UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 EAST LAMAR BOULEVARD
ARLINGTON, TEXAS 76011-4511

June 13, 2018

Mr. John Dent, Jr.
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 DESIGN BASES ASSURANCE (TEAMS) INSPECTION REPORT 05000298/2018011

Dear Mr. Dent, Jr.:

On April 19, 2018, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Cooper Nuclear Station. On April 30, 2018, the NRC inspectors discussed the results of this inspection with Mr. Dan Buman, Director of Nuclear Safety Assurance and other members of your staff. The results of this inspection are documented in the enclosed report.

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

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

If you disagree with a cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region IV; and the NRC resident inspectors at the Cooper Nuclear Station.

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

Sincerely,

/RA/

Thomas R. Farnholtz, Chief
Engineering Branch 1
Division of Reactor Safety

Docket: 50-298
License: DPR-46

Enclosure:
Inspection Report 05000298/2018011
w/ Attachment:
Detailed Risk Analysis for NCV 2018011-02

U.S. NUCLEAR REGULATORY COMMISSION
Inspection Report

Docket Number:	05000298
License Number:	DPR-46
Report Number:	05000298/2018011
Enterprise Identifier:	I-2018-011-0011
Licensee:	Nebraska Public Power District
Facility:	Cooper Nuclear Station
Location:	Brownville, Nebraska
Inspection Dates:	April 2, 2018, to April 30, 2018
Inspectors:	G. George, Senior Reactor Inspector, Region IV, Team Lead W. Cullum, Reactor Inspector, Region IV R. Deese, Senior Reactor Analyst, Region IV S. Hedger, Emergency Preparedness Inspector, Region IV M. Riley, Reactor Inspector, Region II J. Leivo, Contractor, Beckman and Associates M. Yeminy, Contractor, Beckman and Associates
Approved By:	T. Farnholtz, Chief Engineering Branch 1 Division of Reactor Safety

Enclosure

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting a Design Bases Assurance (Teams) inspection at Cooper Nuclear Station in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information. Findings and violations being considered in the NRC's assessment are summarized in the table below.

List of Findings and Violations

<table>
<tr><td colspan="4">Failure to Correct Extent of Condition of Surge Suppression Varistor Failures</td></tr>
<tr><td>Cornerstone</td><td>Significance</td><td>Cross-cutting Aspect</td><td>Report Section</td></tr>
<tr><td>Mitigating Systems</td><td>Green
NCV 05000298/2018011-01
Closed</td><td>None</td><td>71111.21M-Design Bases Assurance Inspection</td></tr>
<tr><td colspan="4">An NRC-identified, Green, Non-cited Violation of Title 10, Code of Federal Regulations Part 50, Appendix B, Criterion XVI, "Corrective Action," occurred when the licensee failed to correct conditions adverse to quality associated with the corrective actions identified in Condition Report RCR 2002-1665 to verify that installed surge suppressor varistors were appropriately sized and that design information was correctly reflected in controlled drawings for the reactor protection system, diesel generator control circuits, and high pressure coolant injection control circuits.</td></tr>
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<tr><td>Mitigating Systems</td><td>Green
NCV 05000298/2018011-02
Closed</td><td>H.6 – Human Performance, Design Margins</td><td>71111.21M-Design Bases Assurance Inspection</td></tr>
<tr><td colspan="4">An NRC-identified, Green, Non-cited Violation of Title 10, Code of Federal Regulations Part 50, Appendix B, Criterion III, "Design Control," occurred for failure to assure that applicable regulatory requirements and the design basis were correctly translated into specifications, drawings, procedures, and instructions. Specifically, the licensee failed to incorporate malfunctions of the residual heat removal (RHR) service water booster pump (SWBP) room cooling temperature switch, which could cause environmental changes leading to functional degradation of system performance, into the design basis to verify the necessary protection system action be retained.</td></tr>
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<tr><td>Cornerstone</td><td>Significance</td><td>Cross-cutting Aspect</td><td>Report Section</td></tr>
<tr><td>Mitigating Systems</td><td>Green
NCV 05000298/2018011-03
Closed</td><td>None</td><td>71111.21M-Design Bases Assurance Inspection</td></tr>
<tr><td colspan="4">An NRC-identified, Green, Non-cited Violation of Title 10, Code of Federal Regulations Part 50, Appendix B, Criterion III, “Design Control,” occurred for the licensee’s failure to ensure design control measures provide for verifying or checking the adequacy of design of the emergency diesel generator room ventilation system by use of alternate or simplified calculation methods, or by a suitable testing program. Specifically, the licensee incorrectly extrapolated the results of the test program, which led to an incorrect room temperature profile. Additionally, the design calculation did not assume potential failures of the CO_2 dampers.</td></tr>
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<tr><td colspan="4">Incorrect Classification of Potential Safety-Related Components</td></tr>
<tr><td>Cornerstone</td><td>Significance</td><td>Cross-cutting Aspect</td><td>Report Section</td></tr>
<tr><td>Mitigating Systems</td><td>Green
NCV 05000298/2018011-04
Closed</td><td>None</td><td>71111.21M-Design Bases Assurance Inspection</td></tr>
<tr><td colspan="4">An NRC-identified, Green, Non-cited Violation of Title 10, Code of Federal Regulations Part 50, Appendix B, Criterion III, “Design Control,” occurred for failure to assure that applicable regulatory requirements and the design basis were correctly translated into specifications, drawings, procedures, and instructions. Specifically, the inspectors identified three examples of the licensee’s failure to properly classify potential safety-related components in the emergency diesel generator ventilation system and RHR service water booster pump room cooling systems.</td></tr>
</table>

INSPECTION SCOPES

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

REACTOR SAFETY

71111.21M—Design Bases Assurance Inspection (Teams)

From April 2, 2018, to April 30, 2018, the inspectors reviewed the following design attributes associated with risk-significant components, permanent plant modifications, and operating experience.

Components (4 Samples)

(1) 4160V Switchgear 1F

a) Component maintenance history and corrective action program reports

b) Electrical distribution and system load flow/voltage drop calculations

c) Procedures for circuit breaker preventive maintenance, inspection, and testing

d) Equipment qualification specifications

e) Surveillance testing of the loss-of-offsite power instrumentation

(2) Emergency Diesel Generator 1B

a) Design requirements within purchase and manufacturing specifications

b) Surveillance and inservice tests of the air starting subsystem and check valves

c) Surveillance tests of the Fuel Oil subsystem including pumps and level switches

d) Surveillance tests and acceptance criteria of the diesel generator room ventilation system

e) Design calculations of the capacity of air starting system, fuel oil supply system, and room ventilation system

f) Component maintenance history and corrective action program reports

g) Failure mode effects of non-safety related pressure relief dampers on diesel generator operation

h) Design piping, electrical, and instrumentation drawings

(3) Residual Heat Removal Pump A

a) Component maintenance history and corrective action program reports

b) Calculations for net positive suction head, maximum flow, and pump performance

c) Inservice test procedures and vendor recommendations for testing frequency

d) Pump curves

e) Heat exchanger thermal performance test procedures and data

(4) Reactor protection system relays for scram discharge volume high level, K1, and reactor vessel low level discharge volume high level, K6A and K6B:

a) Component maintenance history and a sample of corrective action documents

b) Calculations for instrument uncertainties and setpoints

c) Elementary, instrument loop, and instrument piping diagrams

d) Device ratings for the components and auxiliary devices

e) Surveillance procedures and results for calibration, functional tests, and response tests

f) Maintenance procedures and results of preventive maintenance, inspection, and post-maintenance tests

g) Visual inspection of environment and material condition of the relays

<u>Component Large Early Release Frequency (LERF)</u> (1 Sample)

(1) Residual Heat Removal Service Water Booster Pump A:

a) Calculations for room heating, ventilation, and cooling

b) Operating procedures for pump operation during a design bases loss-of-coolant accident coincident with a loss-of-offsite power

c) Pump temperature ratings and pump cooling requirements

d) Calibration and testing history of temperature switches used to indicate high room temperature

e) Qualifications of emergency cooling unit and temperature switch associated with room cooling

Permanent Modifications (4 Samples)

(1) Change Evaluation Document 6024460, “4kV Breaker Auxiliary Switch Removal”

(2) Change Evaluation Document 6028860, “Alternate Decay Heat Removal Subsystem”

(3) Change Evaluation Document 6039040, “Relocation of Shutdown Cooling Isolation Function”

(4) Change Evaluation Document 6039501, “Reactor Protection System Test Box”

Operating Experience (3 Samples)

(1) NRC Information Notice 2010-27, “Ventilation System Preventive Maintenance and Design Issues”

(2) NRC Bulletin BL-88-04, “Potential Safety-Related Pump Loss”

(3) NRC Information Notice 2016-01, “Recent Issues Related to the Commercial Grade Dedication of Allen Bradley 700-RTC Relays”

Evaluation of Inspection Sample Related Operator Procedures and Actions

(1) Control room operator actions resulting from a simulated break in the service water system header while operating RHR service water booster pump and RHR pumps in suppression pool cooling mode.

   a) Control room crew were expected to enter procedures to address flooding in the control building basement.

   b) From receipt time of first control building flooding level alarm, actions to secure the RHR service water booster pump and affected train service water pumps were to be completed within 10 and 30 minutes, respectively.

(2) Operations support center maintenance staff actions resulting from a design basis loss-of-coolant accident to install turbine stop valve shaft adjustment tools as part of aligning the licensee’s alternate leakage treatment pathway configuration.

   a) Qualified staff were expected to implement the procedure attachment for installing the shaft adjustment tools.

   b) Qualified staff were expected to use the designated tools for this task to install the tools within 30 minutes of entering the turbine building.

(3) Operations support center actions resulting from a design basis loss-of-coolant accident or loss-of-offsite power event to establish a natural circulation cooling for the RHR service water booster pump room.

a) Qualified staff were directed to develop natural circulation ventilation for the RHR service water booster pump room using procedure 2.3_R-1, Panel/Window: R-1/E-7, Section 1.3.2.

b) From the receipt of the associated RHR service water booster pump room high temperature alarm, qualified staff were directed by procedure to establish the ventilation flow path within 5 hours.

INSPECTION RESULTS

Failure to Correct Extent of Condition of Surge Suppression Varistor Failures			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000298/2018011-01 Closed	None	71111.21M-Design Bases Assurance Inspection

An NRC-identified, Green, Non-cited Violation of Title 10, *Code of Federal Regulations* Part 50, Appendix B, Criterion XVI, “Corrective Action,” occurred when the licensee failed to correct conditions adverse to quality associated with the corrective actions identified in Condition Report RCR 2002-1665 to verify that installed surge suppressor varistors were appropriately sized and that design information was correctly reflected in controlled drawings for the reactor protection system, diesel generator control circuits, and high pressure coolant injection control circuits.

Description:

The inspectors requested design documents supporting the basis for the application and ratings of the electrical surge suppression varistors shown in the 120 Vac reactor protection system (RPS) control circuits elementary diagram 454005876, Revision 22, for certain General Electric Type HFA relays. The licensee provided the following information. From the following information, the inspectors identified:

- Reactor protection system elementary diagram 454005876, Revision 22, identified a device voltage rating of 175 Vdc. The inspectors identified that no reference to a design specification or a traceable model number was documented to provide critical specifications or qualifications, such as Joule (energy dissipation) ratings.

- Minor design change MDC 74-31, “Voltage Transient Suppression,” performed in 1974, allowed the use of surge suppression metal oxide varistors at the discretion of engineering. The MDC 74-31 identified a “Special Precaution” to verify the General Electric metal oxide varistor is rated for the circuit in which it is applied. The inspectors determined there was no evidence of verification or acceptance criteria for the selection, test, or installation of the devices as stipulated in the “Special Precaution” and “Acceptance Test Procedure” of MDC 74-31.

The inspectors determined that no design specifications or analyses for selection or application were available for surge suppressor varistors, for certain General Electric Type HFA relays used in RPS. The devices were shown symbolically on the RPS diagrams, but the diagrams did not present or reference sufficient detail that would provide a link to the complete specifications or a model number for the device, as well as a design basis for its selection and application.

In addressing the inspectors' concerns, the licensee provided Condition Report RCR 2002-1665, dated September 11, 2002. This condition report documented a NRC-identified, NCV 50-298/0015-02 and associated condition adverse to quality for the, "failure to document and maintain a design standard for surge suppression varistors in the Division 2 emergency diesel [generator] control circuit." As documented in the NCV and Condition Report RCR 2002-1665, "the use of incorrect values for these components caused the generator to frequently trip during the shutdown process and thereby be unavailable for immediate restart." The licensee's apparent cause evaluation concluded that improperly sized and rated metal oxide varistors installed as surge suppressors resulted in a spurious overspeed trip. The immediate corrective action included the replacement of the varistor in the emergency diesel generator control circuit. However, the condition report identified an extent of condition existed beyond the diesel control circuit, as well as programmatic issues, including unresolved design control issues for surge suppressors.

To correct the extent of condition, the two corrective actions identified in Condition Report RCR 2002-1665 were:

- "Revise [Maintenance Procedure] 7.3.16, Low Voltage Relay Removal and Installation, to require Engineering to verify that installed varistors, when found, are appropriately sized and design information is correctly reflected in controlled drawings." The due date was March 1, 2003.

- "Revise the RPS, DG, and HPCI elementary diagrams to include appropriate design information of installed varistors." The due date was May 1, 2003.

Based on the documents provided during the inspection, the inspectors concluded that these corrective actions had never been effectively implemented. The requirement for engineering to verify the suitability of the ratings was not incorporated in the current revision of Maintenance Procedure 7.3.16 or in other revisions since the corrective action due date. Additionally, the appropriate design information for surge suppressors was not completely reflected on drawings either directly or by reference. For example, the voltage rating was identified on the drawings, but additional critical characteristics such as Joule ratings or model numbers were not presented or referenced.

Corrective Actions: The licensee performed a historical search to identify surge suppression varistor applications in safety-related control circuits. The licensee then performed informal and bounding calculations based on first principles and component vendor data for the relays, and installed metal oxide varistors. The team concluded that the results of the informal calculations provided reasonable assurance that the surge suppression varistor ratings had ample margin and were suitable for their application and service.

Corrective Action References: Condition Reports CR-CNS-2018-02331 and CR-CNS-2018-02374

Performance Assessment: Performance Deficiency: The inspectors determined that the failure to correct conditions adverse to quality associated with the corrective actions identified in Condition Report RCR 2002-1665 to verify that installed surge suppressor varistors were appropriately sized and that design information was correctly reflected in controlled drawings was a performance deficiency. Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the design control attribute of Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to establish, implement, and maintain a design standard for surge suppression varistors used in safety-related control circuits could result in unsuitable selection and application of the device, such that failure of the device could result in shorting of a normally energized relay coil changing the systems to an undesirable state. Significance: Using Inspection Manual Chapter 0609, Appendix A, Exhibit 2, dated June 19, 2012, the inspectors determined that this finding was of very low safety significance (Green) because it was a deficiency affecting the design or qualification of a structure, system, or component and operability was maintained. Cross-cutting Aspect: The inspectors determined that this finding did not have a cross-cutting aspect because the most significant contributor did not reflect current licensee performance.
Enforcement: Violation: Title 10, *Code of Federal Regulations* Part 50, Appendix B, Criterion XVI, “Corrective Action,” requires, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected.” Contrary to the above, from September 11, 2002, to April 19, 2018, the licensee failed to establish measures to assure that conditions adverse to quality were promptly corrected. Specifically, the licensee failed to correct conditions adverse to quality associated with the corrective actions identified in Condition Report RCR 2002-1665, to verify that installed surge suppressor varistors were appropriately sized, and that design information was correctly reflected in controlled drawings for the reactor protection system, diesel generator control circuits, and high pressure coolant injection control circuits. Disposition: This violation is being treated as a NCV, consistent with Section 2.3.2 of the Enforcement Policy.

Failure to Ensure Adequate Design Control Measures are in Place Associated with RHR Service Water Booster Pump Room Cooling			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000298/2018011-02 Closed	H.6 – Human Performance, Design Margins	71111.21M-Design Bases Assurance Inspection
An NRC-identified, Green, Non-cited Violation of Title 10, *Code of Federal Regulations* Part 50, Appendix B, Criterion III, "Design Control," occurred for failure to assure that applicable regulatory requirements and the design basis were correctly translated into specifications, drawings, procedures, and instructions. Specifically, the licensee failed to incorporate malfunctions of the RHR service water booster pump room cooling temperature switch, which could cause environmental changes leading to functional degradation of system performance, into the design basis to verify the necessary protection system action be retained.			

Description:

During a design basis loss-of-cooling accident, the licensee operates the RHR service water booster pump system to provide cooling to the RHR system without an uncontrolled release of radioactive material to the environment. The system's design bases, as stated in Cooper Nuclear Station Updated Safety Analysis Report (USAR) Chapter X, Section 8.2.2, includes the following:

- No single system component active failure shall be able to prevent the system from achieving its safety objective.
- The system shall provide an adequate supply of cooling water to the RHR system under all accident and transient conditions.

Standard IEEE 279-1971 provides expectations on what constitutes the plant's design basis and defines the single failure criterion. As part of the design basis minimum elements, Section 3(8) states that the following shall be documented:

> "[T]he malfunctions, accidents, or other unusual events…which could physically damage protection system components or could cause environmental changes leading to functional degradation of system performance, and for which provisions must be incorporated to retain necessary protection system action."

Section 4.2 states:

> "Any single failure within the protection system shall not prevent proper protection system action when required."

Notes further state:

"Single failure' includes such events as the shorting or open-circuiting of interconnecting signal or power cables. It also includes single credible malfunctions or events that cause a number of consequential component, module, or channel failures."

During a design basis accident, the RHR service water booster pump system room is cooled by the RHR service water booster pump room fan coil unit or by manual operator actions. The cooling methods, described in USAR Chapter X, Section 10.3.5.4, include the following:

- The fan coil unit is normally cooled by a recirculation loop to a cooling tower.
- If the recirculation loop is not available, the cooling coils can be manually aligned with service water.
- Acceptable room temperatures can be maintained for a single RHR service water booster pump without forced ventilation through operator action to reduce the room heat load and establishment of a natural ventilation flow path.

The cooling tower described in the USAR has not been functional since the late 1980's. On July 29, 2016, the fan coil unit was tagged out of service, to facilitate removal of the cooling tower, and was still out of service at completion of this inspection. Therefore, the only active means of cooling the RHR service water booster pump room, in all accident and transient conditions, was to take the operator action establishing a natural ventilation flow path.

To implement room cooling under all accident and transient conditions, control room operators initiate actions based on annunciator "RHR SWBP ROOM HIGH/TEMP" (Panel/Window R-1/E-7). This annunciator operates based on a room temperature of 120^0F, as indicated by temperature switch HV-TS-1109. The inspectors questioned whether the events that could cause damage to the temperature switch had been documented and what provisions had been incorporated to retain its ability to provide for protective action. Further, the inspectors questioned whether the postulated failure of the temperature switch would constitute a single active failure and whether it is seismically qualified to function during a design basis earthquake.

Section 9.3.4 of the licensee's original safety evaluation report, (dated February 14, 1973), states that the RHR service water booster pump system is designed as Seismic Class I, and that "RHR Service Water Booster Pump Systems can supply essential cooling water even after failure of all Class II (seismic) systems with which it interfaces and a concurrent failure of an active component in the Class I (seismic) system." No information was provided by the licensee to indicate that the temperature switch had been analyzed during accident conditions, with provisions to retain its ability to provide protective action in those cases. Further, no indication was provided that the temperature switch was seismically qualified to function after a design basis earthquake. Failure of the switch would result in an active failure that would cause the annunciator to not operate. This would result in not ensuring timely operator action to set up cooling for the RHR service water booster pumps. This set of consequential failures would result in degradation of the RHR service water booster pump system to provide adequate cooling water under all accident and transient conditions.

Corrective Actions: In response to this issue, the licensee revised alarm response, system operating, and emergency operating procedures to establish natural ventilation in the RHR service water booster pump system room on the onset of the loss of normal plant ventilation during accident conditions without relying on the temperature switch.

Corrective Action References: Condition Reports CR-CNS-2018-02034 and CR-CNS-2018-02069

Performance Assessment:

Performance Deficiency: The inspectors determined that the licensee's failure to incorporate malfunctions of the RHR service water booster pump room cooling temperature switch, which could cause environmental changes leading to functional degradation of system performance, into the design basis to verify the necessary protection system action be retained, in accordance with IEEE 279-1971 Section 3, "Design Basis," was a performance deficiency.

Screening: The inspectors determined the performance deficiency 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 initiating events to prevent undesirable consequences. Specifically, the licensee's failure to incorporate malfunctions of the temperature switch, which could cause environmental changes leading to functional degradation of system performance, into the design basis would degrade the capability of the RHRSWB system to meet its design function during design basis accidents.

Significance: The inspectors used Inspection Manual Chapter 0609, "Significance Determination Process," Attachment 4, dated October 7, 2016; and the corresponding Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 2, "Mitigating Systems Screening Questions," dated June 19, 2012. The inspectors determined the finding represented a potential loss of function; therefore, the NRC senior reactor analyst completed a detailed risk evaluation. The results of the detailed risk evaluation yielded an estimate of the increase in core damage frequency 1.7E-8/year. Because the increase in core damage frequency was less than 1.0E-7/year, the increase in core damage frequency from external events and the increase in large early release frequency were not analyzed. Based on this evaluation, the performance deficiency screened as having very low safety significance (Green).

Cross-cutting Aspect: The finding had a cross-cutting aspect in the area of human performance associated with Design Margins. Margins are carefully guarded and changed only through a systematic and rigorous process. When the room cooler was tagged out in 2016, the room cooling capability was not evaluated adequately to ensure that design margin was maintained [H.6].

Enforcement:

Violation: Title 10, *Code of Federal Regulations* Part 50, Appendix B, Criterion III, "Design Control" requires, in part, that measure shall be established to assure that applicable regulatory requirements and the design basis for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures and instructions.

Contrary to the above, from July 29, 2016 to present, the licensee failed to assure applicable regulatory requirements and the design basis were correctly translated into the specifications

design of the RHR service water booster pump cooling. Specifically, the licensee failed to incorporate malfunctions of the RHR service water booster pump room temperature switch, which could cause environmental changes leading to functional degradation of system performance, into the design basis to verify the necessary protection system action be retained. Disposition: This violation is being treated as a NCV, consistent with Section 2.3.2 of the Enforcement Policy.

Inadequate Design Basis Calculation for the EDG Rooms Temperature Distribution			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000298/2018011-03 Closed	None	71111.21M-Design Bases Assurance Inspection
An NRC-identified, Green, Non-cited Violation of Title 10, *Code of Federal Regulations* Part 50, Appendix B, Criterion III, "Design Control," occurred for the licensee's failure to ensure design control measures provide for verifying or checking the adequacy of design of the emergency diesel generator room ventilation system by use of alternate or simplified calculation methods, or by a suitable testing program. Specifically, the licensee incorrectly extrapolated the results of the test program, which led to an incorrect room temperature profile. Additionally, the design calculation did not assume potential failures of the CO_2 dampers.			
Description: Design basis calculation NEDC 91-103, "DG Room Cooling Without Cooling Coil," Revision 2, evaluated the capacity of the ventilation system to sufficiently cool the emergency diesel generator room. The calculation was performed in order to approve a design change DC 91-23 that disconnected the flow of service water to cooling coils in the ventilation duct. The calculation determined the required air flow rates and resulting room temperatures of the ventilation system at outside air temperatures of 97°F and 104°F. The calculation was based on an extrapolation of data from the 1988 performance of a Special Test Procedure, which measured temperatures at different locations in the room while the emergency diesel generators were running. The calculation extrapolated the temperatures, assuming maximum outdoor temperatures. The result of the calculation determined that with 97°F inlet air, the temperature of the area around the emergency diesel generator's most limiting component was 120.56°F while the temperature of the air exiting the room will be 134°F, at minimum required air flow. According to the calculation, the limiting component, the alternating current generator, had a limiting temperature of 120.60°F, leaving a design margin of 0.04°F. The Special Test Procedure documented temperature measurements of nine specified locations, where the ninth point was the temperature of the air exiting the room. All eight points measured at different locations around the emergency diesel generator room were substantially lower than the temperature of the air exiting the room, which represents the bulk temperature of the room. Based on the magnitude of temperature differences, the inspectors found the results of the 1988 special test questionable, and as a result, witnessed the surveillance test of the emergency diesel generator at full load on April 16, 2018.			

During this April 16 surveillance test, the inspectors and the licensee measured temperatures in areas previously measured by the 1988 special test and the temperature of the air exiting the room. The resulting temperature of the air exiting the room was generally lower than temperatures measured in the vicinity of the emergency diesel generator for the normal ventilation conditions present during the test. Based on these results, the inspectors and the licensee agreed that the existing heat distribution model for the calculation was not accurate. The licensee's preliminary Gothic model reanalysis and the inspector's hand calculation determined room exit temperature should be 128°F, not 134°F.

Additionally, the calculation failed to consider the potential effects on the ventilation system of the non-safety related CO_2 dampers in the open position. When the ventilation system starts, the supply fan starts, pressurizing the room, and opening the CO_2 dampers installed on each side of the ventilation discharge duct. After a few seconds lag, the discharge fan starts, pressurizing the air inside the discharge duct and closing the CO_2 dampers. If the dampers are in the open position, a significant portion of the hot air would be diverted into the room rather than being discharge outdoors. Therefore, increasing the room temperatures.

Since 2004, the licensee recorded at least 11 failures of the CO_2 dampers to close or found in the open position. These non-safety related dampers did not undergo scheduled preventive maintenance. Furthermore, the emergency diesel generator surveillance test procedure does not have a step to validate that the CO_2 dampers remain closed during the test. The preliminary Gothic model prepared by the licensee determined that if these CO_2 dampers are stuck open, the bulk temperature of the room will be increased by 15 degrees F. This would result in bulk room temperature increasing from 128°F to 143°F.

Based on the increased room temperature, the inspectors requested the licensee complete a review of the emergency diesel generator control components to identify which components' maximum allowable temperatures would be exceeded. The licensee identified that the temperature challenged the design margin of two ventilation control relays for each emergency diesel generator ventilation system. Based on the conservative design margin of the components' maximum allowable temperatures, the components would remain functional.

Corrective Actions: In response to this issue, the licensee completed a preliminary reanalysis to determine that there was no immediate safety concern. The licensee's long term corrective actions will be to correct room temperature model including potential failures of CO_2 dampers.

Corrective Action References: Condition Report CR-CNS-2018-02308

Performance Assessment:

Performance Deficiency: The inspectors determined that the licensee's failure to verify the adequacy of the emergency diesel generator room ventilation by use of alternate or simplified calculation methods, or by a suitable testing program, was a performance deficiency.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the design control attribute of Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee's failure to correctly model the room temperature profile and

potential failures of non-safety related CO_2 dampers had the potential to lead to exceedance of safety-related components' maximum allowable temperatures. Significance: Using Inspection Manual Chapter 0609, Appendix A, Exhibit 2, dated June 19, 2012, the inspectors determined that this finding was of very low safety significance (Green) because it was a deficiency affecting the design or qualification of a structure, system, or component, and operability was maintained. Cross-cutting Aspect: The inspectors determined that this finding did not have a cross-cutting aspect because the most significant contributor did not reflect current licensee performance.
<u>Enforcement</u>: Violation: Title 10, *Code of Federal Regulations* Part 50, Appendix B, Criterion III, "Design Control" requires, 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 19, 2018, the licensee failed to ensure design control measures provide for verifying or checking the adequacy of design of the emergency diesel generator room ventilation system by use of alternate or simplified calculation methods, or by a suitable testing program. Specifically, the licensee incorrectly extrapolated the results of the test program, which led to an incorrect room temperature profile. Additionally, the design calculation did not assume potential failures of the CO_2 dampers. Disposition: This violation is being treated as a NCV, consistent with Section 2.3.2 of the Enforcement Policy.

Incorrect Classification of Potential Safety-Related Components			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000298/2018011-04 Closed	None	71111.21M-Design Bases Assurance Inspection
An NRC-identified, Green, Non-cited Violation of Title 10, *Code of Federal Regulations* Part 50, Appendix B, Criterion III, "Design Control," occurred for failure to assure that applicable regulatory requirements and the design basis were correctly translated into specifications, drawings, procedures, and instructions. Specifically, the inspectors identified three examples of the licensee's failure to properly classify potential safety-related components in the emergency diesel generator ventilation system and RHR service water booster pump room cooling systems.			

Description:

The safety-related (or Essential) classification of structures, systems, and components are governed by the following:

- Cooper Nuclear Station Updated Safety Analysis Report (USAR) Appendix G, Section 2.4.2, “Methodology to Identify Safety-Related Structures, Systems, Components, and Parts”

- Cooper Nuclear Station Procedure 3-EN-DC-167, “Classification of Structures, Systems, and Components,” Revision 4C2

The inspectors identified three examples where the licensee failed follow the methodology listed in Appendix G or Procedure 3-EN-DC-167. The examples are:

Example 1: The licensee failed to classify emergency diesel generator room CO_2 pressure relief dampers as safety-related. The safety-related emergency diesel generator ventilation is equipped with safety-related ducts with CO_2 pressure relief dampers installed in the duct. These CO_2 pressure relief dampers were not designed, manufactured, delivered, installed, or maintained as safety-related components. Failure of a CO_2 damper to close causes some of the discharge air to recirculate back into the room. The combined surface area of the CO_2 dampers located on both sides of the discharge duct is equal to the surface area of the safety-related discharge damper located downstream. Failure of the dampers to close will result in safety-related electrical components exceeding maximum allowable room temperatures. Since 2004, the licensee recorded at least 11 failures of the CO_2 dampers to close or found in the open position. These non-safety related dampers did not undergo scheduled preventive maintenance. Furthermore, the emergency diesel generator surveillance test procedure does not have a step to validate that the CO_2 dampers remain closed during the test. The inspectors determined that the CO_2 pressure relief dampers should be classified safety-related in accordance with USAR Appendix G, Section 2.4.2, paragraph (e) and Procedure 3-EN-DC-167, Section 5.2. Paragraph (e) requires components in non-safety related systems that are required to provide safety-related functions due to interface requirements shall be determined and classified safety-related.
Procedure 3-EN-DC-167 Section 5.2 [2], “Equipment Classification Criteria,” requires Essential (safety-related) classification of equipment if the equipment is not required to function but can fail in a manner that would prevent the system from performing it safety-related function.

Example 2: The licensee failed to classify the RHR service water booster pump room emergency fan coil unit as safety-related. During a design basis accident, the RHR service water booster pumps and motors are cooled by the RHR service water booster pump room emergency fan coil unit or by manual operator actions. This fan coil unit and its associated temperature controller, HV-TC-1109, which starts the cooling unit automatically on high temperature, were not installed and maintained as safety-related components. Failure of the fan cooling unit to cool the RHR service water booster pump room will result in the loss of function of at least one train of RHR function without operator actions to recover. The inspectors determined that the RHR service water booster pump room emergency fan coil unit should be classified safety-related in accordance with USAR Appendix G, Section 2.4.2, paragraph (e) and Procedure 3-EN-DC-167, Section 5.2. Paragraph (e) requires components in non-safety related systems that are required to provide safety-related functions due to interface requirements shall be determined and classified safety-related.

Procedure 3-EN-DC-167 Section 5.2 [2], “Equipment Classification Criteria,” requires Essential (safety-related) classification of equipment if the equipment is not required to function but can fail in a manner that would prevent the system from performing it safety-related function.

Example 3: The licensee failed to classify the RHR service water booster pump room temperature switch HV-TS-1109 as safety-related. During a design basis accident, the RHR service water booster pumps and motors are cooled by the RHR service water booster pump room emergency fan coil unit or by manual operator actions. If the fan coil unit was not available, the USAR credits manual action to provide natural circulation cooling by opening hatches and doors for the room. This natural convection cooling would ensure the safety function of one RHR service water booster pump. The initiation of the manual action occurs on receiving a control room alarm for high room temperature, as indicated by HV-TS-1109. Failure of the switch to function after a design basis loss-of-cooling accident would delay manual actions to initiate natural convection cooling in an appropriate time that ensures the safety function of one RHR service water booster pump. The inspectors determined that the RHR service water booster pump room temperature switch HV-TS-1109 should be classified safety-related in accordance with USAR Appendix G, Section 2.4.2, paragraph (e) and Procedure 3-EN-DC-167, Section 5.2. Paragraph (e) requires components in non-safety related systems that are required to provide safety-related functions due to interface requirements shall be determined and classified safety-related. Procedure 3-EN-DC-167 Section 5.2 [2], “Equipment Classification Criteria,” requires Essential (safety-related) classification of equipment if the equipment is not required to function but can fail in a manner that would prevent the system from performing it safety-related function.

Corrective Actions: In response to Example 1, the licensee completed a preliminary reanalysis to determine that there was no immediate safety concern. The licensee’s long term corrective actions will be to correct the room temperature model including potential failures of CO_2 dampers. In response to Example 2, the licensee had implemented compensatory measures in place while emergency fan coil unit was out of service; therefore, there was no immediate safety concern. In response to Example 3, the licensee revised alarm response, system operating, and emergency operating procedures to establish natural ventilation in the RHR service water booster pump system room on the onset of the loss of normal plant ventilation during accident conditions without relying on the temperature switch.

Corrective Action References: Condition Reports CR-CNS-2018-02034, CR-CNS-2018-02069, and CR-CNS-2018-02075

Performance Assessment:

Performance Deficiency: The inspectors determined that the failure to classify emergency diesel generator ventilation system and RHR service water booster pump room cooling system components as safety-related (Essential), in accordance with USAR Appendix G, Section 2.4.2, “Methodology to Identify Safety-Related Structures, Systems, Components, and Parts” and Cooper Nuclear Station Procedure 3-EN-DC-167, “Classification of Structures, Systems, and Components,” Revision 4C2, was a performance deficiency.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the design control attribute of Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences.

Specifically, the failure to classify components as safety-related (Essential) led to configuration control, maintenance, and problem identification practices which degraded the capability of the emergency diesel generator ventilation system and RHR service water booster pumps to perform their design function. Significance: Using Inspection Manual Chapter 0609, Appendix A, dated June 19, 2012, the team determined that this finding was of very low safety significance (Green) because it was a deficiency affecting the design or qualification of a structures, systems, or components, and operability was maintained. Cross-cutting Aspect: The inspectors determined that this finding did not have a cross-cutting aspect because the most significant contributor did not reflect current licensee performance.
Enforcement: Violation: Title 10, *Code of Federal Regulation*s Part 50, Appendix B, Criterion III, "Design Control" requires, in part, that measure shall be established to assure that applicable regulatory requirements and the design basis for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures and instructions. Contrary to the above, prior to April 19, 2018, the licensee failed to correctly translate the applicable regulatory requirements and design basis to which this appendix applies are correctly translated into specifications, drawings, procedures and instructions. Specifically, the licensee failed to properly classify potential safety-related components in the emergency diesel generator ventilation system and RHR service water booster pump room cooling systems. Disposition: This violation is being treated as a NCV, consistent with Section 2.3.2 of the Enforcement Policy.

EXIT MEETINGS AND DEBRIEFS

The inspectors verified no proprietary information was retained or documented in this report.

On April 19, 2018, the team presented the preliminary results of the Inspection Procedure 71111.21M inspection to Mr. J. Dent, Jr., Site Vice President and Chief Nuclear Officer, and other members of the licensee staff.

On April 30, 2018, the team presented the results of the Inspection Procedure 71111.21M inspection to Mr. D. Buman, Director, Nuclear Safety Assurance and other members of the licensee staff.

DOCUMENTS REVIEWED

IP 71111.21M – Design Bases Assurance Inspection (Teams)

Condition Reports (CR-CNS-)			
1994-00762	2002-01665	2012-09956	2012-10656
2013-02959	2013-05725	2013-06870	2015-00604
2015-02085	2015-02090	2015-02106	2015-02366
2015-02373	2015-02393	2015-02883	2015-02947
2016-00820	2016-06248	2016-06251	2016-06680
2016-06989	2016-08010	2017-05040	2017-07520

Inspection Generated Condition Reports (CR-CNS-)			
2018-01975	2018-01979	2018-01989	2018-02003
2018-02004	2018-02033	2018-02034	2018-02034
2018-02035	2018-02056	2018-02063	2018-02064
2018-02069	2018-02069	2018-02074	2018-02075
2018-02076	2018-02097	2018-02163	2018-02293
2018-02308	2018-02314	2018-02315	2018-02316

Work Orders			
4281699	4551512	4733240	4850159
4908185	4908614	4943761	4944185
4946822	4948722	5030026	5031812
5039857	5056976	5059028	5059032
5059855	5060013	5062381	5072414
5075220	5085176	5089053	5097666
5102533	5113147	5118603	5119820
5121779	5122463	5123397	5123676
5125110	5167111		

Procedures Number	Title	Revision Or Date
15.MS.201	MSIV Leakage Pathway Valve Testing	2, 3, 4
2.0.3	Conduct of Operations	92
2.1.11.1	Turbine Building Data	154
2.1.22	Recovering from a Group I Isolation	60
2.2.59	Plant Air System	74
2.2.69.3	RHR Suppression Pool Cooling and Containment Spray	46
2.2.70	RHR Service Water Booster Pump System	76, 85
2.2.70	RHR Service Water Booster Pump System	82, 85
2.2.73	Standby Gas Treatment System	52
2.3_RHR-GLND-1	RHR Gland Water Supply – Annunciator Panel 1A	5
3.1	Engineering Procedure	10

Procedures Number	Title	Revision Or Date
3-EN-DC-115	Engineering Change Process	15C11
3-EN-DC-147	Engineering Reports	5C1
3-EN-DC-167	Classification of Structures Systems and Components	4C2
3-EN-DC-306	Acceptance of Commercial-Grade Items/Services in Safety-Related Applications	5C1
5.1BREAK	Pipe Break Outside Secondary Containment	17
5.2FUEL	Fuel Failure	20
5.3EMPWR	Emergency Power During Modes 1, 2, or 3	66, 67
5.4FIRE-SD	Procedure -Fire Induced Shutdown From Outside Control Room	70
6.1RPS.307	Surveillance Procedure: Low-High Water Level Channel Calibration (Div 1) [WOs 4850159, 4944185, 5030026]	17
6.1RPS.310	Surveillance Procedure: South SDV High Water Level Switches and Transmitters Examination and Channel Calibration (Div 1) [WOs 4908614, 4946822, 5031812]	25
6.1RPS.311	Surveillance Procedure: RPS Instrument Channel Response Time (Run Mode)(Div 1) [WOs 4908185, 4943761, 5072414]	12
6.1RPS.312	Surveillance Procedure: RPS Instrument Channel Response Time (Shutdown)(Div 1) [WO 4281699]	11
6.1RPS.707	Surveillance Procedure: Reactor Vessel Low-High Water Level Channel Functional Test (Div 1)[WOs, 5085176, 5097666, 5113147]	15
6.1RPS.710	Surveillance Procedure: South SDV High Water Level Switches and Transmitters Channel Functional Test (Div 1) [WOs 5089053, 5102533, 5118603]	20
6.2 DG.105	Diesel Generator Starting Air Compressor Operability (IST) (Div 2)	25
6.2 DG.401	Diesel Generator Fuel Oil Transfer Pump IST Flow Test (Div 2)	31
6.2 DG.402	IST Closure Testing of DGSA Receiver Inlet Check Valves (Div 2)	11
6.2HV.602	Air Flow Test of Fan Coil Unit HV-DG-1D (Div 2)	10
7.1.8	Rigging and Lifting Program	39
7.13.17.1	4160 Breaker Examination	30
7.13.17.3	Replacing 4160V Breakers	14
7.3.16	Low Voltage Relay Removal and Installation	4
7.3.16	Low Voltage Relay Removal and Installation	25
800000023083	Maintenance Plan, Visual Inspection of Relays [WO 4948722]	06/08/15

Procedures Number	Title	Revision Or Date
Alarm Procedure 2.3_B-3	Panel B – Annunciator B-3	37
Alarm Procedure 2.3_R-1	Panel R – Annunciator R-1	16, 18
EOP 5.8, Attachment 1	Secondary Containment Control, Radioactivity Release Control (5A)	16
EOP 5.8, Attachment 1	Primary Containment Control (3A)	17
EOP 5.8.10	Average Drywell Temperature Calculation	8

Drawings (Number)	Title	Revision or Date
2011 sheet 1	Flow Diagram Turbine oil Purification and Transfer System and Diesel oil System	45
2024 sheet 2	Flow Diagram HVAC Miscellaneous Buildings	N38
2040	Sheet 1 Flow Diagram Residual Heat Removal System	82
2040	Sheet 1 Iso Key Flow Diagram Residual Heat Removal System	16
2077	Flow Diagram – Diesel Generator Building Service water Starting Air Fuel Oil Sump System and Roof Drains	N78
2221	HVAC Plan and Sections Diesel Generator Building Heating Boiler Room	N03
28147	RHR Pump Curves	12/06/1973
452006649	Arrangement Drawing, RPS Trip System A	9
452006650	Arrangement Drawing, RPS Trip System B	10
452203102	Scram Discharge Instrument Volume Flow Diagram	N06
452243047	Level Switch Specifications	N01
452243048	Level Switch Specifications	N01
453236930	Cooper Nuclear Station Panel K & R Annunciator Input Loop Diagram ANN-MUX-13	N03
454003632	Flow Diagram, Control Rod Drive Hydraulic System	62
454003919	Cooper Nuclear Station Control Elementary Diagrams	AB
454003974	Cooper Nuclear Station, Turbine Generator Building Operating Floor, Lighting and CCTV Plan (3108, Sh. 1)	AC/21
454003975	Cooper Nuclear Station, Turbine Generator Building Operating Floor - Lighting Plan Sheet 2 (3109)	AB/10

Drawings (Number)	Title	Revision or Date
454005876	Elementary Diagram, Reactor Protection System	22
454006449	Elementary Diagram, Reactor Protection System	N14
454006451	Elementary Diagram, Reactor Protection System	15
454006561	Elementary Diagram, Primary Containment Isolation System	16
454006562	Electrical Diagram, Primary Containment Isolation System	17
454006580	Elementary Diagram, Reactor Protection System	15
454203115	Scram Discharge Volume North Header Instrument Piping	3
454203116	Scram Discharge Volume South Header Instrument Piping	3
729E211BB	Process Diagram Residual Heat Removal System	12
932-71212PI Sheet 2DG	Diesel Generator Building Units 1-HV-DG-1B and 1D	15
CNS-BLDG-456	Floor/Roof Plan 932' – 6" (Critical SW'GR Rooms) Reactor Building	N02
CNS-MS-43	Leakage Path from Outboard MSIV's , Cooper Nuclear Station	N05
DR-2016-0444	EE-PNL-AA3 125 VDC Load & Fuse Schedule	17
DR-2016-0498	Auxiliary One Line Diagram MCC Z SWGR 1A 1B 1E & Critical SWGR 1F 1G	53
DR-2017-0001	Relay Settings for 4160V Bus 1F	36
DR-2017-0196	4160V Switchgear Elementary Diagram	39
M82317	Residual Heat Removal Heat Exchanger	2
M82454	Residual Heat Removal Heat Exchanger	4
MEC3000407 R01-P-MSIV Leakage Pathway	MSIV Leakage Pathway to the Condenser	09/01/11

Calculations Number	Title	Revision or Date
EE-02-078	Engineering Evaluation, Justification for Measuring Tape on Scram Discharge Instrument Volumes	11/04/03
EE-15-048	Engineering Evaluation, DG-REL-DG1(14RY1) Use-as-is Evaluation	0
NEDC 00-003	CNS Aux. Power System Load Flow and Voltage Analysis	9
NEDC 00-029	Post-LOCA MSIV Leakage Path to Main Condenser	7
NEDC 00-111	CNS Auxiliary Power System AC Loads	9
NEDC 01-006	MSIV Leakage Pathway Piping System Walkdown and Evaluation Packages	2

Calculations Number	Title	Revision or Date
NEDC 02-064	Post-LOCA Operator Action Time Limits for Configuration of MSIV Leakage Pathway	7
NEDC 02-068	Review of Westinghouse Letter for Turbine Stop Valve Shaft for Post-LOCA Sealing Force	1
NEDC 07-082	Radiological Dose Analysis for a Loss-of-Coolant Accident (LOCA) at Cooper Nuclear Station	2, 4
NEDC 07-082	Radiological Dose Analysis for a Loss-of-Coolant Accident (LOCA) at Cooper Nuclear Station	4
NEDC 07-090	Fuel Oil Transfer Pump NPSH	2
NEDC 09-010	ECCS Suction Strainer Heat Loss Analysis with Alternate Methodology	0
NEDC 09-102	Internal Flooding – HELB, MELB, and Feedwater Line Break	2
NEDC 11-072	DGSA Accumulator Capacity and Starts	1
NEDC 87-052	DG Storage Tank Fuel capacities	5
NEDC 87-131C	125 VDC Division I Load and Voltage Study	18
NEDC 87-131D	125 VDC Division II Load and Voltage Study	14
NEDC 87-133	NPPD Review of CYGNA Calculation for RHR and CS Pump Motor Acceleration Time	1
NEDC 88-298	Review of S&L Calc 8206-E1 Control Bldg Heat Loads	5
NEDC 88-298	Review of S&L Calc 8206-E1 Control Building Heat Loads	5
NEDC 88-299A	Review of S&L Calc No. COOLC-01 Rev. 6 HVAC Load Calculation for Control Building EL 903' -6"	9
NEDC 88-299A	Review of S&L Calc Number COOLC-01 HVAC Load Calculation for Control Building EL 903'-6"	9
NEDC 89-1828	Max Flow Through the RHR Pumps	0
NEDC 90-068	DG Room Heat Load	2
NEDC 90-384	NBI-LIS-101A, B, C, D, Level 3 Setpoint Calculation	6
NEDC 91-103	DG Room Cooling Without Cooling Coil	2
NEDC 91-290	RHR Loop B Test Calculations	1
NEDC 92-050A	Scram Discharge Instrument Volume High Water Level Setpoint Calculation	4
NEDC 92-050B	Scram Discharge Instrument Volume High Water Level/Rod Withdrawal Block Level Switch Setpoint Calculation	4
NEDC 92-063	Maximum SWBP Room Temperatures with No Cooling from Control Building HVAC	2
NEDC 92-063	Maximum SWBP Room Temperature w/ No Cooling from Control Building HVAC	1, 2

Calculations Number	Title	Revision or Date
NEDC 92-064	Transient Temperature Rise in SWBP Room	3
NEDC 92-202	RHR Loop A Test Calculations	1
NEDC 93-178	Drywell Fibrous Insulation	0
NEDC 94-034D	Small Steam Line Break (SSLB) Analysis	3
NEDC 94-231	RHR Pumps NPSH Maximum Flow Calculation	7
NEDC 97-012	DG Fuel On-Site Storage	3
NEDC 97-044A	NPSH Margins for the RHR and CS Pumps	4
NEDC 97-044A	NPSH Margins for the RHR and CS Pumps	R4C1
NEDC 97-044A	NPSH Margins for the RHR and CS Pumps	R4C2
NEDC 97-044A	NPSH Margins for the RHR and CS Pumps	R4C3
NEDC 97-044A	NPSH Margins for the RHR and CS Pumps	R4C4
NEDC 98-005	Min Flow Line Capacity for RHR Pumps During Single and Parallel Pump Operation	R1
NEDC 98-005	Min Flow Line Capacity for RHR Pumps During Single and Parallel Pump Operation	1
NEDC-32914P	Maximum Extended Load Line Limit and Increased Core Flow for Cooper Nuclear Station	01/2000
NEDO-10320	The General Electric Pressure Suppression Containment Analytical Model	05/1971
NEDO-10329	Loss-of-Coolant Accident & Emergency Core Cooling Models for General Electric Boiling Water Reactors	04/1971
NEDO-21888	Mark I Containment Program Load Definition Report	2
TE-10594270	Technical Evaluation, Dedication Package CGI-10594270 (Amphenol connector)	0
TE-10777780	Technical Evaluation, Dedication Package CGI-10777780 (Amphenol connector)	0

Engineering Changes Number	Title	Revision or Date
EC 6024460	4kV Breaker Auxiliary Switch Removal	11/26/2008
EC 6039040	Relocation of Shutdown Cooling Isolation Function	0
EC 6039501	RPS Test Panels	0
MDC 74-31	Minor Design Change: Voltage Transient Suppression	0

Miscellaneous Number	Title	Revision or Date
	RHR-P-A IST	2015
	RHR-P-A IST	2016

Miscellaneous Number	Title	Revision or Date
	RHR-P-A IST	2017
	RHR-P-A IST	2018
Action Way Tracking #: LO 2016-0035-004	Vulnerability Review: Recent Issues Related to the Commercial Grade Dedication of Allen Bradley 700-RTC Relays [IN 2016-01]	4
CNS Letter CNSS887255	Supplemental Response to IE Bulletin 84-02	06/29/1988
CNS Letter NLS8400018	NPPD Response to IE Bulletin 84-02, Failures of General Electric Type HFA Relays in Use in Class 1E Safety Systems	07/16/1984
CNS-PSA-014	Cooper Nuclear Station Human Reliability Analysis	5
DC 88-053B	Sustained Elevated Ambients in the Critical Switchgear Rooms Reference Letter	02/21/1989
EE 06-025	Implementation of Alternate Source Term LOCA Analysis	1
Engineering Evaluation Number 01-147	Summary of Main Steam Isolation Valve (MSIV) Leakage Pathway to the Condenser Seismic Qualification	2
EPM Report EQDI-3273	Report on Review of Existing Equipment Qualification Master Equipment List – Attachments 1 & 2	09/20/83
Franklin Research Center TER-C5506-69	Technical Evaluation Report, BWR Scram Discharge Volume Long-Term Modifications, NPPD Cooper Nuclear Station [p. 11, surveillance criteria for scram discharge system]]	01/05/82
GE Letter C960911	CNS Setpoint Analysis	09/11/96
Lesson Number COR002-23-02 (1481)	OPS Residual Heat Removal System	34
Lesson Plan: MEC3000407	MSIV Leakage Pathway to the Condenser	1
LO 2016-0035-036	IN 2016-08 Inadequate Work Practices Resulting in Faulted Circuit Breaker Connection	---
LRP-PNL-(9-15)	CNS Screening Evaluation Worksheet Ch A RPIS & RPS Vertical Board 9-15	0
LRP-PNL-(9-17)	CNS Screening Evaluation Worksheet Ch B PCIS & RPS Vertical Board 9-17	0
NEDC-31858P-A	BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems	08/1999
NLS8800347	Response to NRC Bulletin 88-04	07/08/1988
NPPD Submittal to NRC	IE Bulletin 80-14, Degradation of BWR Scram Discharge Volume Capability Proposed Technical Specifications	10/21/1980

Miscellaneous Number	Title	Revision or Date
Procedure Change Request	6.RHR.308	10/09/2014
Procedure Change Request	6.1DG.302	03/15/2017
Procedure Change Request	6.1SW.301	10/10/2017
R3-764-9	Qualification Test Report, ITT Barton Model 764 Differential Pressure Electronic Transmtter	10/05/1982
Screening Evaluation Worksheet	4160V Switchgear	02/09/1996
TAC No. M69439	Plant Specific Evaluation Report for USI A-46 Program Implementation at Cooper Nuclear Station	09/30/1999
TQD # 0655ELE, SAP #552	ERO – Electrical Maintenance OSC Pool Personnel	6
TQD # 0655IAC, SAP #1126	ERO – IAC Maintenance OSC Pool Personnel	3
TQD #0655MEC, SAP # 154	ERO – Mechanical Maintenance OSC Pool Personnel	9
TQD #0655UTL, SAP #953	ERO – Utility Maintenance OSC Pool Personnel	4

Vendor Documents Number	Title	Revision or Date
4500187217	Certificate of Conformance Backdraft Dampers	0
4500187217	Purchase Order Backdraft Dampers	12/01/2015
GE 234A9303NS	Sheet 22: Instrument Data Sheet, Scram Discharge Level	6
GE Application Note 200.60	GE-MOV® Varistors – Voltage Tansient Suppressors	08/75
GE Application Note 200.72	Using GE-MOV® Varistors to Extend Contact Life	08/75
GE Catalog Cut	HFA 100 Multicontact Auxiliary Relays	
GE SIL 044S4R2	HFA Relay Magnetic Coil Assembly Replacement and Relay Adjustments	07/10/84
GE Transient Voltage Suppression Manual	GE-MOV® II Metal Oxide Varistors for Transient Voltage Protection, Ratings and Characteristics Table: L Series [Data Sheet, p. 139]	3rd Edition
GEH 2024F	Multi-Contact Auxiliary Relay Type HFA 51	02/99

Vendor Documents Number	Title	Revision or Date
GEI 68749B	Multi-Contact Auxiliary Relay Types HFA65D & HFA65E	06/95
GEK 45484	Multi-Contact Auxiliary Relay Type HFA151	03/94
GEK 45486A	Multi-Contact Auxiliary Relay Type HFA154	11/88
VM-0004	RHR Pumps	12
VM-1750	GE Relay Composite Manual	16
VM-1763	GE Power Circuit Breaker Composite Manual	7

DETAILED RISK ANALYSIS FOR NCV 2018011-02

Failure to Ensure Adequate Design Control Measures are in Place Associated with RHR Service Water Booster Pump Room Cooling

For the detailed risk evaluation, the analyst changed the base condition of the Cooper SPAR model to reflect a hypothetical, single failure proof configuration where two temperature elements existed for Control Building basement high room temperature annunciation and two temperature elements existed for emergency cooler fan actuation. The temperature elements were modeled with failure rates which were consistent with other temperature elements in the SPAR Template 2015, Version 1, with a value of 1.7E-3. Specific changes the analyst made were:

- Sub-fault trees for Residual Heat Removal Service Water Booster pump failures resulting from high room temperature were added. This sub-fault trees included logic which modeled that the Residual Heat Removal Service Water Booster pumps would fail due to high room temperature in the Control Building basement when the following failed:
    - Normal room cooling,
    - Emergency room cooling, and
    - Alternate room cooling

- The sub-fault tree logic represented normal cooling failing on:
    - Losses of offsite power
    - Failures to run
    - Failures to start
    - Being in test and maintenance

- The sub-fault tree logic represented emergency cooling failing on:
    - Failures to run
    - Failures to start
    - Being in test and maintenance
    - Failures of two temperature elements which start the emergency fan (both independent and common cause failures)

- The sub-fault tree logic represented alternate room cooling failing on:
    - Operators failing to take timely or any action to establish alternate room cooling
    - Failures of two temperature elements which actuate the room temperature annunciator to cue the operator to take action (both independent and common cause failures)

The performance deficiency was then modeled by applying “ignore” failure model logic to the redundant temperature element basic events and the temperature element common cause failure basic events to represent the existing, non-single failure proof configuration. Also, the analyst increased the failure rate of the remaining sole temperature elements to that of the 95th percentile of the distribution of temperature elements in the distribution from SPAR Template 2015, Version 1, with a value of 3.33E-3, to represent a lower quality standard for the existing temperature elements. These changes when applied over a one-year exposure period yielded and estimate of the increase in core damage frequency 1.7E-8/year.

The dominant core damage sequences were losses of offsite power. Operator action to recover suppression pool cooling and operator action to vent containment were the remaining means to mitigate the initiators. These analyses were run on the Cooper SPAR model, Version 8.55, as modified above, ran on SAPHIRE 8.1.6. Because the increase in core damage frequency was less than 1.0E-7/year, the increase in core damage frequency from external events and the increase in large early release frequency were not analyzed.

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