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

February 12, 2020

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 PLANT, UNITS 1 AND 2 – INTEGRATED INSPECTION REPORT 05000315/2019004 AND 05000316/2019004

Dear Mr. Gebbie:

On December 31, 2019, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Donald C. Cook Nuclear Plant, Units 1 and 2. On January 8, 2020, the NRC inspectors discussed the results of this inspection with you and members of your staff. The results of this inspection are documented in the enclosed report.

Four findings of very low safety significance (Green) are documented in this report. Two of these findings involved violations of NRC requirements. Two Severity Level IV violations without an associated finding are also documented in this report. We are treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2 of the Enforcement Policy.

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

If you disagree with a cross-cutting aspect assignment or a finding not associated with a regulatory requirement in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region III; and the NRC Resident Inspector at Donald C. Cook Nuclear Plant, Units 1 and 2.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with Title 10 of the *Code of Federal Regulations* 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

/RA/

Richard A. Skokowski, Chief
Branch 4
Division of Reactor Projects

Docket Nos. 05000315 and 05000316
License Nos. DPR-58 and DPR-74

Enclosure:
As stated

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Letter to Joel P. Gebbie from Richard A. Skokowski dated February 12, 2020.

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 – INTEGRATED INSPECTION REPORT 05000315/2019004 AND 05000316/2019004

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U.S. NUCLEAR REGULATORY COMMISSION
Inspection Report

Docket Numbers:	05000315 and 05000316
License Numbers:	DPR-58 and DPR-74
Report Numbers:	05000315/2019004 and 05000316/2019004
Enterprise Identifier:	I-2019-004-0067
Licensee:	Indiana Michigan Power Company
Facility:	Donald C. Cook Nuclear Plant, Units 1 and 2
Location:	Bridgman, MI
Inspection Dates:	October 01, 2019 to December 31, 2019
Inspectors:	K. Barclay, Resident Inspector J. Ellegood, Senior Resident Inspector M. Garza, Emergency Preparedness Inspector T. Go, Health Physicist M. Holmberg, Senior Reactor Inspector B. Jose, Senior Reactor Inspector J. Mancuso, Resident Inspector T. McGowan, Resident Inspector V. Meghani, Reactor Inspector J. Nance, Senior Resident Inspector
Approved By:	Richard A. Skokowski, Chief Branch 4 Division of Reactor Projects

Enclosure

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting an integrated inspection at Donald C. Cook Nuclear Plant, Units 1 and 2, in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information.

List of Findings and Violations

Unqualified Reactor Vessel Shell Weld Examination			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Initiating Events	Green NCV 05000316/2019004-01 Open/Closed	[H.2] - Field Presence	71111.08P
The inspectors identified a Green finding and an associated Non-Cited Violation of Title 10 of the Code of Federal Regulations (10 CFR), Appendix B, Criterion IX, "Control of Special Processes," for the licensee's failure to ensure the Unit 2 reactor vessel shell weld ultrasonic (UT) examination was controlled using a qualified procedure in accordance with the applicable American Society of Mechanical Engineers (ASME) Section XI Code. Specifically, the licensee approved (with no comments) Revision 9 of vendor procedure WDI-STD-1000 "Remote In-Service Inspection of Reactor Vessel Shell Welds" which established an alternate calibration method for the 45 degrees shear wave UT transducers that resulted in more than a 2 decibel (dB) loss in system sensitivity below that obtained by the ASME Code demonstrated/qualified UT method. Consequently, the UT examination of nine Unit 2 reactor vessel shell welds was not demonstrated as effective for detection and sizing of flaws in these welds.			

Failure to Establish Test Program to Verify Overtemperature Delta Temperature and Overpower Delta Temperature Turbine Runback Would Function When the Modification was Put into Service			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000316/2019004-02 Open/Closed	None (NPP)	71111.18
A finding of very low safety significance (Green) and an associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," was self-revealed on November 5, 2019, when the licensee was performing, for the first time, a complete solid state protection system (SSPS) overlap testing and observed that the signal from the SSPS turbine runback relay did not cause a status change (main turbine runback) on the digital control system (DCS) engineering work station as expected. It was later discovered that two required jumpers were never installed during the implementation of the Unit 2 Main Turbine and Main Feed Pump Turbines Upgrade Control and Protection Project (2-MOD-40046) in 2006. Upon further inspection, it was determined that the licensee did not at the time of the installation of the modification or any time thereafter, until November 5, 2019, perform any end-to-end testing of the U2 Overtemperature Delta Temperature and Overpower Delta Temperature Turbine Runback circuit to ensure that the runback signal was being sent to the DCS and would cause a runback to occur when required to do so.			

Failure to Follow Procedure Results in a Power Transient Due to the Loss of Two Feedwater Heater Drain Pumps and the Isolation of Extraction Steam to the Feedwater Heaters			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Initiating Events	Green FIN 05000315/2019004-03 Open/Closed	[H.1] - Resources	71153
A finding of very low safety significance (Green) was self-revealed on October 23, 2019, when the licensee failed to follow plant procedure 1-OHP-4021-060-014, "Operation of the Heater Drain Pumps," that ensures the proper operation of the pumps and the heater drain system. Specifically, the licensee's failure to correctly implement step 4.19.3 of the procedure and to open the 5A and 5B feedwater heater cross-connect valve 1-HMO-563 when the other cross-connect valve, 1-HMO-564, was already open, resulted in the 5A and 5B feedwater heaters being cross-connected between the heater drain pump suctions. This caused the tripping of both the north and south heater drain pumps and a small power transient.			

Failure to Follow Engineering Change Procedure for Non-Essential Service Water Strainer Design Modification			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Initiating Events	Green FIN 05000316/2019004-04 Open/Closed	[H.4] - Teamwork	71153
A finding of very low safety significance (Green) was self-revealed on July 21, 2019, for the licensee's failure to conduct a failure modes and effects analysis as described in licensee procedure, 12-EHP-5040-MOD-009, Engineering Change Reference Guide, Revision 60. Specifically, licensee procedure, 12-EHP-5040-MOD-009, Engineering Change Reference Guide, Revision 60, on page 89, states, in part that, when modifying or replacing existing components/equipment or adding new components/equipment, the item's failure mechanisms, failure modes, performance history, and reliability shall be considered. It further states, in part, that the team should solicit input from the vendors who may have information relevant to the development of the design change. The licensee documented in its root cause analysis in CR 2019-7047 that the root cause was that the failure modes and effects analysis performed for the NESW strainer design modification lacked rigor, resulting in an unevaluated design vulnerability associated with the strainer backwash feature during high debris ingress events. The licensee also documented in its root cause evaluation that the modification did not address the design change in the backwash piping diameter from 6 to 2 inches and that vendor information was found during the investigation that stated that rapid debris ingress could be problematic with the modification design.			

Main Steam Safety Valves Lift Pressure Setpoints Not Within Technical Specification Lift Setting Tolerance			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Not Applicable	NCV 05000315/2019004-06 Open/Closed	Not Applicable	71153
A self-revealed Severity Level IV Non-Cited Violation (NCV) of Technical Specification (TS) 3.7.1.1 was identified when the licensee operated the plant with two main steam safety valves (MSSVs) which were found to have lift pressure setpoints that were not within TS lift setting tolerance.			

Failure to Make a Timely Report Required by 10 CFR 50.73			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Not Applicable	NCV 05000315/2019004-05 Open/Closed	Not Applicable	71153
The inspectors identified a Severity Level IV violation of 10 CFR 50.73(a)(2)(i)(B) when the licensee failed to submit an Licensee Event Report to the NRC within 60 days of discovery of two conditions prohibited by the plant's Technical Specifications (TSs). Specifically, the Unit 1 #2 steam generator stop valve Train B dump valve was inoperable beginning with the transition to Mode 3 on May 7, 2019, until the valve packing was replaced and retested satisfactorily on May 30, 2019. The licensee submitted LER 315/2019-002-00, Condition Prohibited by Technical Specification Due to an Inoperable Steam Generator Stop Valve Dump Valve, on November 26, 2019.			

Additional Tracking Items

Type	Issue Number	Title	Report Section	Status
LER	05000315/2019-001-00	LER 2019-001-00 for Donald C. Cook Nuclear Plant, Unit 1, Two Main Steam Safety Valve Setpoints Found Outside Technical Specification Limits	71153	Closed
LER	05000316/2019-001-00	LER 2019-001-00 for Donald C. Cook Nuclear Plant, Unit 2, Manual Reactor Trip Due to Non-Essential Service Water System Degraded Condition	71153	Closed
LER	05000315/2019-002-00	LER 2019-002-00 for Donald C. Cook Nuclear Plant, Unit 1, Condition Prohibited by Technical Specification Due to an Inoperable Steam Generator Stop Valve Dump Valve	71153	Closed

PLANT STATUS

Unit 1 began this period at or near 55 percent power due to repairs being made to fix a feedwater leak. On October 2, 2019, the licensee began power ascension to return the unit to full power following repairs. On October 3, 2019, Unit 1 reached full power and remained at or near full power for the remainder of the inspection period.

Unit 2 began the inspection period at or near 50 percent power, in coast down for a refueling outage. On October 1, 2019, the licensee began lowering power to begin the refueling outage. Unit 2 achieved shut down on October 2, 2019. On November 18, 2019, the licensee restarted Unit 2 and commenced power ascension. On November 22, 2019, Unit 2 reached full power and remained at or near full power for the remainder of the inspection period.

INSPECTION SCOPES

Inspections were conducted using the appropriate portions of the inspection procedures (IPs) in effect at the beginning of the inspection unless otherwise noted. Currently approved IPs with their attached revision histories are located on the public website at http://www.nrc.gov/reading-rm/doc-collections/insp-manual/inspection-procedure/index.html. Samples were declared complete when the IP requirements most appropriate to the inspection activity were met consistent with Inspection Manual Chapter (IMC) 2515, “Light-Water Reactor Inspection Program - Operations Phase.” The inspectors performed plant status activities described in IMC 2515, Appendix D, “Plant Status,” and conducted routine reviews using IP 71152, “Problem Identification and Resolution.” The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel to assess licensee performance and compliance with Commission rules and regulations, license conditions, site procedures, and standards.

REACTOR SAFETY

<u>71111.01 - Adverse Weather Protection</u>

<u>Seasonal Extreme Weather Sample (IP Section 03.02) (1 Sample)</u>

(1) The inspectors evaluated readiness for seasonal extreme weather conditions prior to the onset of heavy snow with potential blizzard conditions expected to occur on November 11 - 13, 2019, and which included a National Weather Service (NWS) Winter Storm Warning and a NWS Lakeshore Flood Warning for the following systems:
   - Off-site power sources including Donald C. Cook switch-yard, ISFSI (independent spent fuel storage installation) dry casks, plant fore-bay structure, including trash racks and traveling screens and FLEX (Diverse and Flexible Coping Strategies) equipment.

<u>71111.04Q - Equipment Alignment</u>

<u>Partial Walkdown Sample (IP Section 03.01) (3 Samples)</u>

The inspectors evaluated system configurations during partial walkdowns of the following systems/trains:

(1) Unit 1 Safety Injection System on October 30, 2019

(2) Unit 1 Motor Driven Auxiliary Feed Water Pump after testing on October 31, 2019
(3) Unit 1 AB Emergency Diesel Generator (EDG) while guarded on October 29, 2019

71111.04S - Equipment Alignment

Complete Walkdown Sample (IP Section 03.02) (1 Sample)

(1) Unit 2 Safety Injection

71111.05Q - Fire Protection

Quarterly Inspection (IP Section 03.01) (4 Samples)

The inspectors evaluated fire protection program implementation in the following selected areas:

(1) Unit 1 Control Room on October 30, 2019
(2) Unit 2 Control Room on October 30, 2019
(3) Unit 1 AB EDG Room on December 3, 2019
(4) Unit 1 CD EDG Room on December 3, 2019

71111.06 - Flood Protection Measures

Inspection Activities - Internal Flooding (IP Section 02.02a.) (1 Sample)

The inspectors evaluated internal flooding mitigation protections in the:

(1) Unit 1 Auxiliary Building

71111.08P - Inservice Inspection Activities (PWR)

PWR Inservice Inspection Activities Sample (IP Section 03.01) (1 Sample)

(1) The inspectors verified that the reactor coolant system boundary, steam generator tubes, reactor vessel internals, risk-significant piping system boundaries, and containment boundary are appropriately monitored for degradation and that repairs and replacements were appropriately fabricated, examined and accepted by reviewing the following activities from October 15, 2019 to December 9, 2019:

03.01.a - Nondestructive Examination and Welding Activities.

- Automated ultrasonic (UT) examination for portions of the Unit 2 reactor vessel shell welds - 2-RPV-A, 2-RPV-D, 2-RPV-E and 2-LHM-04
- Manual UT of the safety injection system elbow-to-branch connection weld 2-SI-58-23
- VT-3 visual examination of the Unit 2 core support lug weld (category B-N-1) at the zero-degree azimuth
- VT-3 visual examination of the Unit 2 core barrel former bolts in columns 7 and 8 and 9 of octant No. 1
- EVT-1 visual examination for portions of the Unit 2 core barrel weld No. 3

- Socket welds OW-2, OW-3 and OW-4 fabricated on steam generator blowdown line (2-ABD-R 742) during replacement of valve 2-BD-103-2 (WO 55517263-01)

03.01.c – Pressurized-Water Reactor Boric Acid Corrosion Control Activities.

- AR 2018-4664, Boric Acid on Lower Surface of Unit 2 Reactor Vessel
- AR 2018-4943, 2IFI-54 Active Leak
- AR 2018-3239, U2 Pressure Relief Tank Coating/Corrosion Issue
- AR 2018-2142, Boric Acid Leak Initial Entry (Near 2-PP-38)
- AR 2018-2133, Boric Acid Leak Increase Since Last Report

71111.11A - Licensed Operator Requalification Program and Licensed Operator Performance

Requalification Examination Results (IP Section 03.03) (1 Sample)

(1) The inspectors reviewed and evaluated the licensed operator examination failure rates for the requalification annual operating tests administered from February 25 through March 28, 2019

71111.11Q - Licensed Operator Requalification Program and Licensed Operator Performance

Licensed Operator Performance in the Actual Plant/Main Control Room (IP Section 03.01) (1 Sample)

(1) The inspectors observed and evaluated licensed operator performance in the Control Room during the drain of the Reactor Coolant System to the 619' elevation on October 5, 2019

Licensed Operator Requalification Training/Examinations (IP Section 03.02) (1 Sample)

(1) The inspectors observed and evaluated performance in the Unit 1 Simulator during a licensed operator requalification evaluated exercise on December 3, 2019

71111.12 - Maintenance Effectiveness

Routine Maintenance Effectiveness Inspection (IP Section 02.01) (1 Sample)

The inspectors evaluated the effectiveness of routine maintenance activities associated with the following equipment and/or safety significant functions:

(1) Feedwater heater heat exchanger 6A outlet check valve stuck in a partially closed position; discovered during Unit 1 plant startup from refueling outage U1C29 on May 14, 2019

Quality Control (IP Section 02.02) (1 Sample)

The inspectors evaluated maintenance and quality control activities associated with the following equipment performance activities:

(1) EDG fuel injection pumps were found to have foreign material (abrasive particulate inside pump - April 2016 - resulting in pump seizing during post maintenance test run and paint chips inside pump discovered at receipt inspection - October 2016). Also, another EDG fuel injection pump seized during loss of offsite power / loss of coolant accident surveillance testing on November 7, 2019, which resulted in an emergency shutdown of the running diesel generator.

71111.13 - Maintenance Risk Assessments and Emergent Work Control

Risk Assessment and Management Sample (IP Section 03.01) (5 Samples)

The inspectors evaluated the risk assessments for the following planned and emergent work activities:

(1) Work on power lines over transformers on October 4, 2019
(2) Unit 1 north hotwell pump high risk work activity, on December 2, 2019
(3) Units 1 and 2 CD reserve feed outage on December 3, 2019
(4) Unit 1 elevated risk due to loop delta-T and Tavg channels inoperable during scheduled channel calibrations and normalizations, on December 6, 2019
(5) Unit 1 elevated risk due to swap over of feedwater heater drain pumps with feedwater header flow imbalance caused by feedwater heater 6A outlet check valve (1-FW-109-6A) stuck partially open and right moisture separator drain tank outlet to low pressure turbine A condenser control valve (1-MRV-403) being fully open on December 7, 2019

71111.15 - Operability Determinations and Functionality Assessments

Operability Determination or Functionality Assessment (IP Section 02.02) (7 Samples)

The inspectors evaluated the following operability determinations and functionality assessments:

(1) Unit 1 control room instrumentation distribution channel III (CRID 3) distribution panel (inverter) operability determination
(2) Unit 2 CD EDG missed surveillance with essential service water crossties open
(3) Unit 2 chemical volume control system (CVCS) charging to regenerative heat exchanger train 'B' shutoff valve operability determination
(4) Unit 2 CD EDG declared inoperable due to seized injector fuel oil pump on cylinder 6F
(5) Unit 2 reactor coolant pump (RCP) number 21 floor plug divider barrier operability determination due to a missed surveillance
(6) Unit 2 source range nuclear instrument, N-32, operability determination
(7) Unit 2 steam generator #22 and #23 channel 3 pressure instrument inoperable due to MTI (maintenance technician - instrument) technician recording the wrong number during channel operational test and calibration surveillance and not being detected by MTI supervisor or unit supervisor reviews

71111.18 - Plant Modifications

Temporary Modifications and/or Permanent Modifications (IP Section 03.01 and/or 03.02) (2 Samples)

The inspectors evaluated the following temporary or permanent modifications:

(1) 2-MOD-40046, Unit 2 main turbine digital control system permanent modification
(2) 50.59 Screen 2017-0410-00, Updates to ECP No. 1-2-00-14, EOP Footnotes

71111.19 - Post-Maintenance Testing

Post-Maintenance Test Sample (IP Section 03.01) (7 Samples)

The inspectors evaluated the following post maintenance tests:

(1) Unit 1 west main feed pump on October 1, 2019
(2) CD EDG fuel oil storage tank (FOST) on October 23, 2019
(3) Unit 1 CD fuel oil transfer pump following planned work on October 23, 2019
(4) Unit 2 train B solid state protection system testing following upgrade during U2C25 on November 3, 2019
(5) Unit 2 pressurizer train B pressure relief valve regulator replacement on November 6, 2019
(6) Unit 2 source range neutron flux instrument channel 2 post maintenance tests, on November 18, 2019
(7) South rod drive motor generator set post maintenance test as documented in work order (WO) 55540486

71111.20 - Refueling and Other Outage Activities

Refueling/Other Outage Sample (IP Section 03.01) (1 Sample)

(1) The inspectors evaluated Refueling Outage U2C25 activities from October 2, 2019 to November 19, 2019

71111.22 - Surveillance Testing

The inspectors evaluated the following surveillance tests:

Surveillance Tests (other) (IP Section 03.01) (4 Samples)

(1) 2-OHP-4030-232-217B, DG2AB Load Sequencing & Engineered Safety Features (ESF) Testing, Attachment 9, 3500/600 KW Load Rejection Testing on October 19, 2019
(2) 2-OHP-4030-232-001, Simultaneous Start of AB and CD Diesel Generators, required every 10 years, on October 20, 2019
(3) CD EDG FOST 10-year internal inspection on October 22, 2019
(4) Train B emergency core cooling system (ECCS) testing while defueled on October 28, 2019

Inservice Testing (IP Section 03.01) (2 Samples)

(1) Unit 2 east essential service water system, under WO 55514326
(2) Train A ECCS testing while defueled on October 29, 2019

Containment Isolation Valve Testing (IP Section 03.01) (1 Sample)

(1) Unit 2 containment cooling water to and from excess letdown heat exchanger, on October 12, 2019

FLEX Testing (IP Section 03.02) (1 Sample)

(1) FLEX blended RCS makeup pump inspection as documented in WO 55467357

71114.04 - Emergency Action Level and Emergency Plan Changes

Inspection Review (IP Section 02.01-02.03) (1 Sample)

(1) The inspectors evaluated the following submitted Emergency Action Level and Emergency Plan changes:
   - Evaluation 18-11, 10 CFR 50.54(q) Screening Form, August 9, 2018
   - Evaluation 18-13, 10 CFR 50.54(q) Screening Form, August 22, 2018
   - Evaluation 19-04, 10 CFR 50.54(q) Screening Form, January 28, 2019

   This evaluation does not constitute NRC approval.

71114.06 - Drill Evaluation

Drill/Training Evolution Observation (IP Section 03.02) (1 Sample)

The inspectors evaluated:

(1) Licensed operator requalification training (LORT) with drill exercise performance (DEP) on December 3, 2019

RADIATION SAFETY

71124.01 - Radiological Hazard Assessment and Exposure Controls

Radiological Hazard Assessment (IP Section 02.01) (1 Partial)

The inspectors evaluated radiological hazards assessments and controls.

(1) (Partial)
   The inspectors reviewed the following:

   Radiological Surveys
   - 20191007-9 Auxiliary Building Hallway/Boric Acid Evap Ion Exchanger 633'
   - 20191007-28 Auxiliary Building Hallway 587'
   - 20191015-4 Upper Containment (Ctmt) Spent Fuel Pit 701'

- 20191009-4 RCP 1-4 Pressurizer Walkway Containment 621'
- 20191009-6 Containment Entrance
- 20191018-1 Core Barrel Surveys 650'

Risk Significant Radiological Work Activities

- U-2 Reactor Disassembly Activities
- U-2 Seal Table Non-breach Work
- U-2 Hold-Down Spring Disassembly and Replacement
- U-2 Upper Internal Moves

Air Sample Survey Records

- 20191015-4 Upper Containment Spent Fuel Pit
- 20191009-6 Instrument Room, RCPs, Ctmt Entrance
- 20191015-17 Annulus Quad/Ctmt Quad
- 20191014-21 Removal and Pre & Post Shielding of U-2 Down Spring

Instructions to Workers (IP Section 02.02) (1 Partial)

The inspectors evaluated instructions to workers including radiation work permits used to access high radiation areas.

(1) (Partial)
The inspectors reviewed the following:

Radiation Work Packages

- RWP-192105 Rev. 3 U-2 Former Bolt Inspection Hold Down Spring Replacement
- RWP-192160 Rev. 1 U-2 Instrument Room Seal Table Activities
- RWP-192187 Rev. 0 U-2 Medium Risk; Under Reactor Vessel Inspection

Electronic Alarming Dosimeter Alarms

- There were no Electronic Dosimeter Alarms that occurred during this inspection.

Labeling of Containers

- There were 5 - 8 Dry Active Waste (DAW) Trash Bags and Containers at Various Elevation at Lower and Upper Containment that were Inspected and Surveyed by the inspector.

Contamination and Radioactive Material Control (IP Section 02.03) (1 Partial)

The inspectors evaluated licensee processes for monitoring and controlling contamination and radioactive material.

(1) (Partial)
The inspectors verified the following sealed sources are accounted for and are intact:

- Shepherd 89-2 /89-1 (400/130 mCi of Cs-137)
- Inst. Calibration ICN-2536 (1.0 Ci Cs-137)
- Hopewell Calibrator BX3-1 (797 Ci/1.06 Ci Cs-137 sealed sources)

Radiological Hazards Control and Work Coverage (IP Section 02.04) (1 Partial)

The inspectors evaluated in-plant radiological conditions during facility walkdowns and observation of radiological work activities.

(1) (Partial)
The inspectors also reviewed the following radiological work package for areas with airborne radioactivity:

- RWP-192105 Rev. 3 U-2 Former Bolt Inspection Hold Down Spring Replacement
- RWP-192160 Rev. 1 U-2 Instrument Room Seal Table Activities
- RWP-192187 Rev. 0 U-2 Medium Risk; Under Reactor Vessel Inspection

High Radiation Area and Very High Radiation Area Controls (IP Section 02.05) (1 Partial)

(1) (Partial)
The inspectors evaluated risk-significant high radiation area and very high radiation area controls.

Radiation Worker Performance and Radiation Protection Technician Proficiency (IP Section 02.06) (1 Partial)

(1) (Partial)
The inspectors evaluated radiation worker performance and radiation protection technician proficiency during the containment walkdown.

71124.02 - Occupational ALARA Planning and Controls

Radiological Work Planning (IP Section 02.01) (1 Partial)

The inspectors evaluated the licensee's radiological work planning.

(1) (Partial)
The inspectors reviewed the following activities:

- RWP-192105; U-2 Former Bolt Inspection Hold Down Spring Replacement
- RWP-192160; U-2 Instrument Room Seal Table Activities
- RWP-192187; U-2 Medium Risk; Under Reactor Vessel Inspection

Implementation of ALARA and Radiological Work Controls (IP Section 02.03) (1 Partial)

The inspectors reviewed as low as reasonably achievable practices and radiological work controls.

(1) (Partial)
The inspectors reviewed the following activities:

- RWP-192105 Rev. 3 U-2 Former Bolt Inspection Hold Down Spring Replacement
- RWP-192105 Rev. 3 Task 8 High Risk Lower Internal Movements

Radiation Worker Performance (IP Section 02.04) (1 Partial)

The inspectors evaluated radiation worker and radiation protection technician performance during:

(1) (Partial)
- U-2 Former Bolt Inspection Hold Down Spring Replacement and Lower Internal Movements
- U-2 Core Barrel Move to Vessel/Down Post

71124.06 - Radioactive Gaseous and Liquid Effluent Treatment

Instrumentation and Equipment (IP Section 02.04) (1 Sample)

The inspectors reviewed the following radioactive effluent discharge system surveillance test results:

(1) Auxiliary Building Ventilation Gaseous Effluent Flow Functional Test

OTHER ACTIVITIES – BASELINE

71151 - Performance Indicator Verification

The inspectors verified licensee performance indicators submittals listed below:

MS06: Emergency AC Power Systems (IP Section 02.05) (2 Samples)

(1) Unit 1 (October 1, 2018–September 31, 2019)
(2) Unit 2 (October 1, 2018–September 31, 2019)

MS07: High Pressure Injection Systems (IP Section 02.06) (2 Samples)

(1) Unit 1 (October 1, 2018–September 31, 2019)
(2) Unit 2 (October 1, 2018–September 31, 2019)

MS08: Heat Removal Systems (IP Section 02.07) (2 Samples)

(1) Unit 1 (July 1, 2018–June 30, 2019)
(2) Unit 2 (July 1, 2018–June 30, 2019)

MS09: Residual Heat Removal Systems (IP Section 02.08) (2 Samples)

(1) Unit 1 (July 1, 2018–June 30, 2019)
(2) Unit 2 (July 1, 2018–June 30, 2019)

MS10: Cooling Water Support Systems (IP Section 02.09) (2 Samples)

(1) Unit 1 (July 1, 2018–September 30, 2019)
(2) Unit 2 (July 1, 2018–September 30, 2019)

BI02: RCS Leak Rate Sample (IP Section 02.11) (2 Samples)

(1) Unit 1 (October 1, 2018–September 31, 2019)
(2) Unit 2 (October 1, 2018–September 31, 2019)

71152 - Problem Identification and Resolution

Semiannual Trend Review (IP Section 02.02) (1 Sample)

(1) The inspectors reviewed the licensee’s corrective action program for potential adverse trends in the management of contractors and supplemental personnel on-site that might be indicative of a more significant safety issue.

Annual Follow-up of Selected Issues (IP Section 02.03) (1 Sample)

The inspectors reviewed the licensee’s implementation of its corrective action program related to the following issues:

(1) The inspectors reviewed the licensee's assessment and corrective actions associated with the degraded condition of the unit 1 feedwater heater 6A outlet check valve and its effects on the operation of the plant, its associated systems and associated updated final safety analysis described surveillance requirements.

71153 - Followup of Events and Notices of Enforcement Discretion

Event Report (IP Section 03.02) (3 Samples)

The inspectors evaluated the following licensee event reports (LERs):

(1) LER 315/2019-002-00, Condition Prohibited by Technical Specification Due to an Inoperable Steam Generator Stop Valve Dump Valve (ADAMS accession: ML19333B815). The circumstances surrounding this LER are documented in Inspection Report 05000315/2019003, Section 71152 with a finding and an associated violation. This LER is being closed in this report with a finding documented under IP 71153 for failure to submit this LER in a timely manner.
(2) LER 315/2019-001-00, Two Main Steam Safety Valve Setpoints Found Outside Technical Specification Limits (ADAMS accession: ML19122A171). The inspectors determined that it was not reasonable to foresee or correct the cause discussed in the LER therefore no performance deficiency was identified. The circumstances and an NCV surrounding this LER are documented in the Results section of this report under IP 71153.
(3) LER 316/2019-001-00, Manual Reactor Trip Due to Non-Essential Service Water System Degraded Condition (ADAMS accession: ML19259A022). The circumstances and an NCV surrounding this LER are documented in the Results section of this report under IP 71153.

Personnel Performance (IP Section 03.03) (1 Sample)

(1) The inspectors evaluated the loss of both running feedwater heater drain pumps, the ensuing secondary plant transient, and licensee's performance both on the original loss of both feedwater heater drain pumps and the licensee's recovery attempts on October 23 and 24, 2019.

INSPECTION RESULTS

Unqualified Reactor Vessel Shell Weld Examination			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Initiating Events	Green NCV 05000316/2019004-01 Open/Closed	[H.2] - Field Presence	71111.08P

The inspectors identified a Green finding and an associated Non-Cited Violation of Title 10 of the Code of Federal Regulations (10 CFR), Appendix B, Criterion IX, "Control of Special Processes," for the licensee's failure to ensure the Unit 2 reactor vessel shell weld ultrasonic (UT) examination was controlled using a qualified procedure in accordance with the applicable American Society of Mechanical Engineers (ASME) Section XI Code. Specifically, the licensee approved (with no comments) Revision 9 of vendor procedure WDI-STD-1000, "Remote In-Service Inspection of Reactor Vessel Shell Welds" which established an alternate calibration method for the 45 degrees shear wave UT transducers that resulted in more than a 2 decibel (dB) loss in system sensitivity below that obtained by the ASME Code demonstrated/qualified UT method. Consequently, the UT examination of nine Unit 2 reactor vessel shell welds was not demonstrated as effective for detection and sizing of flaws in these welds.

Description:

On June 8, 2009, the NRC approved a licensee request (ISIR-29) to extend the Code 10-year Inservice Inspection Interval for the Unit 2 reactor vessel welds to a 20-year interval between examinations. Therefore, the last inservice UT examination of the Unit 2 vessel welds occurred in 1996, and the licensee intended to use the results of the 2019 reactor vessel weld UT examination to satisfy the ASME Code Section XI requirements for both the 3rd and 4rth Code intervals. For the 2019 outage UT examination, NRC regulation 10 CFR 50.55a(g)6(ii)(C) – "Augmented ISI requirements" required the licensee to apply the ASME Code Section XI, Appendix VIII, "Performance Demonstration for Ultrasonic Systems." Supplements 4 and 6 of Appendix VIII, established the process for demonstration of a UT system on vessel weld mockups (with weld flaws) to validate a UT system can identify and size flaws that may exist within reactor vessel welds. After a UT system is demonstrated/qualified on a vessel weld mockup, Appendix VIII identifies the essential UT procedure variables (includes the UT system calibration method) that must be controlled within allowable tolerance bands or the UT system is required to be re-demonstrated.

During the 2019 refueling outage, the inspectors identified that the vendor had changed (and the licensee approved) procedure WDI-STD-1000 "Remote Inservice Inspection of Reactor Vessel Shell Welds" to establish a different UT calibration method than applied during the original UT procedure demonstration. Because this change reduced the UT examination system sensitivity to flaws, the inspectors were concerned that this change could result in a

failure to identify, size and/or correct flaws (if they exist) within the reactor vessel shell welds. In Section 8.2.3 of procedure WDI-STD-1000, instructions were provided for calibration of the 45 degrees shear wave transducers that established the examination sensitivity for thinner vessel shell welds (e.g. less than 8 inches in thickness). Specifically, the 45 degrees shear wave transducer sensitivity was established based on the response from a 6-inch deep side drilled hole set at 80 percent full screen height. This calibration method reduced the UT system sensitivity from that established using a 10-inch deep side drilled hole that was applied during the initial procedure demonstration/qualification (when the vendor procedure was identified as PDI-ISI-254, Remote Inservice Inspection of Reactor Vessel Shell Welds, Revision 4).

The inspectors reviewed vendor records to determine why the calibration method for the 45 degrees shear wave transducers was changed and to assess the impact of this change on the UT system examination sensitivity. On January 25, 2005, the vendor issued Revision 6 to PDI-ISI-254 which first incorporated the change in calibration method. On February 14, 2005, the vendor issued “Vendor Technical Basis for Revision 7 of Procedure PDI-ISI-254,” which justified this change for Revision 7 that contained the same instructions first issued in Revision 6 for establishing calibration for the 45 degrees shear wave transducers on the 6-inch deep side drilled hole. In this document, the vendor stated that after initial qualification, field experience with the procedure indicated that when applying this sensitivity adjustment to shell thicknesses in the 5 to 8-inch range, the data presentation was generally saturated or overly sensitive. In saturated data, flaw features have an enlarged, less distinct presentation, requiring a rescan of the area at a lower gain setting to resolve the flaw measurement features according to the image quality requirements defined in the procedure. To address this problem, the vendor elected to establish sensitivity based on a 6-inch deep hole for examination of thinner reactor vessel shell weld sections. In the vendor’s technical basis document, the vendor compared the UT system sensitivity for 45 degrees shear wave search units calibrated on the 10-inch deep side drilled hole to a calibration set up on the 6-inch deep side drilled hole and recorded a reduction in UT examination sensitivity of 9 dB below that established during the original procedure demonstration. However, the vendor did not re-demonstrate the revised UT procedure on a vessel weld mockup (meeting Supplement 6 of Appendix VIII of ASME Code Section XI) to confirm that the procedure with the revised calibration method would still be adequate to detect and size weld flaws. Because the change in calibration method was incorporated into the current UT procedure (WDI-STD-1000) and did not meet the ASME Code Appendix VIII-4300 “Calibration Methods” acceptance criteria of 2 dB, the inspectors determined that the UT examination as completed on the thinner Unit 2 reactor vessel shell welds (2-RPV-D, 2-RPV-E, 2-LHM-01, 2-LHM-02, 2-LHM-03, 2-LHM-04, 2-LHM- 05, 2-LHM-06, 2-LHM-07) was not a demonstrated/qualified UT examination.

Absent NRC identification, the licensee would have relied on an unqualified UT examination of the Unit 2 reactor vessel shell welds, which may have resulted in failure to identify and correct flaws in the reactor vessel. The inspectors reviewed NUREG-1801, Generic Aging Lessons Learned (GALL) Report, and determined that that the most likely degradation mechanisms affecting the reactor shell welds was cumulative fatigue damage (e.g., crack growth due to cyclic loading). If undetected fatigue flaws were left inservice, it is possible the cracks would progress and result in through-wall pressure boundary leakage. Based on operating experience, very small leaks that develop at reactor vessel head weld locations can result in wastage of the vessel steel caused by corrosion and if this condition was not promptly identified and corrected, it would increase the chance for a loss-of-coolant-accident (LOCA) initiating event. Most of the Unit 2 reactor vessel welds were subjected to multiple

inservice UT examinations (including the unqualified 2019 Unit 2 refueling outage examination) without identification of defects, therefore it was unlikely that undetected fatigue cracks exist within the affected vessel welds.

Corrective Actions: The licensee entered the non-conforming UT examination procedure into the corrective action program (CAP) as AR 2019-10017 and was evaluating options to restore compliance for the affected Unit 2 reactor vessel welds. In 2010, the Unit 1 reactor vessel welds were also examined by the same vendor using Revision 7 of the vendor procedure (with the unqualified UT calibration method) and the licensee entered this issue into the CAP as AR 2019-10338. The licensee determined that the Unit 1 vessel was operable because the required Code inspections had been performed and the periodic reactor coolant system leak rate tests had not shown deficiencies which would prevent operability of the Unit 1 reactor vessel. The Wesdyne vendor also entered this issue into their CAP as IR-2019-15837 and if the vendor determines that a generic condition exists, the vendor CAP would trigger a vendor evaluation or notification report to the NRC in accordance with 10 CFR Part 21 requirements. Following examination of the Unit 2 reactor vessel welds, the licensee's vendor successfully demonstrated the alternate calibration method (e.g., 6-inch deep side drilled hole) on an ASME Code Section XI, Appendix VIII compliant mockup weld test block to qualify the UT procedure (reference Electric Power Research Institute (PDQS) No. 932).

Corrective Action References: AR 2019-10017 and AR 2019-10338

Performance Assessment:

Performance Deficiency: The licensee's failure to ensure the Unit 2 reactor vessel shell welds UT examination was controlled using a qualified procedure in accordance with the applicable ASME code was contrary to 10 CFR 50 Appendix B, Criterion IX, "Control of Special Processes," and a performance deficiency.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Human Performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the licensee's failure to ensure the reactor vessel shell weld UT examination was controlled using a qualified procedure rendered the quality of the vessel weld UT examination as indeterminate and potentially inadequate. An inadequate UT examination would increase the likelihood of an undetected weld crack that propagates by fatigue and results in a LOCA initiating event. The inspectors also reviewed the examples of minor issues in IMC 0612, Appendix E, "Examples of Minor Issues," and did not identify any examples related to this issue.

Significance: The inspectors assessed the significance of the finding using Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." The inspectors applied the Exhibit 1, "Initiating Events Screening Questions," and determined this issue did not affect systems used to mitigate a LOCA and after a reasonable assessment of potential degradation it was not likely that the finding would have resulted in exceeding the reactor coolant system leak rate for a small break LOCA. Therefore, the inspectors answered questions A.1 and A.2 of Exhibit 1 as "No" and the performance deficiency screened as very low risk significance (Green).

Cross-Cutting Aspect: H.2 - Field Presence: Leaders are commonly seen in the work areas of the plant observing, coaching, and reinforcing standards and expectations. Deviations from standards and expectations are corrected promptly. Senior managers ensure supervisory and management oversight of work activities, including contractors and supplemental personnel. In this case, the licensee managers failed to ensure adequate oversight of the UT contractor. Specifically, the licensee failed to apply adequate technical rigor during review and approval of the vendor's UT examination procedure.
Enforcement: Violation: Title 10 CFR 50, Appendix B, Criterion IX, "Control of Special Processes," states that "Measures shall be established to assure that special processes, including welding, heat treating, and nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements." The 1995 Edition with 1996 Addenda of the ASME Code Section XI, Appendix VIII, Sub-Article VIII-4300 "Calibration Methods" step (c) stated "The alternative calibration method is acceptable when the system sensitivity is no more than 2 dB below that obtained by the qualified method." Contrary to the above, as of October 22, 2019, the licensee failed to establish measures to assure that the special process for nondestructive testing (e.g., UT examination) of the Unit 2 reactor vessel shell welds was controlled using a qualified procedure in accordance with the applicable codes. Specifically, on August 19, 2019 the licensee approved (with no comments) Revision 9 of vendor procedure WDI-STD-1000 "Remote In-Service Inspection of Reactor Vessel Shell Welds" with an alternate calibration method for the 45 degrees shear wave UT transducers that caused more than a 2 dB loss of system sensitivity below the demonstrated/qualified UT calibration method and was therefore not acceptable. Consequently, on October 22, 2019, the subsequent UT examination of nine Unit 2 reactor vessel shell welds was not demonstrated as effective for detection and sizing of weld flaws. Enforcement Action: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

Failure to Establish Test Program to Verify Overtemperature Delta Temperature and Overpower Delta Temperature Turbine Runback Would Function When the Modification was Put into Service			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000316/2019004-02 Open/Closed	None (NPP)	71111.18
A finding of very low safety significance (Green) and an associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," was self-revealed on November 5, 2019, when the licensee was performing, for the first time, a complete solid state protection system (SSPS) overlap testing and observed that the signal from the SSPS turbine runback relay did not cause a status change (main turbine runback) on the digital control system (DCS) engineering work station as expected. It was later discovered that two required jumpers were never installed during the implementation of the Unit 2 Main Turbine			

and Main Feed Pump Turbines Upgrade Control and Protection Project (2-MOD-40046) in 2006. Upon further inspection, it was determined that the licensee did not at the time of the installation of the modification or any time thereafter, until November 5, 2019, perform any end-to-end testing of the U2 Overtemperature Delta Temperature and Overpower Delta Temperature Turbine Runback circuit to ensure that the runback signal was being sent to the DCS and would cause a runback to occur when required to do so.
Description: Engineering modification 2-MOD-40046 was implemented in 2006. In the modification package under test activities, it states, in part, demonstrate that the various MT (main turbine) runback features function properly. The testing methodology is also described in the modification package and states, in part, that "Test shall demonstrate all means to raise/lower turbine speed/load as provided by the BODD (TS3000 Digital Control System Basis of Design Document D C Cook Nuclear Unit 2 Main Turbine Control System)," "Test shall provide means to tune the dynamic response of each controlled component," and "Test shall be sequenced to conduct testing while minimizing effects to interfacing SSCs (structures, systems, and components)." The inspectors determined through additional inspection activities that the licensee was unable to find any functional testing that was performed following installation of the Unit 2 Main Turbine and Main Feed Pump Turbines Upgrade Control and Protection Project (2-MOD-40046) in 2006. The licensee stated that the design interface was tested, but the referenced prints identified that the place where the technicians connected the test leads was upstream of the wires missing jumpers, which would explain why the results were incorrectly acceptable. The licensee's investigation identified that wiring in panel 'DTU' shown on drawing PS-2-92031 did not exist. The drawing showed two wires between points 'TLD' and 'TLF' for 'TL-1' & 'STDX-1.' Those wires did not exist, thus preventing the runback signal from being sent to the DCS system. Corrective Actions: The licensee installed the correct jumpers as shown on drawing PS-2-92301 and reperformed the required SSPS testing satisfactorily. Corrective Action References: AR 2019-10976
Performance Assessment: Performance Deficiency: The inspectors determined that during modification testing the licensee failed to establish a test program to assure that all testing required to demonstrate that the installed (in 2006) Overtemperature Delta Temperature and Overpower Delta Temperature Turbine Runback function would perform satisfactorily. This was a performance deficiency that was within the licensee's ability to foresee and prevent. Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Procedure Quality attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the procedure used to test the installation of the Overtemperature Delta Temperature and Overpower Delta Temperature Turbine Runback function did not ensure that the logic was connected to the DCS and therefore did not demonstrate that the Overtemperature Delta Temperature and Overpower Delta Temperature Turbine Runback function with its associated main turbine runback would perform satisfactorily in service.

Significance: The inspectors assessed the significance of the finding using Appendix A, “The Significance Determination Process (SDP) for Findings At-Power.” Using Exhibit 2, "Mitigating Systems Screening Questions," the inspectors determined that the finding was of very low safety significance (Green) because they answered 'no' to each question under section C. Reactivity Control Systems.

Cross-Cutting Aspect: Not Present Performance. No cross-cutting aspect was assigned to this finding because the inspectors determined the finding did not reflect present licensee performance.

Enforcement:

Violation: Title 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," states, in part, that a test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents.

Contrary to the above, since the installation of the safety-related Unit 2 Main Turbine and Main Feed Pump Turbines Upgrade Control and Protection Project (2-MOD-40046) in 2006, the licensee failed to establish a test program to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Specifically, during the Unit 2 Main Turbine and Main Feed Pump Turbines Upgrade Control and Protection Project (2-MOD-40046) in 2006, the licensee did not perform any end-to-end testing of the Unit 2 Overtemperature Delta Temperature and Overpower Delta Temperature Turbine Runback circuit or verify complete overlap of portion system testing to ensure that the runback signal was being sent to the Turbine Digital Control System and would cause a runback to occur when required to do so by the modification package.

Enforcement Action: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

Observation: Degraded Condition of the Unit 1 Feedwater Heater 6A Outlet Check Valve and Its Effects on the Operation of the Plant, and Its Associated Systems.	71152

During the restart of Unit 1 following refueling outage U1C29 in May 2019, the licensee discovered that the Unit 1 6A high pressure feedwater heater outlet check valve was not fully open. This was based, in part, on Unit 1 high pressure feedwater heater 6A (1-HE-6A) steam pressure being approximately 171 pounds per square inch gauge (psig) and the high pressure feedwater heater 6B (1-HE-6B) steam pressure being approximately 143 psig. Also, the normal drain valve for feedwater heater 6A, 1-HRV-651, was approximately 24 percent open and the normal drain valve for feedwater heater 6B, 1-HRV-652, was approximately 33 percent open. That occurred because there was more water to condense in the 6B feedwater heater than in the 6A feedwater heater and the automatic level controls were trying to maintain the same level in each of the two feedwater heaters.

With the Unit 1 6A high pressure feedwater heater outlet check valve (1-FW-109-6A) not fully open, there was some blockage of feedwater flow through the A heater string which resulted

in an imbalance in flow between the A and B heater strings. This condition has occurred previously on at least two occasions which were linked to feedwater heater level control anomalies and unit power oscillations. Specifically, as documented in AR 2016-5627 in April 2016, following the replacement of feedwater heaters 6A and 6B, the licensee confirmed that the 6A feedwater heater outlet check valve was stuck in the partially closed position resulting in a significant flow imbalance between the A and B feedwater strings. The flow imbalance was confirmed by numerous indications showing increased flow through the B string. The licensee resolved the issue by slowly closing the valve stem from the full open position until it barely moved the lodged plug. That movement resulted in a change in the delta-pressures across the feedwater heater 6A and feedwater heater 6B tube bundles. After contacting the plug and slightly moving it, the valve was slowly returned to full open with the plug following, which was verified by delta-pressures across the 6A heater and 6B heater. The delta pressure returned to 6.9 psid, indicating a balanced flow. There was a corrective action created at that time to inspect the internals of the 6A high pressure feedwater heater outlet check valve, 1-FW-109-6A, under WO 55484557 during the next refueling outage, U1C28. The licensee stated that the work would be scope-added through the regular scoping process which would determine the need to further inspect the other three discharge check valves. Another action request was generated on January 18, 2018, for an imbalance between feedwater flow through the A and B feedwater strings caused by 1-FW-109-6A sticking in the slightly closed position. This action request referred to the same work order WO 55484557 that was referenced in AR 2016-5627 in April 2016. That work order was removed from the scope of U1C28 refueling outage. AR 2018-0569 stated that the WO would be added to the scope for U1C29. That WO was also removed from the U1C29 scope of work prior to that outage.

The difference between what the licensee discovered during the startup from refueling outage U1C29 and previous events, in regards to 1-FW-109-6A sticky, was that the issue was not resolved when the unit reached 100 percent power. The licensee was unable to move the stem of the valve in the closed direction more than three turns. Previously, it had taken at least 18 turns to actually cause a change in the feedwater flow indication parameters. The licensee states in AR 2019-5226 that, "Due to this unexpected condition with the handwheel, the troubleshooter was aborted and the feedwater flow imbalance was not resolved." AR 2019-5229 was created on May 14, 2019, to address the issue of the imbalanced flow between heater drain strings A and B. Three corrective actions were created to look at creating an Operational Decision Maker. All three action items were closed administratively with no action taken except to scope the investigation and repairs, if needed, into U1C30 refueling outage.

AR 2019-6268, "1-MRV-403 is expected to remain open during current cycle," was written on June 20, 2019, and states that because of the flow imbalance between north and south heater drain strings, the right moisture separator drain tank outlet to low pressure turbine 'A' condenser control valve, 1-MRV-403, will remain open (approximately 25 percent) until refueling outage U1C30 in order to assure there is sufficient heater drain control margin to properly control 5A feedwater heater level. In early September, the licensee reduced power on Unit 1 to affect repairs to one of the main feedwater pump (MFP) casings. Following repairs, the licensee began to raise power and experienced issues with maintaining control of the 5 and 6 A/B feedwater heater levels in automatic. Licensee personnel opened 1-MRV-403 slowly until the valve was 60 percent open. At that point the controllers were able to maintain control of the feedwater heaters as designed and the licensee continued to operate at 100 percent power with no additional issues. What the licensee did not do was take into account the effect that fully opening 1-MRV-403 had on the flow rates between the

north and south heater drain strings. The normal flow rate at 100 percent power is balanced between the north and the south heater drain strings. With the 1-FW-109-6A stuck in a mid-position, from U1C29 startup until the downpower and repair of the MFP casing in early September, the flow rates through the north and south heater drain strings were 4800 gpm and 2900 to 3250 gpm, respectfully. Following the repair to the MFP casing and after returning to 100 percent power, the flow rate through the north string has been 4960 gpm and 1850 gpm through the south string. The position of 1-MRV-403 following the repairs to the MFP casing increased from its original 25 percent open to 60 percent open which was not known to either operations management or senior site managers, until after the October 23 event.

On September 13, 2019, during main turbine valve testing, several unexpected annunciators were received in the control room, including "Heater Drain PLC Trouble," "Right Moisture Separator Drain Tank (MSDT) High," and "Heater 5B Level Low." The control system restored the 5B heater level to set point and the right MSDT level returned to its pre-test condition without any operator action. The main turbine valve testing evolution was then aborted.

On October 23, 2019, at approximately 1508, while restoring from a fill, vent and flush of the middle heater drain pump, in accordance with procedure 1-OHP-4021-060-014, Attachment 8, the header drain string crosstie valve, 1-HMO-563, was opened with 1-HMO-564 (the other header drain string crosstie valve) already open. Due to the significant imbalance between the flow in the A and B heater drain strings, cross-tying the two headers together as well as the 5A and 5B feedwater heaters caused the feedwater heaters levels to begin fluctuating immediately. As documented in AR 2019-10383, the operator opened the south heater drain pump bypass valve, 1-HMO-563, and immediately saw the heater levels start to fluctuate. The operators tried to close the north heater drain pump bypass valve, 1-HMO-564 and alarms began coming in. The At-The-Controls (ATC) operator looked over, called out the unit supervisor's (US) name, who was turned around discussing an upcoming activity. When the US looked over, they saw the Hi-Hi Level Alarm on the feedwater heater. Within seconds, the south heater drain pump had tripped followed by the tripping of the north heater drain pump. The control room crew took the appropriate actions following the tripping of the pumps and the isolating of extraction steam from the turbines to the feedwater heaters.

Due to the increased flow imbalance which was still not well or widely known or understood by the licensee, the oncoming night shift operating crew attempted to restore the two tripped heater drain pumps at approximately 0319 on the morning of October 24, 2019. The crew had started a third condensate booster pump to makeup for having no heater drain pumps in service following the tripping of the south and north heater drain pumps earlier in the evening. The north and south heater drain pumps had been returned to service with reactor power at approximately 96 percent power. Once the plant was stabilized, the middle condensate booster pump was tripped to place the plant back into its normal full power alignment. When the pump was secured, level lowered rapidly in the 5B feedwater heater. The north heater drain pump discharge valve also closed rapidly, but did not close fast enough to prevent the heater drain pump from tripping on low-low level in the 5B feedwater heater. The middle condensate booster pump, which had been secured and placed in auto, restarted as condensate header pressure lowered. The transient propagated to the MFP and the MFP deferential pressure lowered to approximately 100 psid as the feedwater regulator valves went almost full open. Several steam generator high level deviation alarms were received before the system began to stabilize. The licensee believed that the heater drain pump tripped due to the inability of its discharge valve to close fast enough when the middle

condensate booster pump was secured. The licensee further believed that it was not the fault of the level control system but was rather the result of the current higher flow on the north heater drain string.
Following the two events on October 23rd and 24th, the inspectors after the 0630 morning meeting on October 24th asked the licensee's senior leadership team, "What had changed." Senior management was unaware of the changes to the position of 1-MRV-403 and the effect that it had on the flow differential between the south and north heater drain strings since the repair of the MFP casing in September.

On October 27, operations went to the simulator to perform swapping of the U1 heater drain pumps on the alpha string heaters. What they found was that they at that time did not have enough flow on the alpha heater string to swap heater drain pumps. There was only 1850 gpm (approx.) flow on the alpha string (south heater drain string). In their simulation they lost both heater drain pumps on low low level in the feedwater heaters.

Since that time, the licensee has issued a special procedure for swapping heater drain pumps that includes reducing reactor power to 90 percent before proceeding with the evolution. The procedure also has the licensee take the feedwater heater controllers to manual to prevent the large swings in feedwater heater levels due to the imbalance in flow between the heater drain strings.

The licensee has had several opportunities to correct the deficient condition of the Unit 1 6A high pressure feedwater heater outlet check valve (1-FW-109-6A) and to create temporary or special procedures for operating the plant and the feedwater heater drain system with the imbalance of flow between the 'A' and 'B' heater drain strings, all of which were reasonably within the licensee's ability to foresee and that should have been prevented. The inspectors determined that the performance deficiency involved did rise to the level of being of more than minor and was indeed a finding that is documented in this report in section 71153. No violations of NRC requirements were identified by the inspectors. The inspectors concluded that the licensee's actions following the events on October 23 and 24, 2019, have been timely and that the actions the licensee took to troubleshoot the condition have been reasonable and appropriate. Additionally, the inspectors concluded that the licensee had established corrective actions to address the issue that were appropriate for its safety significance and that could reasonably be expected to resolve the issue.

Observation: Adverse Trends in the Management of Contractors and Supplemental Personnel Onsite	71152
During this quarter, the inspectors performed a trend review of action requests (AR) generated from January through November 2019. Early on in the review it became apparent that the licensee was having significant issues with management of contractors and supplemental personnel onsite. During the time period reviewed, over 200 individual ARs were generated for issues with contractors and supplemental workers. These included work performed on wrong (operating unit) equipment during Unit 2 refueling outage, the makeup plant motor control center feeder breaker being inadvertently tripped, as well as OSHA related issues and many other performance near misses. These near misses included improper staging of material in the plant, fire seals being incorrectly repaired/inspected, not wearing proper personal protection equipment, not following foreign material exclusion zone requirements, not following work order requirements and licensee procedures, incorrect lifting of leads during testing, the removal of two solid state protection system circuit cards from site, the use of a contractor procedure without approval, damage to plant fire fighting equipment, unattended combustible material in the Unit 2 auxiliary building, etc.	

During the Unit 2 refueling outage in the fall, the licensee performed a common cause evaluation and documented the results in AR 2019-10634, "Perform Common Cause." The focus of the common cause was on three events, the first being work performed on the Unit 1 closed cooling water (CCW) system instead of the authorized work on the Unit 2 CCW system. The second event was the inadvertent opening of the makeup plant motor control center feeder breaker and the third event was the OSHA lost time accident.

There were two groups that met to review the three events and each recommended actions to take. The first group was made up of first line supervisors (FLS). The team believed that in general the level of preparation and engagement was commensurate with the level of challenge the work group expects to receive. An example they gave was if a work group is going to be challenged by the unit supervisor for work performed in the control room, then that group will take more time to understand their task because they know they will be asked a series of questions before being allowed to go to work. The group also discussed that many low risk jobs are "self-briefed" by the work group and the supervisor is not at the brief. The FLSs recommended the following and implemented those recommendations for two weeks. The actions were, first, before going to the field, each work group would check-in with the supervisor who would then ask two questions: "What safety hazards do you expect to encounter and what are you doing to mitigate/eliminate them?" and, "What status control hazards do you expect to encounter and what are you doing to mitigate/eliminate them?" The second action was to have managers and supervisors acting as "Engagement Rovers" who would be in the field validating that each work group had a good understanding of what they are doing, and asking them the same two questions mentioned above. If the rover finds a work group unable to adequately answer the questions, then the rover would contact the supervisor for a discussion at the work site.

The managers' working group came up with several recommendations, three short-term actions, and four long-term actions. Two of the group recommendations were put into place and ran in parallel with the FLS's recommendations, with one exception, which was to have a Senior Reactor Operator or Reactor Operator walk through the plant telling workers the significance of the equipment they were working on and its impact to plant operations.

Part of the licensee's longer term strategy was to partner with a contractor for its refueling outages that would have the depth and experience to manage not only the actual work being performed during an outage, but to be able to manage the day-to-day activities in the outage control center.

The ability of the licensee to effectively manage its contractors and supplemental workforce is very important and can have significant impact on station performance including safe operations.

Although the licensee has had significant issues with the performance of its contractors, the inspectors determined that the licensee's performance in managing its contractors has begun to improve. The deficiencies involved did not rise to the level of being of more than minor safety significance and, as such, did not constitute a documentable finding. Additionally, no violations of NRC requirements were identified by the inspectors. Following their review, the inspectors concluded that the licensee’s identification of the issues have been timely and that the actions the licensee had taken to troubleshoot the condition have been reasonable and appropriate. Additionally, the inspectors concluded that the licensee had established

corrective actions to address the issues that were appropriate for their safety significances and that could reasonably be expected to resolve the issues.

Failure to Follow Engineering Change Procedure for Non-Essential Service Water Strainer Design Modification

Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Initiating Events	Green FIN 05000316/2019004-04 Open/Closed	[H.4] - Teamwork	71153

A finding of very low safety significance (Green) was self-revealed on July 21, 2019, for the licensee's failure to conduct a failure modes and effects analysis as described in licensee procedure, 12-EHP-5040-MOD-009, Engineering Change Reference Guide, Revision 60. Specifically, licensee procedure, 12-EHP-5040-MOD-009, Engineering Change Reference Guide, Revision 60, on page 89, states, in part that, when modifying or replacing existing components/equipment or adding new components/equipment, the item's failure mechanisms, failure modes, performance history, and reliability shall be considered. It further states, in part, that the team should solicit input from the vendors who may have information relevant to the development of the design change. The licensee documented in its root cause analysis in CR 2019-7047 that the root cause was that the failure modes and effects analysis performed for the NESW strainer design modification lacked rigor, resulting in an unevaluated design vulnerability associated with the strainer backwash feature during high debris ingress events. The licensee also documented in its root cause evaluation that the modification did not address the design change in the backwash piping diameter from six to two inches and that vendor information was found during the investigation that stated that rapid debris ingress could be problematic with the modification design.

Description:

On July 19, 2019, a high differential pressure (D/P) alarm came in for the Unit 2 north non-essential service water (NESW) pump strainer. The pump was removed from service and the south NESW pump was placed in service. On July 21, 2019, the high D/P alarm came in for the Unit 2 south NESW pump strainer. The operations shift reduced power and manually shut down Unit 2, due to degrading conditions in the Unit 2 NESW pump discharge strainers. Further investigation identified the NESW strainers were full of mussels and the backwash lines were plugged internal to the strainers. The inspectors reviewed and evaluated the licensee's root cause analysis as part of the event followup and review of LER 2019-001-00, Donald C. Cook Nuclear Plant, Unit 2, Manual Reactor Trip Due to Non-Essential Service Water System Degraded Condition. Prior to the event there were several relevant activities that occurred.

On August 2, 2016, AR 2016-8839 was initiated to document significant biological debris ingress when opening the circulating water intake tunnel shutoff valve,12-WMO-30, which resulted in significant debris collection in the screen house from the traveling water screens, but it did not result in any impacts to downstream systems, including the non-essential service water system. The AR included an Environmental group's report which identified the debris consisted of dead zebra mussels, quagga mussels, and zebra-quagga hybrid mussels. The report and condition evaluation recommended minimizing intake maintenance requiring the closing of 12-WMO-30 during periods of high lake temperature. High lake temperature would cause a large increase in mussel mortality leading to significant increases in debris which could then pass through the traveling water screens and make their way to

downstream systems such as the NESW system and its strainers. No formal action was created to drive the recommendations from the condition evaluation. The AR was closed on August 26, 2016.
On September 26, 2016, EC-0000054253, a design modification package for the replacement of the current NESW pumps with new pumps was approved by the licensee. The new modification also required the replacement of the pumps' discharge strainers with an alternate design, that reduced the diameter of the backwash pipe from six to two inches. The inspectors reviewed Engineering Change Reference Guide, 12-EHP-5040-MOD-009, Revision 60, which states, in part that, when modifying or replacing existing components/equipment or adding new components/equipment, the item's failure mechanisms, failure modes, performance history, and reliability shall be considered. It further states, in part, that the team should solicit input from the vendors who may have information relevant to the development of the change. The licensee documented in its root cause analysis in AR 2019-7047 that the root cause was that the failure modes and effects analysis performed for the NESW strainer design modification lacked rigor, resulting in an unevaluated design vulnerability associated with the strainer backwash feature during high debris ingress events. The licensee also documented in its root cause evaluation that the modification did not address the change in the backwash piping diameter from six to two inches and that vendor information was found during the investigation that stated that rapid debris ingress could be problematic with the modification design.

On Tuesday, July 16, 2019, lake temperature and forebay temperature averaged above 76 degrees Fahrenheit. At that temperature zebra, quagga, and zebra-quagga hybrid mussels began dying off at a rapid rate. The 12-WMO-30 valve had been closed for some extended time due to the center intake piping inspections and repairs. The mussel population impacted by the high temperatures were in both the center intake structure and in the forebay east and west of the traveling water screens, with the east side of the Unit 2 traveling water screens having an established and dense population of mussels since the last cleaning in May 2018. On July 17, 2019, the inspections and repairs to the center intake were completed.

On July 18, 2019, during the day shift, the licensee was in a hot weather alert. Around noon, the shift manager had a discussion with operations and senior management regarding several current plant concerns, including Unit 1 containment temperature (slowly rising), Unit 2 main feed pump turbine (MFPT) condenser vacuums declining, etc., all due to increased lake water and ambient temperatures. The restoration of flow from the center intake through circulating water intake tunnel shutoff valve, 2-WMO-30, was discussed as potential action to help manage ongoing plant concerns. The shift manager made the decision to proceed with clearance restoration and the partial opening of 12-WMO-30. At 1300, 12-WMO-30 was taken to five percent open. A large influx of debris on the Unit 2 side of the forebay led operators to switch to the outside fish basket on the Unit 2 side. On night shift, the plant continued in a hot weather alert. The plan was to continue to slowly open 12-WMO-30 over the weekend due to current and predicted plant conditions. At 0108, 12-WMO-30 was opened from 5 percent to 10 percent with the debris ingress into the forbay being less than that seen on day shift. At 0234 the north NESW strainer D/P alarm came in and did not clear. The south NESW pump was started and the north NESW pump was secured.

On Friday, July 19, 2019, conditions appeared to stabilize for the NESW pump in service and the forebay. There continued to be plant challenges for the Unit 2 MFPT condenser vacuums and Unit 1 containment temperature. There were also challenges to performing maintenance

on the north NESW pump strainer due to lack of required parts, manpower to erect needed scaffolding, and high heat stress near the pump strainer. The decision was made between Maintenance, Operations, and the Plant Manager to wait until Monday, July 22, to perform any additional maintenance activities.

On Saturday, July 20, 2019, the licensee considered the option to open 12-WMO-30 another 10 percent to 20 percent open. This option was initially challenged by the station management and day shift supervisors. Subsequent discussions resolved the concerns and all individuals agreed to proceed forward with opening 12-WMO-30 to 20 percent open. This was accomplished in two steps, with the final five percent opening occurring near the end of day shift.

On Sunday, July 21, 2019, shortly after midnight the Unit 2 south NESW strainer D/P alarm came in. The Unit 2 north and south NESW pumps were swapped back and forth to try and keep the D/P alarms clear. Phone calls for operations assistance were made at 0030 and at 0330. Shortly after turnover to day shift the decision was made to manually shut down Unit 2. At 0655, a rapid down power of Unit 2 commenced and the Unit 2 reactor was manually tripped at 0826.

Corrective Actions: The licensee has implemented the following corrective actions to prevent recurrence: 1) Revision to Operations procedures to provide additional guidance for operating NESW strainers and system to account for anticipated challenging conditions; 2) Revision to the Standard Design Process Interface procedure to provide guidance for performing Failure Modes and Effects Analysis if certain criteria are met; 3) Creation of event initiated preventive maintenance activities to disassemble, clean and inspect NESW strainers; and 4) Validation of minimum/maximum NESW strainer part levels that will support emergent cleaning.

Corrective Action References: AR 2019-7047 and AR 2016-8839

Performance Assessment:

Performance Deficiency: The licensee's failure to evaluate the failure mode created by the new non-essential service water strainer backwash feature during high debris ingress events was a performance deficiency that was within the licensee's ability to foresee, prevent, and/or correct.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Design Control attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the licensee documented in its root cause analysis in AR 2019-7047 that the root cause was that the failure modes and effects analysis performed for the NESW strainer design modification lacked rigor, resulting in an unevaluated design vulnerability associated with the strainer backwash feature during high debris ingress events. The licensee also documented in its root cause evaluation that the modification did not address the change in the backwash piping diameter from six to two inches and that vendor information was found during the investigation that stated that rapid debris ingress could be problematic with the modification design.

Significance: The inspectors assessed the significance of the finding using Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." The inspectors followed

the guidance in Appendix A and determined that the finding did cause the licensee to trip the reactor but did not cause the loss of any mitigating equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition and therefore the finding screened as Green, very low safety significance. Cross-Cutting Aspect: H.4 - Teamwork: Individuals and work groups communicate and coordinate their activities within and across organizational boundaries to ensure nuclear safety is maintained. Specifically, the licensee failed to communicate and coordinate the findings and activities from the Environmental group's report, found in AR 2016-8839, addressing excessive mussel debris during the opening of the Circulating Water Center Intake Valve 12-WMO-30 within and across organizational boundaries to ensure that nuclear safety was maintained. The report and AR clearly outlined several factors, including high lake and forebay temperatures (>76 degrees F), low or no flow in the center intake tunnel, and the reestablishment of flow from previous no flow conditions, such as the opening of 12-WMO-30, that can have a significant effect on the volume of debris influx and which would have identified another failure mode that should have been evaluated as a design vulnerability.
<u>Enforcement</u>: The inspectors did not identify a violation of regulatory requirements associated with this finding because the NESW was not subject to the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

Failure to Follow Procedure Results in a Power Transient Due to the Loss of Two Feedwater Heater Drain Pumps and the Isolation of Extraction Steam to the Feedwater Heaters			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Initiating Events	Green FIN 05000315/2019004-03 Open/Closed	[H.1] - Resources	71153
A finding of very low safety significance (Green) was self-revealed on October 23, 2019, when the licensee failed to follow plant procedure 1-OHP-4021-060-014, "Operation of the Heater Drain Pumps," that ensures the proper operation of the pumps and the heater drain system. Specifically, the licensee's failure to correctly implement step 4.19.3 of the procedure and to open the 5A and 5B feedwater heater cross-connect valve 1-HMO-563 when the other cross-connect valve, 1-HMO-564 was already open, resulted in the 5A and 5B feedwater heaters being cross-connected between the heater drain pump suctions. This caused the tripping of both the north and south heater drain pumps and a small power transient.			
<u>Description</u>: On October 23, 2019, while operating in Mode 1 at 100 percent power, the licensee was restoring from a fill, vent, and flush of the middle heater drain pump in preparations for placing it in service and securing the south heater drain pump for maintenance. While attempting to align the middle heater drain pump to the suction of the south heater drain pump, the licensee opened the 5A and 5B feedwater heater cross-connect valve, 1-HMO-563, with the other cross-connect valve, 1-HMO-564, already open, which caused the 5A and 5B feedwater heaters to be cross-connected between the heater drain pump suctions. Levels in the 5A and 5B feedwater heaters began to shift. 1-HMO-564 was then closed. Multiple alarms came in due to the level changes in the feedwater heaters and within less			

than a minute after 1-HMO-563 was originally opened the south heater drain pump tripped on low suction pressure. Within one more minute the north heater drain pump tripped on overcurrent. Reactor power increased slightly due to the isolation of extraction steam caused by the high water levels in the feedwater heaters. The licensee reduced power by approximately two percent by reducing turbine load as necessary to maintain power at or below its licensed thermal power limit. It was later determined that control rod bank D had moved past the top of core limit of 228 steps to 230 steps when Unit 1 experienced a power reduction due to the secondary plant transient associated with the heater drain pump trips. The licensee's assessment of the event, documented in AR 2019-10383, "Unit 1 North and South Heater Drain Pumps Tripped at 100 percent Power," identifies numerous human performance errors which led to the event. These human errors include: 1) ineffective pre-job briefs; 2) having a non-licensed operator peer check a licensed operator N/A'ing a step in a procedure that would be performed in the control room by a licensed operator; 3) ineffective turnovers between operating crews; 4) the work load for the on shift operating crew being too cumbersome, and 5) being a first time task for the operator performing the swap over of heater drain pumps online.

Corrective Actions: The licensee has implemented minor procedural changes to ensure that clear direction is given to the operators to not have both feedwater heater suction header cross-ties open at the same time. Several crew and department learnings were published after the event occurred and several items were added to and/or re-enforced during operator training.

Corrective Action References: AR 2019-10383; AR 2019-10375; AR 2019-10716

Performance Assessment:

Performance Deficiency: The licensee's failure to correctly implement step 4.19.3 of procedure 1-OHP-4021-060-014, "Operation of the Heater Drain Pumps," Attachment 8, "Middle Heater Drain Pump Fill and Vent," and opened 1-HMO-563, south heater drain pump bypass valve, when 1-HMO-564, north heater drain pump bypass valve, was open caused the tripping of both the south and north heater drain pumps, the isolation of extraction steam to the feedwater heaters, and a power transient due to colder feedwater introducing positive reactivity to the core and was a performance deficiency that was in the licensee's ability to foresee and prevent.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Human Performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, on October 23, 2019, the licensee did not correctly implement step 4.19.3 of procedure 1-OHP-4021-060-014, "Operation of the Heater Drain Pumps," Attachment 8, "Middle Heater Drain Pump Fill and Vent," and opened 1-HMO-563, south heater drain pump bypass valve, when 1-HMO-564, north heater drain pump bypass valve, was open which caused the tripping of both the south and north heater drain pumps, the isolation of extraction steam to the feedwater heaters, and a power transient due to colder feedwater introducing positive reactivity to the core.

Significance: The inspectors assessed the significance of the finding using Appendix A, “The Significance Determination Process (SDP) for Findings At-Power.” Using the questions in Exhibit 1, "Initiating Events Screening Questions," under "Transient Initiators," the inspectors determined the finding was of very low safety significance (Green) because the finding did not

cause a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition. Cross-Cutting Aspect: H.1 - Resources: Leaders ensure that personnel, equipment, procedures, and other resources are available and adequate to support nuclear safety. Specifically, the licensee failed to ensure that procedure 1-OHP-4021-060-014 was adequate to support successful swap over of heater drain pumps and the licensee failed to provide adequate oversight of activities in the control room to ensure that the evolution was performed successfully and supported nuclear safety.
Enforcement: The Inspectors did not identify a violation of regulatory requirements associated with this finding because the heater drain pump operations was not subject to the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," or TS 5.4 "Procedures."

Failure to Make a Timely Report Required by 10 CFR 50.73			
Cornerstone	Severity	Cross-Cutting Aspect	Report Section
Not Applicable	Severity Level IV NCV 05000315/2019004-05 Open/Closed	Not Applicable	71153
The inspectors identified a Severity Level IV Non-Cited violation of 10 CFR 50.73(a)(2)(i)(B) when the licensee failed to submit a Licensee Event Report to the NRC within 60 days of discovery of two conditions prohibited by the plant's Technical Specifications (TSs). Specifically, the Unit 1 #2 steam generator stop valve Train B dump valve was inoperable beginning with the transition to Mode 3 on May 7, 2019, until the valve packing was replaced and retested satisfactorily on May 30, 2019. The licensee submitted LER 315/2019-002-00, Condition Prohibited by Technical Specification Due to an Inoperable Steam Generator Stop Valve Dump Valve, on November 26, 2019.			

Description:

During inspection activities that assessed the licensee's corrective actions described in the third quarter DC Cook integrated Inspection Report 05000315/2019003 and 05000316/2019003, the inspectors identified that the licensee failed to identify that the Unit 1 #2 steam generator stop valve (SGSV) Train B dump valve and associated #2 SGSV was inoperable from the time Unit 1 entered Mode 3 on May 7, 2019, until the issue was corrected on May 30, 2019. As described in the NRC third quarter inspection report, the licensee initially did not assess the impact of this issue on operability and AR 2019-8511, "Past ODE [Operability Determination Evaluation] Possibly Missed," was generated and an operability assessment was performed. This assessment was completed on September 25, 2019, and concluded that the Unit 1 #2 SGSV Train B dump valve and associated #2 SGSV was inoperable beginning with the transition from Mode 4 to Mode 3 on May 7, 2019, until the valve packing was replaced and retested satisfactorily on May 30, 2019. The licensee submitted LER 315/2019-002-00, 'Condition Prohibited by Technical Specification Due to an Inoperable Steam Generator Stop Valve Dump Valve' on November 26, 2019, 203 days after it determined through its past operability determination evaluation that the #2 SGSV Train B dump valve, 1-MRV-222 was inoperable.

Corrective Actions: The licensee entered the issue into its corrective action system as AR 2019-12428, LER 315/2019-002-00 untimely submittal to the NRC, dated 12/26/2019.

Corrective Action References: AR 2019-5615; AR 2019-8511; AR 2019-12428

Performance Assessment:

The NRC determined that this violation was associated with a previously documented finding assessed using the significance determination process. NCV 05000315/2019003-01

Enforcement:

The ROP's significance determination process does not specifically consider the regulatory process impact in its assessment of licensee performance. Therefore, it is necessary to address this violation which impedes the NRC's ability to regulate using traditional enforcement to adequately deter non-compliance.

Severity: The inspectors determined that the failure to submit a report required by 10 CFR 50.73 is a Severity Level IV violation in accordance with Section 6.9.d.9 of the NRC Enforcement Policy.

Violation: Title 10 CFR 50.73(a)(1) states, in part, a licensee shall submit a Licensee Event Report (LER) for any event of the type described in this paragraph within 60 days after discovery of the event. Paragraph 50.73(a)(2)(i)(B) states, in part, the licensee shall report any operation or condition which was prohibited by the plant's Technical Specifications.

Contrary to the above, on July 7, 2019, the licensee failed to submit a LER within 60 days after the May 7, 2019 discovery of an operation or condition which was prohibited by the plant's Technical Specifications. Specifically, the licensee reported the condition on November 26, 2019, 203 days after the May 7, 2019 determination by the licensee that the Unit 1 #2 steam generator stop valve Train B dump valve MRV-222 was inoperable from

May 7, 2019, until May 30, 2019, which was longer than the plant's Technical Specification required action completion time. In addition, the valve was inoperable when the licensee changed Modes from 4 to 3 on May 7, 2019. Both were conditions prohibited by the plant's TS. This closes LER 315-2019-002-00. Enforcement Action: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

Main Steam Safety Valves Lift Pressure Setpoints Not Within Technical Specification Lift Setting Tolerance			
Cornerstone	Severity	Cross-Cutting Aspect	Report Section
Not Applicable	Severity Level IV NCV 05000315/2019004-06 Open/Closed	Not Applicable	71153
A self-revealed Severity Level IV Non-Cited Violation (NCV) of Technical Specification (TS) 3.7.1.1 was identified when the licensee operated the plant with two main steam safety valves (MSSVs) which were found to have lift pressure setpoints that were not within TS lift setting tolerance.			
Description: On March 4, 2019, Donald C. Cook Unit 1 was in Mode 1 and holding at approximately 49 percent reactor power while performing Technical Specification (TS) surveillance testing on the main steam safety valves (MSSVs) as-found lift setpoints. The as-found setpoint pressure for Steam Generator #2 MSSV, 1-SV-1B-2, was 1109.96 pounds per square inch gauge (psig), which was 4.22 percent above the TS allowable setpoint pressure of +/- three percent of 1065 psig. After adjustment, the valve passed the as-left testing criteria of +/- one percent of the setpoint of 1065 psig. The as-found setpoint pressure for Steam Generator #2 MSSV, 1-SV-2A-2, was 1117.32 psig, which was 3.94 percent above the TS allowable setpoint pressure of +/- three percent of 1075 psig. After adjustment, the valve passed the as-left testing criteria of +/- one percent of the setpoint of 1075 psig. As a result of the test failures, the licensee expanded the testing scope to two additional valves for each failure (four valves in all) as required by the ASME OM Code. All four of the additional valves passed the as-found testing within the required criteria of +/- 3 percent for the setpoint test. The licensee determined that the apparent cause of the failed as-found setpoint pressure testing on both valves was setpoint drift. The licensee documented through historical analysis of 122 previous as-found tests that 72 valves experienced a rise in the as-found setpoint while 50 valves experienced a decrease in as-found setpoint. Of the 122 previous as-found tests, there were five failures. In addition, over the course of ten years, many valves have experienced both an increase and a decrease in as-found setpoint from one test to the next. In the licensee's analysis, this indicates that choosing a setpoint above or below the nominal value would not necessarily decrease the likelihood of failure. Therefore, the licensee believes that no further corrective action is necessary because of the low frequency of failure of the valves and the historical randomness of setpoint drift. Based on a review of the surveillance test results of previous MSSV setpoint tests, the inspectors concluded that an MSSV lifting slightly outside the acceptance tolerance did not indicate a problem with the valve, as-left testing, or surveillance test errors. Specifically, the inspectors found that MSSV performance had been within tolerance as described above and			

there was no indication of valve degradation or testing errors. Therefore, the inspectors determined that the existence of an inoperable MSSV was not reasonably within the licensee's ability to foresee and correct, and therefore, was not a performance deficiency.

Corrective Actions: MSSV 1-SV-1B-2 and 1-SV-2A-2 were adjusted to +/- one percent of their TS required pressures and returned to operable status. Safety valves 1-SV-1A-3, 1-SV-2A-3, 1-SV-3-3, and 1-SV-1A-1 were tested as expanded scope valves. All four valves passed their as-found tests.

Corrective Action References: AR 2019-1993 and AR 2019-1995.

Performance Assessment:

The NRC determined this violation was not reasonably foreseeable and preventable by the licensee and therefore is not a performance deficiency.

Enforcement:

This issue is considered within the traditional enforcement process because there was no performance deficiency associated with the violation of NRC requirements. The NRC Enforcement Manual, Section 1.2.8.E.3 states, in part, that, traditional enforcement is used for violations with no associated performance deficiencies.

Severity: The NRC determined the issue to be more than minor because the issue affected multiple MSSVs and a detailed engineering analysis was required to determine the significance of the issue. The NRC Enforcement Policy, Section 2.2.1 states, in part, that, whenever possible, the NRC uses risk information in assessing the safety significance of violations and assigning severity levels. To determine the significance of the issue, the inspectors screened the condition using IMC 0609, Appendix A, "The Significance Determination Process For Findings At Power," and determined that the condition was of very low safety significance (Green) because in the as-found condition, the MSSVs still could have performed their intended safety function in mitigating the consequences of a postulated accident. The average of the as-found test pressures of all MSSVs was below the design pressure analysis limit. Accordingly, after considering that the condition represented very low safety significance, the inspectors concluded that the violation would be best characterized as Severity Level IV under the traditional enforcement process.

Violation: Technical Specification 3.7.1 requires main steam safety valves to be operable in Modes 1, 2, and 3 with lift settings as specified in TS Table 3.7.1-2. With one or more valves inoperable, action is required to restore the valve(s) to operable status within four hours, otherwise reduce reactor power in accordance with TS or be in Mode 3 within 6 hours and in Mode 4 within the next 12 hours.

TS Table 3.7.1-2 specifies the lift setpoint pressure for setpoint pressure for Steam Generator #2 MSSV, 1-SV-1B-2 as 1065 psig,+/- 3 percent (1033 - 1097 psig) and for Steam Generator #2 MSSV, 1-SV-2A-2 as 1075 psig +/- 3 percent (1043 - 1107 psig).

Contrary to the above, during the plant operation Cycle 29 (November 26, 2017- March 6, 2019), the licensee operated with inoperable main steam safety valves in Modes 1, 2 and 3 since the lift settings were not as specified in TS Table 3.7.1.2 for Steam Generator #2 MSSV, 1-SV-1B-2 Steam Generator #2 MSSV, 1-SV-2A-2, and the licensee failed to take the action required by TS 3.7.1. Specifically, the post Cycle 29 the as-found setpoint pressure for Steam Generator #2 MSSV, 1-SV-1B-2, was 1109.96 psig, which was 4.22 percent above the

TS allowable setpoint pressure of +/- 3 percent of 1065 psig and the as-found setpoint pressure for Steam Generator #2 MSSV, 1-SV-2A-2, was 1117.32 psig, which was 3.94 percent above the TS allowable setpoint pressure of +/- 3 percent of 1075 psig.

Enforcement Action: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

EXIT MEETINGS AND DEBRIEFS

The inspectors confirmed that proprietary information was controlled to protect from public disclosure.

- On January 8, 2020, the inspectors presented the integrated inspection results to Mr. J. Gebbie, Chief Nuclear Officer and other members of the licensee staff.
- On October 17, 2019, the inspectors presented the Rad Hazard Assessment Exposure Control and Occupational ALARA inspection results to Mr. S. Lies, Site Vice President and other members of the licensee staff.
- On October 24, 2019, the inspectors presented the Inservice Inspection inspection results to Mr. S. Lies, Site Vice President and other members of the licensee staff.
- On October 28, 2019, the inspectors presented the Emergency Preparedness Inspection inspection results to Mr. J. Gebbie, Chief Nuclear Officer and other members of the licensee staff.
- On December 19, 2019, the inspectors presented the Licensed Operator Requalification program annual examination results review inspection results to Mr. C. Peak, Licensed Operator Requalification Training Lead and other members of the licensee staff.

DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71111.01	Procedures	12-OHP-4022-001-010	Severe Weather	07/21/2017
		PMP-5055-SWM-001	Severe Weather Guidelines	07/27/2017
71111.04Q	Drawings	OP-1-5106A-64	Flow Diagram Aux-Feedwater Unit 1	64
		OP-2-5151A-59	Flow Diagram Emergency Diesel Generator "AB" Unit No. 2	59
		OP-2-5151B-69	Flow Diagram Emergency Diesel Generator "AB" Unit No. 2	69
	Procedures	1-OH-4021-008-002	Placing Emergency Core Cooling System in Standby Readiness	36
		1-OHP-4021-008-007	Operation of the Safety Injection Pumps	10
		1-OHP-4030-156-017E	East Motor Driven Auxiliary Feedwater System Test	15
		2-OHP-4021-032-008AB	Operating DG2AB Subsystems	36
71111.04S	Procedures	2-OHP-4021-008-002	Placing Emergency Core Cooling System in Standby Readiness	33
71111.05Q	Fire Plans	FZ 15	Fire Area AA14, 1 CD Diesel Generator Room-587'	35
		FZ 16	Fire Area AA15, 1AB Diesel Generator Room - 587'	35
		FZ 53	Fire Area AA 46 - Unit 1 Control Room - 633'	35
		FZ 54	Fire Area AA 47 - Unit 2 Control Room - 633'	35
71111.06	Work Orders	55525881-01	MTE, Unit 1 Wet Cable Aux Building Inspections	08/05/2019
71111.08P	Corrective Action Documents	AR 2016-12195	Foreign Object Identified in SG-24	10/23/2016
		AR 2016-12205	SG-22 Volumetric Indication	10/23/2016
		AR 2016-13865	AFW Piping Found Below Design Limit	12/06/2016
		AR 2018-2133	Boric Acid Leak Increase Since Last Report	03/01/2018
		AR 2018-2142	Boric Acid Leak Initial Entry (near 2-PP-38)	03/01/2018
		AR 2018-3239	U2 Pressure Relief Tank coating/corrosion issue	04/19/2018
		AR 2018-4664	OME-1, Boric Acid on Lower Surface of Unit 2 Reactor Vessel	05/10/2018
		AR 2018-4943	2IFI-54 Active Leak	05/03/2018

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		AR 2019-6618	Components Erroneously Exempted from ISI Examination	07/03/2019
		AR 2019-8520	Arc Strike on 2-GC-L-42	09/05/2019
	Corrective Action Documents Resulting from Inspection	AR 2019-10017	NRC Unit 2 ISI Inspection Issue of Concern	10/17/2019
		AR 2019-10076	NRC Observation of UT Examination	10/18/2019
		AR 2019-10321	NRC Unit 2 ISI Inspection Observation	10/23/2019
		AR 2019-10324	NRC Unit 2 ISI Inspection Observation	10/23/2019
		AR 2019-10326	NRC Unit 2 ISI Inspection Observation	10/23/2019
		AR 2019-10338	NRC Unit 2 ISI Inspection Issue of Concern EOC	10/23/2019
	Miscellaneous		Personnel Certification Statement- UT Level II, M. Smith	08/24/2019
			Personnel Certification Statement – VT-1, 2 and 3 Level III, R. Fuller	08/21/2019
			Data Record - Safety Related Weld Data Block	04/27/2018
			Vendor Technical Basis for Revision 7 of Procedure PDI-ISI-254	02/14/2005
		1.2TS	Weld Procedure Specification	6
		1.8TS	Weld Procedure Specification	6
		232	Weld Procedure Qualification Data Sheet	03/09/1989
		234	Weld Procedure Qualification Data Sheet	03/20/2008
		235	Weld Procedure Qualification Data Sheet	03/30/1989
		255	Weld Procedure Qualification Data Sheet	08/08/1989
		EPRI Letter	EPRI Performance Demonstration Procedures Qualification for WesDyne Procedure PDI-ISI-254 Revision 9	11/26/2019
		PDQS 1067	E. Cawley Wesdyne Level II UT PDI-UT-254 Revision 4	08/26/2016
		PDQS 1149	M. Smith LMT-10-PAUT-002	08/16/2017
		PDQS 405	S. Sabo Wesdyne Level III UT PDI-UT-254 Revision 3	06/10/2014
		PDQS 407	Procedure PDI-ISI-254, Remote Inservice Inspection of Reactor Vessel Shell Welds, Revision 4	08/01/2001
		PDQS 574	C. Wyffels Wesdyne Level III UT PDI-UT-254 Revision 3	06/27/2014
		PDQS 851	E. Overly Wesdyne Level II UT PDI-UT-254 Revision 4	02/07/2018
		PDQS 885	Procedure LMT-10-PAUT-002, Manual Phased Array Ultrasonic Examination of Austenitic and Ferritic Piping Welds	01/03/2018
		PDQS-932	Procedure: PDI-ISI-254, Revision 9 Remote Inservice	11/12/2019

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
			Examination of Reactor Shell Welds	
		Report 51 - 9263363 - 000	DC Cook U2C23 Steam Generator Condition Monitoring and Operational Assessment	0
	NDE Reports		BD/2-BD-103-2 VT-2 Examination Record	05/11/2018
			2-BD-103-2 2 OW2, 3 and 4 Socket Welds- Liquid Penetrant Examination Record	04/28/2018
		U2-VE-19-014	2-SI-58-23 Weld - Ultrasonic Examination Report	10/22/2019
		U2-VE-19-028	Weld 2-RPV-D	12/03/2019
		U2-VE-19-030	Weld 2-RPV-A	12/03/2019
		U2-VE-19-049	Weld 2-N3-I	12/03/2019
	Procedures	LMT-10-PAUT-002	Manual Phased Array Ultrasonic Examination of Austenitic and Ferritic Piping Welds	1
		PDI-ISI-254	Remote Inservice Inspection of Reactor Vessel Shell Welds	4
		PMI-5032	Boric Acid Corrosion Control Program	7
		PMP-5050-001-001	Boric Acid Corrosion Control	23
		WDI-STD-088	Underwater Remote Visual Examination of Reactor Vessel Internals	15
		WDI-STD-1000	Remote Inservice Inspection of Reactor Vessel Shell Welds	9
	Work Orders	55517263-01	Weld 2-BD-103-2	04/28/2018
71111.11A	Miscellaneous		DC Cook Licensed Operator Requalification Program Annual Examination Summary Results for 2019	12/19/2019
71111.11Q	Miscellaneous	RQ-E-4405-U1-A	Period 4405 As Found Simulator Evaluation Unit 1, Simulator Evaluation Exercise Guide	0
71111.12	Corrective Action Documents	AR 2019-5226	1-FW-109-6A Unable to Close More than 3 Handwheel Turns	05/14/2019
		AR 2019-5229	1-FW-109-6A is Suspected to be Partially Closed	05/14/2019
71111.13	Miscellaneous	Cook Nuclear Plant Plan of the Day Meeting package	Unit 1 Online Risk - Rev 0, Cycle 112 Week 10	12/02/2019
	Procedures	1-OHP-SP-443	Removing Heater Drain Pumps From Service In Support of Main Turbine Valve Testing	0
		PMP-2291-WAR-	Work Activity Risk Management Process for Hang/Remove	12/02/2019

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		001	Clearance on North Hotwell (1-PP-5N) #1291338	
		PMP-2291-WAR-001	Work Activity Risk Management Process, Data Sheet 1, Work Activity Risk Evaluation form for U1 Main Turbine and Feed Pump Turbine Valve Testing	12/07/2019
71111.15	Corrective Action Documents	2019-12297	As Found Data Out of Tolerance	12/18/2019
		AR 2019-10307	Unit 1 CRID 3 Inverter Abnormal	10/22/2019
		AR 2019-10605	Malfunction during Stroking of 2-QMO-201	10/29/2019
		AR 2019-11046	Missed Surveillance for RCP #21 Floor Plug	11/07/2019
		AR 2019-11077	Drop in Frequency Noted during LOP/LOCA Testing	11/07/2019
	Miscellaneous	eSOMS Unit 2 Control Room Log	Unit 2 Control Room Log	11/18/2019
	Procedures	2-IHP-4030-251-010	Steam Generators 2 and 3 Steam Pressure Protection Set 3 Channel Operational Test and Calibration	12/17/2019
		2-IHP-4030-251-010	Steam Generators 2 and 3 Steam Pressure Protection Set 3 Channel Operational Test and Calibration	12/18/2019
		PMP-4030-EXE-001	Conduct of Surveillance Testing	25
71111.18	Engineering Changes	2-MOD-40046	Unit 2 Main Turbine and Main Feed Pump Turbines Upgrade Control and Protection Project	0
		ECP No.: 2-T2-07	Unit 2 Main Turbine TS3000 Digital Control System Basis of Design Document	11
	Miscellaneous	DW-04-009	Direct Work Request to Please Consider an Alternate PRZR Spray Termination Criteria In E-3 To Avoid Challenging PRZR and Ruptured SG Overfill	03/04/2009
		E-3	Steam Generator Tube Rupture	3
		E-3	Steam Generator Tube Rupture	3+
		ICP-01776 - 50.59 Screen	Updates to ECP No. 1-2-O0-14, EOP Footnotes	0
		PSBD Revision 24	12-OHP-4023-E-3	24
	Procedures	1-OHP-4023	Steam Generator Tube Rupture	25
		2-IHP-6030-IMP-288	Delta T / T AVG Turbine Runback Test and Timer Calibration	6

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71111.19	Corrective Action Documents	AR 2019-10932	2-NRV-152 Failed to Fully Open on Backup Air	11/05/2019
		AR 2019-11423	2-NRI-32 is not Responding Correctly and is Inoperable	11/14/2019
	Miscellaneous	436-C ASME Procedure Qualification Record	Gas Tungsten Arc	0
		437-C ASME Procedure Qualification Record	Shielded Metal Arc	0
		6.43S - Welding Procedure Specification	Manual Gas Tungsten Arc and Shielded Metal Arc Welding	0
	Procedures	12-THP-6020-CHM-307	Emergency Diesel Fuel Oil	32
		EC-0000054410-TP-002A	U2 Solid State Protection System (SSPS) Train A & B Channel 1 Testing	1
		EC-0000054410-TP-002E	Unit 2 Solid State Protection System (SPSS) Train A & B Multiple Channel Testing	0
		PMP-2350-INS-001	Conduct of Inspection Activities	12
		PMP-2350-INS-0011	Conduct of Inspection Activities	12
		PMP-4043-EQC-001	Equipment Control - CD Fuel Oil	10/21/2019
	Work Orders	55474816 - 30	1-89-WRT-101, 600VAC Welding Receptacle Disconnect Switch	04/12/2019
		55506847 - 29	PRF 040248 - U2 SSPS Upgrade	10/28/2019
		55506847 - 29	PRF 040248 - U2 SSPS Upgrade	11/04/2019
		55538240-01	U1 West Main Feed Pump Casing Drain Weld Leak	09/29/2019
		55538240-06	WLD, 1-PP-1W, Repair Casing Steam Leak & Restore Drain Line	09/30/2019
		55539092-10	MTI, 2-NRI-32 Calibrate Loop	11/18/2019
		55540428-75	MTI, 2-NRI-32; 12-IHP-6030-IMP-011; Test and Swap	11/19/2019

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
			Cables	
		55540428-81	MTI, 2-NRI-36-DWR, Perform COT, 2-IHP-4030-213-230	11/18/2019
		55540486 - 01, 05, 06, 07, 09, 10, 19, 20, 25, 26, 28, 35, and 36	2 CRDMG 2S-MTR, Reactor Rod Control South Motor-Generator Set CRDMG-2S Motor and Supply Breaker	11/17/2019
71111.20	Corrective Action Documents Resulting from Inspection	AR 2019-11225	NRC has Questioned the Thread Engagement on the Prz Bottles	11/10/2019
		AR 2019-11397	Question from the NRC Resident Regarding SG Safety Leakby	11/12/2019
	Miscellaneous		U2C25 Refueling Outage Shutdown Safety Plan Report	0
	Procedures	12-MHP-4050-FHP-023	Reactor Vessel Head Removal with Fuel in the Vessel	1
		12-MHP-5021-001-009	Torque Selection	24
		2-OHP-4030-217-054W	West Residual Heat Removal Train Operability Test - Shutdown	10/28/2019
		PMP-2060-WHL-001	Work Hour Limitation and Fatigue Management	10
71111.22	Corrective Action Documents	2019-10604	2-IMO-225, W CTS Pump Suction from RWST Could not be Stroked	10/29/2019
		2019-10605	Malfunction During Stroking of 2-QMO-201	10/29/2019
		2019-10610	2-IRV-310 Failed Stroke Timing	10/29/2019
	Procedures	2-EHP-4030-234-203	Unit 2 LLRT	28
		2-OHP-4030-208-008R	ECCS Check Valve Test	34
		2-OHP-4030-208-008R	ECCS Check Valve Test	34
		2-OHP-4030-208-053A	ECCS Valve Operability Test - Train A	36
		2-OHP-4030-208-053B	ECCS Valve Operability Test - Train B	33

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		2-OHP-4030-232-001	Simultaneous Start of AB and CD Diesel Generators	10/20/2019
		2-OHP-4030-232-217B	DG2AB Load Sequencing and ESF Testing	56
		2-OHP-4030-232-217B, Attachment 9	3500/600 KW Load Rejection Testing	10/19/2019
		2-OHP-5030-019-002E	East Essential Service Water System Flow Test	10
		OHI-4032	Leakage Monitoring Program	19
	Work Orders	55467357	FLEX Blended RCS Makeup Pump Visual Inspection	01/21/2019
		55521662 Completion Document	2-SV-64 IST Safety Valve Bench Testing, Group 2-CCW-03	10/24/2019
71114.04	Corrective Action Documents	AR2018-10527	DC Cook Emergency Plan Revision Summary Not Submitted to the NRC	11/20/2018
		AR2018-7846	50.54q Screen - Lakeland Medical Document	08/07/2018
	Miscellaneous	Evaluation 18-11	10 CFR 50.54(q) Screening Form	08/09/2018
		Evaluation 18-13	10 CFR 50.54(q) Screening Form	08/22/2018
		Evaluation 19-04	10 CFR 50.54(q) Screening Form	01/28/2019
71124.01	Radiation Surveys	20191007-28	Auxiliary Building Hallway 587'	10/10/2019
		20191007-9	Auxiliary Building Hallway/Boric Acid Evap Ion Exchanger 633'	10/07/2019
		20191009-4	RCP 1-4 Pressurizer Walkway Containment 621'	
		20191015-4	Upper Containment (Ctmt) Spent Fuel Pit 701'	10/15/2019
		20191018-1	Core Barrel Surveys 650'	10/18/2019
	Radiation Work Permits (RWPs)	RWP-192160	U-2 Instrument Room Seal Table Activities	1
		RWP-192187	U-2 Medium Risk; Under Reactor Vessel Inspection	0
		RWP-192105	U-2 Former Bolt Inspection Hold Down Spring Replacement	3

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71124.02	ALARA Plans	RWP-192105	U-2 Former Bolt Inspection Hold Down Spring Replacement	3
		RWP-192160	U-2 Instrument Room Seal Table Activities	1
		RWP-192187	U-2 Medium Risk; Under Reactor Vessel Inspection	2
	Procedures	PMP-6010-ALA-001	ALARA Program Review Of Plant Work Activities	35
71124.06	Work Orders	55511641 01	Auxiliary Building Ventilation Gaseous Effluent Flow Functional Test	03/20/2018
71151	Miscellaneous		MSPI Derivation Report - MSPI Cooling Water System – Unit 2	09/2019
			MSPI Derivation Report - MSPI Heat Removal System – Unit 1	06/2019
			MSPI Derivation Report - MSPI Heat Removal System – Unit 2	06/2019
			MSPI Derivation Report - MSPI Residual Heat Removal System - Unit 1	06/2019
			MSPI Derivation Report - MSPI Residual Heat Removal System - Unit 2	06/2019
			Custom Unavailability Report for the Component Cooling Water System	09/30/2019
			Custom Unavailability Report for the Essential Service Water System	09/30/2019
			Custom Unavailability Report for the Auxiliary Feedwater System	06/30/2019
			Custom Unavailability Report for the ECCS and RHR Systems	06/30/2019
			Operators' Log - Unit 1	07/01/2018 - 09/30/2019
			Operators' Log - Unit 2	07/01/2018 - 09/30/2019
			MSPI Derivation Report - MSPI Emergency AC Power System - Unit 1	12/2018

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
			MSPI Derivation Report - MSPI Emergency AC Power System - Unit 2	12/2018
			MSPI Derivation Report - MSPI Emergency AC Power System - Unit 1	03/2018
			MSPI Derivation Report - MSPI Emergency AC Power System - Unit 2	03/2018
			MSPI Derivation Report - MSPI Emergency AC Power System - Unit 1	06/2019
			MSPI Derivation Report - MSPI Emergency AC Power System - Unit 2	06/2019
			MSPI Derivation Report - MSPI Emergency AC Power System - Unit 1	09/2019
			MSPI Derivation Report - MSPI Emergency AC Power System - Unit 2	09/2019
			MSPI Derivation Report - MSPI Cooling Water System - Unit 1	09/2019
		MSPI Margin Report	MSPI Indicator Margin Remaining in Green - Unit 1	12/2018
		MSPI Margin Report	MSPI Indicator Margin Remaining in Green - Unit 2	12/2018
		MSPI Margin Report	MSPI Indicator Margin Remaining in Green - Unit 2	03/2019
		MSPI Margin Report	MSPI Indicator Margin Remaining in Green - Unit 1	06/2019
		MSPI Margin Report	MSPI Indicator Margin Remaining in Green - Unit 2	06/2019
		MSPI Margin Report	MSPI Indicator Margin Remaining in Green - Unit 1	09/2019
		MSPI Margin Report	MSPI Indicator Margin Remaining in Green - Unit 2	09/2019
	Procedures	PMP-7110-PIP-001	Reactor Oversight Program Performance Indicators and Monthly Operating Report Data	19
		PMP-7110-PIP-001	Reactor Oversight Program Performance Indicators and Monthly Operating Report Data	18

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		PMP-7110-PIP-001	Reactor Oversight Program Performance Indicators and Monthly Operating Report Data	17
71152	Corrective Action Documents	AR 2019-11667	Unit 2 DMIMS Alarms	11/24/2019
		AR 2019-11697	High 50/60 Hz Noise on Channels 3 and 12	11/25/2019
		AR 2019-11698	DMIMS Unit 2 Channels 7 and 8 Rejected Alarms	11/25/2019
		AR 2019-11713	2-DMIMS Alarm	11/26/2019
		AR 2019-4856	11 RCP - Loose Lateral Restraint Bumper Shim	05/05/2019
		AR 2019-5942	Section XI Valves did not Receive NDE and CMTRs	06/10/2019
		GT 2019-11714	Replace All Safety Related Centerline Check Valves	11/26/2019
71153	Corrective Action Documents	AR 2016-8839	Zebra Mussel Release Occurred While Opening 12-WMO-30	08/02/2016
		AR 2019-10375	U-1 North Feedwater Heater Drain Pump Tripped When Removing the Middle Condensate Booster Pump	10/24/2019
		AR 2019-10383	Unit 1 North and South Heater Drain Pumps Tripped At 100% Power	10/23/2019
		AR 2019-5615	1-MRV-222 Failed Surv	05/29/2019
		AR 2019-7047	Received Ann 204 drop 13 South NESW pp strainer DP high	07/21/2019
	Corrective Action Documents Resulting from Inspection	AR 2019-12428	LER 315/2019-002-00 not submitted to NRC in timely manner	12/26/2019
		AR 2019-8511	Past ODE Possibly Missed	09/05/2019
	Engineering Changes	EC-0000054253	NESW Pumps and Discharge Strainers	0
	Procedures	12-EHP-5040-MOD-009	Engineering Change Reference Guide	60