1-OHP-4021-032-001AB; Diesel Generator 1AB Operation; Revision 37
- 1-OHP-4023-ECA-0.0; Loss of All Alternating Current Power; Revision 36
- 1-OHP-4023-ES-0.1; Reactor Trip Response; Revision 30
- AR 2010-10940; Debris Found in 2-OME-1 on the Core Plate; October 16, 2010
- AR 2010-12310; Degraded Baffle Bolt Lock Bars Found After Bolt Replacement; November 15, 2010
- AR 2015-14802; Component Design Bases Inspection 2015 – Emergency Diesel Generators Air Receivers Surveillance Criteria; November 13, 2015
- AR 2015-15019; 2015 Component Design Bases Inspection Fuel Oil Storage Tank Seven Day Supply for Emergency Diesel Generator; November 18, 2015
- AR 2015-15249; Evaluate Need to Update UFSAR for Dual Unit Loss of Offsite Power; November 24, 2015
- AR 2016-10111; NRC Observation on Vibration Related Equipment Failures, September 7, 2016
- AR 2016-11493; Unit 2 Cycle 22 Fuel Defect; November 12, 2016
- AR 2016-12216; U2C23 Baffle Former Bolt UT Inspection Results; October 23, 2016
- AR 2016-12788; 1-DG-255C Valve Leaking By; November 3, 2016
- AR 2016-3776; Foreign Material Identified and Removed from U1 Refuel Cavity, March 30, 2016
- AR 2016-4337; Application of MD-12-DG-014-N to Entire Emergency Diesel Generator Air System; April 9, 2016
- Finding PAO-16-04-02; Foreign Material Exclusion Deficiencies; April 18, 2016
- Management Review Meeting Summary Package; November 18, 2016
- PAO-16-02-01; Identification and Implementation of Augmented Quality Attributes; February 23, 2016
- PAO-16-09-01; Discrepant/Non-Conforming Conditions; September 2, 2016
- Plant Health Committee Top Ten Equipment Issues; December 7, 2016
- Task Interface Agreement – Licensing Basis for D. C. Cook Nuclear Power Plant, Units 1 and 2, During a Steam Generator Tube Rupture Event Coincident with a Loss of Offsite Power, (TIA 2012-11); December 7, 2012
- NF-AE-16-97; D.C. Cook Unit 2 Cycle 22 Baffle Former Bolt Exclusion Relative to Leaking Fuel Location; November 23, 2016

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

- AR 2015-8122; PODE Needed to Ensure Compliance to Technical Specification 3.3.2, Function 6g; June 18, 2015\
- GT 2015-6774; NRC Information Notice 2015-05; May 18, 2015
- GT-2012-10495; EO-CNPOE Number: OE36359, Loss of Main Feedwater Pump Logic Inadvertently Bypassed During Plant Startup; August 26, 2012
- LER 05000315/2015-003-00; Main Feed Pump Technical Specification; August 8, 2015

- NRC Information Notice 2015-05; Inoperability of Auxiliary and Emergency Feedwater Auto-Start Circuits on Loss of Main Feedwater Pumps; May 12, 2015

4OA5 Other Activities

- 12-EHP-5043-EDC-001; Evaluation of Discrepant Conditions; Revision 22
- AR 2014-15685; Potential EP Finding
- AR 2014-2980; 50.54(q) Evaluation Adverse to Quality
- AR 2014-3688; NRC Observation Regarding Performance of 50.59 Products
- AR 2014-3789; Removal of TRM 8.4.3 was Not Adequately Evaluated
- AR 2015-12261; NRC Question on Deletion of TRM 8.4.3
- AR 2015-14259; Determine Current License Basis for U1/U2 HSD Panels
- AR 2015-2386; TPD-600-EPC Revision Not Evaluated
- AR 2015-8008; Review/Compare CEP Items
- AR 2015-9584; Hot Shutdown Panel Tech Spec/UFSAR Removal
- AR 2016-2026; NRC Traditional Enforcement Violations
- AR 2016-3703; ACE May Have Reached Incorrect Conclusion
- AR 2016-9941-3; ACE on NRC Concerns with Completeness and Accuracy
- D. C. Cook Emergency Plan; Revision 37
- DTG-SRT-001; 50.59 Review Team; Revision 0
- Performance Assurance Audit PA-15-02: Emergency Preparedness
- PMI-2351; 10 CFR 50.59 and 10 CFR 72.48 Program Administration; Revision 0
- PMP-2080-EPP-200; Initiating Changes to the Emergency Plan or Emergency Plan Implementing Procedures; Revision 8
- PMP-2350-SES-001; 10 CFR 50.59 Reviews; Revision 18
- PMP-7030-CAP-001; Action Initiation; Revision 35
- PMP-7030-CAP-002; Condition Action and Closure; Revision 30
- PMP-7030-CAP-003; Conduct of Condition Evaluations; Revision 8
- PMP-7030-CAP-004; Conduct of Effectiveness Reviews; Revision 6
- PMP-7030-CAP-005; Conduct of Causal Evaluations; Revision 10
- PMP-7030-CAP-006; Conduct of Beyond Design Basis Evaluations; Revision 1
- PMP-7030-MOP-001; Corrective Action Program Management Oversight Process; Revision 24
- PMP-7030-OE-001; Operating Experience Program; Revision 29
- RMA-2080-EPA-008; Emergency Plan Management; Revision 19

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
ALARA	As-Low-As-Reasonably-Achievable
AR	Action Request
ASME	American Society of Mechanical Engineers
BMV	Bare Metal Visual
CAP	Corrective Action Program
CCDP	Conditional Core Damage Probability
CDBI	Component Design Bases Inspection
CFR	*Code of Federal Regulations*
EC	Engineering Change
EDG	Emergency Diesel Generator
EPRI	Electric Power Research Institute
ESW	Essential Service Water
ET	Eddy Current Testing
IASCC	Irradiation Assisted Corrosion Cracking
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
ISI	Inservice Inspection
LER	Licensee Event Report
LOOP	Loss of Offsite Power
MRP	Materials Reliability Program
MSPI	Mitigating Systems Performance Index
MSR	Moisture Separator Reheater
MT	Magnetic Particle Examination
NCV	Non-Cited Violation
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
OSP	Outage Safety Plan
PI	Performance Indicator
PT	Liquid Penetrant Examination
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RFO	Refueling Outage
SBO	Station Blackout
SDP	Significance Determination process
SG	Steam Generator
SPAR	Standardized Plant Analysis Risk
SRA	Senior Reactor Analyst
SSC	Structures, Systems, and Components
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Examination
VT	Visual Examination
WO	Work Order

Letter to Joel P. Gebbie from Kenneth Riemer dated February 13, 2017

SUBJECT: DONALD C. COOK NUCLEAR POWER PLANT, UNITS 1 AND 2—NRC INTEGRATED INSPECTION REPORT 05000315/2016004; 05000316/2016004; 05000315/2016501; 05000316/2016501

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