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

July 20, 2012

Brian J. O'Grady, Vice President-Nuclear
and Chief Nuclear Officer
Nebraska Public Power – Cooper
Nuclear Station
72676 648A Avenue
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION – NRC COMPONENT DESIGN BASES INSPECTION, NRC INSPECTION REPORT 05000298/2012007

Dear Mr. O'Grady:

On April 4, 2012, the US Nuclear Regulatory Commission (NRC) completed a Component Design Bases Inspection at your Cooper Nuclear Station. The enclosed report documents our inspection findings. The preliminary findings were discussed on April 4, 2012, with Mr. D. Buman, Director of Engineering, and other members of your staff. After additional in-office inspection, a final telephonic exit meeting was conducted on June 8, 2012, with Mr. D. Buman, Director of Engineering, Mr. A. Zaremaba, Director of Nuclear Safety Assurance, and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The team reviewed selected procedures and records, observed activities, and interviewed cognizant plant personnel.

Based on the results of this inspection, the NRC has identified six findings that were evaluated under the risk significance determination process. Violations were associated with all of the findings. All of the findings were found to have very low safety significance (Green) and the violations associated with these findings are being treated as noncited violations, consistent with the NRC Enforcement Policy.

If you contest any of the noncited violations, or the significance of the violations you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the US Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 1600 East Lamar Blvd., Arlington, Texas 76011; the Director, Office of Enforcement, US Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Cooper Nuclear Station. The information you provide will be considered in accordance with Inspection Manual Chapter 0305. In addition, if you disagree with the characterization of the crosscutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV, and the NRC Resident Inspector at Cooper Nuclear Station.

In accordance with Code of Federal Regulations, Title 10, Part 2.390 of the NRC's Rules of Practice, a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

T.R.F.

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

Docket: 50-298
License: DRP-46

Enclosure:
NRC Inspection Report 05000298/2012007
w/Attachment:
Supplemental Information

cc w/Enclosure:
Electronic Distribution for Cooper Nuclear Station

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-298
License:	DRP-46
Report:	05000298/2012007
Licensee:	Nebraska Public Power District
Facility:	Cooper Nuclear Station
Location:	72676 648A Ave Brownville, NE 68321
Dates:	March 5 through June 8, 2012
Team Leader:	R. Kopriva, Senior Reactor Inspector, Engineering Branch 1, Region IV
Inspectors:	R. Latta, Senior Reactor Inspector, Engineering Branch 1, Region IV M. Williams, Reactor Inspector, Engineering Branch 1, Region IV B. Correll, Reactor Inspector, Engineering Branch 2, Region IV S. Garchow, Senior Operations Engineer, Operations Branch, Region IV
Accompanying Personnel:	V. Ferrarini, Structural Contractor, Beckman and Associates G. Morris, Electrical Contractor, Beckman and Associates
Approved By:	Thomas R. Farnholtz, Chief Engineering Branch 1

SUMMARY OF FINDINGS

IR 05000298/2012007; 03/05/2012 – 06/08/2012; Cooper Nuclear Station, baseline inspection, NRC Inspection Procedure 71111.21, "Component Design Basis Inspection."

The report covers an announced inspection by a team of five regional inspectors and two contractors. Six findings were identified. All of the findings were of very low safety significance. The final significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified Findings

Cornerstone: Mitigating Systems

Green. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions." Specifically, prior to March 8, 2012, the licensee failed to incorporate the seismic/barge impact loadings using a +Y (vertical up) component in combination with the lateral loads, which would result in the highest concrete anchor bolt interaction, into Calculation NEDC 12-20 for the service water instrument rack. Also, the calculation incorrectly utilized a factor of safety of four for the anchor bolts, where as the Updated Safety Analysis Report, Appendix C 2, Section 2.10, specified a factor of safety of five. This finding was entered into the licensee's corrective action program as Condition Report CR-CNS-2012-01665.

The team determined that the failure to incorporate the seismic/barge impact loadings using a +Y (vertical up) component in combination with the lateral loads into calculation NEDC 12-20, and using an incorrect safety factor for the instrument rack anchor bolts 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 of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with NRC Inspection Manual Chapter 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," the issue was determined to have very low safety significance (Green) because it was a design deficiency confirmed not to result in loss of operability or functionality. Specifically, the licensee revised the associated calculations to include the correct required standards, and the calculation was acceptable. This finding was determined to have a crosscutting aspect in the area of human performance, associated with the work practices component because the licensee did not ensure that supervisory or management oversight of the work activities, including contractors, were such that nuclear safety was supported [H.4(c)] (Section 1R21.2.5).

Green. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," which states, in part, "A program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate acceptance limits contained in applicable documents." Specifically, prior to April 4, 2012, for the Startup Station Service Transformer (SSST), the licensee did not use the actual measured bus bar resistance values which exceeded the calculated values. This resulted in non-conservative values used in Calculation NEDC 00-003, which did not bound actual plant parameters. Also, for the Emergency Station Service Transformer (ESST), the current procedure has a resistance acceptance tolerance specified as 1 Ohm, and in Condition Report CR-CNS-2011-11750, the licensee found the actual measured value was in the milliohms, which should have been used as the acceptance criteria in the procedure. This finding was entered into the licensee's corrective action program as Condition Reports CR-CNS-2012-02358 and CR-CNS-2012-02359.

The team determined that the failure to provide adequate acceptance criteria for the bus duct resistance for the Emergency Station Service Transformer and the Startup Station Service Transformer was a performance deficiency. This finding was more than minor because it was associated with the test control attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with NRC Inspection Manual Chapter 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," the issue was determined to have very low safety significance (Green) because it was a test deficiency confirmed not to result in loss of operability or functionality. Specifically, the licensee performed an engineering justification for the bus resistance acceptance criteria based on the difference between the as measured resistance values and those values used in the voltage regulation study, and found the values acceptable. This finding was determined to have a cross-cutting aspect in the area of human performance associated with the decision making component because the licensee did not use conservative assumptions in decision making and adopt a requirement to demonstrate that the proposed action is safe in order to proceed rather than a requirement to demonstrate that it is unsafe in order to disapprove the action [H.1(b)] (Section 1R21.2.13).

Green. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," which states, in part, "measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition." Specifically, in 2005, the licensee performed a review of the "C" swing battery charger disconnect switch fuses and their ratings, documented in Condition Report CR-CNS-2005-09378. However, the actions associated with this Condition Report did not evaluate the Updated Safety Analysis Report emergency event function which states that each battery charger shall have adequate capacity to restore its battery to full charge from a totally discharged condition while carrying the normal station steady state direct current load. This finding was entered into the licensee's corrective action program as Condition Report CR-CNS-2012-01611.

The team determined that the failure to adequately assess all design requirements during the review of Condition Report CR-CNS-2005-09378 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 of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with NRC Inspection Manual Chapter 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," the team determined that the finding represented a loss of system safety function requiring a Phase 2 evaluation. The Region IV Senior Reactor Analyst concluded that a Phase 3 evaluation was needed to address the issue because it departed from the guidance provided for Phase 1 or Phase 2. Using NRC Inspection Manual Chapter 0609, and Standardized Plant Analysis Risk (SPAR) model, the Senior Reactor Analyst identified that the frequency of events where the defective swing charger would affect core damage sequences were very low, that a station blackout restored by offsite power within one hour would not be expected to result in a failure of the swing charger, and it would be likely that the other battery charger would successfully charge the associated direct current bus and battery and result in a successful recovery. Therefore, the issue was determined to have very low significance (Green). This finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance (Section 1R21.2.14).

<u>Green</u>. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The 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 4, 2012, the licensee failed to perform an adequate review of the design basis requirements to establish a preventive maintenance program for molded case circuit breakers located in the safety-related station battery chargers and important to safety battery inverters. This finding was entered into the licensee's corrective action program as Condition Reports CR-CNS-2012-1647 and CR-CNS-2012-1664.

The team determined that the failure to adequately review the design basis requirements, and not establishing a preventive maintenance program for molded case circuit breakers located in the safety-related station battery chargers and important to safety battery inverters, was a performance deficiency. This finding was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with NRC Inspection Manual Chapter 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," the issue was determined to have very low safety significance (Green) because it was a design deficiency confirmed not to result in loss of operability or functionality. Specifically, there have not been any failures of these molded case circuit breakers attributed to lack of preventative maintenance. This finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance (Section 1R21.2.15).

Green. The team identified a Green noncited violation of Technical Specification 5.4.1.a, which states, in part, "Written procedures shall be established, implemented, and maintained, covering the procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A.6.w, Acts of Nature (e.g., tornado, flood, dam failure, earthquakes)." Specifically, prior to April 4, 2012, the licensee failed to maintain Procedure 7.0.11, Flood Control Barriers, Revision 24, to ensure the materials required to construct flood protection barriers were correctly listed and inventoried, to effectively protect personnel and equipment doors around the perimeter of the facility. This finding was entered into the licensee's corrective action program as Condition Report CR-CNS-2012-01920.

The team determined that the failure to maintain Cooper Nuclear Station Operations Procedure 7.0.11, "Flood Control Barriers," Revision 24, with an adequate inventory of required materials listed in the procedure, was a performance deficiency. This finding was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems Cornerstone, and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with NRC Inspection Manual Chapter 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," the team determined that the finding was potentially risk significant due to a seismic, flooding, or severe weather initiating event and a Phase 3 analysis was required. A Region IV Senior Reactor Analyst performed a Phase 3 significance determination using NRC Inspection Manual Chapter 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria." In accordance with Appendix M, the Senior Reactor Analyst determined that although it is not certain that the licensee could erect all of the flood barriers within 72 hours, it is likely that they could finish barriers to the emergency diesel generators and emergency core cooling systems in time to provide vital power and injection capabilities within the time required. Also, it is likely that extraordinary efforts could be taken to complete the barriers if the licensee was falling behind their time line, with knowledge of the timing of the arrival of flood waters. The failure of the Missouri River dams would most likely begin with incipient failure symptoms, providing extra time for the licensee to stage and prepare for the erection of barriers. Therefore, the issue was determined to have very low safety significance (Green). This finding was determined to have a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee failed to take appropriate corrective actions to address safety issues and adverse trends in a timely manner, commensurate with their safety significance and complexity [P.1(d)] (Section 1R21.2.16).

Green. The team identified a Green noncited violation, with two examples, 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 instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished." Specifically, prior to April 4, 2012, the licensee did not follow the requirements of Cooper Nuclear Station Operations Manual Administrative Procedure 0.5.OPS, "Operations Review of Condition Reports/Operability Determination," Section 6 "Prompt Determination," Step 6.1.1.6. This step requires the use of Attachment 3, Item 3, which addresses design basis assumptions, descriptions, calculations, or values used in the Cooper Nuclear Station Updated Safety Analysis Report shall be used to ensure all aspects of the condition are addressed. For two, separate, Prompt Operability Determinations, one for the standby liquid control test tank, and the second one for the standby liquid control tank, the licensee had not considered the effect of vertical seismic loading in their calculation as identified in the Updated Safety Analysis Report (Table -3-7 page C-3-73). These findings were entered into the licensee's corrective action program as Condition Reports CR-CNS-2012-001214, CR-CNS-2012-001232, CR-CNS-2012-001651, CR-CNS-2012-001918 and CR-CNS-2012-01962.

The team determined that the failure to follow the requirements of Cooper Nuclear station Operations Manual Administrative Procedure 0.5.OPS, "Operations Review of Condition Reports/Operability Determination," Step 6.1.1.6, was a performance deficiency. This finding was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with NRC Inspection Manual Chapter 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," the issue was determined to have very low safety significance (Green) because it was a design deficiency confirmed not to result in loss of operability or functionality. Specifically, the licensee revised the associated calculations to include the correct required standards, with acceptable results. This finding was determined to have a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee failed to properly classify, prioritize, and evaluate for operability and reportability, conditions adverse to quality [P.1(c)] (Section 1R21.3.5).

B. Licensee-Identified Violations

No findings were identified.

REPORT DETAILS

1 REACTOR SAFETY

Inspection of component design bases verifies the initial design and subsequent modifications and provides monitoring of the capability of the selected components and operator actions to perform their design bases functions. As plants age, their design bases may be difficult to determine and important design features may be altered or disabled during modifications. 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 Bases Inspection (71111.21)

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 Assessment 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.

.1 Inspection Scope

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 bases 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 the review of operator actions, 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 bases 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;

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 samples that include risk-significant and low design margin components, containment-related components, and operating experience issues. The sample selection for this inspection was 17 components, one of which is containment-related, six operating experience items, and four Event Scenario-Based activities. The selected inspection and associated operating experience items supported risk significant functions including the following:

a. Electrical power to mitigation systems: 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 safety-related and important to safety 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 and without offsite power available. As such the team selected:

- 125 Vdc Battery and Battery Charger 1B
- 125 and 250 Vdc Swing Battery Chargers 1C
- 125 Vdc Bus 1B
- Offsite power as a Loss of Offsite Power Initiator
- 4160 Vac Bus 1G
- Portable Emergency Diesel Generator
- Division II Critical 460Vac Motor Control Center Y

b. Mitigating systems needed to attain safe shutdown. The team reviewed components required to perform the safe shutdown of the plant. As such the team selected:

- High Pressure Core Injection Steam Emission Valve (HPCI-MOV-014)
- High Pressure Core Injection Suction Valve From Torus (HPCI-MOV-058)
- High Pressure Core Injection Pump Governor Valve
- Containment Hardened Vent
- B Train Residual Heat Removal Minimum Flow Recirculation Valve (RHR-MOV-16B)
- High Pressure Core Injection Steam Admission Valve (MO14)
- Service Water Pump 1 B
- RHR SW Booster Pump Motor 1B

.2 Results of Detailed Reviews for Components

.2.1 High Pressure Core Injection (HPCI) Steam Emission Valve (HPCI-MOV-014)

Inspection Scope

The team reviewed the updated final safety analysis report, technical specification requirements and limiting conditions for operation, system description, the current system health report, selected drawings, relevant maintenance and test procedures, and condition reports associated with the high pressure core injection steam admission valve, HCPI-MOV-014. Also reviewed were design bases documents, calculations, and conducted component inspection to assess the adequacy of the motor and actuator for the valve. Specifically, the team reviewed control circuit schematics, voltage drop calculations, motor sizing data, and overall condition of the motor actuator. The team also performed system walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its intended design basis function. Specifically, the team reviewed:

- System piping and instrumentation drawings for the high pressure core injection system.
- Valve vendor manual and correspondence file information for the motor.
- System design basis documents and system modifications.
- Preventive maintenance procedures and schedule for valve and motor.
- Valve thrust requirements for design/licensing basis conditions.
- Motor operated valve torque test procedure and torque test trend data.
- Completion of preventive maintenance work orders for motor starter breaker testing.
- Calculations for determining minimum motor terminal voltage under design/licensing basis conditions.
- Calculations for determining minimum contactor terminal voltage under design/licensing basis conditions.
- Calculations for the motor starter breaker and motor thermal overload heater selection.
- Environmental design requirements under design/licensing basis conditions.
- Calculations for limiting component analysis (weak link analysis) including seismic loads for valve HPCI-MOV-014.
- Pipe stress calculation containing valve HPCI-MOV-014.
- Pipe stress isometric drawing.

b. <u>Findings</u>

No findings were identified.

.2.2 <u>High Pressure Core Injection Suction Valve from Torus (HPCI-MOV-058)</u>

a. <u>Inspection Scope</u>

The team reviewed the updated final safety analysis report, technical specification requirements and limiting conditions for operation, system description, the current system health report, selected drawings, relevant maintenance and test procedures, and condition reports associated with the high pressure core injection suction valve from the torus, HCPI-MOV-058. The team also performed system walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its intended design basis function. Further, the team reviewed design bases documents, calculations, and conducted component inspection to assess the adequacy of the motor and actuator for the valve. Included in these reviews were the control circuit schematics, voltage drop calculations, motor sizing data, and overall condition of the motor actuator. Specifically, the team reviewed:

- System piping and instrumentation drawings for the high pressure core injection suction valve from the torus.
- Valve vendor manual and correspondence file information for the motor.
- System design basis documents and system modifications.
- Preventive maintenance procedures and schedule for valve and motor.
- Valve thrust requirements for design/licensing basis conditions.
- Motor operated valve torque test procedure and torque test trend data.
- Completion of preventive maintenance work orders for motor starter breaker testing.
- Calculations for determining minimum motor terminal voltage under design/licensing basis conditions.
- Calculations for determining minimum contactor terminal voltage under design/licensing basis conditions.
- Calculations for the motor starter breaker and motor thermal overload heater selection.
- Environmental design requirements under design/licensing basis conditions.
- Calculations for limiting component analysis (weak link analysis) including seismic loads for valve HPCI-MOV-058.
- Pipe stress calculation containing valve HPCI-MOV-058.
- Pipe stress isometric drawing.
- Piping loads on Torus.

b. <u>Findings</u>

No findings were identified.

.2.3 <u>High Pressure Core Injection Pump Governor Valve</u>

a. <u>Inspection Scope</u>

The team reviewed the updated final safety analysis report, system description, the current system health report, selected drawings, relevant maintenance and test procedures, and condition reports associated with the high pressure core injection pump governor valve. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its intended design basis function. Specifically, the team reviewed:

- Valve assembly drawings and component vendor manual including correspondence file information.
- Preventive and corrective work orders completed.
- Operational history including related industry operational experience and condition reports.

- Mechanical over speed trip device vendor manual and system description information.
- Mechanical over speed trip device operational experience reports.

b. Findings

No findings were identified.

.2.4 Primary Containment Purge, Makeup, and Vent System

a. Inspection Scope

The team reviewed the updated final safety analysis report, system description, selected drawings, relevant emergency operating procedures, and condition reports associated with the primary containment purge, makeup, and vent system. Additionally, the team performed system walkdowns and conducted interviews with system engineering and operations personnel to ensure the capability of this system to perform its intended design basis function. The team also observed operator actions in response to a station blackout event in the simulator. Specifically, the team reviewed:

- Design and installation of the torus hard pipe vent system including seismic supports and equipment maintenance history.
- Operator actions associated with the primary containment purge, makeup, and vent system during station blackout conditions including equipment accessibility.
- Operator knowledge and understanding of alternate core cooling mitigation strategies including primary containment venting and hydrogen control.
- Containment isolation signal override functions associated with the torus purge and vent valve supply line.
- Pipe stress analysis and connection to torus analysis.

b. Findings

No findings were identified.

.2.5 Service Water Pump 1B

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 Service Water Pump 1B. 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:

- Seismic/Barge impact calculations for instrument racks.
- Seismic/Barge impact analysis for pump and pump anchorage.

- Preventive maintenance procedures for the motor.
- Vendor manual, nameplate data, and specifications for the pump.
- System design basis documents.
- Calculations for determining minimum motor terminal voltage under design/licensing basis conditions.
- Calculations for determining minimum contactor terminal voltage under design/licensing basis conditions.
- Calculations for the motor starter breaker and motor thermal overload heater selection.
- Environmental design requirements under design/licensing basis conditions.

b. Findings

Introduction. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," involving the seismic calculation for the service water pump instrument rack. Specifically, on March 6, 2012, the licensee failed to correctly incorporate the required safety factor and torqueing requirements in the bolting of the service water pump instrumentation rack.

Description. As a result of the walkdown activities associated with the inspection, the licensee identified that one of the support legs of the service water instrument rack (Class 1 seismic structure), located in the Service Water Pump room, was corroded to the extent that it would no longer support the rack under all required loading conditions. The system was declared inoperable and a temporary design was implemented per TCC 4881013. Calculation NEDC 12-020 Rev. 0 was issued to justify the temporary modification. The team reviewed the temporary modification support calculation and identified some discrepancies.

Based on the results of this review, the team determined that the calculation did not consider the most limiting loading combination for the structure and its components. The most limiting component in the modified structure are the concrete anchor bolts connecting the modified structure to the concrete floor (all remaining components have a significantly higher safety margin). The calculation did not consider the seismic/barge impact loadings using a +Y (vertical up) component in combination with the lateral loads which would result in the highest concrete anchor bolt interaction. These anchor bolts are a pair of 3/8" Hilti HDI Drop-In anchors and are subject to a factor of safety of 5 as identified in the Updated Safety Analysis Report, Appendix C-2, Section 2.10. However, calculation NEDC 12-020 Rev. 0 incorrectly utilized a factor of safety of 4 for these bolts. The calculation was updated in Rev 1 to include the most limiting load case and utilized a factor of safety of 5 for the concrete anchor bolts.

The team reviewed Rev 1 of the calculation which in addition to the inclusion of the proper load combination and the factor of safety of 5 also included the as-built support review. The installed configuration modified the torque values for four (4) ½ inch bolts from 62 ft/lbs to 22 ft/lbs without justification. The site issued NEDC 12-020 Rev. 2 which satisfactorily addressed the team's concerns and concluded that the structure and anchor bolts were adequate to resist the updated load combinations using the correct factor of safety of 5 and reduced torque.

Analysis. The team determined that the failure to incorporate the seismic/barge impact loadings using a +Y (vertical up) component in combination with the lateral loads into calculation NEDC 12-20, and using an incorrect safety factor for the instrument rack anchor bolts 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 of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with NRC Inspection Manual Chapter 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," the issue was determined to have very low safety significance (Green) because it was a design deficiency confirmed not to result in loss of operability or functionality. Specifically, the licensee revised the associated calculations to include the correct required standards, and the calculation was acceptable. This finding was determined to have a crosscutting aspect in the area of human performance, associated with the work practices component because the licensee did not ensure that supervisory or management oversight of the work activities, including contractors, were such that nuclear safety was supported [H.4(c)].

Enforcement. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions." Contrary to the above, the licensee failed to ensure that measures were established to assure that applicable design basis were correctly translated into specifications, drawings, procedures, and instructions. Specifically, prior to March 8, 2012, the licensee failed to incorporate the seismic/barge impact loadings using a +Y (vertical up) component in combination with the lateral loads, which would result in the highest concrete anchor bolt interaction, into Calculation NEDC 12-20 for the service water instrument rack. Also, the calculation incorrectly utilized a factor of safety of four for the anchor bolts, where as the Updated Safety Analysis Report, Appendix C 2, Section 2.10, specified a factor of safety of five. This finding was entered into the licensee's corrective action program as Condition Report CR-CNS-2012-01665. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a noncited violation consistent with the NRC Enforcement Policy: NCV 5000298/2012007-01, "Failure to Adequately Analyze Seismic Requirements for Service Water Instrument Rack."

.2.6 Residual Heat Removal Service Water Booster Pump Check Valve (CV-20)

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 check valve (CV-20). 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.
- Listing of condition reports for the past three years
- Piping and instrumentation diagram for the residual heat removal service water booster pump check valve (CV-20).

b. Findings

No findings were identified.

.2.7 "B" Train 125 Vdc Battery

a. Inspection Scope

The team reviewed the updated safety analysis report, design bases documents, calculations, corrective and preventative maintenance, and testing of the safety-related, Division 2 250 V battery 1B, performed a walk down of the battery and associated components, and interviewed the system engineer. The team also reviewed alternating and direct current one-line diagrams, protective circuits, coordination curves, vendor manuals, maintenance procedures, pilot cell selection criteria and selection history, and conducted a conference call with the battery vendor on March 28, 2012. The team performed walk downs of the 125 V and 250 V battery chargers. Specifically, the team reviewed:

- Technical specification requirements.
- Battery sizing, voltage drop, and short circuit calculation.
- Electrical schematics.
- Battery installation drawings.
- Service and performance discharge testing.
- Previous three modified performance discharge test results.
- Battery rack and mounting calculation.

b. Findings

No findings were identified.

.2.8 4160 Vac Bus 1G

a. Inspection Scope

The team reviewed the updated safety analysis report, main and 4160 Vac switchgear one-line diagrams, selected 4160 Vac, safety related switchgear circuit breaker elementary diagrams, load sequencing timing relays, undervoltage relay setpoints assorted calculations, manufacturer's information and related American National Standards Institute (ANSI) standards to ensure there was adequate voltage at the 4160 volt buses and adequate interrupting capability in case of a short circuit on the bus. The team conducted walk downs of the switchgear room, the simulator and the main control room. Specifically, the team reviewed:

- Vendor installation instructions.
- Past maintenance records for the last three years.
- Surveillance procedures and surveillance results.
- Listing of condition reports for the past three years.
- System design basis documents and system modifications.
- Environmental design requirements under design/licensing basis conditions.

b. Findings

No findings were identified.

.2.9 Portable Emergency Diesel Generator

a. Inspection Scope

The team performed a walk down of the portable diesel generator, reviewed drawings, the vendor manual and procedures for its use. The team reviewed the diesel generator capacity (175 kW) to ensure it was adequately sized for the intended use with the two Train C (swing) battery chargers. The team confirmed the procedure provided the connection to 480 V buses and that there were pre-staged cables. Specifically, the team reviewed:

- Vendor installation instructions.
- Past maintenance records for the last three years.
- Surveillance procedures and surveillance results.
- Listing of condition reports for the past three years.
- System design basis documents and system modifications.
- Preventive maintenance procedures and schedule for the motor.
- Completion of preventive maintenance work orders for motor starter breaker testing.
- Calculations for determining minimum motor terminal voltage under design/licensing basis conditions.
- Calculations for determining minimum contactor terminal voltage under design/licensing basis conditions.
- Calculations for the motor starter breaker.
- Environmental design requirements under design/licensing basis conditions.

b. Findings

No findings were identified.

.2.10 "B" Train Residual Heat Removal Minimum Flow Recirculation Vave (RHR-MOV-16B)

a. Inspection Scope

The team reviewed the updated safety analysis report, design bases documents, calculations, and conducted component inspection to assess the adequacy of the motor and actuator for the valve. Specifically, the team reviewed:

- Control circuit schematics.
- Voltage drop calculations.
- Motor sizing data.
- Flow and time delay setpoints.
- Operating relay schematics.
- Overall condition of the motor actuator.
- Calculations for limiting component analysis (weak link analysis) including seismic loads for valve RHR-MOV-16B.

b. Findings

No findings were identified.

.2.11 Division II Critical 460 Vac Motor Control Center "Y"

a. Inspection Scope

The team reviewed the updated safety analysis report, design bases documents, calculations, corrective and preventative maintenance, and testing of the essential 460 Vac motor control center Y. Finally, the team performed a visual non-intrusive inspection to assess the installation configuration, material condition, and potential vulnerability to hazards. Specifically, the team reviewed:

- System health reports, component maintenance history and licensee's corrective action program reports to verify the monitoring and correction of potential degradation.
- Calculations for electrical distribution system load flow/voltage drop, short-circuit, and electrical protection and coordination to assess the adequacy and appropriateness of design assumptions and to verify that bus capacity was not exceeded and bus voltages remained above minimum acceptable values to support transmission of power to downstream safety-related 460 Vac.
- The protective device settings and circuit breaker ratings; to ensure adequate selective protection coordination of connected equipment during worst-case, short-circuit conditions to ensure continuity of power to downstream safety-related buses.
- Circuit breaker preventive maintenance inspection and testing procedures; to determine adequacy relative to industry and vendor recommendations.

b. Findings

No findings were identified.

.2.12 Residual Heat Removal Service Water Booster Pump Motor (CED 6025080)

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, the current system health report, selected drawings, maintenance and test procedures, protection relay setting sheets, coordination calculation and condition reports associated with residual heat removal service water booster 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:

- Vendor installation instructions.
- Past maintenance records for the last three years.
- Surveillance procedures and surveillance results.
- Listing of condition reports for the past three years.
- System design basis documents and system modifications.
- Preventive maintenance procedures and schedule for the motor.
- Completion of preventive maintenance work orders for motor starter breaker testing.
- Calculations for determining minimum motor terminal voltage under design/licensing basis conditions.
- Calculations for determining minimum contactor terminal voltage under design/licensing basis conditions.
- Calculations for the motor starter breaker.
- Environmental design requirements under design/licensing basis conditions.

b. Findings

No findings were identified.

.2.13 Offsite Power

a. Inspection Scope

The team reviewed the offsite power interface with the Cooper Nuclear Station as a Loss of Offsite Power initiator. The team reviewed the Operational Interface Agreement (OIA) required by the North American Electric Reliability Corporation (NERC) Reliability Standard NUC-01-02 , "Nuclear Plant Interface," and confirmed the OIA was in place with Nebraska Public Power District transmission department. The team walked down the upgrades to the 345 kV, 161 kV and 69 kV switchyards, the proposed upgrades to the Large Power Transformers (Main Power Transformers) with an installed spare and new fire walls between phases, future replacements planned for the Normal Station Service Transformer, the Start-up Station Service Transformer (SSST) and the Emergency Station Service Transformer (ESST). The team reviewed selected power transformer nameplates, nameplate drawings and transformer test data for comparison with data used in the plant voltage regulation and short circuit calculations. The team also reviewed the condition monitoring of the non-segregated phase bus duct.

b. Findings

Introduction. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," for failure to provide adequate resistance values and acceptance criteria in documentation of the non-segregated phase bus ducts connecting the power transformers to the 4160 volt safety-related switchgear.

Description. During the walkdown of the non-segregated phase bus, which provides the power path for the offsite power to the in-plant electric distribution system, the team questioned the licensee about the preventative maintenance criteria identified in the licensee's surveillance test document, Procedure 7.3.41, "Examination and Meggering of Non-Segregated Phase Bus," Revision 7. The team found that the acceptance criteria for the resistance measurement from the switchgear through the transformer was non-conservative. The value identified in the procedure was three orders of magnitude higher that the resistance used in the voltage regulation studies.

The Startup Station Service Transformer (SSST) bus bar resistances were measured in 2009 (WO4458028), using Procedure 7.3.41. Four (4) bus bar circuits were measured, corresponding to the transformer feeds from the Startup Station Service Transformer to four (4) circuit breakers. The resistance values obtained included the transformer windings. When the transformer winding resistance was subtracted from the measured values, the actual resistance measured increased 998 micro-ohms for each bus bar phase. Calculation NEDC 00-003, "Cooper Nuclear Station Auxiliary Power System Load Flow and Voltage Analysis," used inputs to the bus bar resistance values calculated in NEDC 90-368, "Startup Station Service Transformer 4160 V Bus Impedance," for the Startup Station Service Transformer bus bar. The model inputs the bus undervoltage relay trip values, and then calculates the grid voltage necessary to assure that the grid remains tied to the 4160 volt safety related bus during accident/event conditions. The team identified that the actual measured bus bar resistance values exceeded the calculated values, which resulted in the values used in the NEDC 00-003 being non-conservative, not bounding the actual plant parameters, for bus bar resistance. The licensee revised the model used in NEDB 00-003, using the additional bus bar resistance (998 micro-ohms per bus bar phase) and the results indicate that the existing 168 kV value in Procedure 6.EE.610, "Offsite AC Power Alignment," remained acceptable.

Also, for the Emergency Station Service Transformer, the team identified that the NEDC 00-003, "Cooper Nuclear Station Auxiliary Power System Load Flow and Voltage Analysis." should appropriately match/bound the maintenance procedure criteria. The team identified that in Procedure 7.3.41, the resistance value for the Emergency Station Service Transformer (ESST) bus bar was inappropriate. The current procedure has a resistance acceptance tolerance specified as one (1) Ohm. The licensee had identified in Condition Report CR-CNS-2011-11750 that this resistance was not appropriate, and that actual measured values in milliohms should be used as the acceptance criteria. After the team questioned the licensee about the bus bar resistance acceptance criteria, the licensee noted that the corrective actions identified in the condition report had not been incorporated yet, but if the corrective actions had been followed, there would have been a discrepancy between revised Procedure 7.3.41 and NEDC 00-003, "Cooper

Nuclear Station Auxiliary Power System Load Flow and Voltage Analysis." The licensee has issued Condition Reports CR-CNS-2012-02358 and CR-CNS-2012-0259 to address these issues. The licensee performed calculation NEDC 12-013, "Operability Calculation for SSST Bus Impedance Discrepancy" to demonstrate the measured values for duct resistance remained acceptable.

Analysis. The team determined that the failure to provide adequate acceptance criteria for the bus duct resistance for the Emergency Station Service Transformer and the Startup Station Service Transformer was a performance deficiency. This finding was more than minor because it was associated with the test control attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with NRC Inspection Manual Chapter 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," the issue was determined to have very low safety significance (Green) because it was a test deficiency confirmed not to result in loss of operability or functionality. Specifically, the licensee performed an engineering justification for the bus resistance acceptance criteria based on the difference between the as measured resistance values and those values used in the voltage regulation study, and found the values acceptable. This finding was determined to have a cross-cutting aspect in the area of human performance associated with the decision making component because the licensee did not use conservative assumptions in decision making and adopt a requirement to demonstrate that the proposed action is safe in order to proceed rather than a requirement to demonstrate that it is unsafe in order to disapprove the action [H.1(b)].

Enforcement. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," which states, in part, "A program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate acceptance limits contained in applicable documents." Contrary to the above, the licensee failed to incorporate known acceptance limits into written test procedures. Specifically, prior to April 4, 2012, for the Startup Station Service Transformer (SSST), the licensee did not use the actual measured bus bar resistance values which exceeded the calculated values. This resulted in non-conservative values used in Calculation NEDC 00-003, which did not bound actual plant parameters. Also, for the Emergency Station Service Transformer (ESST), the current procedure has a resistance acceptance tolerance specified as 1 Ohm, and in Condition Report CR-CNS-2011-11750, the licensee found the actual measured value was in milliohms, which should have been used as the acceptance criteria in the procedure. This finding was entered into the licensee's corrective action program as Condition Reports CR-CNS-2012-02358 and CR-CNS-2012-02359. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a noncited violation consistent with the NRC Enforcement Policy: NCV 5000298/2012007-02, "Failure to Provide Adequate Resistance Values for the Preventative Maintenance of the Non-Segregated Phase Bus Duct."

2.14 Swing Battery Charger 200 Amp Fuses

a. Inspection Scope

The team reviewed the updated safety analysis report, alternating current (AC) and direct current (DC) one-line diagrams, protective circuits, coordination curves, vendor manuals and maintenance procedures. The team performed walk downs of the 125 V and 250 V battery chargers.

b. Findings

Introduction. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for failure to address the design bases of the battery chargers following identification of an undersized fused disconnect, connecting the "C" (swing) battery chargers to the DC buses.

Description. The licensee enhanced the 125 and 250 Vdc Class 1E systems with the addition of a "C" battery charger (installed by Design Change 87-073) for each system. The addition of the "C" chargers permits the licensee the flexibility to operate with the "C" charger replacing either the Division I or the Division II battery chargers during plant operation, scheduled maintenance, or outages. Upon review of the direct current (DC) one line diagram, the team noticed that the Division I and Division II 200 Amp battery chargers were connected to the direct current bus through a 300 Amp fused disconnect switch whereas the 200 Amp "C" swing charger was connected through a 200 Amp fused disconnect switch. In Condition Report CR-CNS-2005-09378 the licensee had previously questioned the size of the 200 Amp fuse in this application but their evaluation failed to analyze the Updated Safety Analysis Report, Section VIII-6.2.2 requirement which states that each battery charger shall have adequate capacity to restore its battery to full charge from a totally discharged condition while carrying the normal station steady state direct current load. Under this condition, the battery charger will go into its current limit mode drawing 215 Amps. The manufacturer fuse curve for the 250 Vdc battery charger ends at 1000 seconds (16.67 minutes or 0.28 hours) and for the 125 Vdc battery charger ends at 300 seconds (5.0 minutes or 0.083 hours). Based on the manufacture fuse curve, when extrapolated out past the published information, the 250 Vdc fuses may not remain intact during the recharge period. Based on the licensee's information for recent battery recharge evolutions following a discharge test, the battery charger output current remained high, in excess of 200 amperes for more than 2 hours. Industry standards, such as the National Electrical Code, caution about using fuses above 80% of their rating. The team concluded that the capability to recharge the batteries following an event may be challenged.

The licensee declared the "C" swing battery chargers inoperable until further evaluation of the 200 Amp fuses could be performed. The licensee had sent the fuses to a vendor for testing. The licensee also confirmed that during the last three years, both the 125 Vdc and the 250 Vdc "C" swing chargers had been placed in service while one of the Division I or II battery chargers for were out of service. The 125 Vdc and the 250 Vdc "C" swing chargers had been in service for over 300 hours each. The licensee is currently reviewing actions needed to correct the sizing issue with the 200 Amp fused disconnect switches.

Analysis. The team determined that the failure to adequately assess all design requirements during the review of Condition Report CR-CNS-2005-09378 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 of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with NRC Inspection Manual Chapter 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," the team determined that the finding represented a loss of system safety function requiring a Phase 2 evaluation. The Region IV Senior Reactor Analyst concluded that a Phase 3 evaluation was needed to address the issue because it departed from the guidance provided for Phase 1 or Phase 2. Using NRC Inspection Manual Chapter 0609, and Standardized Plant Analysis Risk (SPAR) model, the Senior Reactor Analyst identified that the frequency of events where the defective swing charger would affect core damage sequences were very low, that a station blackout restored by offsite power within one hour would not be expected to result in a failure of the swing charger, and it would be likely that the other battery charger would successfully charge the associated direct current bus and battery and result in a successful recovery. Therefore, the issue was determined to have very low significance (Green). This finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance

Enforcement. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," which states, in part, "measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition." Contrary to the above, the licensee identified a deviation from the original design and did not assure that the condition was not adverse to quality. Specifically, in 2005, the licensee performed a review of the "C" swing battery charger disconnect switch fuses and their ratings, documented in Condition Report CR-CNS-2005-09378. However, the actions associated with this Condition Report did not evaluate the Updated Safety Analysis Report emergency event function which states that each battery charger shall have adequate capacity to restore its battery to full charge from a totally discharged condition while carrying the normal station steady state direct current load. This finding was entered into the licensee's corrective action program as Condition Report CR-CNS-2012-01611. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a noncited violation consistent with the NRC Enforcement Policy: NCV 5000298/2012007-03, "Failure to Address the Design Bases of the Battery Chargers Following Identification of an Undersized Fused Disconnect Switch Connecting the Swing Battery Chargers to the Direct Current (DC) Buses."

.2.15 Molded Case Circuit Breakers

a. Inspection Scope

The team reviewed the one line diagrams for the battery chargers, inverters and motor control centers, vendor documents, industry standards, industry operating experience and preventive maintenance procedures to assess the condition monitoring performed on molded case circuit breakers.

b. Findings

Introduction. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for failure to appropriately evaluate preventive maintenance activities for molded case circuit breakers internal to safety-related and important to safety components, following the review of operating experience.

Description. The team performed an in-depth review of safety-related battery chargers and the important-to-safety battery inverter 1-A, and their internal molded case circuit breakers. While reviewing maintenance and surveillance procedures for this electrical equipment, the team noted that the licensee did not have an established preventive maintenance program, providing assurance that the equipment would operate and function as designed throughout the life expectancy of the equipment.

The NRC Information Notice IN 93-64 identified certain standard molded case circuit breaker tests (such as individual pole resistance, 300-percent thermal overload, and instantaneous magnetic trip tests), performed periodically, were found to be effective along with the additional techniques of infrared temperature measurement and vibration testing. The information notice also stated that "An example of the industry standards that address periodic testing and preventive maintenance is IEEE Std 308-1974, IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations (endorsed by Regulatory Guide 1.32, Revision 2, February 1977)." Section 6.3, as well as Section 7.4.1 of the current (1991) edition of the standard, recommended that periodic tests be performed at scheduled intervals to detect the deterioration of the equipment and to demonstrate operability of the components that are not exercised during normal operation. The information notice also identified numerous other industry documents identifying concerns and requirements for testing of electrical equipment, specifically, molded case circuit breakers.

The Updated Safety Analysis Report, Chapter VIII, "Electrical Power," addresses inspection and testing of equipment in three locations:

- Updated Safety Analysis Report Section VIII-3, "Emergency Power System," Subsection 3.7, "Inspection and Testing," Part 2, states: "Periodic tests/inspections of equipment is performed as defined in maintenance programs to determine equipment operability and functional performance."
- Updated Safety Analysis Report, Section VIII-4, "Auxiliary Power Distribution System," Subsection 7, "Inspection and Testing," Part 2, states: "Periodic tests of the equipment and the system are conducted to: a.) Detect the deterioration of equipment in the system toward an unacceptable condition, and b.) Demonstrate

the capability of normally de energized equipment to perform properly when energized."

- Updated Safety Analysis Report, Section VIII-8, "120-240 Vital AC Power," Subsection 8, "Inspection and Testing," states: "Periodic tests of the equipment and system are conducted to detect the deterioration of the equipment in the system toward an unacceptable condition."

The team identified that the licensee had reviewed generic communication NRC Information Notice IN 93-64, "Periodic Testing and Preventative Maintenance of Molded Case Circuit Breakers." The licensee's evaluation of the information notice was documented in Inter-District memo, McClure to Moeller, dated January 20,1994. This evaluation referred to an earlier evaluation of molded case circuit breakers in licensee memorandum CNSS915709, dated August 19, 1991, and Nebraska Public Power District inter-district memo Horn to Meacham, dated August 7, 1991, which documents a review of molded case circuit breakers at Cooper Nuclear Station, and ruled out required testing of the safety-related battery chargers and the important-to-safety battery inverter 1-A internal molded case circuit breakers. The evaluation specifically eliminated any requirements to perform preventive maintenance for the internal molded case circuit breakers on the basis that there were other electrical protective devices in the circuit (an upstream coordinated fuse) that would operate if the molded case circuit breakers failed to operate on a fault. The licensee considered the molded case circuit breakers as maintenance disconnects. The licensee's analysis, identified in the Nebraska Public Power District memorandum to M. Unruh from R. Krause, dated August 15, 1991, indicated that the NRC Information Notice 93-64 was to demonstrate that the molded case circuit breakers would trip on the time current curve. The licensee did not analyze all potential molded case circuit breaker failures, such as potential premature operation of the molded case circuit breakers. Premature operation of a thermal magnetic molded case circuit breaker is possible if there was a poor contact at a molded case circuit breaker terminal connection.

The NRC also issued Information Notice IN 2008-02, "Findings Identified During Component Design Bases Inspections," which identifies circuit breakers as a component in which there had been numerous operational concerns. The information notice references thirty nine NRC inspection reports pertaining to the components specifically identified in the information notice. Information Notice IN 20008-02 presented another opportunity for the licensee to review specific components for proper operation, testing and maintenance. The licensee entered this issue in their corrective action program as Condition Report CR-CNS 2012-1647 issued for the molded case circuit breakers located in the battery chargers, and Condition Report CR-CNS 2012-1664 issued for the molded case circuit breakers located in the battery inverter 1A including and extent of condition review.

<u>Analysis</u>. The team determined that the failure to adequately review the design basis requirements, and not establishing a preventive maintenance program for molded case circuit breakers located in the safety-related station battery chargers and important to safety battery inverters, was a performance deficiency. This finding was more than minor because it was associated with the procedure quality attribute of the Mitigating

Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with NRC Inspection Manual Chapter 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," the issue was determined to have very low safety significance (Green) because it was a design deficiency confirmed not to result in loss of operability or functionality. Specifically, there have not been any failures of these molded case circuit breakers attributed to lack of preventative maintenance. This finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance.

Enforcement. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The 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, the licensee failed to provide for verifying or checking the adequacy of design by the performance of a suitable testing program. Specifically, prior to April 4, 2012, the licensee failed to perform an adequate review of the design basis requirements to establish a preventive maintenance program for molded case circuit breakers located in the safety-related station battery chargers and important to safety battery inverters. This finding was entered into the licensee's corrective action program as Condition Reports CR-CNS-2012-1647 and CR-CNS-2012-1664. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a noncited violation consistent with the NRC Enforcement Policy: NCV 5000298/2012007-04, "Failure to Establish a Preventative Maintenance Program for Molded Case Circuit Breakers."

.2.16 Flood Related Structures

a. Inspection Scope

The team reviewed the flood control barriers procedure, flood procedure, the dam break analysis, barrier strength calculations, and condition reports associated with flooding protection. The team also performed a walkdown of all primary and secondary barrier locations, and observed the materials and equipment, and their storage locations, for all barrier construction. Additionally, the team spoke with flood protection program members and system engineering personnel to gain understanding of the procedures and implementation strategy. Specifically the team reviewed:

- Station procedures for Flood, and Flood Control Barriers.
- Dam break flooding barrier analysis, dated April 2, 2012, and Rev 1 dated April 3, 2012.
- Photos of previously constructed barriers from 2009, 2010, and 2011.
- Calculations for strength capacity for materials and installation methods.
- Condition reports related to flooding for the past three years.

b. Findings

Introduction. The team identified a Green noncited violation of Technical Specification 5.4.1.a, for failure to establish adequate procedures involving flooding protection. Specifically, as of April 2, 2012, the licensee failed to establish an adequate procedure to supply adequate materials and manpower to complete the installation of plywood barriers and sandbags in order to protect personnel and equipment doorways from flooding during a 72-hour response time for an upstream dam break.

Description. The team reviewed the licensees' dam break flooding barrier analysis, dated April 2, 2012, and Revision 1 dated April 3, 2012, and found that if one of the upstream dams on the river were to break, the site would have approximately 72 hours in which to respond to the break, and complete the installation of plywood barriers and sandbags in order to protect personnel and equipment doorways from flooding.

The team noted that in Cooper Nuclear Station Operations Procedure 7.0.11, "Flood Control Barriers," Revision 24, paragraph 3.1.3.3, the licensee's analysis for adequate reinforcement of the plywood barriers required 2 inch x 4 inch horizontal members on the top and bottom or the barrier, with an additional 2 inch x 4 inch horizontal support placed midway between vertical supports throughout the span. During the review of the procedure, the team identified that the required materials identified in paragraph 3.1.3.3 of Procedure 7.0.11 were not included in Attachment 5, "Bill of Material," and Attachment 6, "Location and Material Requirements." Additionally, Attachment 5 required only 2200 sand bags to complete the erected barriers. Review of the licensees' photos of previously constructed barriers from 2009, 2010, and 2011, revealed that a significantly larger number of sandbags were used in construction of the flood barriers.

When the team questioned the licensee about the significant differences in required materials identified in Procedure 7.0.11 (the plywood barriers and the sandbags), the licensee produced a white paper outlining a flood barrier construction strategy that included 12,700 sand bags, and requiring 64 people to complete flood barrier installation. Upon review of the white paper document, the team pointed out to the licensee, that in order to complete the tasks outlined in the white paper, onsite teams would have to work consecutive shifts (56 hours) without breaks. The licensee then modified the white paper to include 139 individuals dedicated to flood protection barrier construction and installation in order to meet the 72 hour completion time. On April 26, 2012, with inspection team members present, the licensee provided a demonstration of erecting two typical flood barriers. During the 3 hour demonstration, four teams, of two people each, filled approximately 450 sandbags. This rate of filling sand bags equated to less than half the rate identified in the licensees' revised white paper, which was used to support Procedure 7.0.11.

Analysis. The team determined that the failure to maintain Cooper Nuclear Station Operations Procedure 7.0.11, "Flood Control Barriers," Revision 24, with an adequate inventory of required materials listed in the procedure was a performance deficiency. This finding was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems Cornerstone, and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with NRC Inspection Manual Chapter 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," the team determined that the finding was potentially risk significant due to a seismic, flooding, or severe weather initiating event and a Phase 3 analysis was required. A Region IV Senior Reactor Analyst performed a Phase 3 significance determination using NRC Inspection Manual Chapter 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria." In accordance with Appendix M, the Senior Reactor Analyst determined that although it is not certain that the licensee could erect all of the flood barriers within 72 hours, it is likely that they could finish barriers to the emergency diesel generators and emergency core cooling systems in time to provide vital power and injection capabilities within the time required. Also, it is likely that extraordinary efforts could be taken to complete the barriers if the licensee was falling behind their time line, with knowledge of the timing of the arrival of flood waters. The failure of the Missouri River dams would most likely begin with incipient failure symptoms, providing extra time for the licensee to stage and prepare for the erection of barriers. Therefore, the issue was determined to have very low safety significance (Green). This finding was determined to have a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee failed to take appropriate corrective actions to address safety issues and adverse trends in a timely manner, commensurate with their safety significance and complexity [P.1(d)].

Enforcement. The team identified a Green noncited violation of Technical Specification 5.4.1.a, which states, in part, "Written procedures shall be established, implemented, and maintained, covering the procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A.6.w, Acts of Nature (e.g., tornado, flood, dam failure, earthquakes)." Contrary to the above, the licensee failed to maintain a procedure recommended in Regulatory Guide 1.33, Revision 2, Appendix A.6.w. Specifically, prior to April 4, 2012, the licensee failed to maintain Procedure 7.0.11, Flood Control Barriers, Revision 24, to ensure the materials required to construct flood protection barriers were correctly listed and inventoried, to effectively protect personnel and equipment doors around the perimeter of the facility. This finding was entered into the licensee's corrective action program as Condition Report CR-CNS-2012-01920. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a noncited violation consistent with the NRC Enforcement Policy: NCV 5000298/2012007-05, "Failure to Have an Adequate Procedure for Erecting Flood Barriers."

.2.17 Containment Structures

a. Inspection Scope

The team reviewed the period inspection reports done under the Structures Monitoring program for concrete, masonry, seismic gaps, structural and tanks. The team also reviewed the leakrate testing on the primary and secondary containments, and torus inspection packages from the previous two cycles. The team also spoke with the program owners and system engineering personnel to gain understanding of their documentation and program implementation.

b. Findings

No findings were identified.

.3 Results of Reviews for Operating Experience

.3.1 Inspection of Information Notice 2010-25 – "Inadequate Electrical Connections"

a. Inspection Scope

The team reviewed the licensee's response to the information notice, and reviewed a condition report search for inadequate electrical connections for the past five years. Specifically, the search was associated with individual keywords "electrical," "loose," and "connection," and also "loose and connection."

b. Findings

No findings were identified.

.3.2 Inspection of Information Notice 2010-26 – "Submerged Electrical Cables"

a. Inspection Scope

The team reviewed the underground cable raceway drawings, observed the P3 manhole, and reviewed the results of cable insulation resistance measurements for the Service Water pump motors and the Emergency Diesel Generator 2 cables.

b. Findings

No findings were identified.

.3.3 Significant Operating Experience Report (SOER) 10-1 "Power Transformers"

a. Inspection Scope

The team reviewed the licensee's response to the industry recommendations for maintenance, testing and replacement of large power transformers. The licensee had addressed all nine recommendations through their corrective action program using nine different Condition Reports.

b. Findings

No findings were identified.

.3.4 North American Electric Reliability Corporation (NERC) NUC-01-02 "Reliability Standard for Nuclear Plant Interface"

a. Inspection Scope

The team reviewed the Cooper Nuclear Station response to the North American Electric Reliability Corporation Reliability Standard NUC-01-02, Nuclear Plant Interface Coordination. The standard was developed by an industry working group in response to NRC concerns on grid reliability and offsite power. The standard was reviewed by the NRC before being accepted by the Federal Energy Regulatory Authority (FERC) to require a formal agreement on communication between the transmission entity and the nuclear generator. The Nuclear Plant Interface Requirements (NPIR) contained in the agreement are to ensure a greater degree availability of the offsite power supply to supply adequate voltage to the nuclear plant, particularly following a trip of the nuclear unit as required by General Design Criterion 17, Electrical Power.

b. Findings

No findings were identified.

.3.5 NRC Information Notice 2012-01: "Seismic Considerations – Principally Issues Involving Tanks"

a. Inspection Scope

The team reviewed the licensee's evaluation of NRC Information Notice 2012-01, "Seismic Considerations – Principally Issues Involving Tanks," to verify that the review adequately addressed the industry operating experience. The team verified that the licensee's review, documented in Condition Report CR-CNS-2012-1232, adequately addressed the issues in the information notice. The information notice provides examples and references to events in which licensees failed to recognize various seismic considerations and system alignment issues that could impact safety. The NRC staff had identified recent concerns about Standby Liquid Control test tanks that were not seismically qualified when they contained water. The Standby Liquid Control test tank is not safety-related, but are required to be seismically qualified because they could potentially impact nearby safety-related equipment during a seismic event.

b. Findings

Introduction. The team identified a Green noncited violation, with two examples, of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for failure to follow the requirements of Cooper Nuclear station Operations Manual Administrative Procedure 0.5.OPS, "Operations Review of Condition Reports/Operability Determination," step 6.1.1.6, in that they had not reviewed all design and technical data available to be considered and incorporated into the operability evaluations for the Standby Liquid Control tank and test tank.

Description. On February 21, 2012, the resident inspectors brought to the attention of the licensee, the issuance of NRC Information Notice 2012-01, "Seismic Considerations-Principally Issues Involving Tanks," which provided examples and references to events in which licensees failed to recognize various seismic considerations and system alignment issues that could impact safety. The NRC staff had identified recent concerns about Standby Liquid Control test tanks that were not seismically qualified when they contained water. The operating experience identified may apply to other tanks found on site at nuclear plants. The Standby Liquid Control test tanks described in the information notice were not safety-related but were required to be seismically qualified because they could potentially impact nearby safety-related equipment during a seismic event. Incorrect seismic structural analyses or inadequately reviewed procedure changes have led to licensees using tanks, such as the Standby Liquid Control test tanks, in a manner that left them vulnerable to seismic hazards. The operating experience indicated that it is important to verify that the Standby Liquid Control system test tanks and similar tanks have adequate seismic analysis and are procedurally controlled to ensure that seismic vulnerabilities are appropriately managed and that technical specifications are followed.

Example 1: The licensee issued Condition Report CR-CNS-2012-01232 for the Standby Liquid Control Test Tank in response to the NRC information notice. The licensee also reviewed the potential seismic concerns for the main Standby Liquid Control tank, which is identified in Condition Report CR-CNS-2012-01918. During the initial investigation of the seismic calculations for the different tanks, the licensee identified that they did not have a specific calculation pertaining to the seismic concerns of the Standby Liquid Control test tank. The licensee initiated Design Calculation NEDC 12-015 to evaluate both sliding and overturning at the base of each test tank support leg, in the event of seismic activity. The licensee performed an operability evaluation using the requirements identified in Cooper Nuclear station Operations Manual Administrative Procedure 0.5.OPS, "Operations Review of Condition Reports/Operability Determination." The licensee performed the operability evaluation of the test tank assuming the test tank was full of liquid, and concluded that the tank would remain operable. The team reviewed the licensee's evaluation and identified that the licensee had not considered vertical movement in their calculation, as identified in the Updated Safety Analysis Report (Table -3-7 page C-3-73). Step 6.1.1.6, of Procedure 0.5 OPS requires that the licensee review all design and technical data available to be considered and incorporated into the operability evaluations. The licensee re-performed the calculation, incorporating all available design and technical data, and concluded that the test tank was operable.

Example 2: The licensee issued Condition Report CR-CNS-2012-01918 because they had identified that the original Burns and Roe Calculation, for the main Standby Liquid Control tank, BOOK 35, page 51, used a seismic coefficient of 0.46g, and the source of the coefficient could not be identified. The Cooper Nuclear Station Updated Safety Analysis Report (Table -3-7 page C-3-73) states a value of 0.33g for the operating basis earthquake and a value of 0.66g for the safe shutdown earthquake should be used for the seismic coefficients. The licensee revised the calculation to incorporate the new values and found the tank to still be operable. The team reviewed the licensee's seismic calculation and operability evaluation and found that the licensee had not considered vertical movement in their calculation, as identified in the Updated Safety Analysis

Report (Table -3-7 page C-3-73). Step 6.1.1.6, of Procedure 0.5 OPS requires that the licensee review all design and technical data available to be considered and incorporated into the operability evaluations. The team identified that when vertical movement was incorporated into the seismic calculation, the number of bolts holding the tank in place was insufficient. The number of bolts required per the revised calculation was thirteen, where as there were only twelve bolts holding the tank in place. The licensee noted that the controlled drawing for the tank specified 7/8 inch anchor bolts, and the calculation had specified 3/4 inch bolts. The licensee confirmed that 7/8 inch anchor bolts were actually installed, and when the 7/8 inch anchor bolts were used in the calculation, twelve anchor bolts were more than adequate to hold the tank in place.

Analysis. The team determined that the failure to follow the requirements of Cooper Nuclear station Operations Manual Administrative Procedure 0.5.OPS, "Operations Review of Condition Reports/Operability Determination," Step 6.1.1.6, was a performance deficiency. This finding was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with NRC Inspection Manual Chapter 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," the issue was determined to have very low safety significance (Green) because it was a design deficiency confirmed not to result in loss of operability or functionality. Specifically, the licensee revised the associated calculations to include the correct required standards, with acceptable results. This finding was determined to have a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee failed to properly classify, prioritize, and evaluate for operability and reportability, conditions adverse to quality [P.1(c)].

Enforcement. The team identified a Green noncited violation, with two examples, 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 instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished." Contrary to the above, the licensee failed to accomplish specified steps in accordance with an approved procedure. Specifically, prior to April 4, 2012, the licensee did not follow the requirements of Cooper Nuclear Station Operations Manual Administrative Procedure 0.5.OPS, "Operations Review of Condition Reports/Operability Determination," Section 6 "Prompt Determination," Step 6.1.1.6. This step requires the use of Attachment 3, Item 3, which addresses design basis assumptions, descriptions, calculations, or values used in the Cooper Nuclear Station Updated Safety Analysis Report, shall be used to ensure all aspects of the condition are addressed. For two, separate, Prompt Operability Determinations, one for the standby liquid control test tank, and the second one for the main standby liquid control tank, the licensee had not considered the effect of vertical seismic loading in their calculation as identified in the Updated Safety Analysis Report (Table -3-7 page C-3-73). These findings were entered into the licensee's corrective action program as Condition Reports CR-CNS-2012-001214, CR-CNS-2012-001232, CR-CNS-2012-001651, CR-CNS-2012-001918 and CR-CNS-2012-01962. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a noncited violation consistent with the NRC Enforcement Policy: NCV 5000298/2012007-06, "Failure to Incorporate All Design and Technical Data Available into the Operability Determinations for the Standby Liquid Control Tank and Test Tank."

3.6 Inspection of Information Notice 2006-26, "Failure of Magnesium Rotors in MOV Actuators"

a. Inspection Scope

The team reviewed Information Notice 2006-26, which documented recent failures of motor-operated valve (MOV) actuators as a result of galvanic corrosion, general corrosion, and/or thermally induced stress. These failures highlight vulnerabilities of motor actuators with magnesium rotors, particularly when the motor is located in a high humidity and/or high temperature environment. These motor-operated valve failures illustrate the necessity of adequate inspection and/or preventive maintenance on actuators manufactured with magnesium rotors. The team reviewed current inspection work orders instructions and inspection documentation for inspections performed.

b. Findings

No findings were identified.

.4 Results of Reviews for Operator Actions:

The team selected risk-significant components and operator actions for review using information contained in the licensee's probabilistic risk assessment. This included, but was not limited to, components and operator actions that had a risk achievement worth factor greater than two or Birnbaum value greater than 1E-6. Operator actions that do not have written guidance and are not frequently trained on were also considered.

a. Inspection Scope

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:

- Loss of Electrical Power. Open doors for 125V/250V Switchgear A and control panels that don't have open backs. Requirement - Within 30 minutes from time power has been lost.

- Secure High Pressure Core Injection and start Reactor Core Isolation Cooling. Requirement - Within approximately 10 minutes of High Pressure Core Injection operation.

- Restore Service Water cooling to Diesel Generator(s) by starting Service Water pumps in control room or switchgear room. Requirement – Within 5 min.

- De-energize Security System Inverter feed from NBPP and place Sever Accident Mitigating Guidelines Diesel Generator in service. Requirement - Within 4 hours after start of station blackout.

b. Findings

No findings were identified.

4OA6 Meetings, Including Exit

On April 4, 2012, the team leader presented the preliminary inspection results with Mr. D. Buman, Director of Engineering, and other members of your staff. After additional in-office inspection, a final telephonic exit meeting was conducted on June 8, 2012, with Mr. D. Buman, Director of Engineering, Mr. A. Zaremaba, Director of Nuclear Safety Assurance, and other members of your staff. The licensee acknowledged the findings during each meeting. While some proprietary information was reviewed during this inspection, no proprietary information was included in this report.

4OA7 Licensee Identified Violations

No findings were identified.

Attachments: Supplemental Information

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

A. Alexander, Nuclear Support
J. Anderson, Director of Projects
J. Austin, Manager, System Engineering
T. Barker, Manager, Engineering Support
K. Billesbach, Manager, Materials, Purchasing and Contracts
S. Brown, Manager, Planning, Scheduling & Outage
D. Buman, Director, Engineering
B. Chapin, Assistant Manager, Maintenance
L. Dewhirst, Manager, Corrective Actions and Assessment
R. Estrada, Manager, Design Engineering
M. Ferguson, Manager, Human Resources
J. Flaherty, Senior Staff Engineer, Licensing
D. Goodman, Asst Manager, Operations
B. Hasselbring, Supervisor, Operations Control Room
J. Horn, Supervisor, Mechanical Design Engineering
K. Kreifels, Assessment Leader, Quality Assurance
E. McCutchen, Senior Engineer, Licensing
B. Morris, Superintendent, Maintenance Support
J. O'Connor, Manager, Maintenance
R. Penfield, Manager, Operations
R. Schultz, Audit Engineer, Quality Assurance
K. Sutton, Manager, Nuclear Engineering
K. Tanner, Shift Supervisor, Radiation Protection
R. Thacker, Supervisor, Engineering Support
M. Van Winkle, Supervisor, Electrical Design
D. Van Der Kamp, Manager, Licensing
D. Werner, Acting Manager, Training
B. Wolken, Civil Design Engineering
A. Zaremba, Director, Nuclear Safety Assurance

NRC Personnel

C. Henderson, Resident Inspector
J. Josey, Senior Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000298/2012007-01	NCV	Failure to Adequately Analyze Seismic Requirements for Service Water Instrument Rack (1R21.2.5).
05000298/2012007-02	NCV	Failure to Provide Adequate Resistance Values for the Preventative Maintenance of the Non-Segregated Phase Bus Duct (1R21.2.13).
05000298/2012007-03	NCV	Failure to Address the Design Bases of the Battery Chargers Following Identification of an Undersized Fused Disconnect Switch Connecting the Swing Battery Chargers to the Direct Current (DC) Buses (1R21.2.14).
05000298/2012007-04	NCV	Failure to Establish a Preventative Maintenance Program for Molded Case Circuit Breakers (1R21.2.15).
05000298/2012007-05	NCV	Failure to Have an Adequate Procedure for Erecting Flood Barriers (1R21.2.16).
05000298/2012007-06	NCV	Failure to Incorporate all Design and Technical Data Available into the Operability Determinations for the Standby Liquid Control Tank and Test Tank (1R21.3.5).

LIST OF DOCUMENTS REVIEWED

Calculations

NUMBER	TITLE	REVISION/DATE
0640012-X-226	HPCI Suction Piping from Penetration X226	0
10-073	Evaluation of CNS External Flood Barriers	0
92-050K	HPCI-LSO 74 A/B, HPCI 75 A/B Setpoints	June 26, 1998
Burns & Roe Calculation Civil Structural Book No. 11	Intake Structure, Substructure	April 6, 1970
Burns & Roe Calculation Civil Structural Book No. 12	NPPD – Intake Structure – Barge Impact Study and Fendering System	April 15, 1970
Burns & Roe Calculation Civil Structural Book No. 35	Reactor Building Miscellaneous Items – Standby Liquid Control Tank and Pump Pg. 51 & 52	February 13, 1970
Calculation 681H0441 Byron Jackson	Shock Load Analysis of 28 EXL – 1 Stage VCT	

Calculations

NUMBER	TITLE	REVISION/DATE
– Shock Load Analysis		
NEDC 00-003	Aux Power System Load Flow and Voltage Analysis	7
NEDC 00-003	Auxiliary Power System Load Flow and Voltage Analysis,	7
NEDC 12-015	Standby Liquid Control Test Tank Seismic Evaluation	0
NEDC 12-017	Standby Liquid Control Storage Tank Seismic Evaluation	0
NEDC 86-105B	CNS Critical AC Bus Coordination Study	8
NEDC 86-105B	Critical AC Bus Coordination Study,	8
NEDC 86-105B	480 V Coordination	8
NEDC 86-105E	AC Short Circuit Study	2
NEDC 86-105F	Non-Critical AC Bus Coordination	6
NEDC 87-131B	250 VDC Division II Load and Voltage Study	11
NEDC 87-131B	250 VDC Division II Load and Voltage Study	11
NEDC 87-131B	250 VDC Div II Load and Voltage Study	11
NEDC 87-131D	125 VDC Division II Load and Voltage Study	12
NEDC 87-131D	125 VDC Division II Load and Voltage Study	12
NEDC 87-140	Anchor Bolt Load Calc. For 5000 PSI Concrete	4
NEDC 87-221	125 Volt Battery Racks and Battery Charger Mounting Calculations DC Electrical/Control Building	1
NEDC 88-086B	Second Level Under Voltage Relay Setpoint	10
NEDC 88-086B	Second Level Undervoltage Relay Setpoint	10
NEDC 89-1313	Class IV Service Water Piping Analysis Problem SW-17	6
NEDC 89-149	Class IIN Main Steam Piping Analysis Problem MS-02	6
NEDC 89-1886	CNS Station Blackout Condensate Inventory	2
NEDC 90-367	NSST Bus Duct Impedance	November 30, 1990

Calculations

NUMBER	TITLE	REVISION/DATE
NEDC 90-368	SSST Bus Duct Impedance	December 3, 1990
NEDC 90-369	ESST Bus Duct Impedance	December 3, 1990
NEDC 91-088C	Review of Advent Calc. 96007TR-03 Rev. 2 Limiting Component Analysis for HPCI-MOV-M014	8
NEDC 91-088D	Review of Advent calculation LCA Calculation 96007TR-14 Rev. 0 for HPCI-MOV-MO17 and HPCI-MOV-MO58	2 February 23, 1997
NEDC 91-093	5 KV Penetration Short Circuit and Heat Loss	1
NEDC 91-094	125/250 VDC Battery Charger Analysis	5
NEDC 91-190	Short Circuit Withstand Capability, Rev 2, Attach K	December 1989
NEDC 91-20	UV DV Relay Settings	0
NEDC 91-208	Review of B&R Timing Relay Setpoint Calculation	July 9, 1991
NEDC 91-90	AC Equipment and Cable Short Circuit Withstand Ratings	2
NEDC 91-90 (K)	AMH-.4.76-250 Switchgear Fault Study	December, 1989
NEDC 91-94	125/250 V Battery Charger Analysis	5
NEDC 92-054	Analysis of 24" Torus Purge/Vent Duct for Hard Pipe Vent Loading	0
NEDC 92-074	Analysis of New 10" PC Line To Be Used As Part of Hard Pipe Vent Flow Path	0
NEDC 93-104	Emergency Transformer Permissive Relay Setpoint	5
NEDC 95-003	Determination of Allowable Operating Parameters for CNS MOV Program	27
NEDC 95-211	Maximum Valve Accelerations for 89-10 Program valves	0
NEDC 98-001	Vortex Limit for the ECSTs A and B,	2
NEDC 99-043	Evaluation of 125V DC and 250V DC Racks for CED 1999-0121 and CED 1999-012	7
NEDC-12-020	Service Water Instrument Rack Temporary Post Braces Seismic/Barge Impact Evaluation	0

Condition Reports

CR-CNS-2006-05366	CR-CNS-2012-00059	CR-CNS-2012-01665
CR-CNS-2006-09304	CR-CNS-2012-00276	CR-CNS-2012-01694
CR-CNS-2006-10123	CR-CNS-2012-01104	CR-CNS-2012-01902
CR-CNS-2007-04765	CR-CNS-2012-01179	CR-CNS-2012-01918
CR-CNS-2007-04977	CR-CNS-2012-01214	CR-CNS-2012-01920
CR-CNS-2008-06389	CR-CNS-2012-01232	CR-CNS-2012-01930
CR-JAF-2009-02647	CR-CNS-2012-01306	CR-CNS-2012-01933
CR-CNS-2009-09052	CR-CNS-2012-01308	CR-CNS-2012-01939
CR-CNS-2009-10139	CR-CNS-2012-01310	CR-CNS-2012-01962
CR-CNS-2009-10691	CR-CNS-2012-01326	CR-CNS-2012-01963
CR-CNS-2010-00897	CR-CNS-2012-01563	CR-CNS-2012-01971
CR-CNS-2010-03042	CR-CNS-2012-01566	CR-CNS-2012-01972
CR-CNS-2010-08749	CR-CNS-2012-01587	CR-CNS-2012-01974
CR-CNS-2010-08882	CR-CNS-2012-01588	CR-CNS-2012-01982
CR-CNS-2011-00756	CR-CNS-2012-01594	CR-CNS-2012-01994
CR-CNS-2011-07572	CR-CNS-2012-01611	CR-CNS-2012-02001
CR-CNS-2011-07573	CR-CNS-2012-01647	CR-CNS-2012-02002
CR-CNS-2011-08360	CR-CNS-2012-01649	CR-CNS-2012-02006
CR-CNS-2011-08406	CR-CNS-2012-01650	CR-CNS-2012-02335
CR-CNS-2011-09095	CR-CNS-2012-01651	CR-CNS-2012-02358
CR-CNS-2011-10904	CR-CNS-2012-01664	CR-CNS-2012-02359
		CR-CNS- 2012-02376

Condition Reports Generated During this Inspection

CR-CNS-2012-01566	CR-CNS-2012-01649	CR-CNS-2012-01971
CR-CNS-2012-01587	CR-CNS-2012-01930	CR-CNS-2012-01972
CR-CNS-2012-01588	CR-CNS-2012-01963	CR-CNS-2012-01982
CR-CNS-2012-01594		

Design Basis Documents

NUMBER	TITLE	REVISION/DATE
DCD-01	Emergency Diesel Generator	March 30, 2011

DCD-2, Appendix B	High Pressure Core Injection (HPCI)	March 30, 2011
DCD-04	EEAC	April 1, 2011
DCD-05	EEDC	February 2, 2009
DCD-13, Appendix B	Residual Heat Removal System (RHR)	April 1, 2011
DCD-35	Station Blackout	February 2, 2009

Drawings

NUMBER	TITLE	REVISION/DATE
E501 Sh 44B	RHR-MOV-MO16B Minimum Flow Bypass, RHR Pump B and D	N01
791E261 Sh 7	Residual Heat Removal System - Relay Logic Circuit B	20
791E261 Sh 8	Residual Heat Removal System	23
E501 Sh 32A	HPCI-MOV-MO14 Steam Supply to HPCI Turbine	N01
E501 Sh 33A	HPCI-MOV-MO58 HPCI Pump Suction From Suppression Pool	N01
M08515	"L" Two Step EP3 Racks	N03
M-9315	Battery Arrangement 2 Step EP 3 (2) Sets of (58) LCR-25 Cells	1
3006 Sh 5	Auxiliary One Line Diagram Starter Racks LZ and TZ, MCC's K, L, LX, RA, RX, S, T, TX, X	N75
3007 Sh 6	Auxiliary One Line Diagram Motor Control Centers E, Q, R, RB, & Y	N83
85B-70008 Sh 128	Motor Control Center Y	N06
E150 Sh 22	Relay Settings for Nutherm DC Starter Overload Relays	N02
3001	Main One Line Diagram,	N19
3002	One Lines (Bus 1F, !G, 480 V Bus 1A, 1B, 1E, MCC Z)	N47
3003	MCC One Lines (A,B,F,G)	N47
3004	MCC One Lines (C,D,H,J,DG1, DG2)	N22
3005	MCC One Lines (M,N,P,U,V,W)	N60
3006	Starter and MCC One Lines - MCC-TX	N75
3007	MCC One Lines (E,Q,R,B,Y)	N83

Drawings

NUMBER	TITLE	REVISION/DATE
3008	PMIS Inverter	N20
3009 SH1	12.5 kV Ring Bus One Line	N44
3010 SH1	Vital Power One Line	N75
3010 SH2	Critical Distribution Panel CDP1A	N08
3010 SH15	RPS Power Panel Load and Fuse	N12
3058	DC One Line	N56
3059 SH1	DC Panels	N39
3059 SH13	125 V Load and Fuse Lists	N02
NC66688	345 kV Switchyard One Line Switching Diagram	15
NC66688	345 kV Switchyard One Line Switching Diagram	16
Burns & Roe 2022	Primary Containment Cooling & Nitrogen Integrating System	N78
Terry Corp. 800315C	Suction Hydraulic Trip	May 23, 1983
Terry Corp. 800268D	Assembly of Hydraulic Trip	February 16, 1982
Terry Corp. D-6252	Oil Relay Assembly- Remote Servo	March 27,1968
Terry Corp. C-934-X	Governor Control System	January 20, 1969
Terry Corp. B1042X	Lever Diagram	September 13, 1968

Engineering Reports

NUMBER	TITLE	REVISION/DATE
EE 06-014	Design Basis Stroke Time Requirements for Various Power Operated Valves	0
	250Vdc Load and Voltage Study	11
	Final Report for Reactor Torus IWE Inspection and Corrosion Repair	January 2005
	Final Report for Reactor Torus IWE Inspection and Corrosion Repair	April 2008

Maintenance Work Orders

NPPD # 4170660	NPPD # 4639052	NPPD # 4744583	NPPD # 4803046
NPPD # 4192567	NPPD # 4664076	NPPD # 4746115	NPPD # 4803099
NPPD # 4229616	NPPD # 4664077	NPPD # 4748522	NPPD # 4803367
NPPD # 4289552	NPPD # 4664310	NPPD # 4748527	NPPD # 4803463
NPPD # 4336934	NPPD # 4694669	NPPD # 4749938	NPPD # 4811307
NPPD # 4385313	NPPD # 4702575	NPPD # 4750079	NPPD # 4811310
NPPD # 4441737	NPPD # 4704711	NPPD # 4750080	NPPD # 4812188
NPPD # 4497165	NPPD # 4721931	NPPD # 4753138	NPPD # 4813493
NPPD # 4523336	NPPD # 4723658	NPPD # 4754502	NPPD # 4846462
NPPD # 4561338	NPPD # 4726461	NPPD # 4754509	NPPD # 4848232
NPPD # 4581527	NPPD # 4734975	NPPD # 4754892	NPPD # 4849589
NPPD # 4623906	NPPD # 4734976	NPPD # 4754899	NPPD # 4874132
NPPD # 4625263	NPPD # 4738272	NPPD # 4755012	NPPD # 4878634
NPPD # 4662871	NPPD # 4743377		

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CNS 6.PC.504	Primary Containment Leak Test	November 28, 1998
CNS 6.PC.506	Primary Containment Local Leak Rate Test	February 8, 2012
CNS 6.PC.506	Primary Containment Local Leak Rate Test	January 19, 2012
CNS 6.PC.522	Standby Nitrogen Injection and PC Purge and Vent System Local Leak Rate Tests	January 4, 2012
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6.1 EE.604	125V Battery Charger Performance Test (DIV 2)	February 8, 2011
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CNS 6.PC.504	Primary Containment Leak Test	November 21, 1998
CNS 6.PC.504	Primary Containment Leak Test	December 6, 1998
CNS 6.PC.504	Primary Containment Leak Test	November 28, 1998

CNS 6.PC.506	Primary Containment Local Leak Rate Test	February 8, 2012
CNS 6.PC.506	Primary Containment Local Leak Rate Test	January 19, 2012
CNS 6.PC.522	Standby Nitrogen Injection and PC Purge and Vent System Local Leak Rate Tests	January 4, 2012
CNS 6.PC.530	Primary Containment Leak Test	October 24, 1998
CNS 6.SC.501	Secondary Containment Leak Test	October 21, 2009
CNS 6.SC.501	Secondary Containment Leak Test	April 23, 2008
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System Health Notebooks

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EE-AC_001	AC Power Systems	January 2012
EE-DC_001	DC Power Systems	January 2012
EE-SY_001	Switchyard	January 2012

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C&D Technologies Letter	Subject: Technical Specification Limits for Operability, Intercell Connection Resistance 125 Volt and 250 Volt LCR-25 Batteries	May 1, 1997
C&D Technologies Letter	Pilot Cell Recommendations	March 20, 2012
C&D Technologies	Pilot Cell Recommendations	March 21, 2012

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Letter CED6025080	Replacement of RHR SWBP Motors	
CNSS915709	MCCB Testing	August 19, 1991
CR-2005-9378	Battery Charger Fuses	
CR-2011-11750	Bus Duct Resistance Testing	
DC-91-041	Torus Hard Vent Pipe Vent	December 10, 1992
	MOV Program, Health Report Summary	4^{th} Quarter 2011
EPRI TR-100248, pg 11-3 only	Stationary Battery Guide: Design, Application, and Maintenance	2
Excel Data for HPCI-MO14	Opening/Closing Thrust graphs for date range 5/31/1993 through 5/4/2008	
Excel Data for HPCI-MO58	Opening/Closing Thrust graphs for date range 12/3/1991 through 10/8/2009	
Excel Data for RHR-MO16B	Stroke Times (Open and Close) for date range 1/13/2009 through 1/17/2012	
Excel Data for RHR-MO16B	Opening/Closing Thrust graphs for date range 10/23/1991 through 10/18/2007	
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NUC-001-02	Nuclear Plant Interface	
OE-30458	HPCI Governor Valve Failure to Stroke	January 29, 2010
SOER 10-01	Power Transformers	
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10-01	QA Engineering Audit	June 3, 2010
10-04	QA Audit Operations and Technical Support	October 11, 2010