UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 E. LAMAR BLVD
ARLINGTON, TX 76011-4511

May 1, 2017

Mr. Ken Higginbotham
Vice President-Nuclear and CNO
Nebraska Public Power District
Cooper Nuclear Station
72676 648A Avenue
P.O. Box 98
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION – NRC INTEGRATED INSPECTION REPORT 05000298/2017001

Dear Mr. Higginbotham:

On March 31, 2017, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Cooper Nuclear Station. On April 13, 2017, the NRC inspectors discussed the results of this inspection with Mr. D. Buman, Director, Nuclear Safety Assurance, and other members of your staff. The results of this inspection are documented in the enclosed report.

NRC inspectors documented six findings of very low safety significance (Green) in this report. All of these findings involved violations of NRC requirements. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a of the Enforcement Policy.

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

If you disagree with a cross-cutting aspect assignment 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 IV; and the NRC resident inspector at the Cooper Nuclear Station.

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 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

/RA/

Gregory G. Warnick, Chief
Project Branch C
Division of Reactor Projects

Docket No. 50-298
License No. DPR-46

Enclosure:
Inspection Report 05000298/2017001
w/ Attachments:
1. Supplemental Information
2. Information Request

COOPER NUCLEAR STATION – NRC INTEGRATED INSPECTION REPORT
05000298/2017001 - DATED MAY 1, 2017

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000298
License:	DPR-46
Report:	05000298/2017001
Licensee:	Nebraska Public Power District
Facility:	Cooper Nuclear Station
Location:	72676 648A Ave Brownville, NE
Dates:	January 1 through March 31, 2017
Inspectors:	P. Voss, Senior Resident Inspector C. Henderson, Resident Inspector P. Elkmann, Senior Emergency Preparedness Inspector E. Uribe, Project Engineer I. Anchondo, Reactor Inspector R. Deese, Senior Reactor Analyst J. Watkins, Reactor Inspector
Approved By:	Gregory G. Warnick Chief, Project Branch C Division of Reactor Projects

SUMMARY

IR 05000298/2017001; 01/01/2017 – 03/31/2017; Cooper Nuclear Station; Equip. Alignment, Maint. Effectiveness, Operability Determinations & Functionality Assessments, Problem ID. & Resolution, Follow-up of Events & Notices of Enforcement Discretion.

The inspection activities described in this report were performed between January 1 and March 31, 2017, by the resident inspectors at Cooper Nuclear Station and inspectors from the NRC's Region IV office. Six findings of very low safety significance (Green) are documented in this report. All of these findings involved violations of NRC requirements. The significance of inspection findings is indicated by their color (i.e., Green, greater than Green, White, Yellow, or Red), determined using Inspection Manual Chapter 0609, "Significance Determination Process," dated April 29, 2015. Their cross-cutting aspects are determined using Inspection Manual Chapter 0310, "Aspects within the Cross-Cutting Areas," dated December 4, 2014. Violations of NRC requirements are dispositioned in accordance with the NRC Enforcement Policy. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," dated July 2016.

Cornerstone: Initiating Events

- Green. The inspectors reviewed a self-revealed, non-cited violation of Technical Specification 5.4.1.a, for the licensee's failure to implement Maintenance Procedure 7.3.16, "Low Voltage Relay Removal and Installation," Revision 22, for relay replacement work. Specifically, on October 28, 2016, the licensee failed to evaluate the potential impact of primary containment isolation system relay PCIS-REL-K27 work on shutdown cooling relay PCIS-REL-K30, which was mounted next to K27 and shared a common mounting rail. As a result, the licensee did not identify the potential of losing residual heat removal shutdown cooling, and while installing the K27 relay and snapping it into the mounting rail, workers caused a momentary actuation of relay K30 and a loss of residual heat removal shutdown cooling. Corrective actions to restore compliance included restoration of shutdown cooling, completion of the K27 relay maintenance with shutdown cooling out of service, and an outage risk management procedure change that prohibited work on or near shutdown cooling relays while the system was required to be in service. The licensee entered this deficiency into the corrective action program as Condition Report CR-CNS-2016-07645.

  The licensee's failure to implement Maintenance Procedure 7.3.16, in violation of Technical Specification 5.4.1.a, was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it was associated with the equipment performance attribute of the Initiating Events Cornerstone and affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown operations. Using Inspection Manual Chapter 0609, Appendix G, Attachment 1, "Shutdown Operations Significance Process Phase 1 Initial Screening and Characterization of Findings," dated May 9, 2014, the inspectors determined that the finding did not require a quantitative assessment because the event occurred when the refuel canal/cavity was flooded. Therefore, the finding screened as very low safety significance (Green). The finding had a cross-cutting aspect in the area of human performance associated with work management, because the licensee failed to implement a process of planning, controlling, and executing work activities such that nuclear safety was the overriding priority, including the need for coordination with different work groups or job activities. Specifically, the licensee failed to control, execute, and coordinate safety-related primary containment isolation system relay work activities to ensure residual heat removal shutdown cooling was not adversely impacted [H.5]. (Section 4OA3)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of Technical Specification 5.4.1.a for the licensee's failure to maintain Emergency Procedure 5.1ASD, "Alternate Shutdown," Revision 17, for establishing reactor equipment cooling system flow to the high pressure coolant injection system fan coil unit. Specifically, the licensee failed to maintain Emergency Procedure 5.1ASD with adequate instructions to place the reactor equipment cooling system north or south critical loop in service and verify reactor equipment system flow to the high pressure cooling injection system fan coil unit during some control room evacuation scenarios. The immediate corrective actions were to assess operability of the high pressure coolant injection system during control room evacuations that are not related to fire scenarios, and to revise Emergency Procedure 5.1ASD with instructions to open the critical loop supply valves (REC-MOV-711 or REC-MOV-714) in the control room or locally, and verify reactor equipment system flow to the high pressure coolant injection fan coil unit. The licensee entered this deficiency into the corrective action program as Condition Report CR-CNS-2017-01403.

  The licensee's failure to maintain Emergency Procedure 5.1ASD to establish reactor equipment cooling system flow to the high pressure coolant injection fan coil unit during some control room evacuation scenarios, in violation of Technical Specification 5.4.1.a, was a performance deficiency. The performance deficiency was determined to be more than minor, and therefore a finding, because it was associated with the procedural quality attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events. Specifically, the licensee did not provide instructions to establish reactor equipment cooling system flow to the high pressure coolant injection system fan coil unit, which would have complicated operator response during a control room evacuation. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, the inspectors determined that the finding had very low safety significance (Green) because it: was not a design deficiency; did not represent a loss of system and/or function; did not represent an actual loss of function; did not represent an actual loss of function of at least a single train for longer than its technical specification allowed outage time; and did not result in the loss of a high safety-significant nontechnical specification train. The finding had a cross-cutting aspect in the area of problem identification and resolution associated with identification. Specifically, the licensee failed to implement a corrective action program with a low threshold for identifying issues during the required annual review of emergency procedures [P.1]. (Section 1R04)

- Green. The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," associated with the licensee's failure to identify a condition adverse to quality associated with Station Procedure 2.2.24.1, "250 Vdc Electrical System (Div 1)," Revision 14, in accordance with Station Procedure 0-CNS-LI-102, "Corrective Action Process," Revision 6. Specifically, the licensee failed to identify that Station Procedure 2.2.24.1 contained inadequate instructions to ensure the oncoming charger 1C output voltage was matched with the bus 1A voltage when transferring bus 1A from charger 1A to charger 1C, so that technical specification bus voltage requirements would remain met. This resulted in an unexpected and initially unrecognized decline in voltage on the bus to below the required minimum of 260.4 Vdc.

This condition required the licensee to declare the Division 1 250 Vdc electrical system and Division 1 residual heat removal low pressure coolant injection system inoperable. The immediate corrective action was to adjust the charger 1C float voltage greater than 260.4 Vdc to restore operability of the Division 1 250 Vdc electrical and residual heat removal low pressure coolant injection systems. The licensee entered this deficiency into the corrective action program as Condition Reports CR-CNS-2016-08658 and CR-CNS-2017-00750.

The licensee's failure to identify a condition adverse to quality associated with Station Procedure 2.2.24.1, to ensure technical specification bus voltage requirements were met, in violation of Station Procedure 0-CNS-LI-102, was a performance deficiency. The performance deficiency was determined to be more than minor, and therefore a finding, because it was associated with the procedural quality attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events. Specifically, charger 1C, when in service, did not maintain battery 1A terminal voltage within the requirements of Surveillance Requirement 3.8.4.1, which required the licensee to declare the Division 1 250 Vdc electrical system and the Division 1 residual heat removal low pressure coolant injection system inoperable. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, the inspectors determined that the finding had very low safety significance (Green) because it: was not a design deficiency; did not represent a loss of system and/or function; did not represent an actual loss of function; did not represent an actual loss of function of at least a single train for longer than its technical specification allowed outage time; and did not result in the loss of a high safety-significant, nontechnical specification train. The finding had a cross-cutting aspect in the area of problem identification and resolution associated with evaluation. Specifically, the licensee failed to thoroughly evaluate the charger 1C float voltage issue to ensure that the resolution addressed the cause and extent of condition commensurate with the safety significance [P.2]. (Section 1R12)

- Green. The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," associated with the licensee's failure to identify a condition adverse to quality for Division 1 residual heat removal service water booster pump A, in accordance with Station Procedure 0-CNS-LI-102, "Corrective Action Process," Revision 6. Specifically, on January 5, 2017, the inspectors identified an oil level lower than normally expected, oil on the pump skid, and an oil droplet formed on the Division 1 residual heat removal service water booster pump A inboard bearing sight glass. The inspectors informed the control room of this condition, and the licensee determined the oil leakage from the pump's sight glass would have prevented the pump from operating for the required 30 days during a design basis accident. The immediate corrective action was to repair the Division 1 residual heat removal service water booster pump A inboard bearing sight glass, restoring operability of the pump. The licensee entered this deficiency into the corrective action program as Condition Report CR-CNS-2017-00054.

  The licensee's failure to identify a condition adverse to quality for Division 1 residual heat removal service water booster pump A, in violation of Station Procedure 0-CNS-LI-102, was a performance deficiency. The performance deficiency was determined to be more than minor, and therefore a finding, because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events. Specifically, the oil leakage from the service water booster pump A inboard bearing sight glass would have prevented the pump from operating for its required 30-day mission time during a design basis accident and resulted in the pump being declared inoperable. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, the inspectors determined that the finding had very low safety significance (Green) because it: was not a design deficiency; did not represent a loss of system and/or function; did not represent an actual loss of function; did not represent an actual loss of function of at least a single train for longer than its technical specification allowed outage time; and did not result in the loss of a high safety-significant nontechnical specification train. The finding had a cross-cutting aspect in the area of human performance associated with challenge the unknown because the licensee failed to stop when faced with uncertain conditions and failed to ensure that risks are evaluated and managed before proceeding. Specifically, the licensee did not maintain a questioning attitude during job-site reviews to identify and resolve unexpected conditions, including lower than the expected oil level in the service water booster pump A inboard bearing sight glass, oil on the pump skid, and an oil droplet formed on the bottom of the sight glass [H.11]. (Section 1R15)

- <u>Green</u>. The inspectors identified a non-cited violation of 10 CFR 50.55a(g)(4) for the licensee's failure to use an approved method to disposition an American Society of Mechanical Engineers Code nonconforming condition in the residual heat removal service water system. Specifically, the licensee identified multiple locations with localized pipe thinning below the American Society of Mechanical Engineers Code B31.1 design minimum pipe-wall thickness during an ultrasonic examination but failed to use an approved method to calculate a new acceptable pipe-wall thickness. As a corrective action to restore compliance, the licensee replaced this section of piping on November 1, 2016, during Refueling Outage 29. The licensee entered this issue into the corrective action program as Condition Reports CR-CNS-2016-05558 and CR-CNS-2016-05963.

  The licensee's failure to use an approved method to calculate a new minimum allowable pipe-wall thickness, in violation of 10 CFR 50.55a(g)(4), was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it was associated with the design control attribute of the Mitigating System Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, calculating an allowable minimum pipe-wall thickness value that is below the American Society of Mechanical Engineers code design minimum value reduces the piping's structural integrity, potentially leading to the failure of the piping. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, inspectors determined the finding screened as having very low safety significance (Green) because it: was not a design deficiency; did not represent a loss of system and/or function; did not represent an actual loss of function of at least a single train for longer than its technical specification allowed outage time; and did not result in the loss of a high safety-significant nontechnical specification train. This finding had a cross-cutting aspect in the area of human performance associated with design margins because the licensee failed to operate and maintain the residual heat removal service water system within the American Society of Mechanical Engineers code minimum pipe-wall thickness. Specifically, having identified that the affected pipe location was below the allowable pipe-wall thickness, the licensee opted to calculate and accept a new minimum pipe-wall thickness value that was not consistent with code requirements instead of repairing the affected piping at the time of discovery [H.6]. (Section 4OA2)

- Green. The inspectors reviewed a self-revealed, non-cited violation of Technical Specification 3.0.4 for the licensee's failure to install the correct reactor core isolation cooling pressure control valve, RCIC-AOV-PCV23, mechanical stop and verify proper operation of the system prior to entering a mode of applicability for Technical Specification 3.5.3. This condition resulted in RCIC-AOV-PCV23 going fully open during surveillance testing following Refueling Outage 29, causing a pressure transient. This transient caused a failure of the reactor core isolation cooling turbine lube oil cooler gasket, lifting of a pressure relief valve, and a water leak. The licensee immediately shut down the reactor core isolation cooling system and declared it inoperable. The immediate corrective actions were to restore RCIC-AOV-PCV23 from the closed mechanical stop to the required open mechanical stop and to replace the turbine lube oil cooler gasket to restore operability of the system. The licensee entered this deficiency into the corrective action program as Condition Report CR-CNS-2016-08122 and initiated a root cause evaluation to investigate this condition.

  The licensee's failure to install the correct reactor core isolation cooling pressure control valve, RCIC-AOV-PCV23, mechanical stop and verify proper operation of the system prior to entering a mode of applicability for Technical Specification 3.5.3, in violation of Technical Specification 3.0.4, was a performance deficiency. The performance deficiency was determined to be more than minor, and therefore a finding, because it was associated with the design control attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events. Specifically, the licensee installed RCIC-AOV-PCV23 with the incorrect mechanical stop, and proper valve operation was not verified after installation during Refueling Outage 29, which caused the reactor core isolation cooling system to lose function during surveillance testing. This transient caused a failure of the reactor core isolation cooling turbine lube oil cooler gasket and an associated water leak. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, the inspectors determined that the finding required a detailed risk evaluation because it represented a loss of system and/or function. In the detailed risk evaluation, the analyst assumed the reactor core isolation cooling system was unavailable for 50 hours. The analyst used the Test/Limited Use Version COOPER-DEESE-HCI03 of the Cooper SPAR model run on SAPHIRE, Version 8.1.5. The analyst updated the initiating event frequencies for transients, losses of condenser heat sink, losses of main feed water, grid related losses of offsite power, and switchyard centered losses of offsite power to the more recent values from the 2014 update to the industry data found in INL/EXT-14-31428, "Initiating Event Rates at U.S. Nuclear Power Plants, 1998-2013," Revision 1. From this, the finding was determined to have an increase in core damage frequency of 8.4E-8/year and to be of very low safety significance (Green). Transients, losses of condenser heat sink, and losses of main feed water were the dominant core damage sequences. The automatic depressurization system and the reactor protection system remained to mitigate these sequences. The finding had a cross-cutting aspect in the area of human performance associated with documentation because the licensee failed to create and maintain complete, accurate, and up-to-date documentation associated with RCIC-AOV-PCV23 design drawings and the maintenance procedure for setting and testing the mechanical stop [H.7]. (Section 4OA3)

PLANT STATUS

The Cooper Nuclear Station began the inspection period at full power, where it remained for the rest of the reporting period, except for minor reductions in power to support scheduled surveillances and rod pattern adjustments.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness for Seasonal Extreme Weather Conditions

a. Inspection Scope

On February 7, 2017, the inspectors completed an inspection of the station's readiness for seasonal extreme weather conditions. The inspectors reviewed the licensee's adverse weather procedures for seasonal low temperatures and evaluated the licensee's implementation of these procedures. The inspectors verified that prior to the onset of cold weather, the licensee had corrected weather-related equipment deficiencies identified during the previous cold weather season.

The inspectors selected one risk-significant system that was required to be protected from cold weather:

- Battery Room

The inspectors reviewed the licensee's procedures and design information to ensure the system would remain functional when challenged by adverse weather. The inspectors verified that operator actions described in the licensee's procedures were adequate to maintain readiness of this system. The inspectors walked down portions of this system to verify the physical condition of the adverse weather protection features.

These activities constituted one sample of readiness for seasonal adverse weather, as defined in Inspection Procedure 71111.01.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial Walk-Down

a. Inspection Scope

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

- January 19, 2017, main steam line minimum leakage path
- March 20, 2017, residual heat removal alternate shutdown
- March 27, 2017, reactor core isolation cooling

The inspectors reviewed the licensee's procedures and system design information to determine the correct lineup for the systems. They visually verified that critical portions of the systems were correctly aligned for the existing plant configuration.

These activities constituted three partial system walk-down samples, as defined in Inspection Procedure 71111.04.

b. Findings

Introduction. The inspectors identified a Green, non-cited violation of Technical Specification 5.4.1.a for the licensee's failure to maintain Emergency Procedure 5.1ASD, "Alternate Shutdown," Revision 17, for establishing reactor equipment cooling (REC) system flow to the high pressure coolant injection (HPCI) system fan coil unit (FCU).

Description. The inspectors reviewed Emergency Procedure 5.1ASD, "Alternate Shutdown," Revision 17, for forced evacuation of the control room for any reason other than a fire. From this review, the inspectors identified that the procedure did not contain instructions to establish REC system flow to the HPCI system FCU. Specifically, instructions were not provided to open the REC north or south critical loop valves (REC-MOV-711 or REC-MOV-714) prior to leaving the control room, if time permits, or locally and to verify REC system flow to the HPCI system FCU. The inspectors noted that the REC system function during a forced evacuation of the control room was to supply system flow to the HPCI FCU to support operation of the HPCI system. The inspector informed the licensee of this condition. The inspectors observed that the licensee modified the REC/HPCI system interlocks in June 1993; and as a result, REC system flow would not have automatically aligned to supply the HPCI FCU under all conditions that are encountered when operating in Emergency Procedure 5.1ASD. The immediate corrective actions were to assess operability of the HPCI system for leaving the control room for reasons other than a fire, to revise Emergency Procedure 5.1ASD with instructions to open REC-MOV-711 or REC-MOV-714 in the control room or locally, and to verify REC system flow to the HPCI FCU. The licensee entered this deficiency into the corrective action program as Condition Report CR-CNS-2017-01403.

The inspectors noted that the licensee reviewed the emergency procedures on an annual basis, and that the alternate shutdown panel has an indicating light for REC flow greater than or equal to 12 gallons per minute (gpm) to the HPCI FCU. The inspectors concluded that the licensee's annual emergency procedure review should have identified the procedure deficiency.

Analysis. The licensee's failure to maintain Emergency Procedure 5.1ASD to establish REC system flow to the HPCI FCU during some control room evacuation scenarios, in violation of Technical Specification 5.4.1.a, was a performance deficiency. The performance deficiency was determined to be more than minor, and therefore a finding, because it was associated with the procedural quality attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events. Specifically, the licensee did not provide instructions to establish reactor equipment cooling system flow to the high pressure coolant injection system fan coil unit, which would have complicated operator response during a control room evacuation. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, the inspectors determined that the finding had very low safety significance (Green) because it: was not a design deficiency; did not represent a loss of system and/or function; did not represent an actual loss of function; did not represent an actual loss of function of at least a single train for longer than its technical specification allowed outage time; and did not result in the loss of a high safety-significant nontechnical specification train. The finding had a cross-cutting aspect in the area of problem identification and resolution associated with identification. Specifically, the licensee failed to implement a corrective action program with a low threshold for identifying issues during the required annual review of emergency procedures [P.1].

Enforcement. Technical Specification 5.4.1.a requires, in part, that written procedures be established, implemented, and maintained covering the applicable procedures recommended in Appendix A to Regulatory Guide 1.33, "Quality Assurance Program Requirements," Revision 2, dated February 1978. Regulatory Guide 1.33, Appendix A, Section 6.p, requires procedures for fire in the control room or forced evacuation of the control room. Contrary to the above, from June 1993 to March 2017, the licensee failed to maintain procedures for forced evacuation of the control room. Specifically, the licensee failed to maintain Emergency Procedure 5.1ASD, "Alternate Shutdown," Revision 17, for establishing reactor equipment cooling (REC) system flow to the high pressure coolant injection (HPCI) system fan coil unit (FCU) during some control room evacuation scenarios. The finding resulted in the potential to complicate the licensee's response during control room evacuation scenarios for reasons other than a fire. The immediate corrective actions were to assess operability of the HPCI system for these control room evacuation scenarios, and to revise Emergency Procedure 5.1ASD with instructions to open REC-MOV-711 or REC-MOV-714 in the control room or locally, and verify REC system flow to the HPCI FCU. Because this violation was of very low safety significance (Green) and was entered into the licensee's corrective action program as Condition Report CR-CNS-2017-01403, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the Enforcement Policy.
(NCV 05000298/2017001-01, "Failure to Maintain Alternate Shutdown Emergency Procedure")

.2 Complete Walk-Down

a. Inspection Scope

On March 24, 2017, the inspectors performed a complete system walk-down inspection of the 125 and 250 Vdc battery room. The inspectors reviewed the licensee's procedures and system design information to determine the correct system lineup for the existing plant configuration. The inspectors also reviewed outstanding work orders, open condition reports, in-process design changes, temporary modifications, and other open items tracked by the licensee's operations and engineering departments. The inspectors then visually verified that the system was correctly aligned for the existing plant configuration.

These activities constituted one complete system walk-down sample, as defined in Inspection Procedure 71111.04.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

Quarterly Inspection

a. Inspection Scope

The inspectors evaluated the licensee's fire protection program for operational status and material condition. The inspectors focused their inspection on four plant areas important to safety:

- March 21, 2017, 125 and 250 Vdc battery B room, Fire Area CB-B, Zone 8F
- March 21, 2017, cable spreading room, Fire Area CB-D, Zone 9A
- March 24, 2017, cable expansion room, Fire Area CB-D, Zone 9B
- March 24, 2017, reactor building northeast quadrant, Fire Area RB-A, Zone 1A

For each area, the inspectors evaluated the fire plan against defined hazards and defense-in-depth features in the licensee's fire protection program. The inspectors evaluated control of transient combustibles and ignition sources, fire detection and suppression systems, manual firefighting equipment and capability, passive fire protection features, and compensatory measures for degraded conditions.

These activities constituted four quarterly inspection samples, as defined in Inspection Procedure 71111.05.

b. Findings

No findings were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

On January 30, 2017, the inspectors completed an inspection of the station's ability to mitigate flooding due to internal causes. After reviewing the licensee's flooding analysis, the inspectors chose one plant area containing risk-significant structures, systems, and components that were susceptible to flooding:

- Emergency diesel generator rooms

The inspectors reviewed plant design features and licensee procedures for coping with internal flooding. The inspectors walked down the selected area to inspect the design features, including the material condition of seals, drains, and flood barriers. The inspectors evaluated whether operator actions credited for flood mitigation could be successfully accomplished.

These activities constituted completion of one flood protection measures sample, as defined in Inspection Procedure 71111.06.

b. Findings

No findings were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors completed an inspection of the readiness and availability of risk-significant heat exchangers:

- February 24, 2017, reactor equipment cooling heat exchanger B
- March 8, 2017, reactor equipment cooling heat exchanger A

The inspectors observed performance tests for heat exchangers A and B, reviewed the data from a performance test for heat exchangers A and B, verified the licensee used the industry standard periodic maintenance method outlined in Electric Power Research Institute (EPRI) NP-7552 for heat exchangers A and B, observed the licensee's implementation of biofouling controls for heat exchangers A and B, and observed the licensee's inspection of heat exchangers A and B and the material condition of the heat exchanger internals. Additionally, the inspectors walked down heat exchangers A and B to observe their performance and material condition and verified that heat exchangers A and B were correctly categorized under the Maintenance Rule and were receiving the required maintenance.

These activities constituted completion of two heat sink performance annual review samples, as defined in Inspection Procedure 71111.07.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program and Licensed Operator Performance (71111.11)

.1 Review of Licensed Operator Requalification

a. Inspection Scope

On March 27, 2017, the inspectors observed simulator training for an operating crew. The inspectors assessed the performance of the operators and the evaluators' critique of their performance. The inspectors also assessed the modeling and performance of the simulator. On February 13, 2017, the inspectors observed flow loop simulator training for an operating crew. The training was performed in response to identification of an adverse trend of configuration control events at the station, including the mispositioning of residual heat removal minimum flow valves. The inspectors assessed the performance of the operators and the evaluators' critique of their performance.

These activities constituted completion of one quarterly licensed operator requalification program sample, as defined in Inspection Procedure 71111.11.

b. Findings

No findings were identified.

.2 Review of Licensed Operator Performance

a. Inspection Scope

On February 11, 2017, the inspectors observed the performance of on-shift licensed operators in the plant's main control room. At the time of the observations, the plant was in a period of heightened activity due to a downpower for rod pattern adjustment and surveillance testing.

In addition, the inspectors assessed the operators' adherence to plant procedures, including conduct of operations procedure and other operations department policies.

These activities constituted completion of one quarterly licensed operator performance sample, as defined in Inspection Procedure 71111.11.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

Routine Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed three instances of degraded performance or condition of safety-significant structures, systems, and components (SSCs):

- February 17, 2017, Refueling Outage 29 containment isolation local leak-rate test failures
- February 17, 2017, 250 Vdc distribution system
- March 30, 2017, service water temperature control valve SW-TCV-451 B

The inspectors reviewed the extent of condition of possible common cause SSC failures and evaluated the adequacy of the licensee's corrective actions. The inspectors reviewed the licensee's work practices to evaluate whether these may have played a role in the degradation of the SSCs. The inspectors assessed the licensee's characterization of the degradation in accordance with 10 CFR 50.65 (the Maintenance Rule), and verified that the licensee was appropriately tracking degraded performance and conditions in accordance with the Maintenance Rule.

These activities constituted completion of three maintenance effectiveness samples, as defined in Inspection Procedure 71111.12.

b. Findings

Introduction. The inspectors identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," associated with the licensee's failure to identify a condition adverse to quality associated with Station Procedure 2.2.24.1, "250 Vdc Electrical System (Div 1)," Revision 14, for transferring 250 Vdc, bus 1A, from 250 Vdc, charger 1A, to 250 Vdc, charger 1C, in accordance with Station Procedure 0-CNS-LI-102, "Corrective Action Process," Revision 6.

Description. On December 6, 2016, during operator rounds, the licensee identified that 250 Vdc, bus 1A, did not meet TS 3.8.4, "DC Sources – Operating," Surveillance Requirement (SR) 3.8.4.1. Specifically, SR 3.8.4.1 requires that the 250 Vdc, battery 1A, terminal voltage on a float charge be maintained greater than or equal to 260.4 Vdc. The licensee immediately declared the Division 1, 250 Vdc electrical subsystem and the Division 1, residual heat removal (RHR) low pressure coolant injection (LPCI) system inoperable. The licensee's immediate corrective action was to adjust the 250 Vdc, charger 1C, float voltage to greater than 260.4 Vdc to restore operability of the Division 1, 250 Vdc electrical system and the Division 1, RHR LPCI system. Additionally, the licensee determined that these systems where inoperable for approximately 6 hours, which corresponded to when the licensee transferred 250 Vdc, bus 1A, from the 250 Vdc, charger 1A, to 250 Vdc, charger 1C, in support of 250 Vdc, charger 1A, maintenance. The licensee entered this deficiency into the corrective action program as Condition Report (CR) CR-CNS-2016-08685 and closed this CR to trend because the condition was corrected based on actions taken (CBOAT). Specifically, the licensee did not determine why 250 Vdc, charger 1C, did not maintain 250 Vdc, bus 1A, voltage in accordance with SR 3.8.4.1 when it was placed into service on December 6, 2016.

The inspectors reviewed CR-CNS-2016-08685 and questioned the closure of this CR to trend without additional evaluation. Through NRC questions, the inspectors identified a condition adverse to quality associated with Station Procedure 2.2.24.1, "250 Vdc Electrical System (Div 1)," Revision 14, for transferring 250 Vdc, bus 1A, from 250 Vdc, charger 1A, to 250 Vdc, charger 1C. Specifically, the licensee failed to identify that Station Procedure 2.2.24.1, contained inadequate instructions to ensure the oncoming 250 Vdc, charger 1C, output voltage was matched with 250 Vdc, bus 1A's, voltage in accordance with Station Procedure 0-CNS-LI-102, "Corrective Action Process," Revision 6. This resulted in an unexpected and initially unrecognized decline in voltage on the bus to below the required minimum of 260.4 Vdc. This resulted in the 250 Vdc, battery 1A, terminal voltage on a float charge not meeting SR 3.8.4.1 when the 250 Vdc, charger 1C, was in service. The licensee entered this deficiency into the corrective action program as Condition Report CR-CNS-2017-00750.

Analysis. The licensee's failure to identify a condition adverse to quality associated with Station Procedure 2.2.24.1, "250 Vdc Electrical System (Div 1)," Revision 14, for transferring 250 Vdc, bus 1A, from 250 Vdc, charger 1A, to 250 Vdc, charger 1C, in violation of Station Procedure 0-CNS-LI-102, was a performance deficiency. The performance deficiency was determined to be more than minor, and therefore a finding, because it was associated with the procedural quality attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events. Specifically, 250 Vdc, charger 1C, when in service, did not maintain 250 Vdc, battery 1A, terminal voltage within the requirements of Surveillance Requirement 3.8.4.1, and required the licensee to declare the Division 1, 250 Vdc electrical system and the Division 1, residual heat removal low pressure coolant injection system inoperable. Using Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated June 19, 2012, the inspectors determined that the finding had very low safety significance (Green) because it: was not a design deficiency; did not represent a loss of system and/or function; did not represent an actual loss of function; did not represent an actual loss of function of at least a single train for longer than its technical specification allowed outage time; and did not result in the loss of a high safety-significant, nontechnical specification train. The finding had a cross-cutting aspect in the area of problem identification and resolution associated with evaluation. Specifically, the licensee failed to thoroughly evaluate the 250 Vdc, charger 1C, float voltage issue to ensure that the resolution addressed the cause and extent of condition commensurate with the safety significance [P.2].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that activities affecting quality shall be accomplished in accordance with documented instructions, procedures, or drawings of a type appropriate to the circumstances. Station Procedure 0-CNS-LI-102, "Corrective Action Process,” Revision 6, and Appendix B, quality-related procedure, provides instructions for identifying and reporting problems in the corrective action program. Station Procedure 0-CNS-LI-102, Step 3.1.1 states, in part, that all personnel working at Cooper Nuclear Station are responsible for identifying and reporting problems. Contrary to the above, prior to February 15, 2017, personnel working at Cooper Nuclear Station failed to identify and report problems. Specifically, the licensee failed to identify that Station Procedure 2.2.24.1, “250 VDC Electrical System (Div 1),” Revision 14, contained inadequate instructions to ensure the oncoming charger 1C output voltage was matched with the bus 1A voltage when swapping battery chargers, so that technical specification bus voltage requirements would remain met. This resulted in the licensee not meeting Technical Specification 3.8.4, “DC Sources – Operating,” Surveillance Requirement 3.8.4.1, “verify battery terminal voltage on float charge is greater than or equal to 260.4 Vdc with 250 Vdc, charger 1C, in service.” This condition required the licensee to declare Division 1 250 Vdc electrical system and residual heat removal (RHR) low pressure coolant injection (LPCI) system inoperable. The immediate corrective action was to adjust the charger 1C float voltage greater than 260.4 Vdc to restore operability of the Division 1 250 Vdc electrical system and the Division 1 RHR LPCI system. Because this violation was of very low safety significance (Green) and was entered into the licensee’s corrective action program as Condition Report CR-CNS-2017-00750, this violation is being treated as a non-cited violation (NCV), in accordance with Section 2.3.2.a of the NRC Enforcement Policy.
(NCV 05000298/2017001-02, “Failure to Identify a Condition Adverse to Quality Associated with the 250 Vdc Electrical System”)

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

a. Inspection Scope

The inspectors reviewed three risk assessments performed by the licensee prior to changes in plant configuration and the risk management actions taken by the licensee in response to elevated risk:

- January 27, 2017, residual heat removal maintenance window, Division 2
- February 8, 2017, emergency diesel generator 1 maintenance window
- February 17, 2017, 250 Vdc, bus A, below technical specification minimum voltage

The inspectors verified that these risk assessments were performed timely and in accordance with the requirements of 10 CFR 50.65 (the Maintenance Rule) and plant procedures. The inspectors reviewed the accuracy and completeness of the licensee's risk assessments and verified that the licensee implemented appropriate risk management actions based on the results of the assessments.

The inspectors also observed portions of four emergent work activities that had the potential to cause an initiating event, to affect the functional capability of mitigating systems, or to impact barrier integrity:

- January 30, 2017, reactor core isolation cooling trip and throttle valve RCIC-MOV-14 condition validation and repair
- March 20, 2017, reactor core isolation cooling maintenance window
- February 6, 2017, emergency diesel generator 2 jacket water heater and bypass pump repairs
- March 10, 2017, emergency diesel generator 1 voltage regulator troubleshooting and repairs

The inspectors verified that the licensee appropriately developed and followed a work plan for these activities. The inspectors verified that the licensee took precautions to minimize the impact of the work activities on unaffected structures, systems, and components (SSCs).

These activities constituted completion of seven maintenance risk assessments and emergent work control inspection samples, as defined in Inspection Procedure 71111.13.

b. Findings

No findings were identified.

1R15 Operability Determinations and Functionality Assessments (71111.15)

a. Inspection Scope

The inspectors reviewed seven operability determinations and functionality assessments that the licensee performed for degraded or nonconforming structures, systems, or components (SSCs):

- January 5, 2017, operability determination of fouling of reactor equipment cooling heat exchanger A
- January 12, 2017, operability determination of the 250 Vdc battery A specific gravity
- February 10, 2017, operability determination of reactor core isolation cooling trip and throttle valve degraded/nonconforming condition
- February 17, 2017, functionality assessment of the Z1 sump pump, and the impact on the standby gas treatment system, Division 1
- February 28, 2017, operability determination of the residual heat removal service water booster pump A inboard bearing sight glass oil leakage
- March 10, 2017, operability determination of the Division 1, residual heat removal system minimum flow line manual isolation valves found closed
- March 23, 2017, operability determination of the control rod drive temporary loss of position monitoring system data

The inspectors reviewed the timeliness and technical adequacy of the licensee's evaluations. Where the licensee determined the degraded SSC to be operable or functional, the inspectors verified that the licensee's compensatory measures were appropriate to provide reasonable assurance of operability or functionality. The inspectors verified that the licensee had considered the effect of other degraded conditions on the operability or functionality of the degraded SSC.

These activities constituted completion of seven operability and functionality assessment review samples, as defined in Inspection Procedure 71111.15.

b. Findings

Introduction. The inspectors identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," associated with the licensee's failure to identify a condition adverse to quality for Division 1, residual heat removal (RHR) service water booster pump A (SWBP-A), in accordance with Station Procedure 0-CNS-LI-102, "Corrective Action Process," Revision 6.

Description. On January 5, 2017, during a plant status walk-down, the inspectors identified a lower than expected oil level in the Division 1, residual heat removal SWBP-A inboard sight glass, oil on the pump skid, and an oil droplet formed on the sight glass. The inspectors informed the licensee of this condition. The licensee assessed operability for the pump and determined the oil leakage was three drops per hour. The licensee determined that this leakage rate would have prevented SWBP-A from operating its required 30 days during a design basis accident, and it was declared inoperable. The immediate corrective action was to repair SWBP-A's inboard bearing sight glass to restore operability of the pump. The inspectors determined that the oil leakage started sometime after SWBP-A maintenance was completed on January 3, 2017. The licensee entered this deficiency into the corrective action program as Condition Report CR-CNS-2017-00054.

The inspectors noted that Station Procedure 2.1.11.1, "Turbine Building Data," Revision 159, required operations personnel to perform daily rounds to measure SWBP-A oil levels, and conduct once per shift walk-downs of the area. The inspectors concluded that the licensee's required daily oil measurements, and once per shift walk-downs provided the opportunity to identify the SWBP-A oil leakage prior to January 5, 2017.

Analysis. The licensee's failure to identify a condition adverse to quality for Division 1 residual heat removal SWBP-A, in violation of Station Procedure 0-CNS-LI-102, was a performance deficiency. The performance deficiency was determined to be more than minor, and therefore a finding, because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events. Specifically, the oil leakage from SWBP-A's inboard bearing sight glass would have prevented it from operating for its required 30-day mission time during a design basis accident and resulted in the pump being declared inoperable. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, the inspectors determined that the finding had very low safety significance (Green) because it: was not a design deficiency; did not represent a loss of system and/or function; did not represent an actual loss of function; did not represent an actual loss of function of at least a single train for longer than its technical specification allowed outage time; and did not result in the loss of a high safety-significant nontechnical specification train. The finding had a cross-cutting aspect in the area of human performance associated with challenge the unknown because the licensee failed to stop when faced with uncertain conditions and failed to ensure that risks are evaluated and managed before proceeding. Specifically, the licensee did not maintain a questioning attitude during job-site reviews to identify and resolve unexpected conditions of lower than expected oil level in the SWBP-A's inboard bearing sight glass, oil on the pump skid, and an oil droplet formed on the bottom of the sight glass [H.11].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be accomplished, in accordance with documented instructions, procedures, or drawings of a type appropriate to the circumstances. Station Procedure 0-CNS-LI-102, "Corrective Action Process," Revision 6, an Appendix B, quality-related procedure, provides instructions for identifying and reporting problems in the corrective action program. Procedure 0-CNS-LI-102, Step 3.1.1 states, in part, that all personnel working at Cooper Nuclear Station are responsible for identifying and reporting problems. Contrary to the above, from January 3, 2017, to January 5, 2017, personnel working at Cooper Nuclear Station failed to identify and report problems. Specifically, the licensee failed to identify a condition adverse to quality associated with oil leakage from the Division 1 residual heat removal SWBP-A inboard bearing sight glass, which caused the subsystem to be inoperable. The immediate corrective action was to repair the inboard bearing sight glass to restore operability of the pump. Because this violation was of very low safety significance (Green) and was entered into the licensee's corrective action program as Condition Report CR-CNS-2016-00054, this violation is being treated as a non-cited violation (NCV), in accordance with Section 2.3.2.a of the NRC Enforcement Policy. (NCV 05000298/2017001-03, "Failure to Identify a Condition Adverse to Quality")

1R18 Plant Modifications (71111.18)

a. Inspection Scope

On January 11, 2017, the inspectors reviewed a permanent plant modification to the licensee's inservice inspection program that utilized code reconciliation to allow use of internal pipe stress acceptance criteria from later versions of the American Society of Mechanical Engineers (ASME) code, which affected risk-significant structures, systems, and components (SSCs).

The inspectors reviewed the design and implementation of the modification. The inspectors verified that work activities involved in implementing the modification did not adversely impact operator actions that may be required in response to an emergency or other unplanned event. The inspectors verified that post-modification testing was adequate to establish the operability of the SSC impacted by the modification.

These activities constituted completion of one sample of permanent modifications, as defined in Inspection Procedure 71111.18.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed seven post-maintenance testing activities that affected risk-significant structures, systems, or components (SSCs):

- January 6, 2017, 250 Vdc charger C maintenance
- January 23, 2017, 125 Vdc charger A maintenance
- February 17, 2017, emergency diesel generator 1 maintenance window
- February 17, 2017, residual heat removal maintenance window, Division II
- March 20, 2017, reactor core isolation cooling maintenance window
- March 27, 2017, emergency diesel generator 2 maintenance window
- March 27, 2017, emergency diesel generator 1 voltage regulator repair

The inspectors reviewed licensing- and design-basis documents for the SSCs and the maintenance and post-maintenance test procedures. The inspectors observed the performance of the post-maintenance tests to verify that the licensee performed the tests in accordance with approved procedures, satisfied the established acceptance criteria, and restored the operability of the affected SSCs.

These activities constituted completion of seven post-maintenance testing inspection samples, as defined in Inspection Procedure 71111.19.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed five risk-significant surveillance tests and reviewed test results to verify that these tests adequately demonstrated that the structures, systems, and components (SSCs) were capable of performing their safety functions:

In-service tests:

- March 6, 2017, service water booster pump D

Other surveillance tests:

- February 7, 2017, residual heat removal 2-year comprehensive pump test, Division 1
- February 17, 2017, 125 Vdc and 250 Vdc, charger A, performance testing
- February 17, 2017, essential control building ventilation temperature switch change out and functional test
- March 31, 2017, reactor core isolation cooling turbine overspeed functional test

The inspectors verified that these tests met technical specification requirements, that the licensee performed the tests in accordance with their procedures, and that the results of the test satisfied appropriate acceptance criteria. The inspectors verified that the licensee restored the operability of the affected SSCs following testing.

These activities constituted completion of five surveillance testing inspection samples, as defined in Inspection Procedure 71111.22.

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

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

a. Inspection Scope

The inspector performed an in-office review of the following documents submitted December 5, 2016:

- Emergency Plan Implementing Procedure 5.7.1, “Emergency Classification,” Revision 56

- Emergency Plan Implementing Procedure 5.7.16, “Release Rate Calculation,” Revision 26

These revisions:

- Provided a method to calculate the radiological activity release rate for a release through the hardened containment vent system

- Revised the criteria for the loss of the fuel clad fission product barrier and the potential loss of primary containment from “primary containment flooding is required” to “Severe Accident Guide 1 entry is required,” based on Emergency Preparedness Frequently Asked Question 2015-004, dated July 1, 2015

These revisions were compared to their previous revisions, to the criteria of NUREG-0654, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” Revision 1, to Nuclear Energy Institute Report 99-01, “Emergency Action Level Methodology,” Revision 5, and to the standards in 10 CFR 50.47(b) to determine if the revisions implemented the requirements of 10 CFR 50.54(q)(3) and 50.54(q)(4). The inspector verified that the revisions did not decrease the effectiveness of the emergency plan. This review was not documented in a safety evaluation report and did not constitute approval of licensee-generated changes; therefore, these revisions are subject to future inspection.

These activities constituted completion of two emergency action level and emergency plan change samples, as defined in Inspection Procedure 71114.04.

b. Findings

No findings were identified.

1EP6 Drill Evaluation (71114.06)

Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors observed an emergency preparedness drill on March 28, 2017, to verify the adequacy and capability of the licensee’s assessment of drill performance. The inspectors reviewed the drill scenario, observed the drill from the control room and the emergency operations facility (EOF), and attended the post-drill critique. The inspectors verified that the licensee's emergency classifications, off-site notifications, and protective action recommendations were appropriate and timely. The inspectors verified that any emergency preparedness weaknesses were appropriately identified by the licensee in the post-drill critique and entered into the corrective action program for resolution.

These activities constituted completion of one emergency preparedness drill observation sample, as defined in Inspection Procedure 71114.06.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

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

4OA1 Performance Indicator Verification (71151)

.1 Unplanned Scrams per 7000 Critical Hours (IE01)

a. Inspection Scope

The inspectors reviewed licensee event reports (LERs) for the period of January 1 through December 31, 2016, to determine the number of scrams that occurred. The inspectors compared the number of scrams reported in these LERs to the number reported for the performance indicator. Additionally, the inspectors sampled monthly operating logs to verify the number of critical hours during the period. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, to determine the accuracy of the data reported.

These activities constituted verification of the unplanned scrams per 7000 critical hours performance indicator, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

.2 Unplanned Power Changes per 7000 Critical Hours (IE03)

a. Inspection Scope

The inspectors reviewed operating logs, corrective action program records, and monthly operating reports for the period of January 1 through December 31, 2016, to determine the number of unplanned power changes that occurred. The inspectors compared the number of unplanned power changes documented to the number reported for the performance indicator. Additionally, the inspectors sampled monthly operating logs to verify the number of critical hours during the period. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory

Assessment Performance Indicator Guideline," Revision 7, to determine the accuracy of the data reported.

These activities constituted verification of the unplanned power outages per 7000 critical hours performance indicator, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Routine Review

a. Inspection Scope

Throughout the inspection period, the inspectors performed daily reviews of items entered into the licensee's corrective action program and periodically attended the licensee's condition report screening meetings. The inspectors verified that licensee personnel were identifying problems at an appropriate threshold and entering these problems into the corrective action program for resolution. The inspectors verified that the licensee developed and implemented corrective actions commensurate with the significance of the problems identified. The inspectors also reviewed the licensee's problem identification and resolution activities during the performance of the other inspection activities documented in this report.

b. Findings

No findings were identified.

.2 Annual Follow-up of Selected Issues

a. Inspection Scope

The inspectors selected two issues for an in-depth follow-up:

- March 20, 2017, use of incorrect code allowable stress values for service water piping

  The inspectors assessed the licensee's problem identification threshold, cause analyses, extent of condition reviews, and compensatory actions. The inspectors verified that the licensee appropriately prioritized the planned corrective actions and that these actions were adequate to correct the condition.

- March 24, 2017, reactor core isolation cooling trip and throttle valve, RCIC-MOV-14, pin not installed correctly and pump oil leaks

  The inspectors assessed the licensee's problem identification threshold, cause analyses, extent of condition reviews, and compensatory actions. The inspectors verified that the licensee appropriately prioritized the planned corrective actions and that these actions were adequate to correct the condition.

These activities constituted completion of two annual follow-up samples, as defined in Inspection Procedure 71152.

b. <u>Findings</u>

<u>Introduction</u>. The inspectors identified a Green, non-cited violation of 10 CFR 50.55a(g)(4) for the failure to use an approved method to disposition an ASME code nonconforming condition in the residual heat removal service water system. Specifically, the licensee identified multiple locations with localized pipe thinning below the ASME Code B31.1, design minimum pipe-wall thickness during an ultrasonic examination, but failed to use an approved method to calculate a new acceptable pipe-wall thickness.

<u>Description</u>. Engineering Procedure 3.10.1, "UT Thickness Measurements and Gridding Procedure," Revision 1, provided the examination guidance used to perform pipe-wall thickness measurements. Section 2.4 noted that the procedure does not provide minimum pipe-wall thickness acceptance criteria. Instead, the procedure instructed the testing personnel to provide the program engineer with sufficient information to evaluate the condition of the pipe.

On September 13, 2016, the licensee identified that multiple pipe locations of the residual heat removal service water system exhibited pipe-wall thickness measurements below the minimum design value of 0.218 inches. The licensee proceeded to calculate a new minimum pipe-wall thickness value taking into account the most limiting load that the affected pipe section could potentially experience. This method consisted of entering thickness values into a spreadsheet calculation to verify those thicknesses satisfy the applicable stresses. In this manner, the licensee entered values until the lowest allowable pipe thickness was identified. The inspectors concluded that this approach was not in compliance with any methodology prescribed by the ASME code.

The service water system piping is classified as ASME code Class 3 and was designed to the 1967 Edition of ASME B31.1, "Power Piping." The code calculates design minimum pipe-wall thickness values taking into consideration the expected service loads and the maximum allowable stress for the piping material. Specifically, Section 104, "Pressure Design of Components," provides the equation used to calculate a minimum pipe-wall thickness for a straight pipe under internal pressure (Section 104.1.2.a.1). An alternative formula to calculate the minimum pipe-wall thickness is provided in Section 104.1.2.a.2. The code provides these two as the only code acceptable methods to calculate a design minimum pipe-wall thickness value.

Using the spreadsheet methodology described above, the licensee calculated an allowable minimum pipe-wall thickness value of 0.068 inches. When the inspectors questioned the mathematical formulas used to calculate this value, the licensee identified an error in the equation being used that changed the allowable minimum thickness value to 0.091 inches. Ultimately, the inspectors concluded that the design value of 0.218 inches was the required code minimum pipe-wall thickness as calculated using the equation from ASME B31.1, Section 104. The revised allowable minimum pipe-wall thickness calculated by the licensee represented a nonconservative value. Although the licensee demonstrated that the newly calculated pipe-wall thickness remained structurally sound for the affected pipe location, the licensee failed to use a method prescribed in the ASME code, either the construction code or Section XI or an

NRC approved alternative method, such as described in NRC Inspection Manual Chapter 0326, "Operability Determinations & Functionality Assessments for Conditions Adverse to Quality or Safety."

Analysis. The licensee's failure to use an approved method to calculate a new minimum allowable pipe-wall thickness was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it was associated with the design control attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, calculating an allowable minimum pipe-wall thickness value that is below the ASME code design minimum value reduces the structural integrity of the piping, potentially leading to the failure of the pipe. The performance deficiency was also similar to Examples 3.j and 3.k of NRC Inspection Manual Chapter 0612, Appendix E, in that, implementing an unapproved methodology for calculating pipe-wall-thicknesses represents a significant programmatic deficiency that could lead to worse errors if uncorrected. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, the inspectors determined the finding screened as having very low safety significance (Green) because it: was not a design deficiency; did not represent a loss of system and/or function; did not represent an actual loss of function of at least a single train for longer than its technical specification allowed outage time; and did not result in the loss of a high safety-significant nontechnical specification train. This finding had a cross-cutting aspect in the area of human performance associated with design margins because the licensee failed to operate and maintain the residual heat removal service water system within the ASME code minimum pipe-wall thickness. Specifically, having identified that the affected pipe location was below the allowable pipe-wall thickness, the licensee opted to calculate and accept a new minimum pipe-wall thickness value that was not consistent with Code requirements instead of repairing the affected piping at the time of discovery [H.6].

Enforcement. Title 10 of the Code of Federal Regulations, Part 50, Section 50.55a(g)(4), "Inservice inspection standards requirements for operating plants," requires that components that are classified as ASME code, Class 3, must meet the requirements set forth in Section XI of editions and addenda of the ASME Boiler and Pressure Vessel Code. When an evaluation standard for a particular component, examination category, or examination method are not specified in ASME, Section XI, Subsection IWA-3100(b), the code states that for flaws exceeding the acceptance standards, those flaws need to be evaluated in accordance with the construction code. Contrary to the above, from September 13, 2016, until November 1, 2016, for multiple piping locations that contained flaws, an evaluation standard was not specified in ASME, Section XI, Subsection IWA-3100(b), and the flaws exceeded the acceptance standards, but those flaws were not evaluated in accordance with the construction code. Specifically, the licensee identified multiple locations that contained flaws that were below the construction code minimum design pipe-wall thickness acceptance standards and evaluated those flaws by calculating a new value using a methodology that was not in accordance with the ASME B31.1 construction code. The licensee failed to demonstrate that the measured pipe-wall thickness met the requirements of ASME Code B31.1, Section 104, "Pressure Design of Components," for the required design minimum pipe-wall thickness, and as a result, this condition reduced the structural integrity of the piping, potentially leading to the failure of the pipe. The licensee subsequently replaced this section of piping during Refueling Outage 29, on November 1, 2016. Because this violation was of very low safety significance (Green) and was entered into the licensee's corrective action program as Condition Reports CR-CNS-2016-05558 and CR-CNS-2016-05963, this violation is being treated as a non-cited violation (NCV), in accordance with Section 2.3.2.a of the NRC Enforcement Policy. (NCV 05000298/2017001-04, "Failure to Address Nonconforming Pipe Thinning in Accordance with the ASME Code")

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

.1 (Closed) Licensee Event Report (LER) 05000298/2016007-00, "Isolation of Shutdown Cooling due to Relay Maintenance Results in a Loss of Safety Function"

a. Inspection Scope

On October 28, 2016, during replacement of relay PCIS-REL-K27, the station experienced a loss of shutdown cooling (SDC) while in a refueling outage. Maintenance personnel were installing a new relay onto a shared plastic mounting rail when SDC inadvertently isolated. While snapping the relay into place, the workers disturbed the mounting rail in a manner that caused contacts of the adjacent relay, PCIS-REL-K30, to open. This caused SDC isolation valve RHR-MO-17 to close, which actuated the logic to trip the running 'A' residual heat removal (RHR) pump.

The licensee's root cause evaluation determined that the station did not identify the risk from mechanical agitation during primary containment isolation system (PCIS) relay installation; therefore, the risk was not adequately evaluated or mitigated. The licensee took corrective actions to prevent recurrence, which included revising their shutdown safety risk procedure to identify and list the relays or other devices that could impact SDC when in service.

This issue resulted in a loss of safety function for the RHR system. The licensee reported this failure under 10 CFR 50.72(b)(3)(v) and 10 CFR 50.73(a)(2)(v)(D) as a condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to remove residual heat. The inspectors reviewed the event, including station logs and technical specification (TS) requirements; walked down the affected components; and discussed the events with the licensee. The inspectors also reviewed the root cause evaluation, extent of condition and cause reviews, and the corrective actions associated with the event to ensure they were appropriate.

This licensee event report is closed.

b. Findings

Introduction. The inspectors reviewed a self-revealed, Green, non-cited violation (NCV) of Technical Specification (TS) 5.4.1.a, for the licensee's failure to implement Maintenance Procedure 7.3.16, "Low Voltage Relay Removal and Installation," Revision 22, for relay replacement work. Specifically, the licensee failed to evaluate the potential impact of primary containment isolation system (PCIS) relay PCIS-REL-K27 work on RHR shutdown cooling relay PCIS-REL-K30, which was mounted next to K27 and shared a common mounting rail. As a result, the licensee did not identify the potential of losing RHR SDC, and while installing the K27 relay and snapping it into the mounting rail, workers caused a momentary actuation of relay K30 and a loss of RHR SDC.

Description. On October 28, 2016, the licensee was nearing the end of their planned refueling outage and was in the process of completing a SDC out-of-service work window. Operations personnel were preparing to place the RHR Loop A in service for SDC. Throughout the SDC out-of-service window, the alternate decay heat removal (ADHR) system had been maintaining reactor pressure vessel (RPV) and spent fuel pool temperature. During the outage, the licensee had completed work activities for the replacement of 27 PCIS relay coils, including the relay coil for PCIS-REL-K27. During testing after completion of the work order, maintenance personnel identified that the PCIS-REL-K27 relay did not actuate within expected tolerances. As a result, the licensee revised the work order to direct replacement of the entire relay. This work required more wires to be lifted and the relay to be replaced by removing it from the DIN rail, a plastic snap rail channel that served as a snap-in mounting location for several side by side relays in very close proximity.

Although the PCIS relay work window had been extended, operations personnel placed SDC in service at 8:49 am on October 28, 2016. Subsequently, during replacement of the PCIS-REL-K27 relay at 9:24 am, the action of installing the new relay onto the shared plastic DIN rail disturbed the mounting rail in a manner that caused contact on an adjacent relay, PCIS-REL-K30, to open. This actuation caused an SDC isolation valve to close and tripped the running RHR A pump. Operations personnel declared the RHR A SDC subsystem inoperable and entered technical specification (TS) Limiting Condition for Operation (LCO) 3.9.7, "RHR High Water Level," Conditions A and C. ADHR remained in service throughout the event and the plant remained aligned for natural circulation. Because the alternate cooling methods remained available and in service, there was no notable increase in RPV temperature, and as a result, there was minimal impact to plant operations.

The licensee initiated a root cause evaluation to review the cause of the event. The licensee determined that the root cause of the event was that the station did not identify the risk from mechanical agitation during PCIS relay installation; and therefore, the risk was not evaluated or mitigated. Specifically, the licensee determined that the risk of jarring the common DIN rail mounting was not recognized and the adjacent SDC relay was not protected; as a result, relay work was allowed to continue after SDC was returned to service. In addition, the relay work was not adequately coordinated by operations and outage control center personnel. Specifically, neither the shift manager nor the operations outage control center manager was aware that PCIS relay work was continuing when they gave permission to return SDC to service. As a result of these breakdowns, the licensee took corrective actions to change procedures to prevent work near SDC relays and to protect the associated relays when SDC was required to be in service during outages.

The inspectors reviewed the licensee's root cause evaluation, observed licensee response at the time of the event, reviewed the completed work orders, and evaluated the adequacy of licensee procedures. The inspectors noted that the licensee did not identify any similar events where agitation of a DIN rail had caused unexpected actuations of adjacent relays. The inspectors reviewed Station Procedure 7.3.16, "Low Voltage Relay Removal and Installation," Revision 22. This procedure drives an intrusive analysis of the electrical circuits associated with any relay work that is being planned. The inspectors noted that Step 2.3 of this procedure stated, “the EP&C Relay Component Engineer and Shift Manager shall ensure interaction with other systems or equipment is fully researched prior to installing jumpers, pulling fuses, lifting leads, or removing relay.” In addition, this procedure drove engineering personnel to perform a walk-down of the relay and to fill out a data sheet documenting impacts and concerns associated with work on the relay in question. This procedure also allows the engineer to use previously existing data sheets, rather than filling out new ones.

During their review, the inspectors noted that the licensee’s evaluation did not recognize the importance of Step 3.2.1.1(f) of Station Procedure 7.3.16. Specifically, Step 3.2.1.1(f) stated, “Evaluate potential impact on adjacent components or components that share a common mounting when subject relay is removed/installed.” The inspectors noted that this was a protected step, associated with a commitment for corrective actions from a previously occurring station event. The inspectors followed the commitment reference and discovered that the corrective action had come as a result of an event that was almost identical to the October 28, 2016, event. Specifically, Licensee Event Report (LER) 1999-006, “Inadvertent Half-Group VII Isolation Due to Deenergization of a Relay,” documented a partial Group VII isolation event that was the result of a relay actuation that occurred when workers snapped an adjacent relay into the common DIN rail. The inspectors determined that this previous occurrence and the apparent inadequate Procedure 7.3.16 step should have been evaluated during the performance of the root cause analysis for causal implications. In addition, the inspectors were concerned with the narrow scope of the licensee’s extent of condition review. The inspectors observed that the evaluation should have reviewed potential bumping hazards for other unprotected shutdown cooling (SDC) components, potential bumping hazards for relays mounted on metal mounting rails, potential hazards to other primary containment isolation system (PCIS) relays that weren’t associated with SDC, and other Maintenance Procedure 7.3.16 data sheets to determine if they inadequately addressed Step 3.2.1.1(f) (since existing data sheets could be reused). Finally, the inspectors observed that the licensee’s safety culture evaluation failed to review safety culture implications for the contributing cause identified in the root cause analysis, as required by procedure. In response to the inspectors concerns, the licensee generated condition reports and reopened the root cause evaluation to address its weaknesses.

The inspectors concluded that the licensee failed to implement a process of planning, controlling, and executing work activities such that nuclear safety was the overriding priority, including the need for coordination with different work groups or job activities. Specifically, the licensee did not adequately coordinate work on PCIS-REL-K27, in that, key players in the organization did not maintain awareness of the ongoing work activity when granting permission to return SDC to service. In addition, the licensee did not evaluate the potential impact of the K27 relay work on residual heat removal (RHR) shutdown cooling relay PCIS-REL-K30, which was mounted next to K27 and shared a common mounting rail. As a result, the licensee did not identify the risk of impacts to RHR associated with the work, and while installing the K27 relay and snapping it into the mounting rail, workers caused a momentary actuation of relay K30 and a loss of RHR SDC. The inspectors also observed that the licensee had experienced two inadvertent full and partial Group II (RHR isolation) actuations during the relay work in the weeks and even hours before this event occurred. The inspectors determined that these events should have increased the licensee’s risk recognition associated with the work and its potential impacts on SDC, and that these events served as missed opportunities for the station to avoid the October 28, 2016, loss of SDC event entirely.

Analysis. The licensee's failure to implement Maintenance Procedure 7.3.16, in violation of Technical Specification 5.4.1.a, was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it was associated with the equipment performance attribute of the Initiating Events Cornerstone and affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown operations. Using Inspection Manual Chapter 0609, Appendix G, Attachment 1, "Shutdown Operations Significance Process Phase 1 Initial Screening and Characterization of Findings," dated May 9, 2014, the inspectors determined that the finding did not require a quantitative risk assessment because the event occurred when the refuel canal/cavity was flooded. Therefore, the finding had very low safety significance (Green). The finding had a cross-cutting aspect in the area of human performance associated with work management because the licensee failed to implement a process of planning, controlling, and executing work activities such that nuclear safety was the overriding priority, including the need for coordination with different work groups or job activities. Specifically, the licensee failed to control, execute, and coordinate safety-related primary containment isolation system (PCIS) relay work activities to ensure residual heat removal (RHR) shutdown cooling (SDC) was not adversely impacted [H.5].

Enforcement. Technical Specification 5.4.1.a, requires, in part, that procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2. Section 9.a of Appendix A to Regulatory Guide 1.33, Revision 2, requires, "Procedures for Performing Maintenance." The licensee established Maintenance Procedure 7.3.16, "Low Voltage Relay Removal and Installation," Revision 22, to meet the Regulatory Guide 1.33 requirement. Prior to performing relay work, Step 3.2.1.1(f) of Procedure 7.3.16 required engineering personnel to, "evaluate the potential impact on adjacent components or components that share a common mounting when the subject relay is removed/installed." Contrary to the above, on October 28, 2016, the licensee failed to implement Procedure 7.3.16 when, prior to work on relay PCIS-REL-K27, engineering personnel did not evaluate the potential impact on adjacent components or components that shared a common mounting when the subject relay was removed or installed. Specifically, the licensee did not evaluate the potential impact of the K27 relay work on RHR shutdown cooling relay PCIS-REL-K30, which was mounted next to K27 and shared a common mounting rail. As a result, the licensee did not identify the risk of impacts to the RHR system associated with the work, and while installing the K27 relay and snapping it into the mounting rail, workers caused a momentary actuation of relay K30 and a loss of RHR SDC. Corrective actions to restore compliance included restoration of RHR SDC, completion of the K27 relay maintenance with RHR SDC out of service, and an outage risk management procedure change that prohibited work on or near RHR SDC relays while the system was required to be in service. Because this violation was of very low safety significance (Green) and was entered into the licensee's corrective action program as Condition Report CR-CNS-2016-07645, this violation is being treated as a non-cited violation (NCV) in accordance with Section 2.3.2.a of the NRC Enforcement Policy. (NCV 05000298/2017001-05, "Loss of Shutdown Cooling due to Relay Maintenance")

.2 (Closed) Licensee Event Report (LER) 05000298/2016008-00, "Purchase and Installation of Incorrect Actuator Results in a Condition Prohibited by Technical Specifications"

a. Inspection Scope

On November 8, 2016, the plant was in Mode 1, at normal operating pressure and temperature, following Refueling Outage (RE) 29, when the licensee commenced reactor core isolation cooling (RCIC) surveillance testing. Shortly after starting the RCIC system, the station operations personnel in the field reported a water leak to the control room. The licensee immediately shut down the RCIC system from the control room, and declared the system inoperable. The licensee identified that the water leak was from a failed RCIC turbine lube oil cooler gasket and a pressure relief valve lifting. During the licensee's initial review, the station identified that RCIC pressure control valve, RCIC-AOV-PCV23, was full open causing excessive cooling water pressure to the turbine oil cooler and causing the gasket to fail and the relief valve to open. The licensee's examination of RCIC-AOV-PCV23 revealed that the actuator was supplied with a closed mechanical stop instead of the required open mechanical stop when this valve was replaced in RE 29. The licensee initiated a work order to fabricate and install an open mechanical stop for RCIC-AOV-PCV23, and the RCIC system was declared operable after satisfactorily completing post-maintenance testing and turbine lube oil cooler gasket replacement.

The licensee initiated a root cause evaluation (RCE) to determine the cause of the event. The licensee reported this failure under 10 CFR 50.73(a)(2)(i)(B) as a condition prohibited by technical specifications, because the RCIC system was inoperable when the plant entered the mode of applicability (Mode 2 with reactor steam dome pressure > 150 psig). The inspectors reviewed the event, including station logs and technical specification requirements; walked down the affected components; and discussed the events with the licensee. The inspectors also reviewed the RCE, extent of condition and cause reviews, and the corrective actions associated with the event to ensure they were appropriate.

This licensee event report is closed.

b. Findings

Introduction. The inspectors reviewed a self-revealed, Green, non-cited violation of Technical Specification 3.0.4 for the licensee's failure to install the correct reactor core isolation cooling (RCIC) pressure control valve, RCIC-AOV-PCV23, mechanical stop and verify proper operation of the system prior to entering the mode of applicability.

Description. On November 8, 2016, the plant was in Mode 1, at normal operating pressure and temperature, following Refueling Outage (RE) 29, when the RCIC system failed on demand surveillance testing. During the licensee's initial review of the failed surveillance test, the station identified that RCIC-AOV-PCV23 was full open, causing excessive cooling water pressure to the turbine oil cooler, causing the gasket to fail and the relief valve to open. The licensee's examination of RCIC-AOV-PCV23 revealed that the actuator was supplied with a closed mechanical stop instead of the required open mechanical stop, when this valve was replaced in RE 29. The licensee initiated a work order to fabricate and install an open mechanical stop for RCIC-AOV-PCV23, and the

RCIC system was declared operable after satisfactorily completing post-maintenance testing and turbine lube oil cooler gasket replacement. The licensee entered this deficiency into the corrective action program as Condition Report CR-CNS-2016-08122 and initiated a root cause evaluation (RCE) to investigate this condition.

The RCE concluded that the root cause of this event was that the incorrect air operated valve was purchased because the material master purchase order text and design drawings did not have the required details for RCIC-AOV-PCV23, which required an open mechanical stop. An associated contributing cause was that Station Procedure 7.2.51.1, "Air Operator Valve Actuator Setup/Testing," Revision 21, and its attachment, did not provide sufficient direction to set the mechanical stop to ensure the valve would be able to perform its design function. These conditions resulted in the licensee installing RCIC-AOV-PCV23 with the incorrect mechanical stop and failure to verify proper operation of the valve. This resulted in the RCIC system being inoperable when the plant entered the mode of applicability (Mode 2 with reactor steam dome pressure > 150 psig) in violation of Technical Specification 3.0.4. The inspectors determined this issue was self-revealed because the RCIC system failed when required during on demand surveillance testing.

The licensee initiated corrective actions to revise the material master order text, to revise the associated design drawing to reflect an open mechanical stop for RCIC-AOV-PCV23, and to revise Station Procedure 7.2.51.1 to add information on setting open mechanical stops.

Analysis. The licensee's failure to install the correct RCIC pressure control valve, RCIC-AOV-PCV23, mechanical stop and verify proper operation of the system prior to entering the mode of applicability, in violation of Technical Specification 3.0.4, was a performance deficiency. The performance deficiency was determined to be more than minor, and therefore a finding, because it was associated with the design control attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events. Specifically, the licensee installed RCIC-AOV-PCV23 with the incorrect mechanical stop and proper valve operation was not verified after installation during RE 29, which caused the the RCIC system to lose function during surveillance testing. This transient caused a failure of the RCIC turbine lube oil cooler gasket and a RCIC system water leak. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, the inspectors determined that the finding required a detailed risk evaluation because it represented a loss of system and/or function. In the detailed risk evaluation, the analyst assumed the reactor core isolation cooling (RCIC) system was unavailable for 50 hours. The analyst used the Test/Limited Use Version COOPER-DEESE-HCI03 of the Cooper SPAR model run on SAPHIRE, Version 8.1.5. The analyst updated the initiating event frequencies for transients, losses of condenser heat sink, losses of main feed water, grid related losses of offsite power, and switchyard centered losses of offsite power to the more recent values from the 2014 update to the industry data found in INL/EXT-14-31428, "Initiating Event Rates at U.S. Nuclear Power Plants, 1998-2013," Revision 1. From this, the finding was determined to have an increase in core damage frequency of 8.4E-8/year and to be of very low safety significance (Green). Transients, losses of condenser heat sink, and losses of main feed water were the dominant core damage sequences. The automatic depressurization system and the reactor protection system remained to mitigate these sequences. The finding had a cross-cutting aspect in the area of human performance associated with documentation because the licensee failed to create and maintain complete, accurate, and up-to-date documentation associated with RCIC-MOV-PCV23 design drawings and maintenance procedure for setting and testing the mechanical stop [H.7].

Enforcement. Technical Specification (TS) 3.0.4 requires, in part, that when a limiting condition for operation (LCO) is not met, entry into a mode or other specified condition in the applicability shall only be made when the associated actions to be entered permit continued operation in the mode or other specified condition in the applicability for an unlimited period of time, after performance of a risk assessment addressing inoperable systems and components, or when an allowance is stated. Contrary to the above, on November 8, 2016, with LCO 3.5.3, not met, the station entered into a mode and specified condition in the applicability when the associated actions to be entered did not permit continued operation in the mode and specified condition in the applicability for an unlimited period of time; without performance of a risk assessment addressing inoperable systems and components; and without a stated allowance. Specifically, the licensee transitioned Cooper Nuclear Station into Mode 2 with reactor steam dome pressure > 150 psig with LCO 3.5.3 not met due to the RCIC system being inoperable. The RCIC system was inoperable since the licensee installed RCIC-AOV-PCV23 with the incorrect mechanical stop, and proper valve operation was not verified after installation during Refueling Outage 29. The immediate corrective actions were to restore RCIC-AOV-PCV23 from having a closed mechanical stop to the required open mechanical stop, and to replace the turbine lube oil gasket to restore operability of the system. Because this violation was of very low safety significance (Green) and was entered into the licensee's corrective action program as Condition Report CR-CNS-2016-08122, this violation is being treated as a non-cited violation (NCV) in accordance with Section 2.3.2.a of the NRC Enforcement Policy.
(NCV 05000298/2017001-06, "Failure to Install Correct Mechanical Stop and Verify Proper Operation")

These activities constituted completion of two event follow-up samples, as defined in Inspection Procedure 71153.

4OA5 Other Activities

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

a. Inspection Scope

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

- The licensee identified and discussed with plant staff the lessons-learned from the open phase condition events at the United States operating plants including the Byron Station open phase condition and its consequences. This included conducting operator training for promptly diagnosing, recognizing consequences, and responding to an open phase condition.

- The licensee updated plant operating procedures to help operators promptly diagnose and respond to open phase conditions on off-site power sources credited for safe shutdown of the plant.

- The licensee established and implemented periodic walk-down activities to inspect switchyard and transformer yard equipment such as insulators, disconnect switches, and transmission line and transformer connections associated with the off-site power circuits to detect a visible open phase condition.

- The licensee ensured that routine maintenance and testing activities on switchyard components have been implemented and maintained. As part of the maintenance and testing activities, the licensee assessed and managed plant risk in accordance with 10 CFR 50.65(a)(4) requirements.

b. Findings

No findings were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On February 9, 2017, the inspector presented the Temporary Instruction 2515/192 inspection results to Mr. D. Buman, Director, Nuclear Safety Assurance, and other members of the licensee staff. The licensee acknowledged the issues presented. No proprietary information was identified.

On April 3, 2017, the inspector conducted a telephonic exit meeting to present the results of the in-office inspection of changes to the licensee's emergency plan to Mr. J. Stough, Manager, Emergency Preparedness, and other members of the licensee staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

On April 13, 2017, the inspectors presented the inspection results to Mr. D. Buman, Director, Nuclear Safety Assurance, and other members of the licensee staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

T. Barker, Manager, Engineering Program and Components
L. Bray, Licensing Specialist
D. Buman, Director, Nuclear Safety Assurance
B. Chapin, Manager, Maintenance
T. Chard, Manager, Quality Assurance
L. Dewhirst, Manager, Corrective Action and Assessment
K. Dia, Director, Engineering
M. Dickerson, Electrical Engineer
T. Forland, Engineer, Licensing
G. Gardner, Engineering Design Manager
D. Goodman, Manager, Operations
K. Higginbotham, Vice President, Chief Nuclear Officer
D. Kimball, Director, Nuclear Oversight
J. Reimers, Manager, System Engineering
J. Shaw, Manager, Licensing
J. Stough, Manager, Emergency Preparedness
C. Sunderman, Manager, Radiation Protection
D. Van Der Kamp, Licensing Technical Specialist

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000298/2017001-01	NCV	Failure to Maintain Alternate Shutdown Emergency Procedure (Section 1R04)
05000298/2017001-02	NCV	Failure to Identify a Condition Adverse to Quality Associated with the 250 Vdc Electrical System (Section 1R12)
05000298/2017001-03	NCV	Failure to Identify a Condition Adverse to Quality (Section 1R15)
05000298/2017001-04	NCV	Failure to Address Nonconforming Pipe Thinning in Accordance with the ASME Code (Section 4OA2)
05000298/2017001-05	NCV	Loss of Shutdown Cooling due to Relay Maintenance (Section 4OA3)
05000298/2017001-06	NCV	Failure to Install Correct Mechanical Stop and Verify Proper Operation (Section 4OA3)

Closed

05000298/2016007-00	LER	Isolation of Shutdown Cooling due to Relay Maintenance Results in a Loss of Safety Function (Section 4OA3)
05000298/2016008-00	LER	Purchase and Installation of Incorrect Actuator Results in a Condition Prohibited by Technical Specifications (Section 4OA3)
2515/192	TI	Inspection of the Licensee's Interim Compensatory Measures Associated with the Open Phase Condition Design Vulnerabilities in Electric Power Systems (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Miscellaneous Documents

Number	Title	Revision
12-069	NEDC, Battery Room Low Temperature Study	0
14-012	Engineering Evaluation, Control Building Essential Ventilation System Calculation Corrections	0
14-056	NEDC, Essential Control Building Ventilation Desired Thermostat Setpoints and Hydrogen Concentration Calculation	0, 0C1

Procedures

Number	Title	Revision
2.2.38.2	Portable Heating System	17
2.3_R-1	Panel R – Annunciator R-1	16

Condition Reports (CRs)

CR-CNS-2011-12345	CR-CNS-2012-00724	CR-CNS-2013-04508	CR-CNS-2014-00588
CR-CNS-2016-07295	CR-CNS-2016-08587		

Work Orders

4551512	5047793

Section 1R04: Equipment Alignment

Miscellaneous Documents

Number	Title	Revision
00-029	NEDC, Post-LOCA Leakage Path to Main Condenser	7, 7C2, 7C3

Miscellaneous Documents

Number	Title	Revision
87-131A	NEDC, 250 VDC Division I Load and Voltage Study	13
87-131B	NEDC, 250 VDC Division II Load and Voltage Study	12
87-131C	NEDC, 125 VDC Division I Load and Voltage Study	17
87-131D	NEDC, 125 VDC Division II Load and Voltage Study	13
91-094	NEDC, 125 VDC/250 VDC Battery Charger Analysis	5
94-034H	NEDC, Containment Analysis for Appendix R – Shutdown from Alternate Shutdown Room	3
2027	Burns & Roe, Flow Diagram – Loop B Reactor Recirculation & Suppression Chamber Vent Systems & Connections Cooper Nuclear Station, Sheet 2	15
2040	Burns & Roe, Cooper Nuclear Station Flow Diagram Residual Heat Removal System Loop B, Sheet 2	19

Procedures

Number	Title	Revision
2.2.67	Reactor Core Isolation Cooling System	73
2.2.67A	Reactor Core Isolation Cooling System Component Checklist	22
2.2.67B	Reactor Core Isolation Cooling System Instrument Valve Checklist	3
2.2.69.1	RHR LPCI Mode	30
2.4TOX	Toxic Gas In the Control Room	12
5.1ASD	Alternate Shutdown	17
5.1Incident	Site Emergency Incident	36
5.2Fuel	Fuel Failure	21
5.4Fire-S/D	Fire Inducted Shutdown from Outside Control Room	69

Condition Reports (CRs)

CN-CNS-2012-01614	CR-CNS-2016-08588	CR-CNS-2016-08823	CR-CNS-2017-01437
CR-CNS-2017-01438	CR-CNS-2017-01439	CR-CNS-2017-01465	

Section 1R05: Fire Protection

Miscellaneous Documents

Number	Title	Revision/Date
	CNS List of Active Fire Impairments	March 20, 2017
17-0010	Transient Combustible Permit – RCIC and CS Pump Room	0
91-03	NEDC, Qualification of Fire Barrier Penetration Seal Details	8

Procedures

Number	Title	Revision
0.7.1	Control of Combustibles	40
0-CNS-WM-104A	Online Fire Risk Management Actions	3
6.FP.606	Fire Barrier/Penetration Seal Visual Examination	25
CNS-FP-211	Reactor Building Northeast Quadrant Elevations 881' and 859'	4
CNS-FP-234	Office Building Cable Expansion Room	3
CNS-FP-285	CNS Fire Barrier Penetration Seal Details, Sheet 2	5

Condition Reports (CRs)

CR-CNS-2017-01680 CR-CNS-2017-01682

Work Orders

5115325

Section 1R06: Flood Protection Measures

Miscellaneous Documents

Number	Title	Revision
09-102	NEDC, Internal Flooding – HELB, MELB, and Feedwater Line Break	1

Procedures

Number	Title	Revision
2.3_S-1	Panel S – Annunciator S-1	24

Procedures

Number	Title	Revision
6.1SW.401	Diesel Generator Service Water Check Valve and Sump Test (IST)(Div 1)	2
6.2SW.401	Diesel Generator Service Water Check Valve and Sump Test (IST)(Div 2)	2
FDN-F02	Alert Control Room to Potential Flooding – (DG Room, Control Building, and Man Hole Drains and Sump Alarms)	4

Work Orders

5039789	5039790

Section 1R07: Heat Sink Performance

Miscellaneous Documents

Number	Title	Revision
2016-0265	Barrier Control Permit, REC B Heat Exchanger Work Week 1708	0
6039660	Engineering Change, Reactor Equipment Cooling Time Delay Relay Relocation	2

Procedures

Number	Title	Revision
7.2.42.1	REC Heat Exchanger Maintenance	11
13.15.1	Reactor Equipment Cooling Heat Exchanger Performance Analysis	35

Condition Reports (CRs)

CR-CNS-2017-00002	CR-CNS-2017-00456	CR-CNS-2017-00858	CR-CNS-2017-00956

Work Orders

5064527	5135116	5135117	5135118	5135119
5115304	5115446	5115447		

Section 1R11: Licensed Operator Requalification Program and Licensed Operator Performance

Miscellaneous Documents

Number	Title	Date
30-003	Reactivity Maneuvering Plan, February 11, 2017 Quarterly Downpower	January 26, 2017

Procedures

Number	Title	Revision
6.MS.201	Main Steam Isolation Valve Operability Test (IST)	23
10-EN-RE-215	Reactivity Maneuver Plan	4C3
15.RF.101	RFPT Stop Valve Test	5
OTH015-17-01	OPS Configuration Management and Watchstanding Principles-DLA	0

Condition Reports (CRs)

CR-CNS-2016-06022	CR-CNS-2016-06166	CR-CNS-2016-06347	CR-CNS-2016-07010
CR-CNS-2016-08636	CR-CNS-2016-08744	CR-CNS-2017-00138	CR-CNS-2017-00184
CR-CNS-2017-00444	CR-CNS-2017-00553	CR-CNS-2017-00620	

Section 1R12: Maintenance Effectiveness

Miscellaneous Documents

Title	Revision
IST Basis Document	10

Procedures

Number	Title	Revision
0.5.OPS	Operations Review of Condition Reports/Operability Determination	56
2.2.24.1	250 VDC Electrical System (Div 1)	14
3.40	Primary Containment Leakage Rate Testing Program	12
3.9	ASME OM Code Testing of Pumps and Valves	29
6.2SW.101	Service Water Surveillance Operation	49

Condition Reports (CRs)

CR-CNS-2008-01253	CR-CNS-2008-01617	CR-CNS-2011-01683	CR-CNS-2011-09718
CR-CNS-2011-09748	CR-CNS-2011-10665	CR-CNS-2012-02112	CR-CNS-2012-02130
CR-CNS-2012-09743	CR-CNS-2013-02616	CR-CNS-2014-06953	CR-CNS-2015-01943
CR-CNS-2016-00813	CR-CNS-2015-05766	CR-CNS-2016-03380	CR-CNS-2016-06185
CR-CNS-2016-06731	CR-CNS-2016-08253	CR-CNS-2016-08685	CR-CNS-2016-09021
CR-CNS-2017-00750	CR-CNS-2017-00869	CR-CNS-2017-00885	

Work Orders

5028389	5137384

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Miscellaneous Documents

Number	Title	Revision/Date
	RCIC Window Week 1710 Protected Equipment Program Tracking Form	March 8, 2017
	Week 1704 RHR Div 2 window protected equipment Program Tracking Form	January 25, 2017
87-131A	NEDC, 250 VDC Division 1 Load and Voltage Study	
17003	Probabilistic Risk Assessment, Bounding Risk Profile for RCIC & SW-V-118 Extended Unavailability	March 14, 2017
11343412	Engineering Change, Dedication of MM 2111972	0
11343413	Engineering Change, Dedication of MM 2111971	

Procedures

Number	Title	Revision
0-CNS-WM-100	Work Order Generation, Screening, and Classification	7
0-CNS-WM-104	On-Line Schedule Risk Assessment	3
0-PROTECT-EQP	Protected Equipment Program	36
2.2.17	Emergency Station Service Transformer	64
2.2.18	4160V Auxiliary Power Distribution System	210
2.3_C-1	Panel C – Annunciator C-1	30
2.3_C-2	Panel C – Annunciator C-2	52

Procedures

Number	Title	Revision
2.3_C-3	Panel C – Annunciator C-3	50
2.3_C-4	Panel C – Annunciator C-4	31
5.3Grid	Degraded Grid Voltage	46
6.2DG.102	Diesel Generator Demonstration of Operability Test (DIV 2)	55

Condition Reports (CRs)

CR-CNS-2017-00222	CR-CNS-2017-00223	CR-CNS-2017-00346	CR-CNS-2017-00551
CR-CNS-2017-00552	CR-CNS-2017-00561	CR-CNS-2017-00562	CR-CNS-2017-01144
CR-CNS-2017-01168	CR-CNS-2017-01360	CR-CNS-2017-01370	CR-CNS-2017-01371
CR-CNS-2017-01387	CR-CNS-2017-01405	CR-CNS-2017-01414	CR-CNS-2017-01449
CR-CNS-2017-01453	CR-CNS-2017-01457	CR-CNS-2017-01458	CR-CNS-2017-01462

Work Orders

5028504	5066859	5072905	5115325	5121485
5121816	5127747	5154689	5163417	5173717
5173718	5174199	5176196	5179597	5180525
5180968	5181625			

Section 1R15: Operability Determinations and Functionality Assessments

Miscellaneous Documents

Number	Title	Revision
	CO RHRA-1-RE29 RHRA MAINTENANCE, Clearance Order for RHR A Maintenance Window	0
	Security Badge Reader Logs – October 7, 2016, and November 23 – 29, 2016	
87-131A	NEDC, 250 VDC Division 1 Load and Voltage Study	
3255	Burns & Roe Control Room – Control Panels Connection Wiring Diagram Sheet #39	12
719E353	Probe Buffer – Rod Position Indication System Drawing	10
791E338	Rod Position Indication System Cabinet Drawing	8
IAC750-00-00	Rod Position Information System, CT#924	4

Procedures

Number	Title	Revision/Date
0-EN-HU-103	Human Performance Error Review – RHR Minimum Flow Valves Isolated	February 7, 2017
0.37	Measuring and Test Equipment (M&TE) Calibration Program Guidelines	31
0.5.OPS	Operations Review of Condition Reports/Operability Determination	56
2.0.2	Operations Logs and Reports	108
2.2.69.1	RHR LPCI Mode	30
2.2.69.3	RHR Suppression Pool Cooling and Containment Spray	47
2.2.70	RHR Service Water Booster Pump System	80
2.3_9-3-1	Annunciator Response Procedure – Panel 9-3-1	37
2.4RPIS	Rod Position Indication System Failure	8
5.3EMPWR	Emergency Power During Modes 1, 2, or 3	65
6.1EE.602	Div 1 125V/250V Station Battery 92 Day Check	7
7.2.14	RHR SWBP Overhaul and Replacement	43
13.15.1	Reactor Equipment Cooling Heat Exchanger Performance Analysis	35

Condition Reports (CRs)

CR-CNS-2016-06527	CR-CNS-2016-08588	CR-CNS-2017-00002	CR-CNS-2017-00034
CR-CNS-2017-00054	CR-CNS-2017-00067	CR-CNS-2017-00278	CR-CNS-2017-00346
CR-CNS-2017-00394	CR-CNS-2017-00553	CR-CNS-2017-00558	CR-CNS-2017-00620
CR-CNS-2017-00630	CR-CNS-2017-00810		

Work Orders

5047338	5057195	5057727	5059856	5068552
5170717	5171194	5174199		

Section 1R18: Plant Modifications

Condition Reports (CRs)

CR-CNS-2016-05963	CR-CNS-2016-07649	CR-CNS-2016-06582	CR-CNS-2016-08766
CR-CNS-2016-08783	CR-CNS-2016-08790		

Section 1R19: Post-Maintenance Testing

Miscellaneous Documents

Number	Title	Revision/Date
	Failure Modes and Effects Analysis – RCIC Overspeed Trip Mechanism Actuation	March 11 – 18, 2017
2040	Burns & Roe, Cooper Nuclear Station Flow Diagram Residual Heat Removal System Loop B, Sheet 2	19
2077	Burns and Roe, Flow Diagram – Diesel Generator Building Service Water, Starting Air, Fuel Oil, Sump System and Roof Drains Cooper Nuclear Station	N78
9072400910	Schematic and Interconnection Diagram Series Booster Exciter Voltage Regulator	N08
B-12555	Vendor Manual Drawing – RCIC Overspeed Trip Drawing	1
FMEA-2017-01168	DG1 Voltage Regulator Anomaly 3-6-17 Failure Modes and Effects Analysis	2

Procedures

Number	Title	Revision
0-CNS-WM-104	On-Line Schedule Risk Assessment	3
2.2.67.1	Reactor Core Isolation Cooling System Operations	38
2.2.69.2	Residual Heat Removal System	93
6.EE.601	125/250 V Station and Diesel Fire Pump Battery 7 Day Check	22
6.EE.604	125V Battery Charger Performance Test	21
6.EE.606	250 V Battery Charger Performance Test	23
6.RCIC.102	RCIC IST and 92 Day Test	33
6.RCIC.105	RCIC Turbine Overspeed Testing	19
6.RCIC.311	RCIC Control System Calibration Test	12
6.1DG.101	Diesel Generator 31 Day Operability Test (IST)(DIV 1)	88
6.1DG.105	Diesel Generator Starting Air Compressor Operability (IST)(DIV 1)	23
6.1DG.402	IST Closure Testing of DGSA Receiver Inlet Check Valve (DIV 1)	11
6.1HV.602	Air Flow Test of Fan Coil Unit HV-DG-1C (DIV 1)	8
7.3.1.6	125/250 VDC Station Battery Charger Protective Relays Testing and Calibration	18

Procedures

Number	Title	Revision
7.3.23.6	Battery Charger Clean and Inspect	1
7.3.28.1	Lead Removal/Installation and Lug Installation	30
7.3.51	Electrical Meter Calibration Check	15

Condition Reports (CRs)

CR-CNS-2016-06764	CR-CNS-2016-08586	CR-CNS-2016-08889	CR-CNS-2017-00045
CR-CNS-2017-00222	CR-CNS-2017-00223	CR-CNS-2017-00600	CR-CNS-2017-00610
CR-CNS-2017-00616	CR-CNS-2017-01144	CR-CNS-2017-01168	CR-CNS-2017-01341
CR-CNS-2017-01448	CR-CNS-2017-01449	CR-CNS-2017-01453	CR-CNS-2017-01457
CR-CNS-2017-01458	CR-CNS-2017-01539	CR-CNS-2017-01555	CR-CNS-2017-01589
CR-CNS-2017-01590	CR-CNS-2017-01591		

Work Orders

5039808	5039809	5039810	5040553	5054917
5061169	5066016	5067317	5069263	5072905
5104689	5114968	5114970	5115325	5115336
5115421	5115615	5115616	5121402	5121485
5121816	5127747	5134090	5137384	5154689
5163417	5173717	5173718	5180525	5180968
5181625				

Section 1R22: Surveillance Testing

Miscellaneous Documents

Number	Title	Revision
87-131A	NEDC, 250 VDC Division 1 Load and Voltage Study	13
87-131C	NEDC, 125 VDC Division 1 Load and Voltage Study	17
91-94	NEDC, 125 VDC/250 VDC Battery Charger Analysis	5

Procedures

Number	Title	Revision
2.2.67.1	Reactor Core Isolation Cooling System Operations	38

Procedures

Number	Title	Revision
6.EE.604	125 V Battery Charger Performance Test	21
6.EE.606	250 V Battery Charger Performance Test	23
6.RCIC.102	RCIC IST and 92 Day Test	33
6.RCIC.105	RCIC Turbine Overspeed Testing	19
6.1HV.303	Division 1 Essential Control Building Ventilation Temperature Switch Change Out and Functional Test	16
6.1RHR.101	RHR Test Mode Surveillance Operation (IST)(DIV 1)	35
6.2HV.303	Division 2 Essential Control Building Ventilation Temperature Switch Change Out and Functional Test	16
6.2SWBP.101	RHR Service Water Booster Pump Flow Test and Valve Operability Test (DIV 2)	27

Condition Reports (CRs)

CR-CNS-2013-07137	CR-CNS-2016-08549	CR-CNS-2017-00453	CR-CNS-2017-00586
CR-CNS-2017-00587	CR-CNS-2017-00745	CR-CNS-2017-01291	CR-CNS-2017-01371
CR-CNS-2017-01508	CR-CNS-2017-01999		

Section 1EP6: Drill Evaluation

Procedures

Number	Title	Revision
5.3EMPWR	Emergency Power During Modes 1, 2, or 3	65
5.3SBO	Station Blackout	41

Condition Reports (CRs)

CR-CNS-2017-01805	CR-CNS-2017-01811	CR-CNS-2017-01822	CR-CNS-2017-01832

Section 4OA1: Performance Indicator Verification

Procedures

Number	Title	Revision
0-EN-LI-114	Performance Indicator Process	5C2

Section 4OA2: Problem Identification and Resolution

Miscellaneous Documents

Number	Title	Revision/Date
00-003	NEDC, CNS Aux Power System Load Flow and Voltage Analysis	3C2
2851-3	18” SW-1 Class IVP – Reactor Building Isometric Drawing	N10
2852-8	SW-2 Service Water Class IVP – Reactor Building Isometric Drawing	N16
SW-E-9-2851-3	Ultrasonic Thickness Measurement Report – RHRSW Data	December 13, 2016
SW-E-11-2852-8	Thickness Measurement Report – REC SW Data	August 29, 2016

Procedures

Number	Title	Revision
2.2.18	4160V Auxiliary Power Distribution System	212
6.2DG.302	Undervoltage Logic Functional, Load Shedding, and Sequential Loading Test (DIV 2)	79
7.3.40	Inspection and Meggering of 4160 Volt Buses	25
7.3.41	Examination and High Pot Testing of Non-Segregated Buses and Associated Equipment	11

Condition Reports (CRs)

CR-CNS-2011-03681	CR-CNS-2011-03839	CR-CNS-2015-01731	CR-CNS-2015-01743
CR-CNS-2015-01745	CR-CNS-2015-01746	CR-CNS-2015-01790	CR-CNS-2015-01817
CR-CNS-2014-01072	CR-CNS-2014-01763	CR-CNS-2016-04984	CR-CNS-2016-05123
CR-CNS-2016-05167	CR-CNS-2016-05219	CR-CNS-2016-05558	CR-CNS-2016-05628
CR-CNS-2016-05963	CR-CNS-2016-06582	CR-CNS-2016-07313	CR-CNS-2016-07359
CR-CNS-2016-07649	CR-CNS-2016-08766	CR-CNS-2016-08783	CR-CNS-2016-08790
CR-CNS-2017-00223	CR-CNS-2017-00346	CR-CNS-2017-01018	CR-CNS-2017-01032
CR-CNS-2017-01273			

Work Orders

4499060	50020103	5066859	5120798	5121083
5121085	5121125	5147447	5148640	5149082

Work Orders

5150017	5150817	5173717	5173718	5180895

Section 4OA3: Follow-up of Events and Notices of Enforcement Discretion

Miscellaneous Documents

Number	Title	Revision
00-67	NEDC, Functional and MEDP Evaluation for RCIC-AOV-PCV23	0
1999-006-00	Licensee Event Report, Inadvertent Half-Group VII Isolation due to Deenergization of a Relay	0
791E226	Primary Containment Isolation System As-Built	19
DC 94-332	Residual Heat Removal Minimum Flow Bypass Valve Modification	0

Procedures

Number	Title	Revision
2.2.69.2	Residual Heat Removal System	93
2.4SDC	Shutdown Cooling Abnormal	15
6.2RHR.305	RHR Loop B Pump Low Flow Switch Channel Calibration (DIV 2)	15
7.2.51.1	Air Operator Valve Actuator Setup/Testing	21
7.3.16	Low Voltage Relay Removal and Installation	22
14.25.3	RCIC Auxiliary Cooling Supply Pressure Control LOOP	5

Condition Reports (CRs)

CR-CNS-1999-00486	CR-CNS-2016-05712	CR-CNS-2016-06901	CR-CNS-2016-07634
CR-CNS-2016-07636	CR-CNS-2016-07645	CR-CNS-2016-07654	CR-CNS-2016-08122
CR-CNS-2017-01195	CR-CNS-2017-01227		

Work Orders

454006595	454006596	5056956	5057932	5064335

Section 4OA5: Other Activities

Miscellaneous Documents

Number	Title	Date
2014-0097	Learning Opportunity	
NLS2012081	90-Day Response to NRC Bulletin 2012-01, Design Vulnerability in Electric Power System	October 24, 2012
NLS2014014	Response to Request for Additional Information Regarding Bulletin	February 3, 2014

Procedures

Number	Title	Revision
2.1.11.3	Radwaste and Augmented Radwaste Building Data	77
2.1.12	Control Room Data	107
5.3GRID	Degraded Grid Voltage	46
6.EE.610	Off-Site AC Power Alignment	41

Condition Reports (CRs)

CR-CNS-2012-01136 CR-CNS-2013-07898 CR-CNS-2016-00665* CR-CNS-2017-00666*

* - Initiated due to inspection

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Information Request
January 5, 2017
Notification of Inspection and Request for Information
Cooper Nuclear Station
NRC Inspection Report 05000298/2017001

INSPECTION DOCUMENT REQUEST

Inspection Dates: TBD (Expecting mid to late February)
Inspector: Eduardo Uribe

Documents Requested:

1. Response to NRC Bulletin 2012-01

2. Corrective action documents (in full detail) of the interim corrective actions

3. Corrective action documents (in summary) of the final corrective actions (for my awareness)

4. Any supporting documents for those interim corrective actions (e.g. Ops Procedures, Maintenance Procedures, Work Orders and/or Updated Training Modules).