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

March 28, 2014

Mr. Dennis L. Koehl
President & CEO/CNO
STP Nuclear Operating Company
P.O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT – THE NRC COMPONENT DESIGN BASES INSPECTION REPORT 05000498/2013007; 05000499/2013007

Dear Mr. Koehl:

On February 13, 2014, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the South Texas Project Units 1 and 2. On March 20, 2014, the inspectors discussed the results of this inspection with T. Powell, Site Vice President and other members of your staff. Inspectors documented the results of this inspection in the enclosed inspection report.

NRC inspectors documented seven findings of very low safety significance (Green) in this report. All of these findings involved violations of NRC requirements. Further, inspectors documented a licensee-identified violation which was determined to be of very low safety significance. The NRC is treating these violations as non-cited violations (NCV's) consistent with Section 2.3.2.a of the Enforcement Policy.

If you contest the violations or significance of the NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region IV; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the South Texas Project. The information you provide will be considered in accordance with the NRC Inspection Manual Chapter 0305.

If you disagree with the characterization of the cross-cutting 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 South Texas Project.

In accordance with Title 10 of the *Code of Federal Regulations* 2.390, "Public Inspections, Exemptions, Requests for Withholding," of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC's Public Document Room or from the Publicly Available Records (PARS) component of the NRC's Agencywide Documents Access and Management 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,

/RA/

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

Dockets No.: 50-498; 50-499
Licenses No.: NPF-76; NPF-80

Enclosure: Inspection Report 05000498/2013007; 05000499/2013007
w/ Attachment 1: Supplemental Information
Attachment 2: Detailed Risk Evaluations for the South Texas Project Component Design Bases Inspection

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	50-498; 50-499
Licenses:	NPF-76; NPF-80
Report Nos.:	05000498/2013007; 05000499/2013007
Licensee:	STP Nuclear Operating Company
Facility:	South Texas Project
Location:	Wadsworth, TX
Dates:	January 13 – February 13, 2014
Team Leader:	Wayne C. Sifre, Senior Reactor Inspector, Engineering Branch 1
Inspectors:	John Dixon, Senior Reactor Inspector, Engineering Branch 1 Gwynn Skaggs-Ryan, Reactor Inspector, Engineering Branch 1 Nnaerika Okonkwo, Reactor Inspector, Engineering Branch 2 Chris Cowdrey, Operations Engineer, Operations Branch
Accompanying Personnel:	George Replogle, Senior Reactor Analyst, Division of Reactor Safety Stanley Kobylarz, Contractor, Beckman and Associates John Zudans, Contractor, Beckman and Associates
Approved By:	Thomas R. Farnholtz, Chief, Engineering Branch 1

SUMMARY

IR 05000498/2013007; 05000499/2013007; 01/13/2014 – 02/13/2014; South Texas Project; baseline inspection, NRC Inspection Procedure 71111.21, “Component Design Bases Inspection.”

The report covers an announced inspection by a team of five regional inspectors and two contractors. Seven Green violations were identified. 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 the 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, non-cited 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, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions.” Specifically, prior to February 11, 2014, the licensee failed to adequately verify by analysis that safety-related nuclear steam supply system instrumentation loads would be capable of operating at the minimum inverter output voltage, when the inverter is fed from the station battery, and when considering the actual voltage drop to the load. In response to this issue, the licensee performed a preliminary voltage drop analysis that supported an immediate operability determination. This finding was entered into the licensee’s corrective action program as Condition Report 14-2017.

  The team determined that failure to maintain design control of the nuclear steam supply system instrumentation power supply load was a performance deficiency. This finding was more than minor because if left uncorrected, it would lead to a more significant safety concern. Specifically, the incorrect analysis resulted in a reasonable question of operability of nuclear steam supply system instrumentation to operate at the minimum inverter output voltage, when the inverter is fed from the station battery, and when the actual voltage drop to the load for that condition was considered. In accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated June 19, 2012, Exhibit 2, “Mitigating Systems Screening Questions,” the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. The team determined that this finding did not have a cross-cutting aspect because the most significant contributor did not reflect current licensee performance. (Section 1R21.2.1).

- Green. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," which states, in part, "A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents." Specifically, prior to January 13, 2014, the licensee's preventive maintenance Procedures OPMPO5-NA-002, "4160V Gould Breaker Test," and OPMP05-NA-0018 "4160 Volt Gould HK Breaker Overhaul/Lubrication," failed to assure that the 4160 VAC Gould circuit breakers would perform satisfactorily in service when the licensee performed maintenance prior to completing as-found tests to verify the circuit breakers would function properly. In response to this issue, the licensee validated that the components had passed their required surveillance tests and remained operable. This finding was entered into the licensee's corrective action program as Condition Reports 14-738 and 14-1633.

  The team determined that failure to establish a test and maintenance program which ensures that safety-related 4160 VAC Gould circuit breakers would perform satisfactorily in service was a performance deficiency. This finding was more than minor because if left uncorrected, it would lead to a more significant safety concern. Specifically, the failure to perform as-found tests prior to performing maintenance in preventive maintenance procedures was a significant programmatic deficiency which could cause unacceptable conditions to go undetected. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding had a crosscutting aspect in the area of human performance, documentation component because the licensee failed to create and maintain complete, accurate, and up-to-date documentation. [H.7] (Section 1R21.2.3).

- Green. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," which states, in part, "A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents." Specifically, prior to January 13, 2014, the licensee's preventative and post-maintenance procedures for safety-related 480 VAC Westinghouse DS circuit breakers failed to include manufacturers recommended testing of breaker control circuits at the minimum expected control voltage levels postulated to exist at the device terminals during design basis events. In response to this issue, the licensee validated that the components had passed their required surveillance tests and remained operable. This finding was entered into the licensee's corrective action program as Condition Reports 11-4895 and 14-738.

The team determined that the failure to include manufacturers recommended testing of safety-related circuit breaker control circuits at the voltages postulated to exist at the device terminals during design basis events or to provide justification for not performing the tests was a performance deficiency. This finding was more than minor because if left uncorrected, it would lead to a more significant concern. Specifically, the failure to perform the breaker testing at reduced voltage using minimum expected control voltage levels could cause unacceptable conditions to go undetected. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding had a crosscutting aspect in the area of problem identification and resolution, evaluation component because the licensee failed to thoroughly evaluate the issue to ensure that resolution addressed causes and extent of condition commensurate with their safety significance. [P.2] (Section 1R21.2.5).

- Green. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," which states, in part, "Activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these procedures." Specifically, on November 3, 2013, maintenance personnel performing a maintenance activity change and performing the second party technical review did not initial and date the change that was performed for reactor containment fan cooler 12C backdraft damper as required by Procedure MG-0006, "Work Execution and Closeout Guideline," Revision 11, step 6.2.3. In response to this issue, the licensee initiated revisions to the associated work order instructions and established as-found trend data for backdraft damper 12C. This finding was entered into the licensee's corrective action program as Condition Reports 14-1820 and 14-1836.

  The team determined that failure to follow Procedure MG-0006 to complete the preventative maintenance work order on reactor containment fan cooler 12C as instructed was a performance deficiency. This finding was more than minor because it adversely affected the Mitigating Systems Cornerstone attribute of Procedure Quality and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, not performing a proper procedure change does not ensure a proper technical review of the change and had the potential to challenge the availability and capability of the reactor containment fan cooler. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance, resources component because procedures were not available to ensure successful work performance. [H.1] (Section 1R21.2.9).

- Green. The team identified a Green, non-cited 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 February 13, 2014, documented requirements in purchase specification 3V259VS0005 were not correctly translated into specifications, drawings, and instructions evaluated in calculations MC-06482 and MC-06482A for the safety injection pump room coolers. In response to this issue, the licensee revised the associated calculations and established that the room coolers remained operable. This finding was entered into the licensee’s corrective action program as Condition Report 14-2673.

  The team determined that the failure to maintain design control of the safety injection pump room cooler was a performance deficiency. This finding was more than minor because it adversely affected the Mitigating Systems Cornerstone attribute of Design Control and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, not maintaining design control and performing a proper heat transfer calculation had the potential to challenge the availability, reliability, and capability of the safety injection pump room cooler and in turn the safety function of safety injection pumps. In accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated June 19, 2012, Exhibit 2, “Mitigating Systems Screening Questions,” the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. The team determined that this finding did not have a cross-cutting aspect because the most significant contributor did not reflect current licensee performance. (Section 1R21.2.16).

- Green. The team identified a Green, non-cited 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, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions.” Specifically, prior to January 28, 2014, the licensee failed to adequately verify by analysis that the AF-19 valve motor had adequate voltage available to close the valve when required during postulated high energy line break conditions. In response to this issue, the licensee performed a preliminary battery sizing and voltage analysis and verified that the valve motor had sufficient voltage to close when required by the failure modes and effects analysis. This finding was entered into the licensee’s corrective action program as Condition Report 14-1374.

The team determined that the failure to evaluate and translate the requirements for adequate voltage available at the AF-19 valve motor to close the valve during postulated high energy line break conditions was a performance deficiency. This finding was more than minor because if left uncorrected, it would lead to a more significant safety concern. Specifically, the failure to analyze and translate the relevant requirements resulted in a condition where there was a reasonable question on the capability of the valve to close when required during postulated high energy line break conditions. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. The team determined that this finding did not have a cross-cutting aspect because the most significant contributor did not reflect current licensee performance. (Section 1R21.2.17).

Cornerstone: Initiating Events

- Green. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," which states, in part, "Instructions, procedures, or drawings shall include appropriate qualitative and quantitative acceptance criteria for determining that important activities have been satisfactorily accomplished." Specifically, prior to January 29, 2014, the licensee failed to include appropriate qualitative and quantitative criteria in emergency operating procedures, off-normal operating procedures, and annunciator response procedures that are used during a loss of all seal cooling to a reactor coolant pump to prevent increased risk of a reactor coolant pump seal loss of coolant accident. In response to this issue, the licensee implemented changes to the affected procedures and communicated the changes to the operating staff. This finding was entered into the licensee's corrective action program as Condition Report 14-1635.

  The team determined that the failure to include appropriate qualitative and quantitative criteria in emergency operating procedures, off-normal operating procedures, and annunciator response procedures for a loss of all seal cooling to a reactor cooling pump was a performance deficiency. This finding was more than minor because it adversely affected the Initiating Events Cornerstone attribute of Procedure Quality and affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, operating procedures did not contain appropriate attributes to ensure timely action to prevent an increased likelihood of a reactor coolant pump seal loss of coolant accident. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 1, "Initiating Events Screening Questions," the team determined a detailed risk evaluation was necessary because, after a reasonable assessment of degradation, the finding could result in exceeding the reactor coolant system leak rate for a small loss of coolant accident. Therefore, the senior reactor analyst performed a bounding detailed risk evaluation. The analyst determined that the change to the core damage frequency would be less than 1E-7 per year (Green). This finding had a cross-cutting aspect in the area of human performance, training component because the licensee did not provide training and ensure knowledge transfer to maintain a knowledgeable, technically competent workforce and instill nuclear safety values. [H.9] (Section 1R21.4).

B. Licensee-Identified Findings

- A violation of very low safety significance that was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and associated corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

1 REACTOR SAFETY

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

1R21 Component Design Bases Inspection (71111.21)

To assess the ability of the South Texas Project 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 South Texas Project probabilistic risk assessments and the U.S. Nuclear Regulatory Commission's (NRC) standardized plant analysis risk model. In general, the selection process focused on components that had a risk achievement worth factor greater than 1.3 or a risk reduction worth factor greater than 1.005. The items selected included components in both safety-related and nonsafety-related systems. 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 conditions of the components were consistent with the design basis and that the tested capabilities met the required criteria.

The team reviewed maintenance work records, corrective action documents, and industry operating experience records to verify that licensee personnel considered degraded conditions and their impact on the components. For 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; Title 10 CFR 50.65(a)1 status; operable, but degraded conditions; the resident inspector input of problem equipment; system health reports; industry operating experience; and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense in-depth margins.

The inspection procedure requires a review of 15 to 25 total samples that include risk-significant and low design margin components, containment related components, and operating experience issues. The sample selection for this inspection was 18 components, 4 containment related components, 5 operating experience items, and 4 event based activities associated with the components. 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 and safety-related loads in support of safety system operation in response to initiating events such as loss of offsite power, station blackout, and a loss-of-coolant accident with offsite power available. The team selected the following components:

    - Safety-Related Nuclear Steam Supply System Inverter/Rectifier B IV-1203
    - Reactor Coolant Pump C Underfrequency Relay
    - 4160 VAC Class 1E Switchgear, Bus A
    - Emergency Diesel Generator Output Circuit Breaker B
    - 480 VAC Class 1E, Bus B
    - Steam Generator Power Operated Relief Valve Control Circuit

b. Components necessary to mitigate radiation releases: The team reviewed components required to perform isolation functions to prevent an unmonitored release of radiation. The team selected the following components:

    - Normal and Supplementary Containment Purge Valves
    - Post-Accident Sampling System Containment Isolation Valves
    - Reactor Containment Fan Coolers
    - Containment Electrical Penetrations

c. Mitigating systems needed to attain safe shutdown: The team reviewed components and support systems required to perform the safe shutdown of the plant. The team selected the following components:

- Steam Generator Power Operated Relief Valves FV-7411, FV-7421, FV-7431, and FV-7441

- Technical Support Center Diesel Generator

- Positive Displacement Pump

- Electrical Auxiliary Building Heating, Ventilation, and Air Conditioning

- Auxiliary Feedwater Cross Connect Air Operated Valves

- Safety Injection Pump Room Coolers

- Auxiliary Feedwater System Steam Generator Isolation Motor Operated Valves

- Essential Cooling Water Screen Wash System

.2 Results of Detailed Reviews for Components:

.2.1 Safety-Related Nuclear Steam Supply System Inverter/Rectifier B IV-1203

a. Inspection Scope

The team reviewed the updated safety analysis report, design basis documents, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with safety-related inverter IV-1230 to ensure design basis requirements were met. The team also performed walkdowns and conducted interviews with system and design engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Short circuit calculations, inverter sizing calculations, coordination studies, and voltage drop calculations.

- One-line diagrams and design basis documents for the inverter electrical distribution system to identify requirements and interfaces.

- Preventive maintenance activities to verify the inverter system maintained according to manufacturer recommendations.

- Periodic load testing to demonstrate system capability.

- Vendor documentation to verify distribution panel branch circuit load and load voltage requirements properly translated into inverter sizing and voltage drop calculations.

- Alarm response procedures for monitored conditions and operator response.

- Past modifications associated with the inverter for design basis considerations.

b. Findings

Failure to Properly Evaluate Safety-Related Equipment Electrical Load Requirements when Verifying the Adequacy of Voltage from the Nuclear Steam Supply System Inverter/Rectifier

Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the licensee’s failure to assure that the applicable design basis requirements associated with the safety-related nuclear steam supply system instrumentation electrical load requirements were correctly translated into the plant design.

Description. The team requested the inverter sizing and voltage drop studies to review and verify that the Class 1E inverters and distribution system were capable of providing sufficient voltage and current to the critical loads. The team found that the licensee failed to perform an adequate voltage analysis and design verification to demonstrate that the Class 1E inverters would be capable of providing sufficient voltage to safety-related nuclear steam supply system instrumentation loads fed from inverter backed power distribution panel DP-1203.

The team found that the voltage analysis performed for the nuclear steam supply system instrumentation in power cable sizing verification calculation, EC-5038, did not include the correct instrumentation power supply load information from the nuclear steam supply system vendor. For example, for DP-1203, circuit number 3, nuclear steam supply system process cabinet, the vendor load was 1619 watts at 0.85 power factor and 118 volts, or approximately 1905 volt-amperes. However, the team found that the licensee evaluated only 1190 volt-amperes in calculation EC-5038 and did not evaluate the voltage drop to the load based on the expected load current that the nuclear steam supply system instrumentation cabinet power supplies would require at the minimum inverter output voltage conditions. The team also found that nuclear steam supply system instrumentation loads were incorrectly depicted on the inverter one-line diagram. The one-line diagram incorrectly showed the nuclear steam supply system process cabinet load as 1619 volt-amperes. The value of load volt-amperes on the one-line diagram for DP-1203, circuit number 3, was found by the team to be understated by approximately 286 volt-amperes (1905 volt-amperes minus 1619 volt-amperes) in the example discussed above. Nonetheless, the licensee confirmed that the correct value of load was considered in the inverter sizing analysis and the station battery sizing was performed correctly. The licensee found during their review on this issue that there were other similar errors made on the one-line diagram for DP-1203 where load watts were incorrectly represented as load volt-amperes.

The team determined that the use of incorrect load data and voltage drop methodology contributed both to understating the load current required by the instrumentation power supplies and overestimating the acceptance value for the maximum circuit length for the conductor size that was utilized for the load. The licensee performed a preliminary voltage drop analysis that supported an immediate operability determination that provided assurance the nuclear steam supply system instrumentation power supplies would operate within the manufacturer’s voltage design limit for the minimum conditions of inverter output voltage.

Analysis. The team determined that failure to maintain design control of the nuclear steam supply system instrumentation power supply load was a performance deficiency. This finding was more than minor because if left uncorrected, it would lead to a more significant safety concern. Specifically, the incorrect analysis resulted in a reasonable question of operability of nuclear steam supply system instrumentation to operate at the minimum inverter output voltage, when the inverter is fed from the station battery, and when the actual voltage drop to the load for that condition was considered. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. The team determined that this finding did not have a cross-cutting aspect because the most significant contributor did not reflect current licensee performance.

Enforcement. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions." Contrary to the above, prior to February 11, 2014, the licensee did not assure that applicable regulatory requirements and the design basis were correctly translated into specifications, drawings, procedures and instructions. Specifically, the licensee failed to adequately verify by analysis that safety-related nuclear steam supply system instrumentation loads would be capable of operating at the minimum inverter output voltage, when the inverter is fed from the station battery, and when considering the actual voltage drop to the load. In response to this issue, the licensee performed a preliminary voltage drop analysis that supported an immediate operability determination. This finding was entered into the licensee's corrective action program as Condition Report 14-2017. 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 non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000498/2013007-01, and 05000499/2013007-01, "Failure to Properly Evaluate Safety-Related Equipment Electrical Load Requirements when Verifying the Adequacy of Voltage from the Nuclear Steam Supply System Inverter/Rectifier."

.2.2 Reactor Coolant Pump C Under Frequency Relay

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 the under frequency relay. The team also performed walkdowns, and conducted interviews with system and design engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically the team reviewed:

- Schematics for the under frequency relay.
- Vendor setpoint analysis and acceptance requirements.
- Calculations for determining relay setting and safety analysis limit.
- Surveillance testing to demonstrate relay setting and performance in accordance with vendor requirements.

b. Findings

No findings were identified.

.2.3 4160 VAC Class 1E Switchgear, Bus A

a. Inspection Scope

The team reviewed the updated safety analysis report, design basis documents, the current system health report, calculations, maintenance and test procedures, and condition reports associated with the A Train of 4160 VAC bus E1A. The team also performed walkdowns and conducted interviews with engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Circuit one-line diagrams.
- Bus loading study during normal plant operation and design basis accident load conditions.
- Vendor data on available short circuit current.
- Calculated short-circuit current at loads for the bus.
- Breaker coordination study for the bus.
- Vendor data for the bus and associated circuit breakers.
- Cable sizing requirements and analyses.
- Preventive maintenance and surveillance test procedures.
- Completed surveillance and maintenance documentation.
- Modifications performed.

b. Findings

Improper Sequencing of Maintenance of 4160 VAC Circuit Breakers Prior to As-Found Tests

Introduction. The team identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion XI, "Test Control," involving the licensee's failure to establish a test program which demonstrates that components will perform satisfactorily in service. Specifically, the licensee failed to record as-found test values prior to performing maintenance of cycling, cleaning, and lubricating for 4160 VAC circuit breakers.

Description. The team reviewed five-year preventive maintenance procedures and the overhaul/lubrication procedure for 4160 VAC circuit breakers. During the review, the team identified that Procedure OPMPO5-NA-002, "4160V Gould Breaker Test," and Procedure OPMP05-NA-0018, "4160 Volt Gould HK Breaker Overhaul/Lubrication," directed maintenance personnel to clean, adjust, and manipulate the physical condition of 4160 VAC circuit breaker contacts, insulators, and other critical circuit breaker components before performing an as-found test, to determine if the circuit breakers would have performed their intended design function.

For example, Procedure OPMPO5-NA-002, "4160V Gould Breaker Test," Revision 26, Section 5.16.2 and 5.16.4 directs maintenance personnel to record as-found and as-left readings for breaker closing and opening time. Prior to performing these tests, Sections 5.6 and 5.7 direct the maintenance personnel to perform "contact and insulation cleaning," and "lubrication," respectively. Step 5.12.1 directs the breaker to be cycled. Steps 5.6, 5.7, and 5.12.17 are completed before any as-found tests are performed to verify the functionality of the critical components of the circuit breaker, such as coil operations (electrical operation).

The team reviewed the data sheet resulting from the November 30, 2010, Inspect/Periodic Lube/Test in the preventative maintenance performed on 4160 VAC Standby Diesel Generator 12 output circuit breaker using Procedure OPMPO5-NA-0002, Revision 22. Those results show that maintenance personnel documented the same results for as-found and as-left for the coil opening and closing times tested parameters; therefore, the team determined that the preventive maintenance, as performed, could mask existing conditions such as unacceptable contact resistance, setpoint drift, and mechanical binding. Additionally, the procedure resulted in the inability to verify past functionality of 4160 VAC Gould circuit breakers such as the 4160 VAC Diesel Generator 22 output circuit breaker B2PKSGOE1B14.

Analysis. The team determined that failure to establish a test and maintenance program which ensures that safety-related 4160 VAC Gould circuit breakers would perform satisfactorily in service was a performance deficiency. This finding was more than minor because if left uncorrected, it would lead to a more significant safety concern. Specifically, the failure to perform as-found tests prior to performing maintenance in preventive maintenance procedures was a significant programmatic deficiency which could cause unacceptable conditions to go undetected. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening

Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding had a crosscutting aspect in the area of human performance, documentation component because the licensee failed to create and maintain complete, accurate, and up-to-date documentation [H.7].

Enforcement. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," which states, in part, "A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents." Contrary to the above, prior to January 13, 2014, the licensee failed to establish a test program that assured that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service was identified and performed in accordance with written test procedures which incorporated the requirements and acceptance limits contained in applicable design documents. Specifically, prior to January 13, 2014, the licensee's preventive maintenance Procedures OPMPO5-NA-002, "4160V Gould Breaker Test," and OPMP05-NA-0018 "4160 Volt Gould HK Breaker Overhaul/Lubrication," failed to assure that the 4160 VAC Gould circuit breakers would perform satisfactorily in service when the licensee performed maintenance prior to completing as-found tests to verify the circuit breakers would function properly. In response to this issue, the licensee validated that the components had passed their required surveillance tests and remained operable. This finding was entered into the licensee's corrective action program as Condition Reports 14-738 and 14-1633. 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 non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000498/2013007-02, 05000499/2013007-02, "Improper Sequencing of Maintenance of 4160 VAC Circuit Breakers Prior to As-Found Tests."

.2.4 Emergency Diesel Generator Output Circuit Breaker B

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the B Train 4160 VAC Emergency Diesel Generator output breaker. 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:

- Schematics and control wiring diagrams of record.

- Functional logic diagram of circuit breaker and breaker coordination.

- Preventive maintenance procedures.
- Vendor manual and specifications.
- Load calculations of record and supporting documentation.
- Calculations of record for protection settings and alarms.
- Load Coordination studies.
- Completed preventive maintenance work orders.

b. Findings

No findings were identified.

.2.5 480 VAC Class 1E Bus B

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the B Train 480 VAC Bus E1B1/E1B2. 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:

- Circuit one-line diagrams.
- Bus loading study during normal plant operation and design basis accidents.
- Vendor Data on available short circuit current.
- Vendor installation and maintenance manuals.
- Electrical distribution system load flow/voltage drop, short circuit, and electrical protection and coordination calculations.
- Protective device settings and circuit breaker ratings to confirm operation during worst-case short circuit conditions.
- Circuit breaker preventive maintenance inspection and testing procedures.
- Completed preventive maintenance work orders.

b. Findings

Failure to Establish an Adequate Test Program for Safety-Related 480 VAC Circuit Breakers

Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," involving the licensee's failure to establish an adequate test program for 480 VAC circuit breakers. Specifically, the licensee's preventive maintenance program did not include manufacturers recommended breaker operability testing at reduced voltage using minimum expected control voltage levels.

Description. During a review of documents related to 480 VAC circuit breaker preventive maintenance Procedure OPMP05-NA-0008, "Westinghouse 480 Volt Breaker Test," the team determined that manufactures recommended reduced voltage testing was not included in the procedure and therefore not performed on Westinghouse DS 480 VAC circuit breakers. The licensee utilizes Westinghouse DS 480 VAC type circuit breakers in safety-related 480 VAC electrical systems.

On March 16, 2011, the licensee documented the failure of a 480 VAC 1T-4C reduced voltage test in Condition Report 11-4895. The breaker shunt trip coil failed to operate at 70 VDC. In reviewing the corrective action to this failure, the licensee recommended the revision of safety-related 480 VAC breaker maintenance Procedure OPMP05-NA-0008, "Westinghouse 480 Volt Breaker Test," to include the performance of reduced voltage testing as part of the preventive maintenance program. As of January 12, 2014, this procedure had not been revised to perform this test. The procedure has been used to perform preventive maintenance on some safety-related 480 VAC breakers without performing the reduced voltage test since the failure was identified.

In Condition Report 11-4895, the licensee recognized the importance of performing the reduced voltage test by stating that "reduced voltage testing provides information on the operation of the open and close coils and their interaction with their respective Trip and Close components (Latches, Bushings, Rollers, linkages)." The licensee further stated that "Review of equipment history has found occasions where breakers have failed an as-found low voltage test and have then passed it after the breaker has been overhauled." The licensee has since initiated Condition Report 14-738 and a plan of action to revise the procedure to include the reduced voltage tests.

Analysis. The team determined that the failure to include manufacturers recommended testing of safety-related circuit breaker control circuits at the voltages postulated to exist at the device terminals during design basis events or to provide justification for not performing the tests was a performance deficiency. This finding was more than minor because if left uncorrected, it would lead to a more significant concern. Specifically, the failure to perform the breaker testing at reduced voltage using minimum expected control voltage levels could cause unacceptable conditions to go undetected. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding had a crosscutting aspect in the area of problem identification and resolution, evaluation component because the licensee failed to thoroughly evaluate the issue to ensure that resolution addressed causes and extent of condition commensurate with their safety significance [P.2].

Enforcement. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," which states, in part, "A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents." Contrary to the above, prior to January 13, 2014, the licensee failed to establish a test program that assured that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Specifically, the licensee's preventative and post-maintenance procedures for safety-related 480 VAC Westinghouse DS circuit breakers failed to include manufacturers recommended testing of breaker control circuits at the minimum expected control voltage levels postulated to exist at the device terminals during design basis events. In response to this issue, the licensee validated that the components had passed their required surveillance tests and remained operable. This finding was entered into the licensee's corrective action program as Condition Reports 11-4895 and 14-738. 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 non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000498/2013007-03, 05000499/2013007-03, "Failure to Establish an Adequate Test Program for Safety-Related 480 VAC Circuit Breakers."

.2.6 Steam Generator Power Operated Relief Valve FV-7441 Control Circuit

a. Inspection Scope

The team reviewed the updated safety analysis report, design basis documents, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the steam generator power operated relief valve FV-7441 control circuit. The team also performed walkdowns and conducted interviews with engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Schematic and wiring diagrams.
- One-line and control diagrams for block valve motor starter.
- Calculation for voltage available at valve motor terminal during degraded voltage conditions.

- Cable routing for the power operated relief valve and associated hydraulic pumps.

- Modifications performed on the motor operator and control and starter circuit.

b. Findings

No findings were identified.

.2.7 Normal and Supplementary Containment Purge Valve Operation and Qualification. Compliance with Containment Systems Branch BTP 6-4

a. Inspection Scope

The team reviewed the updated final safety analysis report, the current system health report, selected drawings, motor-operated and air-operated valve qualification calculations, maintenance and test procedures, condition reports and design change packages associated with the normal and supplementary containment purge isolation valves. The team conducted a walk down of the supplementary containment purge valves outside of containment. The team also conducted interviews with engineering personnel to ensure the capability of these components to perform their desired design basis functions. Specifically, the team reviewed:

- Health reports associated with normal and supplementary purge valves for the last several years.

- Maintenance work order history and corrective action program reports from 1989 to 2013 for any common problems or issues.

- Weak link calculations for the normal and supplementary motor operated valves.

- Weak link analyses for qualifying the air/spring operated valves in the supplementary containment purge system.

- Calculation NC-7121, critical mass flow rate through the supplementary purge valve following a loss of coolant accident.

- Calculation 34753/2-48, operability analysis and test report on the supplementary containment purge isolation valves.

- Documentation and drawings provided regarding ducting and piping downstream of supplementary containment purge valves and the potential for downstream piping and supports to become missiles that could damage the purge isolation valves.

b. Findings

No findings were identified.

.2.8 Post Accident Sampling System Containment Isolation Valves

a. Inspection Scope

The team reviewed the updated safety analysis report, design basis documents, system description, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the post-accident sampling system containment isolation valves to ensure design basis requirement specifications were met. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.
- License amendment request and related safety evaluation report for removal of the post-accident sampling system from the technical specifications.
- License amendment request and related safety evaluation report for the approval of the risk informed inservice testing program.
- License amendment request and related safety evaluation report for the graded quality assurance risk exemption.
- Implementation of license commitments from the license amendment requests.
- Implementation of alternate post-accident containment and reactor coolant sampling and the ability to determine associated emergency action levels.

b. Findings

No findings were identified.

.2.9 Reactor Containment Fan Coolers

a. Inspection Scope

The team reviewed the updated safety analysis report, design basis documents, system description, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the reactor containment fan coolers to ensure design basis requirement specifications were met. The team also performed walkdowns and conducted interviews with operations and system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

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

- License amendment request and related safety evaluation report for the approval of the risk informed inservice testing program.

- Purchase specification requirements and comparison with the design basis documents and the heat transfer calculations.

- The original startup flow balance calculation, including annubar flow element constants.

b. Findings

Failure to Follow Preventative Maintenance Procedure for Reactor Containment Fan Cooler

Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” involving the licensee’s failure to perform activities affecting quality prescribed by documented procedures of a type appropriate to the circumstances and accomplished in accordance with these procedures. Specifically, the licensee deviated from Procedure MG-0006, “Work Execution and Closeout Guideline,” Revision 11 without performing a procedure change including justification and adequate review.

Description. On October 30, 2013, Unit 1 control room operators noticed that reactor containment fan cooler 12C inlet temperature was less than outlet temperature by 6 degrees F. Control room logs require a condition report be generated when the temperature difference is greater than 5 degrees F. Consequently, the licensee generated Condition Report 13-12589 to document and evaluate the issue.

On November 1, 2013, the licensee inspected reactor containment fan cooler using preventative maintenance work order 452023. It was determined that the backdraft damper would not return to the closed position and was stiff to operate. Per the work instructions, maintenance personnel were to inspect the damper and operating linkage for issues. Maintenance personnel initiated a one-time change per MG-0006 to change the spring assembly because it did not appear to have the proper tension. Later in the work instructions it requires maintenance to full stroke the backdraft damper. The instructions included a clarifying statement that full stroke is interpreted as against a mechanical stop. Maintenance personnel attempted to perform this step multiple times but the backdraft damper would not go full closed.

At this point maintenance personnel returned to the shop and discussed it with other personnel, and the supervisor, and was informed that the damper only opens “about 20 percent” during normal operation. On November 3, 2013, maintenance personnel returned to the damper and verified that it did close from the 20 percent position and subsequently closed out the work order. The 20 percent position was collective institutional knowledge that did not have any documented basis for this damper or any of the other reactor containment fan cooler backdraft dampers. Due to the orientation of the dampers in the system they will see different flows and different open positions. No technical data or basis is documented to support the 20 percent statement.

In accordance with Procedure MG-0006, step 6.2.3, “Work Instruction Alteration,” a pen and ink change which does not alter the scope may be made by the work supervisor or planner. This change SHALL be initialed and dated by the person performing the change and SHALL receive a second party technical review which is documented by initial and date. It is clear that the maintenance department understood the requirements for changes because a one-time change was performed for replacement of the spring. But because the 20 percent position was collective institutional knowledge it was not recognized as a change to the work order and potentially a change that could impact the operability of the reactor containment fan cooler. The licensee’s corrective actions included generating several feedback forms to revise work order instructions to allow personnel to complete the maintenance instructions as written, documenting the as-found” condition of backdraft damper 12C to allow for trending, and evaluating human performance concerns for using institutional knowledge without basis.

Analysis. The team determined that failure to follow Procedure MG-0006 to complete the preventative maintenance work order on reactor containment fan cooler 12C as instructed was a performance deficiency. This finding was more than minor because it adversely affected the Mitigating Systems Cornerstone attribute of Procedure Quality and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, not performing a proper procedure change does not ensure a proper technical review of the change and had the potential to challenge the availability and capability of the reactor containment fan cooler. In accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated June 19, 2012, Exhibit 2, “Mitigating Systems Screening Questions,” the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance, resources component because procedures were not available to ensure successful work performance [H.1].

Enforcement. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” which states, in part, “Activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these procedures.” Contrary to the above, on November 3, 2013, the licensee failed to ensure that activities affecting quality was prescribed by documented procedures of a type appropriate to the circumstances and failed to accomplish these activities in accordance with these procedures. Specifically, maintenance personnel performing a maintenance activity change and performing the second party technical review did not initial and date the change that was performed for reactor containment fan cooler 12C backdraft damper as required by Procedure MG-0006, “Work Execution and Closeout Guideline,” Revision 11, step 6.2.3. In response to this issue, the licensee initiated revisions to the associated work order instructions and established as-found trend data for backdraft damper 12C. This finding was entered into the licensee's corrective action program as Condition Reports 14-1820 and 14-1836. 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 non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000498/2013007-04, "Failure to Follow Preventative Maintenance Procedure for Reactor Containment Fan Cooler."

.2.10 Containment Electrical Penetrations

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 the Electrical Containment Penetrations. 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:

- One-line diagrams for containment electrical penetrations.
- Cable schedule and routing for the containment electrical penetration cables.
- Cable sizing and material requirements for the penetrations.
- Design documentation and analyses for the seals used for the penetrations.
- Cable protection evaluation for the penetration cables.
- Leak tests and surveillance tests performed on the penetrations and associated cables.
- Preventive maintenance activities performed on the penetrations and associated cables.

b. Findings

No findings were identified.

.2.11 Steam Generator Power Operated Relief Valves FV-7411, FV-7421, FV-7431, and FV-7441

a. Inspection Scope

The team reviewed the updated safety analysis report, the current system health report, selected drawings, maintenance and test procedures, condition reports and design change packages associated with the steam generator power operated relief valves for both units. The team also conducted walkdowns of both units and conducted interviews with engineering personnel to ensure the capability of these components to perform their desired design basis functions. Specifically, the team reviewed:

- Health reports associated with the power operated relief valves for the last several years.

- Maintenance work order history and corrective action program reports from 1989 to 2013 for any common problems or issues.

- Design change packages, DCP 08-9595-10 and DCP 08-9595-11, steam generator power operated relief valve fail close modification and associated 10 CFR 50.59 screenings, evaluations, and updated final safety analysis report changes.

- Operability determinations associated with failure of power operated relief valves to close and the effects of leakage on valve operability (Condition Reports 10-18770-1, 10-18770-16, and 12-21808-4).

- Surveillance test procedures and test results.

b. Findings

No findings were identified.

.2.12 Technical Support Center Diesel Generator

b. Inspection Scope

The team reviewed the updated safety analysis report, the current system health report, selected drawings, calculations, maintenance and test procedures, condition reports and design change packages associated with the Technical Support Center diesel generators. The team also conducted walk downs of the Technical Support Center diesel generator in Unit 2. In addition, the team conducted interviews with engineering personnel to ensure the capability of these components to perform their desired design basis functions. Specifically, the team reviewed:

- Schematics for the emergency diesel generator start and trip circuits and for generator breaker close and trip circuits.

- Maintenance work order history and corrective action program reports from 1989 to 2013 for any common problems or issues.

- One-line diagrams for the electrical distribution system to identify requirements and interfaces.

- Vendor nameplate and equipment specification data.

- Calculations for determining diesel generator load under design basis conditions.

- Calculations and supporting documentation for determining horsepower and other power system loads on the diesel generator.

- Preventive maintenance procedures to verify the diesel generator is maintained according to manufacturer recommendations.

- Analysis of flooding potential for the Technical Support Center diesel generator building in both units and the effects on plant requirements for Technical Support Center diesel generators.

- Periodic load testing to demonstrate system capability for design basis conditions.

b. Findings

No findings were identified.

.2.13 Positive Displacement Pump

a. Inspection Scope

The team reviewed the updated safety analysis report, the current system health report, selected drawings, calculations, maintenance and test procedures, condition reports and design change packages associated with the positive displacement pumps. The team also conducted walk downs of the positive displacement pump in Unit 2. The team also conducted interviews with engineering personnel to ensure the capability of these components to perform their desired design basis functions. Specifically, the team reviewed:

- Maintenance work order history and corrective action program reports from the past five years.

- Surveillance test results for the positive displacement pumps. Emphasis was placed on the capability of the positive displacement pump to be powered from the Technical Support Center diesel generator.

- Procedures utilized to power the positive displacement pump from the Technical Support Center diesel generator.

b. Findings

No findings were identified.

.2.14 Electrical Auxiliary Building Heating Ventilation and Air Conditioning

b. Inspection Scope

The team reviewed the updated safety analysis report, the current system health report, selected drawings, calculations, technical specifications, maintenance and test procedures, condition reports, operating procedures, and design change packages associated with the electrical auxiliary building heating, ventilation, and air conditioning system. The team also reviewed operating procedures that direct operations personnel to take certain actions upon loss of all heating, ventilation, and air conditioning in the electrical auxiliary building. The team conducted walk downs of the electrical auxiliary building heating, ventilation, and air conditioning systems. The team also conducted interviews with engineering, probabilistic risk assessment, and operations personnel to ensure the capability of these components to perform their desired design basis functions. Specifically, the team reviewed:

- System health reports associated with the electrical auxiliary building heating, ventilation, and air conditioning systems and components.

- Maintenance work order history and corrective action program reports from the past five years.

- Procedures 0POP04-HE-0001, “Loss of EAB or Control Room HVAC” and 0POP10-HE-0001, “Loss of EAB HVAC” for procedure logistical planning, validation, and implementation.

- Separation criteria for the electrical auxiliary building heating, ventilation, and air conditioning chilled water piping, as described in the South Texas Project updated final safety analysis report.

b. Findings

No findings were identified.

.2.15 Auxiliary Feedwater Cross Connect Air Operated Valves

a. Inspection Scope

The team reviewed the updated safety analysis report, design basis documents, system description, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the auxiliary feedwater cross connect air operated valves. The team also performed walkdowns and conducted interviews with system engineering and operations personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

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

- Operations procedures and training for positioning or determining that the valves are in the locked neutral position.

b. Findings

No findings were identified.

.2.16 Safety Injection Pump Room Coolers

a. Inspection Scope

The team reviewed the updated safety analysis report, design basis documents, system description, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the safety injection pump room coolers to ensure design basis requirement specifications were met. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.
- License amendment request and related safety evaluation report for the approval of the risk informed inservice testing program.
- Purchase specifications requirements and comparison with the design basis documents and the heat transfer calculations.
- The original startup flow balance calculation, including annubar flow element constants.

b. Findings

Failure to Maintain Design Control of Safety Injection Pump Room Cooler

Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," involving the licensee's failure to assure that the applicable design basis requirements, associated with the safety injection pump room coolers, were correctly translated into the plant design.

Description. While conducting a review of the safety injection pump room cooler heat transfer properties and assumption used in Calculations MC-06482, "Essential Chilled Water / EAB HVAC Design Basis Loads with Capacity of 300 Tons Per Train," Revision 3, and MC-06482A, "Essential Chilled Water Minimum Flow Requirements for EAB, CRE, FHB, and MAB Coolers," Revision 0, the team noted a discrepancy in one of the design parameters. The team determined that the purchase specification for the room cooler, 3V259VS0005, "Specification for Safety Class Air Handling Units," Revision 2, identifies the numbers of fins per inch as a maximum of 8. However, calculations MC-06482 and MC-06482A both use 11 fins per inch. The assumption in the calculation specifies that all coolers have 8 fins per inch except the safety injection pump room coolers which have 11. The licensee initiated Condition Report 14-2673 and determined that the purchase specification for the spare room cooler appeared to have been used incorrectly as the source data for the heat transfer calculations.

The original design was to have a separate cooler for each safety injection pump, but was changed during construction to only use one cooler. The change was not verified to be correctly translated to the design documentation that performed the heat transfer calculations to determine the maximum room air temperature to ensure the operability of the safety injection pumps. Ultimately, in addition to the number of fins per inch being incorrect, it was identified that the number of tubes per row, the face area, the height of the face area, and the tube length were also incorrect. These are the original calculations of record and, as such, have been incorrect prior to February 2014. The licensee's corrective actions included correcting the errors in calculations MC-06482 and MC-06482A and determining that the room coolers remained operable as design margin existed between the calculated maximum room air temperature, 95 degrees F, and the design room air temperature, 120 degrees F.

Analysis. The team determined that the failure to maintain design control of the safety injection pump room cooler was a performance deficiency. This finding was more than minor because it adversely affected the Mitigating Systems Cornerstone attribute of Design Control and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, not maintaining design control and performing a proper heat transfer calculation had the potential to challenge the availability, reliability, and capability of the safety injection pump room cooler and in turn the safety function of safety injection pumps. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. The team determined that this finding did not have a cross-cutting aspect because the most significant contributor did not reflect current licensee performance.

Enforcement. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "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, prior to February 13, 2014, the licensee failed to establish measures to assure that applicable regulatory requirements and the design basis were correctly translated into specifications, drawings, procedures, and instructions. Specifically, documented requirements in purchase specification 3V259VS0005 were not correctly translated into specifications, drawings, and instructions evaluated in calculations MC-06482 and MC-06482A for the safety injection pump room coolers. In response to this issue, the licensee revised the associated calculations and established that the room coolers remained operable. This finding was entered into the licensee's corrective action program as Condition Report 14-2673. 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 non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000498/2013007-05, 05000499/2013007-05, "Failure to Maintain Design Control of Safety Injection Pump Room Cooler."

.2.17 Auxiliary Feedwater Steam Generator Isolation Motor Operated Valves

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 the Auxiliary Feedwater Steam Generator isolation motor operated valves. 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:

- Maximum expected differential pressure, required stem thrust, and stroke time calculations.
- Calculations and design basis documents to ensure acceptance criteria for tested parameters were valid to support operation under accident conditions.
- Component maintenance history and corrective action program reports to verify that degraded conditions were being appropriately addressed.
- Procedures for preventive maintenance, inspection, and testing.
- Calculations for determining minimum motor terminal voltage under design and licensing basis conditions.
- Calculations for determining minimum contactor terminal voltage under design and licensing basis conditions.
- Environmental design requirements under design and licensing basis conditions.

b. Findings

Failure to Evaluate the Adequacy of Voltage Available at AF-19 Valve Motor to Close the Valve During Postulated High Energy Line Break Conditions

Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," involving the licensee's failure to assure that applicable regulatory requirements and the design basis, are correctly translated into specifications, drawings, procedures, and instructions. Specifically, the licensee failed to evaluate the adequacy of voltage available at the AF-19 valve motor operator to close the valve during postulated high energy line break conditions.

Description. The team found that valve AF-19 was required to close to terminate auxiliary feedwater flow under certain accident scenarios (for example, for high energy line break conditions, such as main steam and steam generator blowdown system line breaks) as stated in the Updated Final Safety Analysis Report, Table 10.4-3a, "HELB Failure Modes and Effects Analysis of Auxiliary Feedwater System Electrical Equipment in IVC." However, on review of the Class 1E DC system battery sizing and system voltage calculation, the team found that closing valve AF-19, when required, was not modeled in the analysis and the adequacy of voltage at the valve motor for closing the valve was not determined. The licensee performed a preliminary battery sizing and voltage analysis during the inspection to address this error and verified that the valve motor had sufficient voltage to close when required by the failure modes and effects analysis.

Analysis. The team determined that the failure to evaluate and translate the requirements for adequate voltage available at the AF-19 valve motor to close the valve during postulated high energy line break conditions was a performance deficiency. This finding was more than minor because if left uncorrected, it would lead to a more significant safety concern. Specifically, the failure to analyze and translate the relevant requirements resulted in a condition where there was a reasonable question on the capability of the valve to close when required during postulated high energy line break conditions. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. The team determined that this finding did not have a cross-cutting aspect because the most significant contributor did not reflect current licensee performance.

Enforcement. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions." Contrary to the above, prior to January 28, 2014, the licensee failed to establish measures to assure that applicable regulatory requirements and the design basis were correctly translated into specifications, drawings, procedures, and instructions. Specifically, the licensee failed to adequately verify by analysis that the AF-19 valve motor had adequate voltage available to close the valve when required during postulated high energy line break conditions. In response to this issue, the licensee performed a preliminary battery sizing and voltage analysis and verified that the valve motor had sufficient voltage to close when required by the failure modes and effects analysis. This finding was entered into the licensee's corrective action program as Condition Report 14-1374. 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 non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy:
NCV 05000498/2013007-06, 05000499/2013007-06, "Failure to Evaluate the Adequacy

of Voltage Available at AF-19 Valve Motor to Close the Valve During Postulated High Energy Line Break Conditions."

.2.18 Essential Cooling Water Screen Wash System

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 the essential cooling water screen wash system. 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:

- License Amendments to remove the screen wash system from technical specification surveillances.
- System differential pressure start setpoint verification calculation and procedure.
- Management of bio-fouling in the essential cooling water pond.
- Procedures for preventive maintenance and inspection.
- Component maintenance history and corrective action program reports to verify that degraded conditions were being appropriately addressed.

b. Findings

No findings were identified.

.3 Results of Reviews for Operating Experience:

.3.1 Inspection of NRC Information Notice 1989-54 "Potential Overpressurization of the Component Cooling Water System"

b. Inspection Scope

The team reviewed the licensee's evaluation of Information Notice 1989-54 "Potential Overpressurization of the Component Cooling Water System" to verify the licensee performed an applicability review and took appropriate corrective actions, if appropriate, to address concerns that could result from the failure of the component cooling water tubing within the thermal barrier heat exchanger of the reactor coolant pump. The team verified that the licensee's review adequately addressed the issues in the information notice.

b. Findings

No findings were identified.

.3.2 <u>Inspection of NRC Information Notice 1990-26 "Inadequate Flow of Essential Service Water to Room Coolers and Heat Exchangers for Engineered Safety-Feature Systems"</u>

a. <u>Inspection Scope</u>

The team reviewed the licensee's evaluation of Information Notice 1990-26 "Inadequate Flow of Essential Service Water to Room Coolers and Heat Exchangers for Engineered Safety-Feature Systems" to verify that the licensee performed an applicability review and took appropriate corrective actions, if appropriate, to address the concerns described in the information notice. This information notice discusses potential problems resulting from using the wrong flow and pressure drop relationship in establishing adequate flow of essential service water to room coolers for engineered safety-feature systems and from failing to establish or maintain balanced flow in essential service water systems. The team verified that the licensee's review adequately addressed the issues in the information notice.

b. <u>Findings</u>

No findings were identified.

.3.3 <u>Inspection of NRC Information Notice 2010-23, "Malfunctions of Emergency Diesel Generator Speed Switch Contacts"</u>

a. <u>Inspection Scope</u>

The team reviewed the licensee's evaluation of Information Notice 2010-23, "Malfunctions of Emergency Diesel Generator Speed Switch Contacts," to verify that the licensee performed an applicability review and took appropriate corrective actions, if appropriate, to address the concerns described in the information notice. The team verified that the licensee's review in Condition Report 10-24261 adequately addressed the issues in the information notice. Additionally, the team reviewed actions completed in Condition Report 11-11508 to verify that corrective actions were being implemented.

b. <u>Findings</u>

No findings were identified.

.3.4 <u>Inspection of NRC Information Notice 2012-11, "Age Related Capacitor Degradation"</u>

a. <u>Inspection Scope</u>

The team reviewed the licensee's evaluation of Information Notice 2012-11, "Age Related Capacitor Degradation" to verify that the licensee performed an applicability review and took appropriate corrective actions, if appropriate, to address the concerns described in the information notice. This information notice discusses potential problems resulting from adversely affected capacitors due to age causing the epoxy insulation to harden and crack over time. This degrades the capacitor, allowing a high flow of current and excessive heating. The excessive heat can then ignite the epoxy. The team verified that the licensee's review adequately addressed the issues in the information notice.

b. Findings

No findings were identified.

.4 Results of Reviews for Operator Actions:

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

a. Inspection Scope

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

The selected operator actions were:

- Scenario 1, Part 1: The scenario was designed to place the crew in a situation where they will need to trip a Reactor Coolant Pump that had lost all seal cooling and place the positive displacement pump in service to restore seal cooling.

- Scenario 1, Part 2: The scenario used time compression to move the crew later in the event timeline. Issues with the positive displacement pump resulted in a loss of seal cooling to reactor coolant pump 1B for a period of time and resulted in a small loss of coolant accident. During the implementation of procedure 0POP05-EO-EO10, "Loss of Reactor or Secondary Coolant," Step 20, the crew determines that cold leg recirculation capability cannot be verified and transitions to procedure POP05-EO-EC11, "Loss of Emergency Coolant Recirculation." The crew must then perform a blend to refill the refueling water storage tank.

- Scenario 2: The scenario was designed to place the crew in a situation where reactor containment fan coolers are needed and do not operate automatically such that operator action is needed to place the reactor containment fan coolers in service. To accomplish this, the scenario initiates a loss of coolant accident in containment followed by reduced containment spray availability. Containment pressure rises and the reactor containment fan coolers are needed to control containment pressure.

- In-plant job performance measure: This job performance measure was designed for a plant operator to demonstrate the correct field actions required to refill the refueling water storage tank due to a loss of emergency coolant recirculation following a loss of coolant accident.

b. Findings

Failure to Develop Adequate Procedures for Loss of All Seal Cooling to a Reactor Coolant Pump

Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," involving the licensee's failure to have procedures with appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee failed to develop adequate procedures for responding to a loss of all seal cooling to a reactor coolant pump.

Description. On January 29, 2014, two operating crews were observed during simulator scenarios that required operators to respond to a loss of all seal cooling to a reactor coolant pump. As evidenced by the performance of the crews, Procedures 0POP05-EO-EO00, "Reactor Trip or Safety Injection," Revision 22, 0POP04-RC-0002, "Reactor Coolant Pump Off Normal," Revision 32, and 0POP09-AN-04M8, "Annunciator Lampbox 04M8 Response Instructions," Revision 39, were inadequate for an accident sequence that involved the loss of component cooling water to a reactor coolant pump thermal barrier followed by a loss of the only available centrifugal charging pump. This simulated condition resulted in a complete loss of seal cooling to one reactor coolant pump. Specifically, both operating crews failed to restore seal injection to the affected reactor coolant pump using the positive displacement pump within the six minute timeframe outlined by the licensee's probabilistic risk assessment to prevent the increased risk of a reactor coolant pump seal loss of coolant accident. Further, the operating crew who performed the scenario validation in the simulator on January 13, 2014, also failed to restore seal injection using the positive displacement pump.

Additionally, one crew took actions that would have further degraded the potential seal failure by failing to stop the affected reactor coolant pump within one minute and then initiating seal injection with seal inlet temperature above 230 degrees F, which is contrary to the direction provided by Procedure 0POP04-RC-0002, "Reactor Coolant Pump Off Normal," Revision 32. To restore seal injection, the operating crew utilized Procedure 0POP09-AN-04M8, "Annunciator Lampbox 04M8 Response Instructions," Revision 39, which did not contain the same caution to avoid restoring seal injection once seal inlet temperature exceeded 230 degrees F.

The associated procedures did not include sufficient direction to ensure that reactor coolant pump seal cooling was restored within the risk-significant timeframe. Further, Procedure 0POP09-AN-04M8, "Annunciator Lampbox 04M8 Response Instructions," Revision 39, did not contain adequate direction to prevent further degradation of the reactor coolant pump seal.

Analysis. The team determined that the failure to include appropriate qualitative and quantitative criteria in emergency operating procedures, off-normal operating procedures, and annunciator response procedures for a loss of all seal cooling to a reactor cooling pump was a performance deficiency. This finding was more than minor because it adversely affected the Initiating Events Cornerstone attribute of Procedure

Quality and affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, operating procedures did not contain appropriate attributes to ensure timely action to prevent an increased likelihood of a reactor coolant pump seal loss of coolant accident. In accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated June 19, 2012, Exhibit 1, “Initiating Events Screening Questions,” the team determined a detailed risk evaluation was necessary because, after a reasonable assessment of degradation, the finding could result in exceeding the reactor coolant system leak rate for a small loss of coolant accident. Therefore, the senior reactor analyst performed a bounding detailed risk evaluation. The analyst determined that the change to the core damage frequency would be 1E-7 per year (Green). This finding had a cross-cutting aspect in the area of human performance, training component because the licensee did not provide training and ensure knowledge transfer to maintain a knowledgeable, technically competent workforce and instill nuclear safety values [H.9].

Enforcement. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” which states, in part, “Instructions, procedures, or drawings shall include appropriate qualitative and quantitative acceptance criteria for determining that important activities have been satisfactorily accomplished.” Contrary to the above, prior to January 29, 2014, the licensee failed to include appropriate qualitative and quantitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee failed to include appropriate qualitative and quantitative criteria in emergency operating procedures, off-normal operating procedures, and annunciator response procedures that are used during a loss of all seal cooling to a reactor coolant pump to prevent increased risk of a reactor coolant pump seal loss of coolant accident. In response to this issue, the licensee implemented changes to the affected procedures and communicated the changes to the operating staff. This finding was entered into the licensee’s corrective action program as Condition Report 14-1635. 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 non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy: 05000498/2013007-07, 05000499/2013007-07, “Failure to Develop Adequate Procedures for Loss of All Seal Cooling to a Reactor Coolant Pump.”

4 OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

The team reviewed actions requests associated with the selected components, operator actions, and operating experience notifications. Any related findings are documented in prior sections of this report.

4OA6 Meetings, Including Exit

On February 14,2014, the team leader presented the preliminary inspection results to T. Powell, Site Vice President, and other members of the licensee’s staff. On March 20, 2014, the inspectors discussed the final results of this inspection with T. Powell, Site

Vice President and other members of the licensee’s staff. The licensee acknowledged the findings during the meeting. While some proprietary information was reviewed during this inspection, no proprietary information was included in this report.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of the NRC Enforcement Policy for being dispositioned as a non-cited violation.

- Title 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, “Activities affecting quality shall be prescribed by procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these procedures.” Contrary to the above, on January 22, 2014, an activity affecting quality was not accomplished in accordance with procedures. Specifically, the licensee failed to follow Procedure 0PGP03-ZX-0002A, “Condition Report Process Implementation,” Revision 1, step 4.4 to ensure that a prompt operability determination was completed on two reactor containment fan coolers with backdraft dampers found approximately 50 percent open. The finding was determined to be of very low safety significance because the safety function was never lost and was determined to be operable but degraded. The licensee entered this issue in their corrective action program as Condition Reports 14-1102, 14-1106, and 14-2726.

ATTACHMENT 1 – SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel
C. Albury, Supervisor, Nuclear Fuel and Analysis
M. Berg, Manager, Design Engineering
M. Berg, Engineer, Maintenance Engineering
C. Bowman, General Manager, Engineering
B. Brown, Supervisor, Maintenance Engineering
J. Cook, Supervisor, Design Engineering
R. Dunn, Manager, Nuclear Fuel and Analysis
K. Frazier, Supervisor, Systems Engineering
C. Georgeson, Supervisor, Design Engineering
D. Gore, Supervisor, Nuclear Fuel and Analysis
W. Harris, Engineer, Systems Engineering
A. Hasan, Engineer, Systems Engineering
E. Heacock, Electrical Engineer, DP Engineering
G. Hildebrandt, Manager, Operations
Q. Huynh, Mechanical Engineer, Design Engineering
R. Kersey, Supervisor, Design Engineering
R. Lacey, Electrical Engineer, Design Engineering
H. Leon, Electrical Engineer, Design Engineering
M. Meier, Vice President, Corporate Services
J. Milliff, Manager, Operations Support
J. Morris, Engineer, Regulatory Affairs
L. Peter, Plant General Manager
T. Powell, Site Vice President
R. Savage, Engineer, Regulatory Affairs

NRC personnel
F. Sanchez, Senior Resident Inspector
N. Hernandez, Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000498/2013007-01, 05000499/2013007-01	NCV	Failure to Properly Evaluate Safety-Related Equipment Electrical Load Requirements when Verifying the Adequacy of Voltage from the Nuclear Steam Supply System Inverter/Rectifier
05000498/2013007-02, 05000499/2013007-02	NCV	Improper Sequencing of Maintenance of 4160 VAC Circuit Breakers Prior to As-Found Tests
05000498/2013007-03, 05000499/2013007-03	NCV	Failure to Establish an Adequate Test Program for Safety-Related 480 VAC Circuit Breakers

<u>Opened and Closed</u>

05000498/2013007-04	NCV	Failure to Follow Preventative Maintenance Procedure for Reactor Containment Fan Cooler
05000498/2013007-05, 05000499/2013007-05	NCV	Failure to Maintain Design Control of Safety Injection Pump Room Cooler
05000498/2013007-06, 05000499/2013007-06	NCV	Failure to Evaluate the Adequacy of Voltage Available at AF-19 Valve Motor to Close the Valve During Postulated High Energy Line Break Conditions
05000498/2013007-07, 05000499/2013007-07	NCV	Failure to Develop Adequate Procedures for Loss of All Seal Cooling to a Reactor Coolant Pump

LIST OF DOCUMENTS REVIEWED

Calculations

NUMBER	TITLE	REVISION/DATE
EC-5030	Class 1E Diesel Generator Protection	5
EC-5000	Voltage Regulation Study	15
EC-5001	Fault Analysis	8
EC-5002	Electrical Auxiliary Power Distribution System Model	16
MC-6462	DVAC Calculation for DC Motor MOVs	0
EC-5008	Class 1E DC System Scenario, Battery/Charger/Inverter Sizing & System Voltage Calculation	14
EC-5039	Control Cable Voltage Drop Verification	5
EC-5018	Short Circuit Current Analysis – Class 1E 125 VDC Non-Class 1E 250, 125 and 48 VDC Systems	8
EC-6053	Protective Device Study for Appendix R, Unit 2	3
EC-5053	Protection of Electrical Penetration Conductors	5

Calculations

NUMBER	TITLE	REVISION/DATE
EC-5037	Maximum Allowable Length of AC Power Cables	5
EC-5028	Protection 13.8 KV Switchgear	7
EC-5038	Power Cable Sizing Verification	9
14926-4014-00031-AAA	Cooling Coil Performance Data for Reactor Containment Fan Coolers Based on Specification 2V211VS0001	December 3, 1985
14926-4384-00021-AKJ	Annubar Flow Calculations	March 13, 1985
1-CH-P-01	Essential Chilled Water System	1
5V019VQ1031	HVAC Cooling/Heating Load Calculations	5
NC-7008	Pressurizer Surge Line Break P/T Analysis	3
MC-06482	Essential Chilled Water / EAB HVAC Design Basis Loads with Capacity of 300 Tons per Train	3
MC-06482A	Essential Chilled Water Minimum Flow Requirements for EAB, CRE, FHB, and MAB Coolers	0
34753/2-48	Operability Analysis Test Report, Purge Containment Isolation Valve	April 17, 1987
NC-6013	Control Room, TSC, and Offsite LOCA Radiation Doses	November 28, 2006
NC-7121	Critical Mass Flowrate Through Supplementary Purge Valve Following LOCA	July 30, 2003
MC-5872	IVC,AFT, Verification that the Auxiliary Feedwater System can initiate flow within 60 seconds	2
MC-6507	AF, MEDP and Stem Thrust Calculations for AFT System MOVs	0
6458-00085-UZ	Failure Mode Analyses Size 4 Class 1080 Non-Return Globe Valve figure number 2006JMPQTY	A
4L529TB1000	Design Specification for ASME Section III Gate, Globe, and Check Valves 2-1/2 Inches and Larger	4

Procedures

NUMBER	TITLE	REVISION/DATE
0POP05-EO-EO00	Reactor Trip or Safety Injection	22
0POP05-EO-EO10	Loss of Reactor or Secondary Coolant	21
0POP04-RC-0002	Reactor Coolant Pump Off Normal	32
0POP02-CV-0001	Makeup to the Reactor Coolant System	43
0POP04-AE-0001	First Response to Loss of Any or All 13.8 KV or 4.16 KV Bus	44
0POP05-EO-EC00	Loss of All AC Power	23
0POP05-EO-EC11	Loss of Emergency Coolant Recirculation	18
0POP09-AN-04M7	Annunciator Lampbox 4M07 Response Instructions	28
0POP04-HE-0001	Loss of EAB or Control Room HVAC	11
0POP10-HE-0001	Loss of EAB HVAC	0
OPSP03-AF-0010	Auxiliary Feedwater System Valve Operability Testing	2
OPSP11-ZE-0006	MOV IST Margin and Test Frequency Determination	0
OPSP11-ZE-0001	Check Valve Inspection Unit 1, PM 94000755	10
0POP01-ZO-0004	Extreme Weather Guidelines	33
0PGP03-ZE-0037	Generic Letter 89-10/96-05 Motor Operated Valve Program	6
0PGP03-ZE-0080	Essential Cooling Water System Reliability Program	0
0PMP05-ZE-0312	Limitorque MOV Actuator Lubrication	26
0POP05-EO-FRC1	Response to Inadequate Core Cooling	15

Procedures

NUMBER	TITLE	REVISION/DATE
0PSP07-VE-0005	Gaseous Effluent Dose Assessment	7
0PGP03-ZX-0002C	Common Cause Analysis and AFI Investigations	2
0POP07-CV-0001	Positive Displacement Charging Pump Functional Verification	10
0PGP03-ZA-0514	Controlled System or Barrier Impairment	11
0POP02-DB-0005	Technical Support Center Diesel Generator	35
0PSP03-HC-0003	Reactor Containment Building Supplementary Purge System Valve Operability Test	10
0POP02-HC-0003	Supplementary Containment Purge	25
0POP04-RC-0006	Shutdown LOCA	16
ENG-0007	Predictive Maintenance Administrative Guideline	0
0POP05-EO-EO20	Faulted Steam Generator Isolation	11
0POP05-EO-EO30	Steam Generator Tube Rupture	26
0POP05-EO-ES11	SI Termination	11
0POP05-EO-ES12	Post LOCA Cooldown and Depressurization	17
0POP05-EO-ES33	Post-SGTR Cooldown Using Steam Dump	16
1POP09-AN-03M2	Annunciator Lampbox 1-03M-2 Response Instructions	30
0POP09-AN-0729	TSC DG Local Annunciator Lampbox 1(2)-729 Response Instructions	11
0PMP05-VA-0007	120 VAC NSSS Vital Inverter/Rectifier (10KVA) Performance Test	13
0PMP05-VA-0006	120 VAC NSSS Vital 10KVA Inverter/Rectifier Maintenance	14

Procedures

NUMBER	TITLE	REVISION/DATE
OPMP05-NA-0002	4160V Gould Breaker Tests	26
OPSP11-PH-0001	LLRT: Electrical Penetration	10
3E269ES0029	Specification for Electrical Penetration Assemblies	3
0PMP05-PK-1001	4160 Volt Class 1E Switchgear Maintenance	11
0PMP05-PK-0018	4160 Volt Gould Breaker Overhaul/Lubrication	14
OPEP06-ZA-0002	Infrared Thermography Program Description	6
0PEP06-ZG-0013	Infrared Thermography Data Collection	9
OPGP03-ZM-0021	Control Configuration Changes	19
OPMP05-NA-0014	ABB Type L2 Switch Inspection and Refurbishment	4
OPMP05-PM-4800	Motor Control Center Maintenance ITE Gould	19
OPMP05-NA-0009	G.E. Magne-Blast Breaker Overhaul Lubrication	26
OPMP05-NA-0001	General Electric 13.8KV Breaker Tests	35
5E189ES1004	Specification for Cable Splicing, Termination, and Supports	13
OPMP08-ZI-0011	Generic Temperature Switch Calibration (Filled Element)	20
OPMPO5-PL-1051	Switchgear Maintenance-Class 1E Westinghouse Type DS 480V Load Centers	21
OPSP03-MS-0001	Main Steam System Valve Operability Test	41
OPGP04-ZA-0607	Cable Management Program	0
OPMP05-NA-0004	Molded Case Circuit Breaker Tests	33

Procedures

NUMBER	TITLE	REVISION/DATE
OPGP03-ZX-0002	Condition Reporting Process	47
OPGPO3-ZX-0002A	Condition Reporting Process Implementation	1
OPMP05-NA-0017	480 Volt Type K Breaker Overhaul/Lubrication (Generic)	9
OPMP05-NA-0019	480 Volt ITE LK Breaker Overhaul/Lubrication (Generic)	4
OPMP05-RS-0003	Westinghouse 480 Volt Trip Breaker Test	21
OPMP05-RS-0004	Reactor Trip Breaker Overhaul/Lubrication	2
OPGP03-ZM-0002	Preventive Maintenance Program	37
OPMP05-NA-0018	4160 Volt Gould HK Breaker Overhaul/Lubrication	14
OPMP05-NA-0007	480 Volt DS Breaker Overhaul/Lubrication	13
OPSP06-NZ-0006	Molded Case Breaker Functional Tests and Inspection	16
OPMP05-NA-0020	480 Volt ITE Breakers	19
PSP06-NZ-0005	480 Volt ITE Test and Inspection	13
PMI-EM-NA-003	Gould 480 Volt Type K Breaker Test	11
0POP09-AN-03M3	Annunciator Lampbox 3M03 Response Instructions	31
0ERP01-ZV-IN01	Emergency Classification	9
0PCP08-AP-0003	Post-Accident Sampling of Liquids and RCB Atmosphere at PASS	8
0PEP02-ZG-0007	Post-Accident Failed Fuel Guidelines	5
0PGP03-ZA-0010	Performing and Verifying Station Activities	34

Procedures

NUMBER	TITLE	REVISION/DATE
0POP02-CC-0001	Component Cooling Water	48
0POP02-CH-0001	Essential Chilled Water	24
0POP02-HC-0001	Containment HVAC	23
0POP04-RA-0001	Radiation Monitoring System Alarm Response	29
0PRP11-ZR-0006	Compensatory Monitoring For RT-8050/8051 Inoperable	3
0PSP03-CC-0001	Component Cooling Water Pump 1A(2A) Inservice Test	17
0PSP03-CC-0007	Component Cooling Water System Train 1A(2A) Valve Operability Test	22
0PSP03-CC-0008	Component Cooling Water System Train 1B(2B) Valve Operability Test	18
0PSP03-CC-0009	Component Cooling Water System Train 1C(2C) Valve Operability Test	20
0PSP03-CH-0001	Essential Chilled Water Pump 11A(21A) Inservice Test	20
0PSP03-CH-0002	Essential Chilled Water Pump 11B(21B) Inservice Test	19
0PSP03-CH-0003	Essential Chilled Water Pump 11C(21C) Inservice Test	19
0PSP03-SP-0009A	SSPS Actuation Train A Slave Relay Test	38
0PSP03-SP-0009B	SSPS Actuation Train B Slave Relay Test	43
0PSP03-SP-0009C	SSPS Actuation Train C Slave Relay Test	38
0PSP03-SP-0010A	Train A ESF Load Sequencer Manual Local Test	25
0PSP03-SP-0010B	Train B ESF Load Sequencer Manual Local Test	
0PSP03-SP-0010C	Train C ESF Load Sequencer Manual Local Test	26

Procedures

NUMBER	TITLE	REVISION/DATE
0PSP03-SP-0013A	Train A ESF Actuation and Response Time Test	14, 15, 20
0PSP03-SP-0013B	Train B ESF Actuation and Response Time Test	17, 22
0PSP03-SP-0013C	Train C ESF Actuation and Response Time Test	22, 24
0PSP03-ZG-0001	Valve Remote Position Indicator Verification Test	23
0PSP03-ZQ-0028	Operator Logs	134
0PSP11-CC-0016	CCW Return From RCP Thermal Barrier Flow Valve Testing	12

Drawings

NUMBER	TITLE	REVISION/DATE
3C019S1604	Structural Reactor Containment Building Internal R.A. Riser Elevations Units No. 1 & 2	9
3V119V10004	Piping & Instrumentation Diagram – HVAC Essential Chilled Water System	9
4C361PCC407	Component Cooling Water “CC”	1
5C159Z00205	Instrument Piping Reactor Containment Building Plan at EL. (-) 2’-0”	9
5C159Z00206	Instrument Piping Reactor Containment Building Plan at EL. (-) 2’-0”	8
5C159Z00207	Instrument Piping Reactor Containment Building Plan at EL. (-) 2’-0”	8
5C159Z00208	Instrument Piping Reactor Containment Building Plan at EL. (-) 2’-0”	6
5Q069F05030	Piping & Instrumentation Diagram Radioactive Vent & Drain System Sump Pumps	16
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PD89272	Hydraulic Schematic	B
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4041-01215CE	Control Schematic (Generator) (High Schematic and Regulator)	K
4041-01217CE	Control Schematic (Generator) (Voltage Regulator & Tripping)	L
52769-E0334	Gould Control Diagram	HC4
D72 12400 760	Interconnection Diagram	E
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B03050--00008H4	Schematic Drawing Production 10kVA Inverter 125V DC 120V AC 1-phase 60HZ	B
B03050--00010H4	Outline 10KVA Inverter Front Panel and Connection Details	A
B03050--00009H4	Outline Production 10KVA Inverter	A
B03050--00011H4	Vendor Technical Document Ametek/Solid State Controls	C
9E0PCAC Sh. 1	Single Line Diagram 13.8 KV Switchgear 1H (TGB)	14
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Design Basis Document

NUMBER	TITLE	REVISION
9E519EB1117	Design Basis Document 13.8kV Auxiliary Power System (PC) System	2
2V149VQ1014	Reactor Containment Fan Cooler System	3
2V211VS0001	Specification for Reactor Containment Fan Cooler	2
3V259VS0005	Specification for Safety Class Air Handling Units	2
5A050GAAF01	Auxiliary Feedwater System Risk Significance Basis Document	5
5A050GACC01	Component Cooling Water System Risk Significance Basis Document	5
5A050GACH01	Essential Chilled Water System Risk Significance Basis Document	5
5A050GAEW01	Essential Cooling Water System Risk Significance Basis Document	5
5A050GAHC01	Reactor Containment Building HVAC System Risk Significance Basis Document	5
5S149MB01016	Auxiliary Feedwater System	6
5V219SQ1008	RCFC Duct and Support Structures	2
MRBD	Maintenance Rule System Scoping Basis Report	March 8, 2012
STI 31316240	South Texas Project Units 1 and 2 1.4-Percent Power Uprate Project NSSS Engineering Report	July 2001
5V119VB01022	HE/HE (CRE) Systems	May 3, 2005
2V149VB00114	RCB HVAC System	4
5N049EB01118	Station Blackout	2
5R170MD1017	Chemical and Volume Control - System Description	January 24, 1990

Design Basis Document

NUMBER	TITLE	REVISION
9Q010MD0119	Non-Class 1E Diesel Generators	2
5V119VD0106	EAB HVAC System Description	April 24, 1985
5R289MB01006	Essential Cooling Water System	6
5S109MB01026	Main Steam System	4

Design Change Packages

NUMBER	TITLE	REVISION/DATE
11-31799-1	Revise VTD-A363-0021 to Add Unit 2 Factory Test Data for Inverters	0
10-123-429	Alternate Part for Use In AMETEK Inverter	0
13-50-18	Alternate Oscillator Board for the AMETEK 10KVA Inverter	0
98-687-9	Replace Obsolete Class 1E MCC E1A2 and E1A4 Motor Controller and Circuit Breaker Units	1
08-9595-10	Steam Generator Power Operated Relief Valves Failed Closed Modification	0
04-7140-10	Replace ESF E2A Agastat Relays and add Load Cell Switch Contacts	0
95-1927-2	Evaluate Valve Packing for Valve 2S141TAF0085	February 28, 1995
95-1927-3	Upgrade MOV Database to Reflect Correct Replacement Spring Packs for AF MOVs	March 28, 1995
07-15455-3	Issue one AFW System MOV Weak Link Document	June 26, 2008

Condition Reports

98-1-33	98-1579	99-405	99-11588	00-8027
00-11657	01-8355	02-143	02-5868	04-7175
05-1321	05-3718	05-8716	05-10609	05-10665
05-11008	05-14683	06-4668	06-10644	06-12426
06-16339	07-340	07-17728	08-4027	08-9595
08-15713	08-15764	09-952	10-9122	10-9239
10-1107	10-11730	10-18357	10-18770	10-22327
10-23772	10-23832	10-24261	11-2472	11-3599
11-4995	11-6081	11-6220	11-6599	11-9699
11-7422	11-11508	11-13155	11-14082	11-20355
11-21295	11-23043	12-11263	12-13333	12-23446
12-24238	12-24662	12-25148	12-26039	12-27328
12-27417	12-28350	12-28974	12-29402	12-31619
12-31893	13-677	13-766	13-4452	13-4633
13-5568	13-6074	13-6851	13-7881	13-9254
13-9325	13-10381	13-10827	13-11120	13-11777
13-12589	13-14068	13-15253	13-15337	14-1102
14-1106	14-1369	14-2459		

Condition Reports Generated During the Inspection

13-13170	14-571	14-618	14-641	14-642
14-643	14-738	14-768	14-807	14-1036
14-1091	14-1189	14-1325	14-1354	14-1373
14-1374	14-1393	14-1394	14-1571	14-1633
14-1772	14-1818	14-1820	14-1836	14-1354
14-1862	14-2017	14-2438	14-2441	14-2458
14-2673	14-2726	14-2734		

Work Orders

465920	464316	417617	427467	443375
429267	409571	410948	385146	231280
377345	413292	426829	420092	246377
439403	439374	351603	409950	385148
377757	427648	418089	419817	246053
464316	417906	407286	383823	415942
415949	239758	360702	415950	368248
362018	392745	270842	314052	257849
360702	327449	189820	251246	455546
408155	431155	458221	430271	129743
454062	441917	437973	442202	445202

Work Orders

446565	532107	359850	429140	426491
418659	267544	384630	422456	449195
457852	290751	391875	425740	449225
459804	295950	396172	425783	449623
487433	327959	396243	428236	451473
489934	358266	400699	431257	452023
489935	358364	402822	446085	453984
378602	409111	447385	456672	378608
418849	447483	457309	384628	421117
448154	457606			

Vendor Documents

NUMBER	TITLE	REVISION/DATE
VTD-B455-0042 (IB 6.2.1.7D)	Installation/Maintenance Instructions, Medium Voltage Power Circuit Breaker Type 5HK 1200 Thru 3000 Amperes 5000 Volts Supplier Number 1B.6.2.1.7D	4
VTD-G080-0209	Instruction and Recommended parts for Maintenance Magne-Blast Circuit Breaker Type AM-13.8 750-5 1200 & 2000 Ampere W/ML-13 General Electric GEK-7345	2
088004-LD2-5	Layout Drawing of 18” NEMA Size 1 FVNR MCC Cubicle	0
088004-WD2-054	Wiring Diagram f or MCC E1B1 Cub. A2	1
088004-BM2-054	Bill of Material For MCC E1B1 Cub. A2	1

Vendor Documents

NUMBER	TITLE	REVISION/DATE
WD-59066-B-01, Sh.1	Size 1&2 FVNR Starter	BC-1
VTD-W120-0250	Vendor Technical Document Westinghouse Maintenance Program Manual For Safety Related Type DS Low Voltage Metal Switchgear (ST 32545065)	1
VTD-B455-0017	Vendor Technical Document for Installation/Maintenance Instruction for Indoor & Outdoor Dry and Cast Transformers 1121/2 Thru 10,000 KVA Supplier No. IBXFI-00, Supplier: Name ABB	3
VTD G080-0079	Machine Field Ground Detector Relay Type PJG12B	0
VTD-B455-0047	Vendor Technical Document for Installation/Maintenance Instruction Metal Clad Medium-Voltage Power Switchgear Type 5HK, 7.5HK and 15HK 5000, 7500 and 15000 Volts	3
NP-30482	Name Plate, Class IE Load Center E1B1 Transformer	1
NP-30479	Name Plate, Class IE Load Center E1B2 Transformer	1
4443-000480B	TSC D/G Vendor Spec Data Sheet	February 23, 1987
VTD-W120-0300	Qualified Display Processing System	0
VTD-W120-0502	7300 Series Power Supply	0
VTD-C600-0001	Operation and Maintenance Instructions Drag Velocity Control Element	6
VTD-U055-0001	Instruction Manual for the Positive Displacement Charging Pumps	3

Miscellaneous

NUMBER	TITLE	REVISION/DATE
9Q149MS104	Specification TSC Diesel Generator	4
WCAP-11273	Vendor Technical Information for Westinghouse Setpoint Methodology for Protection Systems	3
4141-00078WE	Quality Test RPT-Low Volt PWR Ctrl & Inst	June 11, 1987
G32.04 CR-24662	STP Response to NRC IN 2012 Age Related Capacitor Degradation	March 27, 2012
4146-00009tEQ/I	Design Quality Test RPT-Low Volt PWR Ctrl & Inst	November 2, 1984
IMT805.01.HO.02	Steam Generator Power Operated Relief Valve (S/G PORV)	
5E540 EL 7000	Fuse and Relay List Data Base	0
3E179ES1150	Specification for Combination Plug-in Controller Units for Class 1E Motor Control Centers	2
4446-00009FOT (PS-1220)	Design Qualification for Low Voltage Instrumentation Electrical Penetration Assemblies	C
4141-00078-FEW	Qualification Test Report- Low voltage Power Control & Instrumentation Electrical Penetration	3
ACE CR-13-5568-2	Apparent Cause Investigation, Pressurizer Heater Backup Group 1A Feeder Breaker E1A1 Cubicle 4E failed to open remotely and locally	May 6, 2013
LOT201.13	Essential Cooling Water Lesson Plan	15
NLO100.29	Essential Cooling Water and Ventilation System	15
NOC-AE-000719	Proposed Amendment to Technical Specification 3/4.7.4 to Revise the Surveillance Requirements for the Essential Cooling Water System	March 17, 2000
CRM1070	CRM System Guidelines Essential Cooling Water System	4
STI 31289329	Revision to Surveillance Requirements for the Essential Cooling Water System	April 30, 2001

Miscellaneous

NUMBER	TITLE	REVISION/DATE
85-003	Deficiency Evaluation Report	0
AAF-TR-7101	Design and Testing of Fan Cooler-Filter Systems for Nuclear Applications	
LOT202.02	Licensed Operator Training	10
MG-0006	Work Execution and Closeout Guideline	11
NLO 100.07	Non-Licensed Operator Training	8
NLO 200.29	Non-Licensed Operator Training	8
ST-HL-AE-1254	Final Report Concerning the Component Cooling Water System Design	May 30, 1985
STI 32764629	Inservice Testing Program Bases Document	5

ATTACHMENT 2 – DETAILED RISK EVALUATIONS FOR THE SOUTH TEXAS PROJECT COMPONENT DESIGN BASES INSPECTION

Section 1R21.4: NCV 05000498/2013007-07, 05000499/2013007-07; Failure to Develop Adequate Procedures for Loss of All Seal Cooling to a Reactor Coolant Pump

The team leader performed the initial significance determination and used Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings at-Power," Exhibit 1, "Initiating Event Screening Questions," to evaluate this issue. The finding required a detailed risk evaluation because the performance deficiency could lead to a reactor coolant pump seal failure and a loss of coolant accident.

The team's Operations Examiner had observed operator exercises in the South Texas Project simulator. The examiner identified that all three control room operator teams failed to start the positive displacement pump within six minutes following the total loss of seal cooling to reactor coolant pump D. The failure to perform the action in time increased the likelihood that the reactor coolant pump seals could overheat and become damaged, thus inducing a reactor coolant pump seal loss of coolant accident. These accidents can range from 21 gallons per minute to over a few hundred gallons per minute.

The analyst performed a detailed risk evaluation. The analyst noted that, for most initiating events, operators would have sufficient time to perform the action. This was because of operational and procedural differences that allowed operators additional time between event initiation and seal cooling failure. The analyst therefore evaluated each set of sequences individually. First the analyst evaluated the original scenario and then evaluated the most risk significant sequences that were identified using the NRC's Standardized Plant Analysis Risk (SPAR) model for South Texas Project, Revision 8.17. Those initiating events included station blackout, the loss of essential cooling water, the loss of component cooling water, and the loss of all safety related room cooling. The analyst also evaluated external events.

The analyst used event and component data from the SPAR model and from NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," dated February 2007.

The analyst noted a SPAR model error. The model included a reactor coolant pump seal failure mode that was not appropriate for South Texas Project. Specifically, the o-ring extrusion failure mode was active in this model. Since South Texas Project had upgraded to the high temperature reactor coolant pump seals, this failure mode did not apply. The analyst used a change set to set this failure probability to zero. This was used consistently in the calculations for both the nominal case (the case without the performance deficiency) and the current case (the case that included the performance deficiency).

Original Scenario: This event started with the spurious operation of the thermal barrier isolation valve (frequency = 1.3E-3/year). Operators were unable to reopen the closed valve (probability = 1.1E-3). Operators then experienced the failure of the in-service centrifugal charging pump (fail to run probability 8.5E-5 in same 24 hour period). The opposite train centrifugal charging pump was out of service for planned maintenance. To capture similar sequences, the analyst also added in the failure to start and failure to run probabilities (total probability = 4.8E-3). Considering the above, the combined loss of seal cooling initiating event frequency was:

Frequency = 1.3E-3 * 1.1E-3 * 8.5E-5 * 4.8E-3 = 5.8E-13

This value bounded the change to the core damage frequency for this event. Therefore, the analyst did not consider this scenario further.

Loss of All Alternating Current (AC): This initiating event is commonly referred to as a station blackout. During the inspection, the Operations Examiner reviewed the station blackout procedure and determined that there is reasonable assurance that operators could successfully address reactor coolant pump seal cooling during a station blackout. The third step of the procedure directed operators to place the positive displacement pump into service. Therefore, the performance deficiency did not affect these sequences.

Loss of Essential Cooling Water: The initiating event frequency for this event was 2.5E-4 per year. The loss of essential cooling water event cascaded into a loss of component cooling water. The loss of component cooling water then caused the loss of reactor coolant pump thermal barrier cooling as well as the loss of charging pump lubricating oil and room cooling. Within six minutes of the second charging pump failure, operators would need to place the positive displacement pump into service.

Procedure 0POP04-EW-0001, “Loss of Essential Cooling Water,” Revision 1, specified:

- A loss of component cooling water to the centrifugal charging pump supplemental cooler may cause respective centrifugal charging pump motor failure in as little as four minutes.

- A loss of component cooling water to the centrifugal charging pump lube oil cooler may cause pump failure in as little as eight minutes.

While these cautionary statements were true for a total loss of component cooling (including system water flow), operators would have substantially more time to place the positive displacement pump into service if the component cooling water system continued to pump water through the system coolers. Without essential cooling water, the component cooling water system would continue to heat up, but this would allow the operators additional time.

The analyst observed a simulator run that mimicked the loss of essential cooling water event. At the event initiation, component cooling water temperature was 78 degrees Fahrenheit. The temperature increased to 105 degrees Fahrenheit in six minutes and to 110 degrees Fahrenheit in a total of 20 minutes. The analyst concluded that the in-service charging pump would remain functional for at least 20 minutes. Charging pump room temperatures would not be expected to exceed 120 degrees Fahrenheit and the pump should remain functional at that temperature. If the in-service charging pump failed, operators would place the standby charging pump into service. The standby pump would likely take less time to fail because the pump room temperature was already elevated. Nonetheless, considering the procedure flow and the placement of the required steps, operators had more than enough time to evaluate the plant conditions and to place the positive displacement pump into service. This would likely occur before the first charging pump failed.

Based on the above, the analyst determined that for the loss of essential cooling water sequences, the identified performance deficiency would not result in a quantifiable increase to the core damage frequency.

Loss of Component Cooling Water: As discussed in the prior section of this evaluation, the total loss of component cooling water (including system flow) could result in the early failure of the charging pumps. However, the failures would occur sequentially. The first pump would fail and operators would then place the standby pump into service. The second pump would run for at least four minutes before failing.

The time-line for this set of sequences was similar to the first scenario above. In that scenario, both charging pumps failed approximately 12 minutes into the exercise.

To evaluate the loss of component cooling water sequences, the analyst set the basic event for the positive displacement pump operator action to a probability of 1.0. The change to the core damage frequency (Delta-CDF) was 1.1E-7/year.

Loss of Switchgear Cooling: This initiating event is the loss of all switchgear room cooling. This can result in cascading equipment failures as well as a loss of reactor coolant pump seal cooling. However, this does not include the loss of essential cooling water and/or the loss of component cooling water events, which are modeled separately. South Texas Project uses chillers to support switchgear room cooling and the failure of all chillers is the most likely initiator.

The loss of switchgear cooling would not result in immediate equipment failures. There is considerable time, on the order of hours, before equipment would be expected to start failing. Operators would have more than sufficient time to place the positive displacement pump into service. The analysis for the loss of essential cooling water bounded these sequences. Therefore, there was no quantitative increase to the core damage frequency for loss of switchgear cooling events.

External Events: The analyst reviewed the South Texas Project "Level 2 Probabilistic Safety Assessment and Individual Plant Examination," dated August 1992. This document also included the licensee's evaluation of external events.

As noted earlier, certain events were not risk significant. That was because:

1) the positive displacement pump was not needed for mitigation;

2) operators had sufficient time to place the positive displacement pump into service; or

3) the failure to promptly place the positive displacement pump into service was not risk significant when quantified using the NRC's SPAR model.

For external events the same philosophy held true. For example, seismic and fire initiators did not completely fail the seal cooling function, so operators were not required to start the positive displacement pump.

The licensee identified one initiator that could challenge reactor coolant pump seal cooling. This event included a tornado that failed offsite power and brought sufficient debris into the essential cooling water ponds to fail the essential cooling water pumps. The frequency for this event was 3.6E-6 per year. The resultant plant equipment failures would be the same as a station blackout. As noted previously, during a station blackout, it is expected that operators would have sufficient time to place the positive displacement pump into service.

Total Change to the Core Damage Frequency: The total change to the core damage frequency associated with this performance deficiency was approximately:

Delta-CDF = 1.1E-7/year

The dominant core damage sequences included loss of component cooling water events combined with the failure to start the positive displacement pump within six minutes. Additional time was available during most other initiating events, which helped to minimize the risk.

Large Early Release Frequency: To address the contribution to conditional large early release frequency, the analyst used NRC Inspection Manual Chapter 0609, Appendix H, "Containment Integrity Significance Determination Process," dated May 6, 2004. Since the performance deficiency did not contribute directly to a steam generator tube rupture or an intersystem loss of coolant accident, the condition was not risk significant to the large early release frequency.