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

June 22, 2015

EA-15-089

Mr. Oscar A. Limpias
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 COMPONENT DESIGN BASIS INSPECTION REPORT NO. 05000298/2015007 AND NOTICE OF VIOLATION

Dear Mr. Limpias:

On May 8, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the Cooper Nuclear Station. The NRC inspectors discussed the results of this inspection with you and other members of your staff. Inspectors documented the results of this inspection in the enclosed inspection report.

During this inspection, the NRC staff examined activities conducted under your license as they relate to public health and safety to confirm compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC has determined that a cited violation is associated with this inspection. The violation is being cited because Cooper Nuclear Station failed to restore compliance with NRC requirements within a reasonable time after a previous violation was identified in NRC Inspection Report 05000298/2010007 (issued December 3, 2010). This is consistent with the NRC Enforcement Policy; Section 2.3.2.a, which states, in part, that a cited violation will be considered if the licensee fails to restore compliance within a reasonable time after a violation is identified.

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

In addition, the NRC has determined that a Severity Level IV violation of NRC requirements occurred. The NRC also identified two additional issues that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has also determined that violations are associated with these issues. These violations are being treated as non-cited violations (NCVs), consistent with Section 2.3.2.a of the NRC Enforcement Policy. These NCVs are described in the subject inspection report.

If you contest the violations or significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to: (1) the Regional Administrator, Region IV; (2) the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and (3) the NRC Senior Resident Inspector at the Cooper Nuclear Station.

If you disagree with a cross-cutting aspect assignment or a finding not associated with a regulatory requirement in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV; and the NRC Senior Resident Inspector at the Cooper Nuclear Station.

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

Sincerely,

/RA/

Anton Vegel, Director
Division of Reactor Safety

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

Enclosures:
1. Notice of Violation
2. Inspection Report No. 05000298/2015007
   w/Attachment: Supplemental Information

cc: Electronic Distribution

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OFFICE	SRI:EB1	RI:EB2	TTC	SRI:EB1	BC:EB1	BC:DRP/C	ACES	D:DRS
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Letter to Oscar A. Limpias from Anton Vegel, dated June 22, 2015

SUBJECT: COOPER NUCLEAR STATION - NRC COMPONENT DESIGN BASIS INSPECTION REPORT NO. 05000298/2015007 AND NOTICE OF VIOLATION

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ROPreports

NOTICE OF VIOLATION

Nebraska Public Power District	Docket No. 50-298
Cooper Nuclear Station	License No. DPR-46
	EA-15-089

During an NRC inspection conducted April 6 through May 8, 2015, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in 10 CFR 50.2 and as specified in the license application, for those components to which this appendix applies, are correctly translated into specifications, drawings, procedures, and instructions.
>
> Contrary to the above, since July 2010 the licensee failed to assure that applicable regulatory requirements and the design basis were correctly translated into specifications, drawings, procedures, and instructions. Specifically, the licensee failed to correctly translate regulatory and design basis requirements associated with tornado and high wind-generated missiles into design information necessary to protect the emergency diesel generator fuel oil storage tank vent line components.

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

Pursuant to the provisions of 10 CFR 2.201, Nebraska Public Power District is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001 with a copy to the Regional Administrator, Region IV, and a copy to the NRC Senior Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation; EA-15-089" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

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

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's ADAMS, accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any

personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Dated this 22nd day of June 2015

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000298
License:	DPR-46
Report No.:	05000298/2015007
Licensee:	Nebraska Public Power District
Facility:	Cooper Nuclear Station
Location:	P.O. Box 98 Brownville, NE 68321-0098
Dates:	April 6 – May 8, 2015
Team Leader:	W. Sifre, Senior Reactor Inspector, Engineering Branch 1
Inspectors:	R. Latta, Senior Reactor Inspector, Engineering Branch 1 N. Okonkwo, Reactor Inspector, Engineering Branch 2 M. Emrich, Senior Reactor Technology Instructor, Technical Training Center
Accompanying Personnel:	C. Barron, Contractor, Beckman and Associates S. Kobylarz, Contractor, Beckman and Associates
Approved By:	Thomas R. Farnholtz Branch Chief, Engineering Branch 1 Division of Reactor Safety

Enclosure 2

SUMMARY

IR 05000298/2015007; 04/06/2015 – 05/08/2015; Cooper Nuclear Station; Component Design Basis Inspection.

The inspection activities described in this report were performed between April 6, 2015, and May 8, 2015, by three inspectors from the NRC's Region IV office, one instructor from the NRC's Technical Training Center, and two contractors. Four findings of very low safety significance (Green) are documented in this report. Three of these findings involved violations of NRC requirements and one of these violations was determined to be Severity Level IV under the traditional enforcement process. The significance of inspection findings is indicated by their color (Green, White, Yellow, or Red), which is determined using Inspection Manual Chapter 0609, "Significance Determination Process." Their cross-cutting aspects are determined using Inspection Manual Chapter 0310, "Aspects Within the Cross-Cutting Areas." Violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process."

Cornerstone: Initiating Events

- Green. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," which states, in part, "Activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these procedures." Specifically, prior to April 6, 2015, the licensee failed to follow Procedure .05.OPS, "Operations Review of Condition Reports/Operability Determination," to ensure that an operability review was performed for Condition Report CR-CNS-2015-01268, which was initiated during the self-audit for the Component Design Bases Inspection to document that Cooper Nuclear Station has under-voltage relays that could be affected by harmonics. In response to this issue, the licensee performed an operability review and an operability evaluation for the under-voltage relays. This finding was entered into the licensee's corrective action program as Condition Report CR-CNS-2015-02337.

  The team determined that failure to perform an operability review associated with Condition Report CR-CNS-2015-01268 was a performance deficiency. This finding was more than minor because it was associated with the human performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown, as well as power operations. Specifically, the licensee failed to perform the required operability review for the identified condition. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 1, "Initiating Event Screening Questions," the issue screened as having very low safety significance (Green) because the finding did not cause a reactor trip and it did not involve the loss of mitigation equipment. This finding had a cross-cutting aspect in the area of human performance associated with teamwork because individuals and work groups failed to communicate and coordinate their activities across organizational boundaries to ensure nuclear safety is maintained [H.4]. (Section 1R21.2.7)

Cornerstone: Mitigating Systems

- Green. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, that "design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Specifically, prior to April 6, 2015, the licensee failed to maintain procedure changes to periodically monitor and add nitrogen to fire protection system headers in the reactor building to mitigate the effects of water hammer. In response to this issue, the licensee determined that the fire protection system remained functional without nitrogen based on empirical evidence suggesting that the system was capable of absorbing the shockwave from a water hammer event. This finding was entered into the licensee's corrective action program as Condition Report CR-CNS-2015-02085.

  The team determined that the failure to adequately maintain control of the fire protection system design to prevent water hammer events was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to events to prevent undesirable consequences. Specifically, the licensee failed to maintain procedure changes to periodically monitor and add nitrogen to fire protection system headers in the reactor building. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. The team determined that this finding did not have a cross-cutting aspect because the most significant contributor did not reflect current licensee performance. (Section 1R21.3.1)

- SL IV. The team identified three examples of a Severity Level IV, non-cited violation, of 10 CFR 50.71, "Maintenance of Records, Making of Reports," Section (e), which states, in part, "each person licensed to operate a nuclear power reactor under the provisions of 10 CFR 50.21 or 10 CFR 50.22 shall update periodically the final safety analysis report (FSAR) originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed. This submittal shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the licensee since the submittal of the original FSAR, or as appropriate, the last update to the FSAR under this section." Specifically, in January 2012 and February 2015, the licensee failed to update the Updated Safety Analysis Report for changes made to their Anticipated Transient Without Scram analyses and plant conduct of operations procedures. This finding was entered into the licensee's corrective action program as Condition Reports CR-CNS-2015-02106, CR-CNS-2015-02090, and CR-CNS-2015-02393.

  The team determined that the failure to update the Final Safety Analysis Report to assure that the information included in the report contains the latest information developed was a

performance deficiency. This finding was evaluated using traditional enforcement because it had the potential for impacting the NRC's ability to perform its regulatory function. This finding was more than minor because each example potentially rendered portions of the safety analyses for Anticipated Transient Without Scram events described in the Updated Safety Analysis Report less conservative or contradicted previous information regarding the licensee's flooding analysis contained in the Updated Safety Analysis Report. The traditional enforcement violation was determined to be a Severity Level IV violation consistent with the example in paragraph 6.1.d(3) of the NRC Enforcement Policy. Since this was a traditional enforcement violation, no cross-cutting aspects were assigned per the guidance contained in Inspection Manual Chapter 0612, Section 07.03(c). (Section 1R21.4)

- Green. The team identified a Green, cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "Design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Specifically, since July 2010 the licensee failed to verify the adequacy of design of the vents for the emergency diesel generator 1 and 2 fuel oil storage tanks to withstand impact from a tornado driven missile hazard, or to evaluate for exemption from missile protection requirements using an approved methodology. This finding was entered into the licensee's corrective action program as Condition Report CR-CNS-2015-02366.

  The team determined that the failure to evaluate the lack of missile protection on the emergency diesel generator 1 and 2 fuel storage tank vents was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to events to prevent undesirable consequences. Specifically, the licensee failed to evaluate a design nonconformance on the emergency diesel generator 1 and 2 fuel storage tanks for lack of missile protection. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," this finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance associated with conservative bias because individuals failed to use decision making practices that emphasize prudent choices over those that are simply allowable [H.14]. (Section 4OA2)

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

This inspection of component design bases verifies that plant components are maintained within their design basis. Additionally, this inspection provides monitoring of the capability of the selected components and operator actions to perform their design basis functions. As plants age, modifications may alter or disable important design features making the design bases difficult to determine or obsolete. The plant risk assessment model assumes the capability of safety systems and components to perform their intended safety function successfully. This inspectable area verifies aspects of the Initiating Events, Mitigating Systems, and Barrier Integrity cornerstones for which there are no indicators to measure performance.

1R21 Component Design Basis Inspection (71111.21)

.1 Overall Scope

To assess the ability of the Cooper Nuclear Station, equipment and operators to perform their required safety functions, the team inspected risk-significant components and the licensee's responses to industry operating experience. The team selected risk-significant components for review using information contained in the Cooper Nuclear Station, probabilistic risk assessments and the U.S. Nuclear Regulatory Commission's (NRC) standardized plant analysis risk model. In general, the selection process focused on components that had a risk achievement worth factor greater than 1.3 or a risk reduction worth factor greater than 1.005. The items selected included components in both safety-related and nonsafety-related systems including pumps, circuit breakers, heat exchangers, transformers, and valves. The team selected the risk-significant operating experience to be inspected based on its collective past experience.

To verify that the selected components would function as required, the team reviewed design basis assumptions, calculations, and procedures. In some instances, the team performed calculations to independently verify the licensee's conclusions. The team also verified that the condition of the components was consistent with the design basis and that the tested capabilities met the required criteria.

The team reviewed maintenance work records, corrective action documents, and industry operating experience records to verify that licensee personnel considered degraded conditions and their impact on the components. For selected components, the team observed operators during simulator scenarios, as well as during simulated actions in the plant.

The team performed a margin assessment and detailed review of the selected risk-significant components to verify that the design basis have been correctly implemented and maintained. This design margin assessment considered original design issues, margin reductions because of modifications, and margin reductions identified as a result

of material condition issues. Equipment reliability issues were also considered in the selection of components for detailed review. These included items such as failed performance test results; significant corrective actions; repeated maintenance; Title 10 CFR 50.65(a)1 status; operable, but degraded conditions; NRC resident inspector input of problem equipment; system health reports; industry operating experience; and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense in-depth margins.

The inspection procedure requires a review of 15 to 25 total samples that include risk-significant and low design margin components, components that affect the large early release frequency (LERF), and operating experience issues. The sample selection for this inspection was 16 components, 2 components that affect LERF, and 4 operating experience items. The selected components and associated operating experience items supported risk-significant functions including the following:

a. <u>Electrical power to mitigation systems</u>: The team selected several components in the electrical power distribution systems to verify operability to supply alternating current (ac) and direct current (dc) power to risk-significant and safety-related loads in support of safety system operation in response to initiating events, such as loss of offsite power, station blackout, and a loss of coolant accident with offsite power available. As such, the team selected:

- 125 Vdc Battery 1A
- 125 Vdc Charger 1A
- 125 Vdc Bus 1A
- Standby Liquid Control Pump Motor Protection
- Core Spray Pump Motor Protection
- 480 Vac Safety-Related Motor Control Center K
- 4160 Vac Safety-Related Switchgear 1F
- Startup Station Service Transformer

b. <u>Components that affect LERF</u>: The team reviewed components required to perform functions that mitigate or prevent an unmonitored release of radiation. The team selected the following components:

- Main Steam Isolation Valve MS-AOV-AO80C
- Suppression Chamber Spray A Inboard Throttle Valve RHR-MOV-MO38A

c. <u>Mitigating systems needed to attain safe shutdown</u>: The team reviewed components required to perform the safe shutdown of the plant. As such, the team selected:

- Core Spray Pump CS-P-A
- Standby Liquid Control Pump SLC-P-A
- Residual Heat Removal Heat Exchanger B
- Residual Heat Removal Service Water Booster Pump 1C
- Residual Heat Removal Heat Exchanger B Bypass Valve

- Residual Heat Removal Service Water Motor Operated Valve 89B

.2 Results of Detailed Reviews for Components

.2.1 125 Vdc Battery 1A

a. Inspection Scope

The team reviewed the updated safety analysis report, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with 125 Vdc Battery 1A. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

- Calculations for electrical system load flow/voltage drop to verify that system voltages remained within minimum acceptable limits.

- Calculations to verify design loading, input assumptions, and environmental parameters are appropriate and that the battery cell is sized to perform the battery design basis function in accordance with the technical specifications.

- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance.

- Results of completed surveillance testing in accordance with technical specifications.

b. Findings

No findings were identified.

.2.2 125 Vdc Charger 1A

a. Inspection Scope

The team reviewed the updated safety analysis report, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with 125 Vdc Charger 1A. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

- Calculations for sizing to verify that charger capacity supported design basis requirements in accordance with technical specifications.

- Calculations for the electrical protection to verify the charger protective devices satisfied design basis requirements.

- Procedures for preventive maintenance, inspection, and testing to verify vendor guidance and design requirements were adequately incorporated.

- Results of completed preventative maintenance and surveillance testing in accordance with technical specifications.

b. Findings

No findings were identified.

.2.3 125 Vdc Bus 1A

a. Inspection Scope

The team reviewed the updated safety analysis report, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with 125 Vdc Bus 1A. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

- Calculations for electrical distribution, system load flow/voltage drop, short-circuit, and electrical protection to verify that bus capacity and voltages remained within minimum acceptable limits.

- The protective device ratings to ensure adequate selective protection coordination of connected equipment during worst-case short circuit conditions.

- Procedures for preventive maintenance, inspection, and testing to verify vendor guidance and design requirements were adequately incorporated.

b. Findings

No findings were identified.

.2.4 Standby Liquid Control Pump Motor Protection

a. Inspection Scope

The team reviewed the updated safety analysis report, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with the standby liquid control pump motor. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.
- The protective device ratings to ensure adequate motor circuit protection and selective protection coordination of connected equipment during worst-case short circuit conditions.

b. Findings

No findings were identified.

.2.5 Core Spray Pump Motor Protection

a. Inspection Scope

The team reviewed the updated safety analysis report, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with the core spray pump motor. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.
- Calculations for electrical protection to verify adequate overcurrent relay settings for motor circuit design basis requirements.
- Protective relay test and calibration results to verify motor overcurrent relays performed in accordance with acceptable setting tolerances.

b. Findings

No findings were identified.

.2.6 480 Vac Safety-Related Motor Control Center K

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with 480 Vac Safety-Related Motor Control Center K. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

- Calculations for electrical distribution, system load flow/voltage drop, short-circuit, and electrical protection to verify that bus capacity and voltages remained within minimum acceptable limits.

- The protective device settings and circuit breaker ratings to ensure adequate selective protection coordination of connected equipment during worst-case short circuit conditions.

- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance; including the cable aging management program.

- Results of completed preventative maintenance on switchgear and breakers, including breaker tracking.

b. Findings

No findings were identified.

.2.7 4160 Vac Safety-Related Switchgear 1F

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with 4160 Vac safety-related switchgear 1F. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

- Calculations for electrical distribution, system load flow/voltage drop, short-circuit, cables routing, and electrical protection to verify that bus capacity and voltages remained within acceptable limits.

- The protective device settings and circuit breaker ratings to ensure adequate selective protection coordination of connected equipment during worst-case short circuit conditions.

- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance; including the cable aging management program.

- Results of completed preventative maintenance on switchgear and breakers, including breaker tracking.

b. Findings

Failure to Perform an Operability Review of a Condition Report

Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the licensee’s failure to accomplish an operability review in accordance with Procedure .05.OPS, “Operations Review of Condition Reports/Operability Determination.” Specifically, the licensee failed to ensure that an operability review was performed for Condition Report CR-CNS-2015-01268, which was initiated to investigate how plant equipment is effected due to harmonics on under-voltage relays.

Description. The team performed a review of corrective actions associated with 4160 Vac safety-related switchgear 1F. Under-voltage relays (27-XX) are used to monitor voltage levels (degraded or loss of voltage) in 4160 Vac switchgear. Relay 27-1F2 is the under-voltage relay for 4160 Vac Switchgear 1F. In January 2015 Cooper Nuclear Station personnel performed a Component Design Basis Inspection focused self-assessment in advance of the April 2015 NRC Component Design Basis Inspection. During this assessment, it was found that the impact of harmonics on under-voltage relays had not been considered in Calculation NEDC 88-086B, “Setpoint Determination of Second Level Under-Voltage Relays”. Condition Report CR-CNS-2015-01268 was then initiated to evaluate the impact of harmonics on under-voltage relays. Harmonics are available on medium voltage switchgear through breaker functions and rotating machinery. It is stated in the condition report that “Harmonics may affect when the undervoltage relays trip, causing the essential buses to be shed earlier or not at all when it is desired.”

In reviewing the corrective action specified in Condition Report CR-CNS-2015-01268, the team determined that, though an operability review was stated to be required, none had been performed. The licensee also concluded that the harmonics “condition does not affect installed plant equipment.” The team questioned the validity of this conclusion when no operability review was performed and it is known that the type of under-voltage

relays installed at Cooper Nuclear Station are susceptible to harmonic effect. In response, the licensee initiated Condition Report CR-CNS-2015-02337 to document the deficiency and perform the operability review in accordance with Procedure .05.OPS, “Operations Review of Condition Reports/Operability Determination.” The result of the operability review was to perform an operability evaluation in accordance with Procedure 0.5.OPS to support a prompt operability determination.

Analysis. The team determined that failure to perform an operability review associated with Condition Report CR-CNS-2015-01268 was a performance deficiency. This finding was more than minor because it was associated with the human performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown, as well as power operations. Specifically, the licensee failed to perform the required operability review for the identified condition. In accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated June 19, 2012, Exhibit 1, “Initiating Event Screening Questions,” the issue screened as having very low safety significance (Green) because the finding did not cause a reactor trip and it did not involve the loss of mitigation equipment. This finding had a cross-cutting aspect in the area of human performance associated with teamwork because individuals and work groups failed to communicate and coordinate their activities across organizational boundaries to ensure nuclear safety is maintained [H.4].

Enforcement. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” which states, in part, “Activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these procedures.” Contrary to the above, prior to April 6, 2015, the licensee failed to ensure that activities affecting quality as prescribed by documented procedures of a type appropriate to the circumstances were accomplished in accordance with those procedures. Specifically, the licensee failed to follow Procedure .05.OPS, “Operations Review of Condition Reports/Operability Determination,” to ensure that an operability review was performed for Condition Report CR-CNS-2015-01268, which was initiated during the self-audit for the Component Design Bases Inspection to document that Cooper Nuclear Station has under-voltage relays that could be affected by harmonics. In response to this issue, the licensee performed an operability review and an operability evaluation for the under-voltage relays. This finding was entered into the licensee’s corrective action program as Condition Report CR-CNS-2015-02337. Because this finding was of very low safety significance and has been entered into the licensee’s corrective action program, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy:
NCV 05000298/2015007-01, “Failure to Perform an Operability Review of a Condition Report.”

.2.8 Startup Station Service Transformer

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with startup station service transformer. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- The design bases document and updated safety analysis report to verify design bases requirement for the transformers.

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

- Calculations for electrical distribution, system load flow/voltage drop, short-circuit, and electrical and mechanical protection to verify transformer loading and voltage capacity limits.

- The transformer protective device settings and tap changer settings to ensure adequate selective protection coordination of connected equipment during worst-case short circuit conditions.

- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance; including the cable and segregated bus aging management program.

- Transformer dissolved gas analysis test reports to evaluate the result and trend to ensure the health of the transformer oil and insulations.

b. Findings

No findings were identified.

.2.9 Main Steam Isolation Valve MS-AOV-AO80C

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with main steam isolation valve MS-AOV-AO80C. The team also conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

- Valve stroke time and leakage test procedures, acceptance criteria, and recent test results.

- Evaluation of the impact of instrument air pressure, steam flow, and building pressure on the valve closing time.

- Calculation inputs associated with main steam isolation valve minimum and maximum allowable closing times.

b. Findings

No findings were identified.

.2.10 Suppression Chamber Spray A Inboard Throttle Valve RHR-MOV-MO38A

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with suppression chamber spray A inboard throttle valve RHR-MOV-MO38A. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

- Calculations for the required thrust to operate the motor-operated valve under the most limiting conditions.

- Motor-operated valve test procedures, acceptance criteria, and recent test results.

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

- Calculations for motor-operated valve design basis operating conditions to verify acceptable methodology for the selection of motor thermal overload protection.

- Periodic testing for motor thermal overload relays to verify there was no unacceptable deterioration for the relays not bypassed during design basis conditions.

b. Findings

No findings were identified.

.2.11 Core Spray Pump CS-P-A

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with core spray pump CS-P-A. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.
- Calculations for the required pump flow, head, minimum flow, and net positive suction head under the most limiting conditions, including under and over-frequency conditions.
- Pump test procedures, acceptance criteria, and recent test results.

b. Findings

No findings were identified.

.2.12 Standby Liquid Control Pump SLC-P-A

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with standby liquid control pump SLC-P-A. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.
- Calculations for the required pump flow, head, minimum flow, and net positive suction head under the most limiting conditions, including postulated Anticipated Transient Without Scram events.

- Pump test procedures, acceptance criteria, and recent test results.

b. Findings

No findings were identified.

.2.13 Residual Heat Removal Heat Exchanger B

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with residual heat removal heat exchanger B. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Work orders and corrective action program documents for the last three years.

- System design criteria and system health reports.

- Corrective action program reports to verify the monitoring and correction of potential degradation, operability evaluations, and apparent cause evaluations.

- Piping and instrumentation diagrams.

- Residual heat removal heat exchanger plugged tube map.

b. Findings

No findings were identified.

.2.14 Residual Heat Removal Service Water Booster Pump 1C

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with residual heat removal service water booster pump 1C. The team also performed walkdowns, and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically the team reviewed:

- Past maintenance records for the last three years.

- Surveillance test results, procedures and preventive maintenance work orders.

- Associated condition reports for the past three years.
- System design basis documents and system modifications.
- Preventive maintenance procedures and completed maintenance work orders.

b. Findings

No findings were identified.

.2.15 Residual Heat Removal Heat Exchanger B Bypass Valve

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with residual heat removal heat exchanger B bypass valve. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically the team reviewed:

- Vendor installation instructions.
- Past maintenance records for the last three years.
- Surveillance procedures and surveillance results.
- Leak rate testing for last three years.
- Associated condition reports for the past three years.
- Piping and instrumentation diagram for the residual heat removal heat exchanger B bypass valve.

b. Findings

No findings were identified.

.2.16 Residual Heat Removal Service Water Motor Operated Valve 89B

a. Inspection Scope

The team reviewed the updated safety analysis report, design bases documents, selected drawings, maintenance and test procedures, and condition reports associated with motor-operated valve 89B. The team also performed system walkdowns and conducted interviews with the system engineering personnel to ensure the capability of

this component to perform its desired design basis function. Specifically, the team reviewed:

- Technical specifications and basis documents.
- Motor sizing data.
- System design criteria and operating instructions.
- Corrective action program documents and system health reports for the last three years.
- Piping and instrumentation diagrams.
- Component surveillance test results and trend reports.
- Maintenance records and operational history.

b. Findings

No findings were identified.

.3 Results of Reviews for Operating Experience

.3.1 Inspection of NRC Information Notice 98-31, "Fire Protection System Design Deficiencies and Common-Mode Flooding of Emergency Core Cooling System Rooms at Washington Nuclear Project Unit 2"

a. Inspection Scope

The team reviewed the licensee's evaluation of Information Notice 98-31, "Fire Protection System Design Deficiencies and Common-Mode Flooding of Emergency Core Cooling System Rooms at Washington Nuclear Project Unit 2," to verify the licensee performed an applicability review and took corrective actions, if appropriate, to address the concerns described in the information notice. This information notice addressed the rupture of a fire protection system valve at WNP-2 station in 1998, resulting in the flooding of emergency core cooling system rooms in the reactor building.

b. Findings

Failure to Adequately Maintain Design Modifications to Prevent Fire Protection System Water Hammer

Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to adequately maintain design modifications that were implemented to prevent fire protection system water hammer events. Specifically, the licensee failed to maintain procedure changes to

periodically monitor and add nitrogen to fire protection system headers in the reactor building.

Description. The inspectors reviewed Cooper Nuclear Station's evaluation of NRC Information Notice 98-31, "Fire Protection System Design Deficiencies and Common-Mode Flooding of Emergency Core Cooling System Rooms at Washington Nuclear Project Unit 2." This information notice addressed the rupture of a fire protection system valve at WNP-2 station in 1998. At WNP-2, a water hammer event resulted in the failure of a fire protection system valve in the reactor building and in the flooding of two emergency core cooling system equipment rooms. In response to the information notice, Cooper Nuclear Station implemented design change CED 1998-0060 to introduce nitrogen bubbles into three of the five fire protection risers in the reactor building. The design change included revisions to station procedures to periodically monitor and add nitrogen to the fire protection system headers. This design change was based on calculation NEDC 00-097, which determined the forces associated with a water hammer event would be significantly reduced by the addition of nitrogen. The modification included procedure changes to periodically monitor and add nitrogen to the headers.

In response to the inspectors' questions, Cooper Nuclear Station personnel determined that the requirement to monitor and add nitrogen to these fire protection risers was no longer included in the plant procedures. They also determined that the technical basis for removing this requirement had not been documented and that no design change was initiated to implement this procedure change. Cooper Nuclear Station personnel initiated Condition Report CR-CNS-2015-02085 during the inspection to address the issue. The condition report recommended that the station either introduce nitrogen into the fire protection risers in accordance with the previous revision of the plant procedure or develop a technical basis for the removal of the nitrogen. The licensee determined that the fire protection system remained functional without the nitrogen based on technical input from Cooper Nuclear Station engineering personnel. This technical input was partially based on calculation NEDC 00-097, which determined the maximum peak riser pressure in the northwest corner of the reactor building and determined the peak force in the fire protection piping system. The technical input also included a discussion of a similar transient that occurred at Cooper Nuclear Station in 1995; this transient did not result in a catastrophic water hammer or piping system damage. The technical input determined that empirical evidence suggested that the system was capable of absorbing the shockwave from a water hammer event.

Analysis. The team determined that the failure to adequately maintain control of the fire protection system design to prevent water hammer events was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to events to prevent undesirable consequences. Specifically, the licensee failed to maintain procedure changes to periodically monitor and add nitrogen to fire protection system headers in the reactor building. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions,"

the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. The team determined that this finding did not have a cross-cutting aspect because the most significant contributor did not reflect current licensee performance.

Enforcement. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, that "design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Contrary to the above, prior to April 6, 2015, the licensee failed to verify or check the adequacy of the fire protection system to remain functional in the event of a water hammer event through calculational methods or through a suitable testing program. Specifically, the licensee failed to maintain procedure changes to periodically monitor and add nitrogen to fire protection system headers in the reactor building to mitigate the effects of water hammer. In response to this issue, the licensee determined that the fire protection system remained functional without nitrogen based on empirical evidence suggesting that the system was capable of absorbing the shockwave from a water hammer event. This finding was entered into the licensee's corrective action program as Condition Report CR-CNS-2015-02085. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000298/2015007-02, "Failure to Adequately Maintain Design Modifications to Prevent Fire Protection System Water Hammer."

.3.2 Inspection of NRC Information Notice 2013-017, "Significant Plant Transient Induced by Safety-Related Direct Current Bus Maintenance at Plant"

a. Inspection Scope

The team reviewed the licensee's evaluation of NRC Information Notice 2013-017, "Significant Plant Transient Induced by Safety-Related Direct Current Bus Maintenance at Plant," to verify the licensee performed an applicability review and took corrective actions, if appropriate, to address the concerns described in the information notice. This information notice discusses significant plant transients induced by safety-related direct current bus maintenance at the plant. The licensee documented the evaluation under LO 2013-0088-032. The team verified that the licensee's review adequately addressed the issues in the information notice.

b. Findings

No findings were identified.

.3.3 Part 21 No. 2015-0014: Defect Identified in ABB K-Line Breaker Secondary Close Latches (Part Number 716610K01)

a. Inspection Scope

The team reviewed the licensee’s evaluation of Part 21 No. 2015-0014: “Defect Identified in ABB K-Line Breaker Secondary Close Latches (Part Number 716610K01)” to verify the licensee performed applicability and vulnerability review of the defect identified in ABB K-Line Breaker Secondary Close Latches (Part Number 716610K01). The Operating Experience reviews were evaluated and documented by the licensee in LO 2013-0087-075, LO 2014-0082-009, and LO 2014-0082-048. It was documented that Cooper Nuclear Station does not use the ABB K-line circuit breakers in plant systems. The team verified that the licensee’s review adequately addressed the issues in the Part 21 report.

b. Findings

No findings were identified.

.3.4 Inspection of NRC Information Notice 2012-11, “Age-Related Capacitor Degradation”

a. Inspection Scope

The team reviewed the licensee’s evaluation of Information Notice 2012-11, “Age-Related Capacitor Degradation,” to verify the licensee performed an applicability review and took corrective actions, if appropriate, to address the concerns described in the information notice. This information notice discusses age-related degradation of capacitors that results from epoxy insulation hardening and cracking over time that allows for a high flow of current and excessive heating. The team verified that the licensee’s review adequately addressed the issues in the information notice.

b. Findings

No findings were identified.

.4 Results of Reviews for Operator Actions

a. Inspection Scope

The team selected risk-significant components and operator actions for review using information contained in the licensee’s probabilistic risk assessment. This included components and operator actions that had a risk achievement worth factor greater than two or Birnbaum value greater than 1E-6.

For the review of operator actions, the team observed operators during simulator scenarios associated with the selected components as well as observing simulated actions in the plant.

The selected operator actions were:

- Scenario 1, Part 1: Inadvertent Main Steam Isolation Valve closure caused a reactor scram. A hydraulic Anticipated Transient Without Scram prevented the reactor from being shut down. The operating crews were expected to enter Emergency Operating Procedures 6A and 7A to control reactor pressure vessel pressure and shut down the reactor using boron injection from the Standby Liquid Control system. The operating crews were also expected to insert control rods into the reactor per Procedure 5.8.3, "Alternate Rod Insertion Methods." This portion of the scenario specifically evaluated the licensee's ability to successfully initiate boron injection within 2 minutes as referenced in Updated Safety Analysis Report XIV, Section 5.9.3.4.4.1 and Procedure 2.0.1.3, "Time Critical Operator Action Control and Maintenance."

- Scenario 1, Part 2: As a result of the Anticipated Transient Without Scram conditions described above, the operators were expected to take actions per Emergency Operating Procedure 3A to mitigate adverse primary containment parameters. Specifically, this portion of the scenario evaluated the licensee's ability to successfully place both loops of the residual heat removal system in suppression pool cooling mode of operation within 30 minutes as referenced in Procedure 2.0.1.3, "Time Critical Operator Action Control and Maintenance."

- In-plant job performance measure (JPM) #1: This job performance measure was designed to evaluate the licensee's ability to perform subsequent actions for ensuring emergency ventilation to essential equipment during a station blackout event. Specifically, the job performance measure evaluated the ability of an operator to open control room panel doors and 125V/250V dc switchgear room doors in accordance with Procedure 5.3SBO, "Station Blackout" within 30 minutes as specified by 5.3SBO and Procedure 2.0.1.3, "Time Critical Operator Action Control and Maintenance."

- In-plant job performance measure (JPM) #2: This job performance measure was designed to evaluate the licensee's ability to perform actions to inject boron into the reactor pressure vessel using the reactor core isolation cooling system in accordance with Procedure 5.8.8, "Alternate Boron Injection and Preparation."

b. Findings

Failure to Update the Final Safety Analysis Report

Introduction. The team identified three examples of a Severity Level IV, non-cited violation of 10 CFR 50.71, "Maintenance of Records, Making of Reports," for the licensee's failure to update the Final Safety Analysis Report originally submitted as part of the application for the license to assure that the information included in the report contains the latest information developed. Specifically, in April 2008, January 2012, and February 2015 the licensee made changes to their operating procedures, Anticipated

Transient Without Scram analyses, and plant conduct of operations procedures that were not subsequently reflected in their Updated Safety Analysis Report.

Description. The first example was identified on April 9, 2015, during the team's review of the licensee's time critical operator actions associated with Anticipated Transient Without Scram events. In August 2014 the licensee identified that contrary to Cooper Nuclear Station Procedure 2.0.1.3, "Time Critical Operator Action Control and Maintenance," the control room operators could not successfully place the residual heat removal system in suppression pool cooling mode of operation during a 100 percent Anticipated Transient Without Scram event with main steam isolation valves closed within the required 11 minutes (per Cooper Nuclear Station Updated Safety Analysis Report Chapter XIV, Section 5.9.3.3.2.c). The licensee generated Condition Report CR-CNS-2014-4453 requesting an engineering review of the time critical operator action. As a result of the engineering review (ER 15-003, Revision 0, dated February 4, 2015), the time critical operator action time for placing residual heat removal in suppression pool cooling mode was changed to 30 minutes. The licensee did not subsequently change Updated Safety Analysis Report, Chapter XIV, Section 5.9.3.3.2.c, to reflect the new time critical operator action time to reflect the updated Anticipated Transient Without Scram analysis that resulted from the engineering review. The inspection team identified that the most recent Updated Safety Analysis Report was submitted for approval on April 24, 2015, and covered changes to the Updated Safety Analysis Report through the 24-month period ending March 10, 2015. The licensee documented this issue in the corrective action program as Condition Report CR-CNS-2015-02090.

The second example was identified on April 22, 2015, during the team's review of the licensee's time critical operator actions associated with Anticipated Transient Without Scram events. In January 2012 the NRC issued Amendment No. 240 to Renewed Facility Operating License No. DPR-46 for Cooper Nuclear Station, concerning changes to the Technical Specification 3.4.3, "Safety/Relief Valves and Safety Valves." As part of the analyses submitted with the proposed amendment, Cooper Nuclear Station used bounding Anticipated Transient Without Scram analysis values for safety/relief valve lift setpoints of +3 percent of nominal lifting setpoint pressure. Cooper Nuclear Station Updated Safety Analysis Report, Chapter XIV, Table XIV-5-4, was not updated to reflect the +3 percent safety/relief valve lift setpoint pressure (the table displays nominal lift setpoint pressures). The licensee documented this issue in the corrective action program as Condition Report CR-CNS-2015-02393.

The third example was identified on April 10, 2015, during the team's review of the internal flooding analysis and related Updated Safety Analysis Report section. The Updated Safety Analysis Report, Section X-8.2.8.1, Flooding, stated that "Two 3" service water system lines provide an emergency cooling water source for the control room air conditioner. There is normally no flow in these lines since they have normally closed manually operated valves in the lines below the 903'6" passageway elevation. Therefore, these lines pose no problems." However, a corrective action associated with Condition Report CR-CNS-2007-07623 changed normal valve positions in Procedure 2.2.76A. As a result, the service water lines were pressurized and were a potential source of flooding. This change did not invalidate the results of the flooding

analysis, but the statement in the Updated Safety Analysis Report was not correct. The licensee documented this issue in the corrective action program as Condition Report CR-CNS-2015-02106.

Analysis. The team determined that the failure to update the Final Safety Analysis Report to assure that the information included in the report contains the latest information developed was a performance deficiency. This finding was evaluated using traditional enforcement because it had the potential for impacting the NRC's ability to perform its regulatory function. This finding was more than minor because each example potentially rendered portions of the safety analyses for Anticipated Transient Without Scram events described in the Updated Safety Analysis Report less conservative or contradicted previous information regarding the licensee's flooding analysis contained in the Updated Safety Analysis Report. The traditional enforcement violation was determined to be a Severity Level IV violation consistent with the example in paragraph 6.1.d(3) of the NRC Enforcement Policy. Since this was a traditional enforcement violation, no cross-cutting aspects were assigned per the guidance contained in Inspection Manual Chapter 0612, Section 07.03(c).

Enforcement. The team identified three examples of a Severity Level IV, Green, non-cited violation, of 10 CFR 50.71, "Maintenance of Records, Making of Reports," Section (e), which states, in part, "each person licensed to operate a nuclear power reactor under the provisions of 10 CFR 50.21 or 10 CFR 50.22 shall update periodically the final safety analysis report (FSAR) originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed. This submittal shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the licensee since the submittal of the original FSAR, or as appropriate, the last update to the FSAR under this section." Contrary to the above, in January 2012 and February 2015, the licensee failed to update periodically the Final Safety Analysis Report (FSAR) to contain all the changes necessary to reflect information and analysis since the last update to the Final Safety Analysis Report. Specifically, the licensee failed to update the Updated Safety Analysis Report for changes made to their Anticipated Transient Without Scram analyses and plant conduct of operations procedures. This finding was entered into the licensee's corrective action program as Condition Reports CR-CNS-2015-02106, CR-CNS-2015-02090, and CR-CNS-2015-02393. Because this finding was not repetitive or willful and has been entered into the licensee's corrective action program, this Severity Level IV violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000298/2015007-03, "Failure to Update the Final Safety Analysis Report (FSAR)."

4. OTHER ACTIVITIES

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

4OA2 Problem Identification and Resolution (71152)

Component Design Basis Review

a. Inspection Scope

The team reviewed condition reports associated with the selected components, operator actions, and operating experience notifications. The team also reviewed corrective actions associated with items identified in previous inspections. Specifically, the team reviewed the updated safety analysis report, system description, design basis documents, selected drawings, maintenance and test procedures, and condition reports associated with the emergency diesel generator day and storage tank vents. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform the desired design function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.
- Design basis adverse weather protection requirements.
- Normal and alternate diesel fuel oil fill procedures.
- Detailed plant drawings and operating, preventive maintenance, and testing procedures.

b. Findings

Failure to Evaluate the Lack of Missile Protection on the Emergency Diesel Generator 1 and 2, Fuel Oil Storage Tank Vents

Introduction. The team identified a Green, cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the licensee’s failure to evaluate the lack of missile protection on the emergency diesel generator 1 and 2 fuel storage tank vents.

Description. The Cooper Nuclear Station Safety Evaluation Report (SER) and Updated Safety Analysis Report states the following with regard to General Design Criteria and the emergency diesel generators:

- Updated Safety Analysis Report, Appendix F, states that the licensee complies with Draft General Design Criteria GDC-2, published July 11, 1967, and the Draft General Design Criteria GDC-2 requires that the systems and components

needed for accident mitigation remain fully functional before, during, and after a tornado event.

- Updated Safety Analysis Report, Chapter I-5, Section 5.2, defines Class I structures and equipment as, "Structures and equipment whose failure could cause significant release of radioactivity or which are vital to a safe shutdown of the plant and removal of decay and sensible heat."

- Safety Evaluation Report, Section 3.5, states that Class I structures were designed to withstand the effects of a spectrum of tornado generated missiles of low level origin, including a 35 foot long utility pole with a 14 inch butt, with an impact velocity of 200 miles per hour.

- Updated Safety Analysis Report, Chapter XII-2, Section 2.1.2.3, specifically identifies the Standby Diesel Generator System and Auxiliaries as Class I equipment.

- Updated Safety Analysis Report, Chapter XII, Section 2.3.3.2.1, states that Class I structures are designed to provide protection against tornado generated missiles.

On December 3, 2010, NRC Component Design Basis Inspection (CDBI) Report 05000298/2010007 (ML103370640), documented Non-cited Violation 05000298/2010007-04, "Inadequate Design Control," for the licensee's failure to establish design control measures, involving the performance of a design review, or the use of alternate or simplified calculational methods, or the performance of a suitable testing program to verify that the emergency diesel generator fuel oil storage and day tank vent lines were adequately protected from tornado generated missiles. The licensee entered this deficiency into their corrective action program as Condition Report CR-CNS-2010-05211 and generated Engineering Evaluation (EE) 10-060, "Evaluation of the Diesel Generator Fuel Oil Tanks."

Subsequently, on February 13, 2014, NRC Inspection Report 05000298/2013005 (ML14044A105) documented Non-cited Violation 05000298/2013005-01, "Failure to Promptly Identify and Correct a Condition Adverse to Quality," for the licensee's failure to promptly identify and correct Non-cited violation 05000298/2010007-04, "Inadequate Design Control." Specifically, inspectors determined that Engineering Evaluation EE 10-060 did not evaluate the vent lines with regard to their ability to withstand tornado generated missiles. Instead, it assumed that if impacted by a missile, there would be no damage to the fuel oil storage tank and discussed manual actions that could be implemented if the vent lines were to be damaged by a tornado generated missile. The licensee entered this deficiency into their corrective action program as Condition Report CR-CNS-2014-00146.

In response to Non-cited Violation 05000298/201305-01, the licensee provided a reply contained in a letter from Mr. O. Limpias to the NRC, dated May 20, 2014, which disputed the use of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," as

the basis for the non-cited violation. As stated in this letter, the licensee denied that a violation of NRC requirements had occurred, in that, the licensee had previously evaluated this condition as documented in Condition Report CR-CNS-2010-05211, which was initiated in response to a nonconforming condition identified during the 2010 Component Design Basis Inspection. The licensee also indicated that they had re-evaluated these results and concluded the original evaluation remained valid. Specifically, Engineering Evaluation EE 10-060, "Evaluation of the Diesel Generator Fuel Oil Tank Vents After a Tornado Strike," Revision 0, was performed to establish the basis for conformance with the pre-General Design Criteria 2, contained in Appendix F, of the Updated Safety Analysis Report.

In conclusion, the licensee's violation denial letter stated that the previous NRC Component Design Basis question related to the diesel generator fuel oil storage tank vent's ability to withstand a tornado missile strike was adequately resolved under Condition Report CR-CNS-2010-05211 and appropriately evaluated in a timely manner commensurate with 10 CFR 50, Appendix B, Criterion XVI.

<u>Violation Denial Review</u>

Consistent with the guidance provided in Policy Guide 0560-3, "Region IV Enforcement Procedures," the NRC staff performed an independent review of the documentation associated with this finding. Based on the results of this review it was determined that the requirements of the draft General Design Criteria, Criterion 2, clearly establish the design function of systems and components of reactor facilities which are essential to the prevention of accidents which could affect public health and safety or mitigation of their consequences. These systems and components are required to be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as tornados. Furthermore, the system design basis requirements contained in the Cooper Nuclear Station Updated Safety Analysis Report, Chapter XII, Section 2.3.3.2.2, "Tornado Generated Missiles," specifies that all Class I structures are designed to provide protection against tornado generated missiles including:

- A 35-foot long utility pole with a 14-inch butt with an impact velocity of 200 miles per hour.

- A one-ton missile such as a compact-type automobile with an impact velocity of 100 miles per hour and a contact area of 25 square feet.

- A two-inch extra heavy pipe, 12 feet long.

- Any other missile resulting from failure of a structure or component or one which has potential of being lifted from storage or working areas at the site.

Additionally, the Cooper Nuclear Station Design Basis for the Diesel Generator Fuel Oil system includes the following requirements:

- The standby diesel generator system must be capable of withstanding the most severe conditions anticipated at the location of the plant. The design basis events are described in IEEE-308-1970, Table I. This table includes postulated earthquake, wind, hurricane, and tornado effects as natural phenomena design basis. Additionally, Table I of IEEE-308-1973, lists accident-generated missiles as one of the events that the emergency diesel system must be designed to withstand.

- The fuel oil subsystem must provide sufficient fuel to operate the standby diesel generator under all postulated conditions.

- The safety classification of the essential emergency diesel system including the diesel fuel oil tank vents is Seismic Class I.

The NRC concluded that the diesel generator fuel oil storage tank Seismic Class I vents were not assured to be designed, fabricated, and erected to withstand the additional forces imposed by natural phenomena such as tornados, as required by the licensing basis stated above. Specifically, the licensee’s evaluation performed in accordance with Condition Report CR-CNS-2010-05211 and the associated Engineering Evaluation EE-10-060 did not adequately demonstrate that the diesel generator fuel oil storage tank vent lines would maintain its ability to withstand a postulated tornado missile impact without loss of function. Although the evaluation references the location of the vents, the area of exposure of the vents to missile impact, and generally discusses the material composition of the vents and the inferred minimal load transferred to the diesel generator fuel oil storage tanks, no definitive analytical basis was identified for concluding that the vent lines would not be damaged by the postulated tornado generated missile and they would remain functional. While the licensee’s compensatory actions dealt with the initial operability condition, the requisite corrective and preventive measures failed to address the nonconforming design condition, concerning the diesel generator fuel oil storage tank vents’ tornado missile protection, initially identified as a performance deficiency in NRC Component Design Basis Inspection Report 05000298/2010007.

Based on these reviews, it was concluded that the finding and non-cited violation for failing to assure that an identified condition adverse to quality was promptly corrected to meet the requirements in 10 CFR Part 50, Appendix B, Criterion XVI, as documented in NRC Inspection Report 05000298/2013005, were valid. The failure to perform a proper engineering evaluation of the diesel generator fuel oil storage tank vents to demonstrate the ability to perform its specified safety function as required by the licensing bases in the event of a tornado generated missile has not been documented.

<u>Current Evaluation Results</u>

During the performance of the 2015 Component Design Basis Inspection, the team met with the licensee’s engineering and licensing staff to establish the current status of the diesel generator fuel oil storage tank vents. As a result of these discussions, it was determined that calculation NEDC 13-046, “Diesel Generator Storage Tank Vent Line

Tornado Missile Durability," had been developed to demonstrate that the current design was acceptable without further action. Specifically, this calculation evaluated the ability of the diesel generator fuel oil storage tank vents to remain operable following an impact from design basis tornado generated missiles. As described in NEDC 13-046, the diesel generator fuel oil storage tank vent lines are part of the fuel oil subsystem of the emergency diesel generator system. Each component of this system is classified as Seismic Class 1S. As described in the Updated Safety Analysis Report, "Class I structures are designed to provide protection against tornado generated missiles." The calculation determines the maximum deflection the vent line can experience without permanent deformation, the amount of deflection the vent line can experience without fracture and the amount of force that would be required to fracture the vent line. Forces from the three Updated Safety Analysis Report specified tornado generated missiles were determined for comparison to the maximum allowable force the vent line can withstand. Specifically, using the worst case tornado generated missile, the calculation concluded that "Due to the overwhelming magnitude of the force and the very short duration of the impact, the vent pipes will shear off or fracture rather than bend and crimp."

Based on the review of NEDC 13-046, the team determined that the calculation did not provide an adequate design analysis that would assure that the diesel generator fuel oil vent lines could maintain an open vent path during a postulated tornado event under all missile scenarios. Specifically, the calculation failed to provide a bounding analysis that demonstrated the vent lines would not crimp subsequent to a tornado generated missile strike from a range of objects "… which has potential of being lifted from storage or working areas at the site."

Analysis. The team determined that the failure to evaluate the lack of missile protection on the emergency diesel generator 1 and 2 fuel storage tank vents was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to events to prevent undesirable consequences. Specifically, the licensee failed to evaluate a design nonconformance on the emergency diesel generator 1 and 2 fuel storage tanks for lack of missile protection. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," this finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance associated with conservative bias because individuals failed to use decision making practices that emphasize prudent choices over those that are simply allowable [H.14].

Enforcement. The team identified a Green, cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "Design control measures shall provide for verifying or checking the adequacy of design, such as by the

performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Contrary to the above, since July 2010, the licensee failed to verify the adequacy of the applicable design control measures. Specifically, the licensee failed to verify the adequacy of design of the vents for the emergency diesel generator 1 and 2 fuel oil storage tanks to withstand impact from a tornado driven missile hazard, or to evaluate for exemption from missile protection requirements using an approved methodology. This finding was entered into the licensee's corrective action program as Condition Report CR-CNS-2015-02366. VIO 05000298/2015007-04, "Failure to Evaluate the Lack of Missile Protection on the Emergency Diesel Generator 1 and 2 Fuel Oil Storage Tank Vents."

4OA6 Meetings, Including Exit

Exit Meeting Summary

On May 8, 2015, the inspectors presented the inspection results to Mr. O. Limpias, Vice President-Nuclear and Chief Nuclear Officer, 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

A. Able, Instrumentation and Controls Supervisor, Design Engineering
T. Barker, Manager, Engineering Programs and Components
M. Bergmeier, Operations
D. Buman, Director, Engineering
T. Chard, Manager, Quality Assurance
L. Dewhirst, Manager, Corrective Actions and Assessment
K. Dia, Manager, Systems Engineering
M. Dickerson, Electrical Engineer, Design Engineering
L. DuBois, Emergency Preparedness
J. Ehlers, Electrical/Instrumentation and Controls Supervisor, System Engineering
R. Estrada, Manager, Design Engineering
J. Flaherty, Senior Staff Engineer, Licensing
G. Gardener, Supervisor, NSSS
D. Goochman, Manager, Operations
K. Higginbotham, General Manager, Plant Operations
D. Kimball, Director, Nuclear Oversight
O. Limpias, Senior Vice President, Nuclear, and Chief Nuclear Officer
E. Nelson, Supervisor, Emergency Preparedness
T. Ocken, Supervisor, Design Engineering
C. Pelchat, Manager, Projects
R. Penfield, Director, Nuclear Safety Assurance
J. Shaw, Manager, Licensing
J. Stough, Manager, Information Technology
C. Sunderman, Manager, Radiation Protection
K. Tom, Assistant to the Director, Engineering
A. Walters, Manager, Chemistry

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000298/2015007-04	VIO	Failure to Evaluate the Lack of Missile Protection on the Emergency Diesel Generator 1 and 2 Fuel Oil Storage Tank Vents (Section 1R21.4OA2)

Opened and Closed

05000298/2015007-01	NCV	Failure to Perform an Operability Review of a Condition Report (Section 1R21.2.7)
05000298/2015007-02	NCV	Failure to Adequately Maintain Design Modifications to Prevent Fire Protection System Water Hammer (Section 1R21.3.1)
05000298/2015007-03	NCV	Failure to Update the Final Safety Analysis Report (FSAR) (Section 1R21.4)

LIST OF DOCUMENTS REVIEWED

Calculations

Number	Title	Revision
NEDC 84-050GH	Cable Tray Fill Calculation, Tray C35A, Section 134	1
NEDC 84-050GJ	Cable Tray Fill Calculation Tray C35A, Section 136	1
NEDC 87-047K	Motor Control Center “K” Load Summary	3
NEDC-86-105B	4160 Volt Switchgear Critical Bus 1F	10
B&R Calculation 2.05.01	Cable Sizing Main Motors & Feeders Cable	1
B&R Calculation 2.05.02	Cable Sizing summary – 4160 V Motors & Feeders	1
NEDC 91-190	Cable Withstand Evaluation	3
NEDC 00-003	Cooper Nuclear Station Auxiliary Power System Load Flow and Voltage Analysis	8
NEDC-08-044	Assessment of SSST with Loss of Fan Cooling for PRA	0
NEDC 87-153	SLC Capacity	0
NEDC 87-158	NPSH Calculation for SLC Pumps	0
NEDC 87-167	SLC Operating Pressure with Two Pumps	2
NEDC 88-190	Essential Pump Minimum Flow	0
NEDC 88-286	MSIV Accumulator Capacity Evaluation	0
NEDC 89-1886	CNS Station Blackout (SBO) Condensate Inventory	3
NEDC 91-185	MOV Thermal Overload Heater Sizing	6
NEDC 92-015	SLC Pump NPSH	1
NEDC 93-125	Stem Nut Wear Evaluation	0
NEDC 94-142	Core Spray Flows with Minimum Flow Bypass (MFB) Valve Open	5
NEDC 94-190	Core Spray Pump Miniflow Analysis	0
NEDC 95-003	Determination of Allowable Operating Parameters for CNS MOV Program MOVs	29
NEDC 96-030	SLC Vortex Limit	0
NEDC 97-044A	NPSH Margins for the RHR and CS Pumps	4
NEDC 97-071	Design Report RAL-1030 Spherical Disk Piston-Flite flow Main Steam Isolation Valve	0

Calculations

Number	Title	Revision
NEDC 00-041	Limiting Component Analysis for Containment Spray Motor-Operated Valves	2
NEDC 00-042	AC MOV Program MOVs	2
NEDC 00-047	Determination of Allowable Operating Parameters for RHR-MOV-MO38A and RHR-MOV-MO38B	0
NEDC 00-080	Flood Door Gap Analysis	1
NEDC 00-110	MOV Program Valve Margin Determination	8
NEDC 01-064	8-hour ECST Volume Requirements for an Isolated Reactor	2
NEDC 01-072	ECCS Pump NPSH/ Vortex Limit with Suction from CST	1
NEDC 09-102	Internal Flooding – HELB, MELB, and Feedwater Line Break	1
NEDC 10-053	EDG High/Low Frequency Effect on ECCS Pumps	2
NEDC 87-131C	125 VDC Division 1 Load and Voltage Study	15
NEDC 86-105C	CNS DC Short Circuit Study	4
NEDC 89-2163	Control Building HVAC	5
NEDC 86-105D	CNS Critical DC Bus Coordination Study	8
NEDC 87-131C	Service Test Profile	15
NEDC 91-094	125 VDC/250 VDC Battery Charger Analysis	5
NEDC 91-185	MOV Thermal Overload Heater Sizing	6
NEDC 99-083	Proto Power Calculation 99-056 Service Water Booster Pump System Hydraulic Analysis	1
NEDC 99-056	Evaluation of Residual Heat Removal Service Water Booster Pump Needed Differential Pressure	0
NEDC 97-044A	Net Positive Suction Head Margins Residual Heat Removal & Core Spray Pumps	4
NEDC 94-190	Core Spray Pump Miniflow Analysis	0
NEDC 94-142	Core Spray Flows with Min Flow Bypass Valve Open	5
NEDC 93-184	Residual Heat Removal Heat Exchangers Thermal Performance &Tube Plug Margin	3
NEDC 93-125	Stem Nut Thread Wear Evaluation, Generic Letter 89-10 MOVs	0
NEDC 00-47	Allowable Operating Parameters RHR-MOV38A & 38B	0

Condition Reports (CRs)

2009-03704	2015-02061	2010-03876	2015-01268	2011-06146
2013-0088-032	2013-08099	2014-08465	2014-01680	2014-08318
2012-09382	2012-06657	2012-01997	2015-02024	2012-01647
2014-05947	2008-00666	2007-00773	2010-06302	2012-04960
2014-06054	2010-05211	2014-00146		

Condition Reports (CRs) Generated during the Inspection

2015-02366	2015-02007	2015-02407	2015-02337	2015-02384
2015-02747	2015-02736	2015-02752	2015-02330	2015-02034
2015-02441	2015-02085	2015-02650	2015-02089	2015-02106
2015-02115	2015-02358	2015-02366	2015-02384	2015-02395
2015-02408	2015-02409	2015-02440	2015-02090	2015-02393
2015-02085				

Work Orders

4458028	4699196	4698778	4442920	4699195
4748604	4983674	5072695	11117931	MWR 99-3212
4951675	4999222	5023453	5023550	5025370
5023592	4951419	4801811	4740703	

Drawings

Number	Title	Revision/Date
3001	Main One Line Diagram	AC/24
3002, Sh. 1	Auxiliary One line Diagram MCC Z, SWGR Bus 1A, 1B, 1E, and Critical Bus 1F 1G	AC/51
3006, Sh. 5	Auxiliary One line Diagram Starter Racks LZ and TZ, MCC's K, L, LX, RA, S, T, TX, X	AE/83
3127, Sh. 6	Turbine Generator Building Cable tray loading schedule	N11
2018	Flow Diagram Turbine Generator Bldg. & Control Bldg. Heating and Ventilating Cooper Nuclear Station	AD/40
932-71212PI, Sh. 1C	Control Building H & V Unit 1-HV-C-1A	N06
3752, Sh. 1	Annunciator Loop Diagram ANN-MUX-02	N04
3750, Sh. 1	Annunciator Loop Diagram ANN-MUX-00	N05

Drawings

Number	Title	Revision/Date
3157	Reactor Building Elev. 931'-6" conduit and Tray plan	AG/27
3013	Generator Tripping Schedule	9
3012, Sh. 2	Main three Line Diagram	N09
3253, Sh. K1	460V Motor Control Center K Connection Wiring Diagram	N19
3253, Sh. DT4	460V Motor Control Centers Wiring Details, Connection Wiring Diagram	N20
E506	Turbine Generator Building Connection Wiring Diagram, Sheet 64	N04
3255, Sh. 38	Control Room-Control Panels Connection Wiring Diagram	N11
3156, Sh. 1	Reactor Building, El 903'-6" Conduit and Tray Plan	AC/35
E501, Sh. 48A	Integrated Control Circuit Diagram, SW-MOV-M089B, RHR HX B Service Water outlet	AC/04
3007, Sh. 6	Auxiliary One Line Diagram, Motor Control Centers, E, O, R, RB & Y	N83
INV-3C-70048, Sh. 2 of 2	Schematic Diagram ARR 130K200F	N02
3058	DC One Line Diagram	64
152D009	250V & 125V DC Switchgear One Line & Schematic	N03
48K7A.STK	MCC-K Circuit 7A	March 18, 1994
2040, Sh. 2	Flow Diagram Residual Heat Removal System Loop B	AB/19
791E264, Sh. 3	Elementary Diagram RCIC System Cooper	N21
791E264, Sh. 7	Elementary Diagram RCIC System Cooper	N15
791E271, Sh. 3	Elementary Diagram HPCI System Cooper	N23
791E271, Sh. 8	Elementary Diagram HPCI System Cooper	N20
2049, Sh. 2	Flow Diagram Condensate Supply System	AC/38

Procedures

Number	Title	Revision/Date
7.3.17	4160 Breaker Maintenance	36
7.3.17-1	4160 Breaker Examination	29
7.3.17.4	4160V Vacuum Bottle Breaker Maintenance	1
3.11	Vendor Manual Control and Use	26

Procedures

Number	Title	Revision/Date
0-EN-OE-100	Operating Experience Program	16CS
14.35.1	Electrical Equipment Instrument Calibration	10
7.3.41	Examination and High-pot testing of Non-Segregated Buses and Associate Equipment	10
2.2.38	HVAC Control Building	40
7.3.2.1	Westinghouse DB-50 Breaker Maintenance and Testing	19
7.3.13	Motor Control Center Examination and Maintenance	22
2.1.11.1	Turbine Building Data	148
0.5.OPS	Operations Review of Condition Reports/Operability Determination	53
6.SWBP.201	Surveillance Procedure SW-MO-89A/B Full Stroke Operability (IST)	6
3-EN-DC-126	Engineering Calculation Process	3C2
2.3-C-2	Operator Observation and Action Startup Transformer Trouble Panel Window C-2/F-9	45
14.11.16	IAC Procedure Yokogawa Recorder DX Series Calibration Check	73
MNT118-00-00	Lesson on Calibration Tool Issues	07
2.2.15	Startup Transformer	54
6.EE.610	Off-Site AC Power Alignment	37
0-CNS-LI-102	Corrective Action Process	0
7.0.2	Preventive Maintenance Program Implementation	53
7.3.2	DC DB-25 and DB-50 Fused Disconnect Testing and Maintenance	22
7.3.39	Inspection of 125/250 VDC Buses and Switchgear A and B	4
7.3.27.1	125V Station Battery Equalizing Charge	13
7.3.31.3	125V/250V Battery Terminal Cleaning and Torqueing	14
2.2.25.1	125 VDC Electrical System (Div. 1)	19
7.3.1.6	125/250 VDC Station Battery Charger Protective Relays Testing and Calibration	18
7.3.14	Thermal Examination of Plant Components	10
7.3.23.6	Battery Charger Clean and Inspect	1
2.0.1.3	Time Critical Operator Action Validation	4

Procedures

Number	Title	Revision/Date
2.1.20.3	RPV Refueling Preparation (Wet Lift of Dryer and Separator)	52
2.2.99	Supplemental Diesel Generator System	5
3-EN-DC-304	MOV Thrust/Torque Setpoint Calculations	1C0
5.8.6	RPV Flooding Systems (Table 6)	32
5.8.7	Primary Containment Flooding/Spray Systems	30
6.1CS.101	Core Spray Test Mode Surveillance Operation (IST) (DIV 1)	28
6.2CS.101	Core Spray Test Mode Surveillance Operation (IST) (DIV 2)	25
6.MS.201	Main Steam Isolation Valve Operability Test (IST)	17
6. PC.513	Main Steam Local Leak Rate Tests	24
6.SDG.101	SDG Test Mode Surveillance Operation	4
6.SLC.101	SLC Pump Operability Test	23
6.SLC.102	SLC Test Mode Surveillance Operation (IST)	27
7.2.24.2	MSIV Speed Adjustment	2
89-176	MSIV Closing Test with Instrument Air Valved Out	May 25, 1989
O-BARRIER	Barrier Control Process	16
O-BARRIER-CONTROL	Control Building	5
5.3SBO	Station Blackout	34
5.8	Emergency Operating Procedures (EOPs)	36
5.8, Attachment 1	1A – RPV Control	16
5.8, Attachment 1	6A – RPV Pressure (Failure to Scram) / Reactor Power (Failure to Scram)	16
5.8, Attachment 1	7A – RPV Level (Failure to Scram)	17
5.8, Attachment 1	3A – Primary Containment Control	15
5.8, Attachment 2	EOP / SAG Graphs	15
5.8.8	Alternate Boron Injection and Preparation	16
2.1.5	Reactor Scram	71
5.8, Attachment 4	Stop and Prevent Hard Card	36

Procedures

Number	Title	Revision/Date
5.8, Attachment 6	Failure to Scram Actions Hard Card	36
2.1.22	Recovering From A Group Isolation	59
0.29.1	License Basis Document Changes	34
0.29.2	USAR Control and Maintenance	19
0-EN-LI-103	Operating License Amendments	7C0
2.0.1.3, Attachment 1	Time Critical Operator Action Validation	May 1, 2015
2.0.1.3, Attachment 1	Time Critical Operator Action Validation	July 21, 2014

Vendor Documents

Number	Title	Revision/Date
VM-1188	Vendor Manual 125 & 250 Volt Batteries and Chargers	12
0109D4798	Bus Duct Arrangement	01
022-3-R-0558, Sh. 22	Power and Control Circuits line-up 04 Unit-1	2
IC1000-K240-A164-X-4AUS	Siemens System and Service Instruction Manual – Vacuum Circuit Breaker (vehicle) Type GEH 4.16KV-250MVA, 4.16KV-250MVA upgraded to 350MVA	
E50001-F710-A251-V1-4A00	Siemens Instruction Manual Type 3AH3 and 3AHc-Vacuum Circuit Breaker operator modules 4.16KV to 38KV	
GEK-41905	GE Instruction and Recommended Parts for Maintenance Magne-Blast Circuit Breaker Type AMH-4.76-250-2D 1200 & 2000 Amperes with ML–13 Mechanism	
GEK-88771-D	GE Instruction Magne-Blast Circuit Breaker Type AMH-4.76-250-0D AMH-4.76-250-1D	
F-1329-D-0460	Connection Diagram, LT IB/24/30MVA OA/FA/FA	N01
791E262, Sh. 1	Standby Liquid Control, System	N17
791E252, Sh. 1	Nuclear Boiler Process Inst.	N12
TR-109641	Guidance on Routine Preventive Maintenance for Magne-Blast Circuit Breaker, Supplement to N	October 1998
VM-0986	Limitorque Valves Composite manual	33
CD 7.4.1.7-7	ABB High Accuracy Voltage Relays ITE-27N Undervoltage Relay; ITE-59N Overvoltage Relay	A

Vendor Documents

Number	Title	Revision/Date
IB 7.4.1.7-7	ABB Single Phase Voltage Relays Type 27N High Accuracy Undervoltage Relay; 59N High Accuracy Overvoltage Relay	D
IB 7.4.1.7-7	ABB Single Phase Voltage Relays Type 27N High Accuracy Undervoltage Relay; 59N High Accuracy Overvoltage Relay	E
VMCF 9 3-228	FPE Transformer Installation, Operations and Maintenance Instructions IN-T-415	

Design Basis Documents

Number	Title	Date
DCD-05	DC Electrical System Design Criteria Document EEDC1	February 2, 2009
DCD-04	AC Electrical System Design Criteria Document EEAC1	October 30, 2014
DCD-12	Core Spray System - Design Criteria Document	October 30, 2014
DCD-19	Standby Liquid Control (SLC) System - Design Criteria Document	January 22, 2010

Miscellaneous

Number	Title	Revision/Date
13-004	Engineering Evaluation - Electrical Bus Outage Maintenance Plans – 24 Month Refueling Cycle Review	1
EC-4899459	1200 A, 4160V Vacuum Bottle Circuit Breaker Replacement	0
10776733	Notification - Evaluate for Preventive maintenance	December 8, 2010
2LE6SJ	Project - Hitachi Overhaul AMH 4,76-250 1200 Amp Circuit Breaker S/N:0224A6208-001	0
061-15288	NLI Overhaul AMH-4.76-250-1D 1200Amp Circuit Breaker S/N:224A6204-008	August 16, 2011
LO 2014-0130	CDBI Focused Self-Assessment Report	January 16, 2015
800000042081	Operation 0050, Transformer General Maintenance	
IEEE Std 279-1971	IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations	
CED 1998-0060	Nitrogen Cushion Installation into Fire Protection System High Points	1

Miscellaneous

Number	Title	Revision/Date
CED 6029940	Supplemental Diesel Generator	May 25, 2010
	Core Spray System Health Report	January, 2015
	Main Steam System Health Report	January, 2015
	Standby Liquid Control System Health Report	January, 2015
EE 01-030	Flood Door Gap Analysis	0
ER 2015-011	Sensitivity Analysis of Diesel Generator Storage Tank Vent Function Following a Tornado Missile Strike	0
SIL No. 482	MSIV Closure Testing Requirement	February 22, 1989
NUMARC 87-00	Guidelines and Technical Bases for NUMARC Initiatives	1
OPL-3A	Input Parameters Verification For ATWS Analyses (Cycle 27)	0
TAC NO. ME5287	Cooper Nuclear Station – Issuance of Amendment Re: Technical Specification 3.4.3 To Reduce The Number of Safety Relief Valves Required To Be Operable For Overpressure Protection	January 31, 2012
SW06	Simulator Cause and Effect Malfunction - Service Water Leakage in Control Building Basement	01.00