UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
612 EAST LAMAR BLVD, SUITE 400
ARLINGTON, TEXAS 76011-4125

January 10, 2011

Matthew W. Sunseri, President and
Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, KS 66839

SUBJECT: WOLF CREEK – NRC COMPONENT DESIGN BASES INSPECTION NRC INSPECTION REPORT 05000482/2010007

Dear Mr. Sunseri:

On November 18, 2010, the U.S. Nuclear Regulatory Commission (NRC) completed the onsite portion of a component design bases team inspection at the Wolf Creek Generating Station. The enclosed report documents our inspection findings. The team discussed the preliminary findings on October 8, 2010, with Mr. M. Sunseri, President and Chief Executive Officer and other members of your staff. After additional in-office inspection, the team leader conducted a final telephonic exit on November 18, 2010, with Mr. T. Garrett, Vice President, Engineering and other members of your staff.

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

This report documents five NRC identified findings of very low safety significance (Green). The findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these findings as noncited violations, consistent with the NRC Enforcement Policy. If you contest the noncited violations or the significance of the noncited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 612 E. Lamar Blvd, Suite 400, Arlington, Texas, 76011-4125; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Wolf Creek facility. In addition, if you disagree with the crosscutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV, and the NRC Resident Inspector at Wolf Creek.

In accordance with 10 CFR 2.390 of the NRC's Rules of Practice, a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS).

ADAMS is accessible from the NRC Website 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

Docket: 50-482
License: NPF-42

Enclosure:
Inspection Report 05000482/2010007
w/Attachment 1: Supplemental Information
Attachment 2: Wolf Creek CCW Seismic Finding Significance Determination Process

cc: w/Enclosure
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ADAMS: ☐ No ☒ Yes	☒ SUNSI Review Complete	Reviewer Initials: WCS/TRF
	☒ Publicly Available	☒ Non-Sensitive
	☐ Non-publicly Available	☐ Sensitive

RI:DRS\EB1	RI\EB1	RI\EB2	OE\OB	C:DRS\EB1
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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000482
License:	NPF-42
Report Nos.:	05000482/2010007
Licensee:	Wolf Creek Nuclear Operating Corporation
Facility:	Wolf Creek Generating Station
Location:	1550 Oxen Lane NE Burlington, Kansas
Dates:	September 7–10, 2010, On site September 13-17, 2010, In office September 20-24, 2010, On site September 27 through October 1, 2010, In office October 4-8, 2010, On site November 18, 2010, In office
Team Leader:	W. Sifre, Senior Reactor Inspector, Engineering Branch 1
Inspectors:	J. Adams, Reactor Inspector, Engineering Branch 1 B. Correll, Reactor Inspector, Engineering Branch 2 D. Strickland, Operations Engineer, Operations Branch C. Baron, NRC Contractor, Beckman and Associates S. Kobylarz, NRC Contractor, Beckman and Associates
Approved By:	Thomas R. Farnholtz, Branch Chief Engineering Branch 1

SUMMARY OF FINDINGS

IR 05000482/2010007, On site September 7-10, September 13-17, and October 4–8, 2010; In office September 13–17, September 27 – October 1, and October 4-8, 2010, Wolf Creek Generating Station: baseline inspection, NRC Inspection Procedure 71111.21, "Component Design Bases Inspection."

The report covers an announced inspection by a team of four regional inspectors and two contractors. Five violations of significance were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process," and the crosscutting aspect was determined using Inspection Manual Chapter 0310, "Components within the Cross Cutting Areas." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC Identified Findings

Cornerstone: Mitigating Systems

- Green. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, which states, in part, that "measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions." Specifically, as of October 8, 2010, the licensee failed to incorporate design seismic requirements into the design calculations and actual system operation. This finding was entered into the licensee's corrective action program as Condition Report 00028237.

  The team determined that the failure to adequately analyze the isolation between the safety related and nonsafety-related portions of the component cooling water system was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the design basis analysis did not ensure that the affected train of component cooling water would perform its required functions after the failure of nonsafety-related component cooling water piping. The inspectors evaluated the issue using Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings." This finding affected the Mitigating Systems Cornerstone because seismic protection was degraded. The inspectors determined that this finding represented the degradation of equipment and functions specifically designed to mitigate a seismic event and that during an earthquake the deficiency would degrade one train of component cooling water, a system that supports a safety system or function. Therefore, this finding was potentially risk significant to seismic initiators and a Phase 3 analysis was required. A Region IV senior reactor analyst performed the Phase 3 significance determination. The change in core damage frequency was calculated to be 7.0×10^{-8} indicating that this finding was of very low safety significance (Green). The dominant risk sequence included a seismic initiating event, loss of offsite power, loss of reactor coolant pump seal cooling, and a failure of high pressure recirculation. This finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance (Section 1R21.2.1).

- <u>Green</u>. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action, which states, in part, that "measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected." Specifically, as of October 8, 2010, the licensee failed to assure that the identified emergency diesel generator room and the service water pump room tornado damper testing deficiency was effectively corrected. This finding was entered into the licensee's corrective action program as Condition Report 00028185.

  The inspectors determined that the failure to implement this corrective action was a performance deficiency. This finding was more than minor because, if left uncorrected, the performance deficiency would have the potential to lead to a more significant safety concern. Specifically, failure to implement this corrective action would have resulted in a failure to periodically test tornado dampers required to protect both the emergency diesel generator room and the essential service water pump room ventilation system. In accordance with Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," a significance determination screening was performed and determined this finding was of very low safety significance (Green) because it was not a design or qualification deficiency, did not represent a loss of system safety function, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The team determined that there was a crosscutting aspect in the area of human performance resources because the licensee failed to provide complete, accurate, and up-to-date work packages [H.2(c)](Section 1R21.2.2).

- <u>Green</u>. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, which states, in part, that "measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions." Specifically, as of October 8, 2010, the design calculations associated with the auxiliary feedwater system line break analysis was not consistent with the actual system operation. This finding was entered into the licensee's corrective action program as Condition Report 00006250.

  The team determined that the failure to adequately analyze a postulated failure of the piping from the condensate storage tank to the auxiliary feedwater pumps was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the design basis analysis did not ensure that the turbine-driven auxiliary feedwater pump would perform its required functions after the failure of nonsafety-related piping from the condensate storage tank. In accordance with NRC Inspection Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," a significance determination screening was performed and determined this finding was of very low safety significance (Green) because it did not represent a loss of system safety function and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The licensee's operability evaluation demonstrated that the auxiliary feedwater system was operable. This finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance (Section 1R21.2.3).

- Green. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, which states, in part, that "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 September 29, 2010, the licensee failed to ensure that the design bases inputs in the emergency diesel generator equipment specification were bounded by expected operational values. The licensee failed to evaluate the effects of the identified design basis maximum local meteorological conditions on the rating for the emergency diesel generators which could have affected the capability of safety-related equipment to respond to initiating events. This finding was entered into the licensee's corrective action program as Condition Report 00028695.

  The team determined that failure to properly incorporate the licensing design basis for extreme local meteorological temperature conditions as a design input in the emergency diesel generator equipment specification was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the design basis analysis did not ensure that the diesel generators could perform their design safety function at the maximum design temperature. In accordance with NRC Inspection Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," a significance determination screening was performed and determined this finding was of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in loss of operability or functionality. This finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance (Section 1R21.2.11).

- Green. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, which states in part, that "measures shall be established to assure that applicable regulatory requirements and the design bases are correctly translated into specifications, drawings, procedures, and instructions." Specifically, September 22, 2010, two out of two operating crews failed to satisfy the minimum time requirement for the transfer of suction of the residual heat removal pumps and the containment spray pumps to the containment recirculation sumps following a large break loss of coolant accident with the worst single active failure as described in Table 6.3-12 of the Updated

Safety Analysis Report. This finding was entered into the licensee's corrective action program as Condition Report 00028276.

The team determined that the failure to translate design requirements into operating procedures was a performance deficiency. This finding was more than minor because it was associated with the Mitigating Systems Cornerstone attribute of design control and affected the associated cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to translate design requirements into Procedure EMG ES-12, "Transfer to Cold Leg Recirculation." In accordance with NRC Inspection Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," a significance determination screening was performed and determined that this finding was of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in loss of operability or functionality. This finding had a crosscutting aspect in the area of human performance resources because the operating personnel were not trained to complete the transfer to cold leg recirculation within the minimum time to ensure the equipment was available to assure nuclear safety [H.2(b)](Section 1R21.4).

B. Licensee-Identified Violations

None.

REPORT DETAILS

1 REACTOR SAFETY

Inspection of component design bases verifies the initial design and subsequent modifications and provides monitoring of the capability of the selected components and operator actions to perform their design bases functions. As plants age, their design bases may be difficult to determine and important design features may be altered or disabled during modifications. The plant risk assessment model assumes the capability of safety systems and components to perform their intended safety function successfully. This inspectable area verifies aspects of the Initiating Events, Mitigating Systems and Barrier Integrity Cornerstones for which there are no indicators to measure performance.

1R21 Component Design Bases Inspection (71111.21)

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

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

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

The team performed a margin assessment and detailed review of the selected risk significant components to verify that the design bases have been correctly implemented and maintained. This design margin assessment considered original design issues, margin reductions because of modifications, and margin reductions identified as a result of material condition issues. Equipment reliability issues were also considered in the selection of components for detailed review. These included items such as failed performance test results; significant corrective actions; repeated maintenance; 10 CFR 50.65(a)1 status; operable, but degraded, conditions; NRC resident inspector input of problem equipment; system health reports; industry operating experience; and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense in-depth margins.

The inspection procedure requires a review of 20 to 30 total samples that include 10 to 20 risk-significant and low design margin components, 3 to 5 relatively high-risk operator actions, and 4 to 6 operating experience issues. The sample selection for this inspection was 15 components, 5 operator actions, and 5 operating experience items.

The selected inspection items supported risk significant functions as follows:

1. Electrical power to mitigation systems: The team selected several components in the offsite and onsite 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 also reviewed the licensee's response to Information Notice 2007-09, "Equipment Operability Under Degraded Voltage Conditions," and Generic Letter 2007-01, "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients." As such the team selected:

   - Engineered safety features transformer number 2 XNB02
   - Emergency diesel generator room heating ventilation and air conditioning system
   - 125 Vdc battery NK14
   - 125 Vdc bus NK04
   - Essential service water pump B supply breaker NB0215
   - Emergency diesel generator exhaust dampers

2. Initiating events minimization:

   - Safety injection pump B motor DPEM01B
   - Essential service water pump B motor PEF01B
   - Essential service water to auxiliary feedwater valves HV-30, HV-31, HV-32, and HV-33
   - Safety injection pump valves EJHV8804A and EJHV8804B
   - Essential service water pump B 1PBEF01B
   - Safety Injection Pump B, PEM01B
   - Essential service water to service water crosstie isolation valve EF40

3. Decay heat removal:

- Residual heat removal pump B motor DPEJ01B

- Component cooling water surge tanks and level instruments TEG01A and TEG01B

.2 Results of Detailed Reviews for Components

.1 Component Cooling Water Surge Tanks and Level Instruments TEG01A and TEG01B

a. Inspection Scope

The team reviewed the Updated Safety Analysis Report, system design criteria, current system health report, selected drawings, operating procedures, and past corrective action documents for the component cooling water surge tanks and level instruments TEG01A and TEG01B. The team performed interviews with design and system engineering personnel. The inspection team performed component walkdowns to ensure the capability of these components to perform their required function. Specifically, the team reviewed:

- Calculations, procedures, and test results related to tank capacity and instrument performance

- Calculations addressing the capability of the tanks to ensure continued system operability with the postulated failure of non-safety related component cooling water piping

- The capacity of the component to perform its required function with a postulated single failure

b. Findings

Introduction. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, in that, the licensee did not adequately analyze the isolation between the safety-related and nonsafety-related portions of the component cooling water system. Specifically, the inspectors determined that the current design calculation was not bounding because it did not ensure the continued operability of the affected component cooling water train in the event of a failure in the nonsafety-related portion of the system. As a result, the affected component cooling water pumps could be subject to reduced suction pressure, cavitation, and potential air ingestion.

Description. The component cooling water system design included a nonsafety-related section that supplied cooling for equipment in the radiation waste building. This portion of the system was designed to be automatically isolated from the safety-related portion of the system by automatic air-operated valves (two in series on the supply line and two in series on the return piping). These isolation valves were designed to close on either a high component cooling water system flow or a low component cooling water surge tank level, indicating a potential break in the system. The intent of the design was to preserve sufficient inventory in the surge tank to keep the system full and operating after the postulated failure of this nonsafety related piping.

The team reviewed Calculation M-EG-12, “Component Cooling Water System Pipe Break.” This calculation showed that there would be sufficient volume available in the surge tank to accommodate the most limiting pipe break, assuming the isolation valves automatically closed within 10 seconds. However, the team determined that the calculation did not address the capacity of the 4-inch surge tank pipe to transfer water from the surge tank to the pump suction at the same rate as the postulated pipe break flow (assumed to be 14,000 gpm in the calculation). As a result, the team questioned if pump suction piping would have a negative pressure during the transient, resulting in pump cavitation, or if air from the postulated break would enter the component cooling water system. In response to these questions, the licensee initiated Condition Report 00028237 on September 22, 2010. The licensee also declared the affected train on the component cooling water system inoperable. Compensatory actions were then taken (administratively closing the isolation valves) and the system was returned to operable status within one day. During the inspection period, the licensee continued to evaluate this condition to determine if the affected train of component cooling water was inoperable prior to administratively closing the system isolation valves and to determine what corrective actions will be required.

Analysis. The team determined that the failure to adequately analyze the isolation between the safety-related and nonsafety-related portions of the component cooling water system was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the design basis analysis did not ensure that the affected train of component cooling water would perform its required functions after the failure of nonsafety related component cooling water piping. The inspectors evaluated the issue using Manual Chapter 0609.04, “Phase 1 - Initial Screening and Characterization of Findings.” This finding affected the Mitigating Systems Cornerstone because seismic protection was degraded. The inspectors determined that this finding represented the degradation of equipment and functions specifically designed to mitigate a seismic event and that during an earthquake the deficiency would degrade one train of component cooling water, a system that supports a safety system or function. Therefore, this finding was potentially risk significant to seismic initiators and a Phase 3 analysis was required. A Region IV senior reactor analyst performed the Phase 3 significance determination. The change in core damage frequency was calculated to be 7.0×10^{-8} indicating that this finding was of very low safety significance (Green). The dominant risk sequence included a seismic initiating event, loss of offsite power, loss of reactor coolant pump seal cooling, and a failure of high pressure recirculation. The details of the Phase 3 analysis are included as Attachment 2 to this inspection report. This finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance.

Enforcement. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, which states, in part, that “measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions.” Contrary to the above, the licensee failed to ensure that measures were established to ensure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Specifically, as of

October 8, 2010, the licensee failed to incorporate design seismic requirements into the design calculations and actual system operation. This finding was entered into the licensee's corrective action program as Condition Report 00028237. Because this violation was of very low safety significance (Green) and has been entered into the licensee's corrective action program, this violation is being treated as a noncited violation, consistent with the NRC Enforcement Policy: NCV 05000482/2010007-01, "Inadequate Design of Component Cooling Water Safety/Nonsafety Isolation."

.2 Emergency Diesel Generator Heating, Ventilation, and Air Conditioning Exhaust Tornado Dampers

a. Inspection Scope

The team reviewed the Updated Safety Analysis Report, design bases documents, calculations, and recent corrective and preventive maintenance of the emergency diesel generator heating, ventilation, and air conditioning exhaust tornado dampers. This review included interviews with the cognizant design and system engineering personnel and component walkdowns to verify the configuration and capability of these components to perform their required function. Specifically, the team reviewed:

- Past condition reports related to damper performance
- Vendor documents, procedures, and test results related to damper design, maintenance, and performance
- The capacity of the dampers to perform their required function in the event of a postulated tornado event

b. Findings

Introduction. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action, in that the licensee did not adequately test the spring-loaded tornado dampers located in the emergency diesel generator room and essential service water pump room ventilation systems. The licensee identified that the spring-loaded tornado dampers were not being periodically tested as required by their Updated Safety Analysis Report and failed to implement appropriate corrective actions to assure that the required periodic testing would occur.

Description. The Wolf Creek emergency diesel generator room and essential service water room ventilation system design included four spring-loaded dampers that were required to automatically close in the event of high differential pressures associated with a design basis tornado. The safety function of these dampers is to protect the heating ventilation and air conditioning system ductwork and components from postulated high-pressure differentials. In 2008, Wolf Creek personnel identified that these dampers had been incorrectly classified as passive components and were not being periodically tested; Condition Report 2008-003276 was initiated to revise Procedure MPE VD-001, "Ventilation Damper Maintenance." Later in 2008, the procedure was updated and the corrective action was closed. However, no action was taken to ensure that the required testing would be performed as part of the scheduled preventative maintenance activities. At the time of the inspection, there was no indication that this testing would have been performed. No damper tests had actually been missed due to this error, but the next damper preventative maintenance activities were scheduled to begin in December 2010 and there was no documentation requiring testing of the spring-loaded dampers to be included in the preventative maintenance activities. Also, at the time of the inspection the required test methods had not been developed and the required tools to accomplish the testing had not been obtained. As a result of these concerns, the licensee initiated Condition Report 00028185 on September 20, 2010, which stated that the procedure change was never communicated to the planners and that there was no corrective action initiated to write a work order for the testing.

Analysis. The inspectors determined that the failure to implement corrective action was a performance deficiency. This finding was more than minor because, if left uncorrected, the performance deficiency would have the potential to lead to a more significant safety concern. Specifically, the failure to implement this corrective action would have resulted in a failure to periodically test tornado dampers required to protect both the emergency diesel generator room and the essential service water pump room ventilation systems. In accordance with Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," a significance determination screening was performed and determined this finding was of very low safety significance (Green) because it was not a design or qualification deficiency, did not represent a loss of system safety function, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The team determined that there was a crosscutting aspect in the area of human performance resources because the licensee failed to provide complete, accurate, and up-to-date work packages [H.2(c)].

Enforcement. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action, which states, in part, that "measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected." Contrary to the above, the licensee failed to assure that conditions adverse to quality were promptly identified and corrected. Specifically, as of October 8, 1010, the licensee failed to assure that the identified emergency diesel generator room and the service water pump room tornado damper testing deficiency was effectively corrected. This finding was entered into the licensee's corrective action program as Condition Report 00028185. Because this violation was of very low safety significance (Green) and has been entered into the licensee's corrective action program, this violation is being treated as a noncited violation, consistent with the NRC Enforcement Policy: NCV 05000482/2010007-02, "Failure to Perform Tornado Damper Testing."

.3 Essential Service Water to Auxiliary Feedwater Valves HV-30, HV-31, HV-32, HV-33

a. Inspection Scope

The team reviewed the Updated Safety Analysis Report, operating procedures, current system health report, selected drawings, operating procedures, and corrective action documents associated with the essential service water to auxiliary feedwater valves. This review included interviews with the cognizant design and system engineering personnel and component walkdowns to verify the configuration and capability of these components to perform their required function. Specifically, the team reviewed:

- Past condition reports related to the performance of these valves

- Calculations, procedures, and test results related to valve performance including inservice testing

- Valve thrust calculations and stroke test results to verify the capability of the valves to perform their function under the most limiting conditions

- Calculations to verify the capability of the valves to transfer the auxiliary feedwater pump suction supply from the condensate storage tank to the essential service water system when the tank is empty or unavailable

b. Findings

Introduction. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, in that, the licensee did not adequately analyze a postulated failure of the piping from the condensate storage tank to the auxiliary feedwater pumps. Specifically, the inspectors determined that the performance deficiency previously addressed by unresolved item URI 05000482/2007006-09, “Potential for Damage to Turbine Driven Auxiliary Feedwater Pump and Essential Service Water System During Condensate Storage Tank Unavailability,” was a noncited violation.

Description. The previous component design basis inspection team reviewed the design calculations concerning the automatic transfer of the suction source for the auxiliary feedwater pumps from the condensate storage tank to the essential service water system. The team determined that the turbine-driven auxiliary feedwater pump and emergency diesel generators started immediately in response to a loss of offsite power, on an undervoltage signal. Vital loads are then automatically sequenced after the emergency diesel generator is ready to accept load. The team questioned a condition in which the condensate storage tank would not be available, such as a seismic event, and determined that the essential service water system would automatically realign to the auxiliary feedwater pump suction by a low auxiliary feedwater suction pressure signal. Considering the 12-second maximum start time for the diesel and subsequent load sequence times, the team found that the first essential service water pump might not reach full flow until about 32 seconds after a loss of offsite power, and the second essential service water pump might not reach full flow until about 37 seconds. Therefore, if the condensate storage tank was not available, and essential service water system was the auxiliary feedwater pump source, the water initially available to the turbine-driven auxiliary feedwater pump suction would be limited to the useable volume in the suction piping. The previous component design basis inspection team initiated an unresolved item associated with the issue because they did not have sufficient information to make a determination.

In response to the previous inspection team’s concerns, the licensee initiated Condition Report 00006250 on July 18, 2007. The condition report addressed several nonconservative inputs and assumptions in Calculation AN-97-1157, which had concluded that the auxiliary feedwater pump suction piping would have adequate volume for the postulated event. The licensee issued Operability Evaluation AL-07-003 on July 18, 2007. The evaluation determined that the turbine-driven auxiliary feedwater pump might be damaged during start-up if the condensate storage tank was not available, and concluded that the auxiliary feedwater system remained operable. The licensee implemented compensatory actions to protect the turbine-driven auxiliary feedwater pump when the condensate storage tank volume is reduced. The licensee also initiated corrective actions to eliminate this concern; that modification had not been completed at the time of the inspection. Based on the information developed by the licensee since the previous component design basis inspection, the inspectors identified a performance deficiency related to the automatic transfer of the suction source for the auxiliary feedwater pumps. Condition Report 00006250 and Operability Evaluation AL-07-003, initiated on July 18, 2007, addressed the performance deficiency and identified the required compensatory actions and corrective actions.

Analysis. The team determined that the failure to adequately analyze a postulated failure of the piping from the condensate storage tank to the auxiliary feedwater pumps was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the design basis analysis did not ensure that the turbine-driven auxiliary feedwater pump would perform its required functions after the failure of nonsafety-related piping from the condensate storage tank. In accordance with NRC Inspection Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," a significance determination screening was performed and determined this finding was of very low safety significance (Green) because it did not represent a loss of system safety function and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The licensee's operability evaluation demonstrated that the auxiliary feedwater system was operable. This finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance.

Enforcement. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, which states, in part, that “measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions.” Contrary to the above, the licensee failed to establish measures to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Specifically, as of October 8, 2010, the design calculations associated with the line break analysis was not consistent with the actual system operation. This finding was entered into licensee’s corrective action program as Condition Report 00006250. Because this violation was of very low safety significance (Green) and has been entered into the licensee’s corrective action program, this violation is being treated as a noncited violation, consistent with the NRC Enforcement Policy: NCV 05000482/2010007-03, “Inadequate Auxiliary Feedwater Pump Suction Line Break Analysis and Design.”

.4 Safety Injection Pump Suction Valves EJHV8804A and EFHV8804B

a. Inspection Scope

The team reviewed the Updated Safety Analysis Report, operating procedures, current system health report, selected drawings, operating procedures, and corrective action documents associated with the safety injection suction valves. These valves were designed to be opened by the operators under post-accident conditions to align the suction of the safety injection pumps to the residual heat removal pump discharge. The inspection team performed interviews with design and system engineering personnel and performed component walkdowns to ensure the capability of the valves to perform their required function. The team also verified that the emergency core cooling system valve position interlocks were appropriately tested to verify the system's capability to perform its function with a postulated single failure. Specifically, the team reviewed:

- Past condition reports associated with the valves
- Calculations, procedures, and test results related to valve performance including inservice testing
- Valve thrust calculations and stroke test results to verify the capability of the valve to perform its function under the most limiting conditions
- Calculations and procedures to verify the capability of the valves to align the safety injection pump suction supply to the residual heat removal pump discharge

b. Findings

No findings were identified.

.5 Essential Water System Train B Pump 1PEF01B

a. Inspection Scope

The team reviewed the design bases documents, Updated Safety Analysis Report, technical specification, calculations, corrective and preventative maintenance, and testing of the essential service water pump. Specifically, the team reviewed:

- Inservice testing data as related to pump degradation and the pump's ability to satisfy the design basis requirements. This review included pump vibration, flow rate and head to verify compliance with technical specifications and code requirements.
- Design basis documentation to verify that the pump performance satisfied worst case design accident conditions, including vendor data
- Motive power source, including undervoltage condition, and emergency diesel generator sequencing
- System health reports for the essential service water system, including the pump

b. Findings

No findings were identified.

.6 Safety Injection Train B Pump PEM01B

a. Inspection Scope

The team reviewed the design bases documents, Updated Safety Analysis Report, technical specification, calculations, corrective and preventative maintenance, and testing of the safety injection pump. Specifically, the team reviewed:

- Design basis documentation and calculations to verify that the pump will satisfy its design safety function and that the pump will not be adversely affected by potential voids in the suction line, either through gas binding or pressure surges

- Inservice testing procedures and test data for flow, vibration, and differential pressure to verify that the pump performance has not degraded, including vendor data

- Motive power source, including undervoltage condition, and emergency diesel generator sequencing

b. Findings

No findings were identified.

.7 Essential Service Water Train B to Service Water System Isolation Valve EFHV0040

a. Inspection Scope

The team reviewed the design bases documents, Updated Safety Analysis Report, technical specification, calculations, corrective and preventative maintenance, and testing of the essential service water to service water isolation valve. Specifically, the team reviewed:

- Inservice test procedures and test data to verify monitoring of valve degradation, including leakage and stroke time. The team also verified that the frequency of testing satisfied the minimum technical specifications and code requirements.

- Design basis, including safety function to close upon a safety injection actuation or loss of offsite power. The team also verified that the valve would isolate the essential service water system and ensure heated return flow is directed to the ultimate heat sink.

- Motive power sources including undervoltage conditions, and emergency diesel generator sequencing

- Vendor data to ensure valve is capable of meeting design basis safety function

b. Findings

No findings were identified.

.8 Safety Injection Pump B Motor DPEM01B

a. Inspection Scope

The team reviewed the Updated Safety Analysis Report, system description, calculations, condition reports, and testing and calibration of the motor overcurrent relays. The team performed a walkdown to assess the configuration, nameplate information, and material condition of the motor and electrical cables and raceways. The team verified the motor capability to provide rated power at design basis pump flow conditions. Specifically, the team reviewed:

- Vendor and licensee drawings
- Calculations that established motor protection and breaker coordination, pump brake horsepower requirements, motor capability at degraded voltage conditions
- Short circuit analysis for the motor power feeder cable to verify that design bases and design assumptions were appropriately translated into design calculations

b. Findings

No findings were identified.

.9 Residual Heat Removal Pump B Motor DPEJ01B

a. Inspection Scope

The team reviewed the Updated Safety Analysis Report, system description, calculations, and condition reports. The team verified the motor capability to provide rated power at design basis pump flow conditions. Specifically, the team reviewed:

- Vendor drawings, a sample of condition reports and motor nameplate information
- Calculations, condition reports, testing and calibration procedures, and data for the motor overcurrent relays
- Calculations that establish motor protection and circuit breaker coordination
- Pump brake horsepower requirements
- Motor capability at degraded voltage conditions
- Short circuit analysis for the motor power feeder cable to verify that design bases and design assumptions were appropriately translated into design calculations

b. Findings

No findings were identified.

.10 Engineered Safety Features Transformer XNB02

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. This review included the licensee's design basis documentation as well as various calculations, procedures, and test results. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its required design basis function. Specifically, the team reviewed:

- Periodic maintenance, surveillance testing, and Doble test results

- Oil quality, dissolved gas trending, and transformer oil samples

b. Findings

No findings were identified.

.11 Emergency Diesel Generator NE002 Diesel Generator Room Heating Ventilation and Air Conditioning System

a. Inspection Scope

The team reviewed the Updated Safety Analysis Report, system description, calculations, and corrective maintenance of the diesel generator room supply fan. The team performed a walkdown of the emergency diesel generator heating ventilation and air conditioning system. Specifically, the team reviewed:

- Emergency diesel generator building heating ventilation and air conditioning calculations

- Supply fan vendor performance curves and requirements for worse case design basis operating conditions

- Supply fan motor protection and coordination calculation

- Emergency diesel generator technical specification and vendor rating datasheets

- A sample of condition reports, and supply fan motor and emergency diesel generator nameplate information

b. Findings

Introduction. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, in that the licensee did not adequately analyze the effects of the identified design basis maximum local meteorological conditions on the rating for the emergency diesel generators. Specifically, this condition could have affected the capability of safety-related equipment to respond to initiating events.

Description. The team identified that the licensee failed to evaluate and specify the worst case design basis local meteorological conditions as stated in the Updated Safety Analysis Report when determining the maximum air inlet temperature for the required emergency diesel generator load capability. The team found that the Updated Safety Analysis Report stated in Section 2.3.2.3, "Local Meteorological Conditions for Design and Operating Bases," that the site extreme outside air temperature was assumed to vary from -60°F to +120°F. The team reviewed the licensee's diesel generator building heating ventilation and air conditioning Calculation GM-320 and found that the licensee calculated 122°F as the maximum diesel building ambient temperature, which was also the maximum diesel engine inlet air temperature because the diesel engine derives inlet air for combustion from inside the engine room. However, the maximum diesel building ambient air temperature calculated in GM-320 was based on a 97°F outside air temperature. On review of Specification Number 10466-M-018, "Technical Specification for Standby Diesel Generators," for the standardized nuclear unit power plant system, the team found that 122°F was also the temperature that was specified for the maximum diesel air intake temperature. Based on discussions with the licensee's engineers, the team determined that the impact of 120°F outside air temperature had not been evaluated when determining the maximum engine inlet air temperature. Since the licensee had not analyzed the effect of the site extreme meteorological maximum temperature condition on the engine's air inlet temperature, the inspectors questioned the capability and operability of the diesel at the maximum local design basis environmental conditions. During the inspection, the licensee determined the diesel generator was operable based on the outside air temperature which was then expected to be less than 80°F. The licensee also evaluated past operability based on meteorological data for the highest recorded local area temperature of 109°F at John Redmond Lake, 106.25°F was the highest recorded temperature recorded at the Wolf Creek site meteorological tower, and recently recorded data for diesel generator building temperature indicated that the average temperature difference between the room and outside ambient temperature was approximately 10°F, which would result in approximately 119°F for the engine air inlet temperature, which was less than the specified diesel maximum air inlet temperature of 122°F. Based on the data provided, the team found the licensee's determination of past operability acceptable.

The capability of the diesel engine for an air inlet temperature based on the 120°F extreme local meteorological design basis temperature was under review by the licensee and the diesel generator manufacturer during the inspection. The team also reviewed Technical Requirements Manual 3.7.21 and Shift Log STS CR-001 which addressed the requirement for diesel generator room temperature to be less than or equal to 119°F, and found that the log provided no direction or guidance to operators for compensatory actions to be taken to restore area temperature to within allowable limits should temperature be greater than 119°F. The licensee entered Condition Report 00028695 into the corrective action system to address the deficiency in the shift log on the lack of direction for compensatory actions when diesel building temperature is greater than 119°F.

Analysis. The team determined that failure to properly incorporate the licensing design basis for extreme local meteorological temperature conditions as a design input in the emergency diesel generator equipment specification was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the design basis analysis did not ensure that the diesel generators could perform their design safety function at the maximum design temperature. In accordance with NRC Inspection Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," a significance determination screening was performed and determined this finding was of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in loss of operability or functionality. This finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance.

Enforcement. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, which states, in part, that “measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures and instructions.” Contrary to the above, the licensee failed to establish measures to assure that applicable design bases were correctly translated into specifications, drawings, procedures and instructions. Specifically, prior to September 29, 2010, the licensee failed to ensure that the design bases inputs in the emergency diesel generator equipment specification were bounded by expected operational values. The licensee failed to evaluate the effects of the identified design basis maximum local meteorological conditions on the rating for the emergency diesel generators which could have affected the capability of safetyrelated equipment to respond to initiating events. This finding was entered into the licensee’s corrective action program as ConditionReport 00028695. Because this finding was determined to be of very low safety significance (Green) and was entered into the licensee’s corrective action program, this violation is being treated as a noncited violation consistent with the NRC Enforcement Policy: NCV 05000482/2010007-04, “Emergency Diesel Generator Specified Rating did not Address Engine Operation at Design Basis Extreme Meteorological Temperature Conditions.”

.12 125 Vdc Class 1E Battery NK14

a. Inspection Scope

The team reviewed the Updated Safety Analysis Report, sizing calculations, short circuit calculations, design specifications, installation drawings, one-line diagrams, modifications made to the battery and battery rack, battery vendor manual, maintenance activities, and held discussions with battery system engineering personnel. The team performed visual inspection of the battery and the environs to assess material condition and to verify the battery and battery rack meet installation design requirements. The team also interviewed battery system engineering personnel to assess the design, installation, testing configuration, and maintenance of the battery. Specifically, the team reviewed:

- Discharge testing methodologies to verify the batteries are being tested to ensure that design requirements are being met. The licensee performs a modified performance test, which encompasses the most limiting currents of the station blackout profile and the loss of coolant accident profile.

- Pilot cell selection criteria and historical data to ensure pilot cells were being selected and maintained in accordance with industry and NRC guidance

- Corrective actions associated with previous violations associated with intercell connection resistance technical specification value (Condition Report 00702492) and battery surface cleanliness (Condition Report 200702580) to ensure corrective actions have been implemented

b. Findings

No findings were identified.

.13 125 Vdc Distribution Bus NK04

a. Inspection Scope

The team reviewed the Updated Safety Analysis Report, voltage drop calculations, short circuit calculations, and coordination studies. The team also reviewed one-line diagrams, maintenance documents, Quality Assurance audit reports, and vendor manuals. Specifically, the team reviewed:

- Vendor manuals to ensure adequate sizing of bus components were utilized

- Maintenance activities to ensure the components are being maintained in accordance with vendor recommendations

- Calculations for short circuit analysis to ensure proper bus and cable sizing met design requirements

- Coordination studies to ensure proper fuse and circuit breaker coordination were satisfied, and properly sized to protect cables from short circuit faults

- Voltage drop calculations to verify sufficient voltage available at individual component control devices, and that the most limiting battery voltage was used to determine the minimum voltage available at the device

b. Findings

No findings were identified.

.14 Essential Service Water Pump Motor DPEF02B

a. Inspection Scope

The team reviewed the Updated Safety Analysis Report, vendor manual for motor ratings, maintenance activities, design ratings, operating experience information, motor performance data, one-line diagrams, selected condition reports, and preventive maintenance activities to ensure the motors are designed and being maintained to meet design conditions and assumptions. Specifically, the team reviewed:

- Speed-torque curves to verify the motor is capable of providing the required pump torque during degraded voltage conditions

- Voltage drop analysis to ensure sufficient voltage is available at the motor terminals to provide the required motor torque during design basis events

- Cable and circuit breaker sizing and maintenance to ensure design requirements are being met

- Visual inspection of the motor to assess the physical condition and environmental conditions to ensure the motor is being maintained capable to perform its design function

- Circuit breaker relay settings appropriately set to ensure motor protection while remaining capable of meeting design requirements

- Circuit breaker relay setting surveillance activities to ensure the relays are being properly maintained to perform their safety function

b. Findings

No findings were identified.

.15 Essential Service Water Pump Motor DPEF02B

a. Inspection Scope

The team reviewed the Updated Safety Analysis Report, system description, calculations, condition reports, and testing and calibration of the motor overcurrent relays. The team verified the motor capability to provide rated power at design basis pump flow conditions. The team also conducted a walkdown of the motor to assess the configuration, nameplate information and observable material condition of the motor and the electrical cables and raceways. Specifically, the team reviewed:

- The vendor drawings, a sample of condition reports

- Calculations that establish motor protection and breaker coordination

- Pump brake horsepower requirements

- Motor capability at degraded voltage conditions and short circuit analysis for the motor power feeder cable to verify that design bases and design assumptions were appropriately translated into design calculations

b. Findings

No findings were identified.

.3 Results of Reviews for Operating Experience:

.1 NRC Information Notice 2006-29, "Potential Common Cause Failure of Motor-Operated Valves as a Result of Stem Nut Wear"

a. Inspection Scope

The team reviewed the licensee's evaluation and disposition of NRC Information Notice 2006-29, "Potential Common Cause Failure of Motor-Operated Valves as a Result of Stem Nut Wear," which documented multiple instances where excessive stem nut wear resulted in motor operated valves becoming inoperable. The licensee reviewed the information notice and performed a comprehensive review of all of their motor operated valves to determine which could be susceptible to this phenomenon. The licensee concluded that they were less susceptible to this wear because of the use of a very pure lubricant. The inspectors determined by review that the licensee's inspections of their stem nuts, to date, support this conclusion. The licensee has also instituted additional inspection requirements, including zero-transition time measurements and direct physical measurements of stem nut thread wear, to further minimize their susceptibility to this phenomenon.

b. Findings

No findings were identified.

.2 NRC Generic Letter 2007-01, "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients"

a. Inspection Scope

In Generic Letter 2007-01, "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients," the NRC requested the status of all cable failures for those cables that were inaccessible or underground as well as a description of inspection, testing, and monitoring programs for these cables. The team reviewed the licensee's response to the generic letter, which reported one cable failure in 2004 for startup transformer XMR01. The team reviewed drawings, cable design and testing specifications, work instructions for sump pumps, and megger test data, and had discussions with design engineering personnel responsible for submerged cables.

The team observed the opening of essential service water manholes MHE3B and MHE4B to check the level of water intrusion since the previous manhole inspection. Although water was observed inside the manholes, the water level was below all cable trays inside the manholes. The inspection and dewatering frequency performed by the licensee appears to be adequate for managing the water intrusion for safety related manholes, such that the cable trays do not become submerged.

The team also reviewed the licensee's actions taken to address nonsafety-related manhole water intrusion. The team had discussions with licensee and contractor personnel regarding performing hydrology studies to address these nonsafety-related manholes.

b. Findings

No findings were identified.

.3 NRC Information Notice 2007-09, "Equipment Operability Under Degraded Voltage Conditions"

a. Inspection Scope

The team reviewed the licensee's evaluation of NRC Information Notice 2007-09, "Equipment Operability Under Degraded Voltage Conditions," to verify that the review adequately addressed the industry operating experience. The team verified that the licensee's review documented on Condition Report 2007-001617 adequately addressed the issues in the information notice. The team verified that the licensee assured that emergency diesel generators service water pump motors had adequate voltage at the starter circuit to ensure operability under degraded voltage conditions, and that the specified surveillance requirement minimum required voltage was above the calculated minimum voltage required for component operability.

b. Findings

No findings were identified.

.4 NRC Information Notice 1996-06, "Design and Testing Deficiencies of Tornado Dampers at Nuclear Power Plants"

a. Inspection Scope

The team reviewed NRC Information Notice 1996-06, "Design and Testing Deficiencies of Tornado Dampers at Nuclear Power Plants," which addressed the potential for inoperability of tornado dampers because of either inadequacies in damper testing or deficiencies in damper design. In response to Information Notice 1996-06, the licensee evaluated the design and testing of tornado damper. The team reviewed this evaluation to verify that the plant was adequately protected against postulated tornado events.

b. Findings

The team's finding associated with tornado damper testing is addressed in Section 1R21.2.2 of this report.

.5 NRC Regulatory Issue Summary 2006-23, "Post-tornado Operability of Ventilating and Air-Conditioning Systems Housed in Emergency Diesel Generator Rooms"

a. Inspection Scope

The team reviewed NRC Regulatory Issue Summary 2006-23, "Post-tornado Operability of Ventilating and Air-Conditioning Systems Housed in Emergency Diesel Generator Rooms," which addressed loading effects caused by natural phenomena on ventilating and air conditioning systems housed inside a structure partially exposed to the outside environment. In response to Regulatory Issue Summary 2006-23, the licensee evaluated the capability of the emergency diesel generator rooms to withstand tornado pressure effects. The team reviewed this evaluation to verify that the plant was adequately protected against postulated tornado events.

b. Findings

The team's finding associated with tornado damper testing is addressed in Section 1R21.2.2 of this report.

4. Results of Reviews for Operator Actions

a. Inspection Scope

The team reviewed five risk significance operator actions as follows:

- Terminate Auxiliary Feedwater Flow to a Faulted Steam Generator following a Steam Line Break inside Containment: The team observed licensed operator crews perform a simulator scenario consisting of a large steam line break inside containment in accordance with Procedure EMG E-0, "Reactor Trip or Safety Injection," including foldout page Item 3, "Faulted S/G Isolation Criteria."

- Post Loss of Coolant Accident Refuel Water Storage Tank Switchover to Cold Leg Recirculation: The team observed licensed operator crews perform a simulator scenario consisting of a large break LOCA with a failed open refueling water storage tank suction valve to Train B residual heat removal pump in accordance with Procedure EMG ES-12, "Transfer to Cold Leg Recirculation."

- Shed Nonessential DC Loads during Station Blackout: The team observed nonlicensed operators perform in-plant job performance measures to shed large nonessential loads in accordance with Procedure EMG C-0, Loss of All AC Power," Attachment C, "DC Load Shedding."

- Provide Alternate AC Power Source to 120 Vac Instrument Bus: The team observed nonlicensed operators perform in-plant job performance measures to locally restore power to a 120 Vac instrument bus in accordance with Procedure OFN NN-021, "Loss of Vital 120 VAC Instrument Bus," Attachment A, "Loss of Vital Instrument Bus NN01."

- Manually Isolate a Ruptured Steam Generator: The team observed nonlicensed operators perform Procedure EMG E-3, Steam Generator Tube Rupture, Step 3.c and Step 3.d to locally isolate flow from a ruptured steam generator.

b. Findings

Introduction. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, in that the licensee did not adequately translate design requirements to operating procedures. Specifically, the inspectors determined that two of two crews in the simulator were unable to transfer suction of the residual heat removal pumps and the containment spray pumps to the containment sumps within the allotted time specified in the Updated Safety Analysis Report.

Description. The team identified that on two occasions, with two different operating crews, the licensee was unable to satisfy the time requirements specified in Table 6.3-12 of the Updated Safety Analysis Report with a single active failure as described in the Updated Safety Analysis Report. The single active failure was the inability to close, either manually or automatically, the valve from the refueling water storage tank to the suction of the residual heat removal pump. The time required actions which the operating crews were unable to satisfy were:

- Transfer residual heat removal pump suction to cold leg recirculation within eight minutes and nine seconds of receipt of Alarm ALR 00-047D, "Refueling Water Storage Tank Level LoLo 1 Auto Transfer (36 percent)"

- Transfer containment spray pump suction to cold leg recirculation within two minutes and ten seconds of receipt of Alarm ALR 00-047C, “Refueling Water Storage Tank Level LoLo 2 Auto Transfer (11 percent)”

The actions described above were all associated with emergency core cooling system train B. The licensee had an analysis which determined that the core cooling requirements would be satisfied with a single train. As a result, the plant was not in an unanalyzed condition.

Analysis. The team determined that the failure to translate design requirements into operating procedures was a performance deficiency. This finding was more than minor because it was associated with the Mitigating Systems Cornerstone attribute of design control and affected the associated cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to translate design requirements into Procedure EMG ES-12, “Transfer to Cold Leg Recirculation.” In accordance with NRC Inspection Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," a significance determination screening was performed and determined this finding was of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in loss of operability or functionality. This finding had a crosscutting aspect in the area of human performance resources because the operating personnel were not trained to complete the transfer to cold leg recirculation within the minimum time to ensure the equipment was available to assure nuclear safety [H.2(b)].

Enforcement. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, which states in part, that “measures shall be established to assure that applicable regulatory requirements and the design bases are correctly translated into specifications, drawings, procedures, and instructions.” Contrary to the above the licensee failed to establish measures to assure that applicable design bases were correctly translated into specifications, drawings, procedures, and instructions. Specifically, on September 26, 2010, two out of two operating crews failed to satisfy the minimum time requirement for the transfer of suction of the residual heat removal pumps and the containment spray pumps to the containment recirculation sumps following a large break loss of coolant accident with the worst single active failure as described in Table 6.3-12 of the Updated Safety Analysis Report. This finding was entered into the licensees corrective action program as Condition Report 00028276. Because this finding was determined to be of very low safety significance (Green) and was entered into the licensees corrective action program, this violation is being treated as a noncited violation consistent with the NRC Enforcement Policy:
NCV 05000482/2010007-05, “Failure to Translate Design Requirements to Operating Procedures for the Transfer of Residual Heat Removal and Containment Spray Suction to the Containment Recirculation Sumps.”

4OA6 Meetings, Including Exit

On October 8, 2010, the team leader presented the preliminary inspection results to Mr. M. Sunseri, President and Chief Executive Officer, and other members of the licensee’s staff.

On November 18, 2010, the team leader conducted a telephonic final exit meeting with Mr. T. Garrett, Vice President, Engineering and other members of the licensee's staff. The licensee acknowledged the findings during each meeting. While some proprietary information was reviewed during this inspection, no proprietary information was included in this report.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

M. Sunseri, President and Chief Executive officer
T. Berland. Engineer, Design Engineering
J. Cherry, Human Resources
B. Dale, Maintenance, Manager
D. Dandreo, Engineer, Design Engineering
D. Dixon, Design Engineering
D. Edgecomb, Licensing
D. Erbe, Manager, Security
R. Flannigan, Manager, Regulatory Affairs
R. Foust, Design Engineering
T. Garrett, Vice President Engineering
S. Hedges, Site Vice President
S. Henry, Acting Plant Manager
R. Hobby, Licensing
D. Hooper, Supervisor, Licensing
T. Jamar, Design Engineering
T. Jensen, Manager, Chemistry
W. Kennamore, Nuclear Manager
B. Ketchum, Probablistic Safety Assesment
S. Koenig, Manager, Corrective Actions
J. Luedke, System Engineering
B. Masters, Supervisor, Design Engineering
D. Meredith, Design Engineering
B. Muilenburg, Licensing
W. Norton, Manager, IPS and Scheduling
G. Pendergrass, Director, Plant Engineering
R. Ptacek, Licensing
E. Ray, Manager, Quality
L. Ratzlaff, Manager, Support
B. Smith, Project Engineering
T. Smither, Records
A. Stull, Vice President and Chief Administrative Officer
J. Suter, Acting Manager, Design Engineering
B. Vickery, Manager, Financial Services
B. Williams, Design Engineering
C. Williams, System Engineering
M. Westman. Manager, Training
J. Yunk, Manager, Human Resources

NRC Personnel

C. Long, Senior Resident Inspector
C. Peabody, Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000482/2010007-01	NCV	Inadequate Design of Component Cooling Water Safety/Nonsafety Isolation (Section 1R21.2.1)
05000482/2010007-02	NCV	Failure to Perform Tornado Damper Testing (Section 1R21.2.2)
05000482/2010007-03	NCV	Inadequate Auxiliary Feedwater Pump Suction Line Break Analysis and Design (Section 1R21.2.3)
05000482/2010007-04	NCV	Emergency Diesel Generator Specified Rating did not Address Engine Operation at Design Basis Extreme Meteorological Temperature Conditions (Section 1R21.2.11)
05000482/2010007-05	NCV	Failure to Translate Design Requirements to Operating Procedures for the Transfer of Residual Heat Removal and Containment Spray Suction to the Containment Recirculation Sumps (Section 1R21.3.2)

Closed

05000482/2007006-09	URI	Potential for Damage to Turbine Driven Auxiliary Feedwater Pump and Essential Service Water System During Condensate Storage Tank Unavailability (Section 1R21.2.3)

LIST OF DOCUMENTS REVIEWED

PROCEDURES

NAME	TITLE	REVISION / DATE
AP 14A-003	Scaffold Construction and Use	17
STS PE-049C	A Train Underground Essential Service Water System Piping Leakage Test	6
STS EF-1OOB	Essential Service Water System Inservice Pump B & Essential Service Water B Discharge Check Valve Test	35
STS VT-001	Verification of OMN-L, Motor Operated Valve Exercise Requirements	3B
STN PE-040	Essential Service Water Crosstie Valve Leak Test	6
STS KJ-001B	Integrated Diesel Generator and Safeguards Actuation Test Train B	36

NAME	TITLE	REVISION / DATE
EMG E-0	Reactor Trip or Safety Injection	20
EMG ES-12	Transfer to Cold Leg Recirculation	17
STS EM-100B	Safety Injection Pump "B" Inservice Pump Test	23
AI 2D-003	Motor Operated Valve Trending and Periodic Verification Program	1B
P-01B-007-CNOOI	Evaluation of the Essential Service Water Piping System for Higher Pressure (from 1750 psig to 1825 psig).	January 7, 2008
MCE DC-001	DC Ground Location	8
MPE-E009Q-01	13.8 kV and 4.16 kV Switchgear Inspection and Testing	12
CKL ZL-009	Site Readings Sheets	68
0400	Westar Energy, Inc. Transmission Operations Procedure	March 30, 2010
OFN AF-025	Unit Limitations	29
I-ENG-003	Vibration Monitoring and Analysis	5A
I-ENG-004	Lubricating Oil Analysis	3A
MGE-EOOP-05	Insulation Resistance Testing	18
MPE-ML-001	Motor Sampling and Lubrication PM Activity on Various Equipment	13
MPE-E009Q-03	Inspection and Testing of Siemens Vacuum Circuit Breakers	4
STS MT-018	Weekly Inspection of 125 VDC Lead-Calcium Batteries	20B
STS MT-019	125 VDC Class 1E Quarterly Battery Inspection	
STS MT-021	Service Test for 125 VDC Class 1E Batteries	15A
STS MT-058	Modified Performance Test for 125 VDC Class 1E Batteries	5A
ALR 00-051D	Component Cooling Water Surge Tank A Level HI/LO	6A
ALR 00-053D	Component Cooling Water Surge Tank B Level HI/LO	5A
ALR 00-127D	Condensate Storage Tank Level LOLO 2	8

NAME	TITLE	REVISION / DATE
ALR 00-127E	Condensate Storage Tank Level LOLO 1	10a
AP-M-01	Minimum Condensate Storage Tank Water Level	0
EMG E-0	Reactor Trip or Safety Injection	25
EMG ES-12	Transfer to Cold Leg Recirculation	18
MPE VD-001	M627A Ventilation Damper Maintenance	11
MPU-FU-001	Fan Unit Maintenance	15A
OFN EG-004	Component Cooling Water System Malfunctions	11
STN OQT-001A	Operations A Train Quarterly Tasks	30
STN OQT-001B	Operations A Train Quarterly Tasks	38
STS IC-560	Channel Calibration Auxiliary Feedwater Pump Low Suction Pressure	9A
STS PE-007	Periodic Verification of Motor Operated Valves	3
STS PE-050A	Component Cooling Water System Train A and Auxiliary Building Service Loop Pressure Test	6
STS PE-050B	Component Cooling Water System Pump C Pressure Test	5
STS PE-050C	Component Cooling Water System Train B Pressure Test	5
STS PE-050D	Component Cooling Water System Pump D Pressure Test	6
STS PE-050E	Component Cooling Water System Inside Containment Service Loop Pressure Test	8
RNM C-0577	General Electric IAC Induction Disc Overcurrent Relay	4
STS CR-001	Shift Logs for Modes 1, 2, and 3	76

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NAME	TITLE	REVISION
NK-E-001	Class 1E DC Voltage Drop	2
NK-E-002	Class 1E Battery Sizing	4
NK-E-003	Class 1E 125V DC Batteries Short-Circuit Study	1

NAME	TITLE	REVISION
XX-E-009, Attachment BC, Sheet 3 of 7	Speed Torque Current Curves (Essential Service Water pump motor)	10
10466-EF-32	Essential Service Water Pump Motor Thermal Growth	0
AL-30-WC	Auxiliary Feed Water System Setpoints	3
AN-02-010	Emergency Diesel Generator Room Temperature	0
AP-M-001	Minimum Condensate Storage Tank Water Level	0
BN-20	Refueling Water Storage Tank Volumes and Level Setpoints	3
BN-21	Maximum Flow Rate from Refueling Water Storage Tank to Containment	0
BN-M-013	Time Available for Injection, Emergency Core Cooling System, and Containment Spray Pumps Transfer	2
EG-04-W	Determine Acceptable Surge Tank Vacuum	0
EG-21	Component Cooling Water Surge Tanks Thermal Growth	0
EG-26	Component Cooling Water System Pipe Break	0
EG-M-30	Component Cooling Water Surge Tank Relief Valves EGV0159 and EGV0170 Sizing Review	2
EJ-M-07	EJHV8804A/B Motor Operated Valve Bounding Pressure Conditions Determination	2
EJ-M-015	Thrust Torque Calculation for EJHV8804A and EJHV8804B	5
GM-99	Diesel Building Heating Ventilation and Air Conditioning	A
GM-249	Diesel Building Heating Ventilation and Air Conditioning	0
J-K-EG01	Instrument Uncertainty Estimate and Safety Related Setpoints	1
LE-M-004	Flooding in Class 1E Switchgear Rooms	0
M-EG-5	Component Cooling Water System	1
M-EG-12	Component Cooling Water System Pipe Break	1
SA-90-064	Analysis of Containment Pressure Temperature Response to the Postulated Loss of Coolant Accident	1

NAME	TITLE	REVISION
EF-21	Essential Service Water Pumphouse and Yard Piping for Wolf Creek.	1
EF-24	Essential Service Water System Effluent Temperature	3
EF-29	EF-HV-043 and EF-HV-44 Maximum Flow After Pipe Break of Non-Nuclear Piping Downstream	0
EF-MH-002	Reconciliation of the Essential Service Water Pump Performance and Underground Leakage Test Data	0
EF-35	Essential Service Water Pump Head Requirement	2
EF-M-021-000-CN001	Component Cooling Water Pump 07416 Replaces the Existing Jamesbury 30" Butterfly Valves with Crane Butterfly Valves	0
XX-E-009	System NB, NG, PG Undervoltage/ Degraded Voltage Relay Setpoints	1
XX-E-012	Safety-Related MCC Control Circuit Allowable Wire Lengths	2
E-H-8	System NB Protective Relays	5
GM-320	Diesel/Generators Building Heating Ventilation and Air Conditioning	0
XX-E-006	Evaluate the Performance of the WCNOC Station Auxiliary System	5
E-11005A	Emergency Diesel Generator Loading Data	1
CN-SEE-I-08-67	Wolf Creek/Callaway Mechanical Seal Evaluation for Emergency Core Cooling System Pumps	0
XX-M-075	Wolf Creek Generating Station Emergency Core Cooling System Piping Void Analysis	0
XX-M-076	Startup Pressure Pulse Analysis for Wolf Creek Generating Station Emergency Core Cooling System Discharge Piping	0
EM-M-002	Safety Injection Pump Discharge Pressure	0

DRAWINGS

NAME	TITLE	REVISION
12937-13	Schedule for Q-Listed Tornado Dampers	A
E-12NF01	Load Shedding and Emergency Load Sequencing Logic	4

E-13EJ04A	Schematic Diagram Residual Heat Removal Pump 1 to Charging Pump Valve	5
E-13EJ04B	Schematic Diagram Residual Heat Removal Pump 2 to Safety Inj. Pump Valve	7
J-12AL04	Control Logic Diagram	1
J-12EG02	Component Cooling Water System Demineralized Water Make-Up to Component Cooling Water Surge Tank	7
KP1909W	Logic Block Diagram Engineered Safety Features Actuation System	0
M-11EG01	System Flow Diagram Component Cooling Water	2
M-12AL01	Auxiliary Feedwater System	10
M-12BN01	Borated Refueling Water Storage Tank	14
M-12EG01	Component Cooling Water System	16
M-12EG02	Component Cooling Water System	19
M-12EG03	Component Cooling Water System	9
M-12EJ01	Residual Heat Removal System	43
M-12EM01	High Pressure Coolant Injection System	37
M-12GM01	Diesel Generator Building Heating Ventilation Air Conditioning	1
M-13EF06	Piping Isometric	15
E-11NK02	Class 1E 125V DC System Meter and Relay Diagram	8
E-11NB02	Lower Medium Voltage System Class 1E 4.16 kV Single Line Meter and Relay Diagram	2
E-13GM04-A	Schematic Diagram Diesel Generator Building Exhaust Damper	0
E-018-00176	Motor Control Center Layout Specifications	11
J-12GM01A	Diesel Generator Building HVAC Supply Fan	1
E-018000177	Motor Control Center Layout Specifications	12
E-13GM01	Schematic Diagram Diesel Generator Ventilation Supply Fan	4

E-13GM01A	Schematic Diagram Diesel Generator Ventilation Supply Fan	4
E-11NB01	Lower Medium Voltage System Class 1E 4.16 KV Single Line Diagram Meter and Relay Diagram	2
E-11PG20	Low Voltage System Non-Class 1E Motor Control Center Summary	105

ENGINEERING REPORTS

NUMBER	TITLE	REVISION / DATE
10-00620	Evaluation of Guillotine Break in Non-Safety Wolf Creek Nuclear Operating Company Piping	October 26, 2010
E-009B-00003	Evaluation of Siemens 3EF1 036 Surge Limiters	W02
VTS-021111-R	Fairbanks Morse, Wolf Creek Nuclear Power Station, Heat Exchanger Analysis Supplemental Report	November 11, 2002
M-018-01502	Load Table Analysis Submittal W01– Frequency and Voltage Dip Study – 2001	January 4, 2002

VENDOR MANUALS

NUMBER	TITLE	REVISION
13364	Tornado Damper – Inlet	G
14113	Tornado Damper – Exhaust	0
E-009B-00009	Siemens Type 3AF-GER Vertical Lift Direct Replacement Vacuum Circuit Breakers	W05
E-050A-00011	Lucent Technologies LINEAGE 2000 Round Cell Battery	W01
E-012.2-00014	Instruction Manual for Essential Service Water Pump Motors	W06
10466-E-029	Technical Specification for 5000 Volt and 15000 Volt Single-Conductor Power Cable	9

CONDITION REPORTS

00006250	00028237	00029601	00027947	00028399
00011247	00028246	00029602	00002492	00028179
00028059	00028281	00029603	00028201	00028271
00028136	00028335	00029605	00027912	00028204

00028139	00028343	00026446	00027990	00028290
00028185	00028399	00028170	00005944	00028695
00028210	00029252	00002580	00002658	00028451
00028215	00029389	00028628	00002597	00001617

MAINTENANCE WORK ORDERS

99-210151-000	08-304080-000	08-310457-000	08-309902-000	09-321589-010
05-274434-007	08-305212-005	10-330309-000	09-318549-000	09-321589-006
06-291396-001	08-306365-000	04-261091-000	10-326301-000	10-324569-000
06-291562-005	08-308644-001	10-326276-000	10-326024-001	08-310554-000
07-294733-000	08-308653-001	06-291340-000	07-300602-000	09-316335-000
07-298329-000	08-308668-001	09-321444-000	10-323452-000	10-323767-000
07-299121-000	08-308682-001	09-315426-000	06-290427-000	08-303897-000
08-203681-001	08-309276-001	10-326024-000	10-325906-000	10-323770-000
10-326288-000				

MISCELANEOUS

NUMBER	TITLE	REVISION / DATE
96-0521	Performance Improvement Request	February 21, 1996
M-627A	Technical Specifications for Dampers	12
OE03	Operating Experience Detail Report	September 24, 2008
WCNOC-09-0001/12957	Auxiliary Feedwater Supply Surge Tank	January 6, 2009
09-02-ENG	Quality Assurance Audit Report – Engineering Program	April 3, 2009
ET 07-0007	90 Day response to NRC Geleric Letter 2007-01, Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients	May 2, 2007
ET 06-0038	Application for Renewed Operating License	September 27, 2006
07-00575	Request for Additional Information on the Response to Generic Letter 2007-01	October 25, 2007

ET 07-0053	Response to NRC Request for Additional Information on the Response to Generic Letter 2007-01	November 13, 2007
08-00605	Closeout of Generic Letter 2007-01	October 28, 2008
WO 10-0045	Revision to Response to NRC Generic Letter 2007-01	June 30, 2010
EF-09-0002	Operability Evaluation – Essential Service Water Cables Functional Checks Train A	0
E-10NK(Q) System Description	125 – Volt DC System (Class 1E Power System)	3
M-10EF System Description	Essential Service Water System	6
10466-E-00NB(Q) System Description	Lower Medium Voltage System – 4.16 kV (Class 1E Power System)	7
10466-E-012.2-054-03	Large Induction Motors 250 Horsepower and Larger	3
J-435 (Q)	Design Specification for Orifice Plate for Nuclear Class 2 and 3 Piping Systems for the Wolf Creek Generating Station	17
10466-M-018	Technical Specification for Standby Diesel Generators for the Standardized Nuclear Unit Power Plant System	6

ATTACHMENT 2
Wolf Creek CCW Seismic Finding
Significance Determination Process

A. Brief Description of Issue

The inspectors identified a performance deficiency related to isolation between the safety related and non-safety related portions of the CCW system. The CCW system design included a non-safety related section that supplies equipment in the radiation waste building; this portion of the system was designed to be automatically isolated from the safety related portion by automation air-operated valves (two in series on the supply and two in series on the return piping). These isolation valves were designed close on either high CCW system flow or low CCW surge tank level, indicating a potential break in the system. The intent of the design was to preserve sufficient inventory in the surge tank to keep the system full and operating after the postulated failure of this non-safety related piping.

The team reviewed calculation M-EG-12, Component Cooling Water System Pipe Break, revision 1; this calculation showed that there would sufficient volume available in the surge tank to accommodate the most limiting pipe break, assuming the isolation valves automatically closed within 10 seconds. However, the team determined that the calculation did not address the capacity of the 4-inch surge tank pipe to transfer water from the surge tank to the CCW pump suction at the same rate as the postulated pipe break flow (assumed to be 14,000 gpm in the calculation). As a result, the team questioned if pump suction piping would have a negative pressure during the transient, resulting in pump cavitation, or if air from the postulated break would enter the CCW system.

As a result of these concerns, the licensee initiated Condition Report 00028237 on September 22, 2010. The licensee also declared the affected train on CCW inoperable; compensatory actions were then taken (administratively closing the isolation valves) and the system was returned to operable status within one day. During the inspection period, the licensee continued to evaluate this condition to determine if the affected train of CCW was inoperable prior to administratively closing the system isolation valves and to determine what corrective actions will be required.

B. Statement of Performance Deficiency

The licensee failed to adequately analyze the interface between the safety-related and non-safety related portions of the CCW system. Specifically, the inspectors determined that the current design calculation was not bounding because it did not ensure the continued operability of the affected CCW train in the event of a failure in the non-safety related portion of the system. As a result, the affected CCW pumps could be subject to reduced suction pressure, cavitation, and potential air ingestion.

C. Significance Determination Basis

1. Reactor Inspection for IE, MS, BI Cornerstones

a. <u>Phase 1 Screening Logic, Results and Assumptions</u>

In accordance with NRC Inspection Manual Chapter 0612, Appendix B, "Issue Screening," the inspectors determined that the failure to adequately analyze the isolation between the safety related and non-safety related portions of the CCW system was a performance deficiency that was reasonably within Wolf Creek's ability to foresee and prevent. The finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the design basis analysis did not ensure that the affected train of CCW would perform its required functions after the failure of non-safety related CCW piping. Also, the inspectors determined that the finding was similar to Examples 3.j and 3.k of Manual Chapter 0612, Appendix E, in that there was a reasonable doubt of the operability of the component based on the existing analyses.

The analyst evaluated the issue using the Significance Determination Process (SDP) Phase 1 Screening Worksheet for the Initiating Events, Mitigating Systems, and Barriers Cornerstones provided in Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings." This finding affected the Mitigating Systems Cornerstone because seismic protection was degraded. The analyst determined that the finding represented the degradation of equipment and functions specifically designed to mitigate a seismic event and that during an earthquake the deficiency would degrade one train of component cooling water, a system that supports a safety system or function. Therefore, the finding was potentially risk significant to seismic initiators and a Phase 3 analysis was required.

b. <u>Phase 2 Estimation</u>

The risk-informed inspection notebooks estimate the risk for internal initiators. The plant-specific SDP worksheets do not currently include initiating events related to fire, flooding, severe weather, seismic, or other external initiating events. The subject performance deficiency only affected the plant response to seismic events. Therefore, a Phase 2 estimation was not applicable to this inspection finding.

c. <u>Phase 3 Evaluation for External Events</u>

The analysts determined that, for the subject performance deficiency to affect the core damage frequency, a seismic event must result in a gross piping failure of the nonsafety-related portion of the component cooling water system. Additionally, the analyst assumed that such an earthquake would result in an unrecoverable loss of offsite power (LOOP). Additionally, the analyst determined that, normal operating alignments of the system would result in the failure of 1 but not both of the component cooling water trains. To quantify the increase in core damage frequency (ΔCDF) caused by this design deficiency, the analyst also must know the change in core damage probability assuming that the above postulated conditions occurred.

As such, the analysts evaluated the subject performance deficiency by determining each of the following parameters for any seismic event producing a given range of median peak-ground acceleration "a" [SE(a)]:

The frequency of the seismic event SE(a) ($\lambda_{SE(a)}$) ;
The probability that a LOOP occurs during the event ($P_{LOOP\text{-}SE(a)}$);
The probability that nonsafety CCW piping fails during the event ($P_{CCW\text{-}SE(a)}$);
The probability that CCW pumps fail before the system isolates ($P_{Failure\text{-}SE(a)}$); and
The conditional change in core damage probability ($\Delta CCDP_{SE(a)}$)

The ΔCDF for the acceleration range in question ($\Delta CDF_{SE(a)}$) can then be quantified as follows:

$$\Delta CDF_{SE(a)} = \lambda_{SE(a)} * P_{LOOP\text{-}SE(a)} * P_{CCW\text{-}SE(a)} * P_{Failure\text{-}SE(a)} * CCDP_{SE(a)}$$

Given that each range “a” was selected by the analysts specifically to be independent of all other ranges, the total increase in risk, ΔCDF, can be quantified by summing the $\Delta CDF_{SE(a)}$ for each range evaluated as follows:

$$\Delta CDF = \sum_{a=.05}^{1} \Delta CDF_{SE(a)}$$

over the range of SE(a).

Frequency of the Seismic Event

NRC research data indicated that seismic events of 0.05g or less have little to no impact on internal plant equipment. Therefore the analyst assumed that seismic events less than 0.05g do not directly affect the plant. The analyst assumed that seismic events greater than 1.0g lead to core damage. The analysts therefore examined seismic events in the range of 0.03g to 1.0g.
The analysts divided that range of seismic events into segments (called "bins" hereafter); specifically, seismic events were binned between each of the following median accelerations 0.05g, 0.08g, 0.15g, 0.25g, 0.30g, 0.40g, 0.50g, 0.65g, 0.80g and 1.0g, respectively.

In order to determine the frequency of a seismic event for a specific range of peak ground motion (g values), the analysts used the frequencies of exceedance published in Table 4A-1, “Seismic Hazard Vectors for the 72 SPAR Plants,” of the Risk Assessment of Operational Events Handbook, Volume 2, “External Events,” Revision 1.01. These values indicated the frequency of the seismic event that generates a level of ground motion that exceeds the lower value in each of the bins. The analysts then calculated the difference in these "frequency of exceedance" values to obtain the frequency of seismic events for the binned seismic event ranges.

For example, according to the Handbook, the frequency of exceedance for a 0.3g seismic event is estimated at 9.86 x 10^{-6}/yr and a 0.4g seismic event at 4.67 x 10^{-6}/yr. The frequency of seismic events with median acceleration in the range of 0.3g to 0.4g [SE(0.3-0.4)] equals the difference, or 5.19 x 10^{-6}/yr.

Probability of a Loss of Offsite Power

The analysts assumed that a seismic event severe enough to break the ceramic insulators on the transmission lines will cause an unrecoverable loss of offsite power.

The analysts obtained data on switchyard components from the Risk Assessment of Operating Events Handbook; Volume 2, "External Events," Revision 1.01, which referenced generic fragility values listed in:

> NUREG/CR-6544, 'Methodology for Analyzing Precursors to Earthquake-Initiated and Fire-Initiated Accident Sequences," April 1998; and
>
> NUREG/CR-4550, Vols 3 and 4 part 3, ‘Analysis of Core Damage Frequency: Surry / Peach Bottom’ 1986

The references describe the mean failure probability for various equipment using the following equation:

$$P_{fail}(a) = \Phi\ [\ \ln(a/a_m) / (\beta_r^2 + \beta_u^2)^{1/2}]$$

Where Φ is the standard normal cumulative distribution function and

a = median acceleration level of the seismic event;
a_m= median of the component fragility;
β_r= logarithmic standard deviation representing random uncertainty;
β_u= logarithmic standard deviation representing systematic or modeling uncertainty.

In order to calculate the LOOP probability given a seismic event the analysts used the following generic seismic fragilities:

a_m = 0.3g
β_r = 0.30
β_u = 0.45

Using the above normal cumulative distribution function equation the analysts determined the conditional probability of a LOOP given a seismic event. For each of the bins the calculation was performed substituting for the variable "a" (median acceleration level) the acceleration levels obtained from the bins described above. The following table shows the results of the calculation for various acceleration levels.

Median Acceleration Level/Probability of LOOP							
0.08g	3.1E-2		0.3g	6.0E-1		1.0g	1.0
0.25g	4.3E-1		0.8g	9.8E-1			

Probability That Nonsafety-Related CCW Piping Fails

In order to calculate the probability that nonsafety-related CCW piping ruptures given a seismic event, the analysts used the following generic seismic fragilities (using an NRC assessment of industrial facility pipe breaks):

$a_m = 0.50g$
$\beta_r = 0.30$
$\beta_u = 0.35$

Using the above standard normal cumulative distribution function equation, the analysts determined the conditional probability that nonsafety-related CCW piping ruptures given a seismic event for each of the bins. The calculation was performed substituting for the variable "a" (median acceleration level) the acceleration levels obtained from the bins described above. The following table shows the results of the calculation for various acceleration levels.

Median Acceleration Level/ Probability of Nonsafety-Related CCW Piping Rupture							
0.08g	4.9E-4		0.3g	2.1E-1		1.0g	7.5E-1
0.25g	9.6E-2		0.8g	9.0E-1			

Probability of Single-Train Failure of CCW PumpsGiven the lack of evidence that the pumps would not cavitate and air bind during the isolation of a secondary system pipe break, the analyst assumed that 1 train of component cooling water would fail given a seismic rupture of the piping. Therefore, the analyst determined that the best available approximation of the probability of failure ($P_{Failure-SE(a)}$) was 1.0.

Conditional Change in Core Damage Probability

The analyst evaluated the spectrum of seismic initiators to determine the resultant impact on the reliability and availability of mitigating systems affecting the subject performance deficiency.

The analyst used the Wolf Creek SPAR Model, Revision 3.51, to perform the Phase 3 evaluation. The analysts first created a baseline case by setting the initiating event probability for a LOOP to 1.0 and all other initiating event probabilities in the SPAR model to zero. Offsite power was assumed to be non-recoverable following seismic events that break the ceramic insulators (low fragility components) on the transmission lines. Therefore, the analysts set the non-recovery probabilities for offsite power to 1.0. The SPAR model showed the resultant core damage probability as 3.01×10^{-3}, which represented the baseline case that is used in the above equation.

The SPAR Model showed that loss of a train of component cooling water during an unrecoverable LOOP indicated a conditional core damage probability of 2.37×10^{-2}. Therefore, the change in core damage probability is

$$\Delta CCDP_{SE(a)} = 2.37 \times 10^{-2} - 3.01 \times 10^{-3} = 2.1 \times 10^{-2}$$

Phase 3 Seismic Results

Considering the factors described above, namely,

- The frequency of the seismic event;
- The probability that a LOOP occurs during the event;
- The probability that that nonsafety CCW piping fails during the event;
- The probability that CCW pumps fail before the system isolates; and
- The conditional change in core damage probability

The total increase in risk, ΔCDF, can be quantified by summing the $\Delta CDF_{SE(a)}$ for each bin as follows:

$$\Delta CDF = \sum_{a=.03}^{6} \Delta CDF_{SE(a)}$$

over the range of SE(a).

The analyst's quantification was performed using an issue-specific spreadsheet and the resulting ΔCDF was 7.0×10^{-8}. This indicates that the risk of the subject finding is best characterized as Green.

d. Risk Contribution from Large Early Release Frequency (LERF

Using IMC 0609 Appendix H, the SRA determined that this was a Type A finding (i.e., LERF contributor) for a large dry containment. For PWR plants with large dry containments, only findings related to accident categories ISLOCA and SGTR have the potential to impact LERF. In addition, an important insight from the IPE program and other PRAs is that the conditional probability of early containment failure is less than 0.1 for core damage scenarios that leave the RCS at high pressure (>250 psi) at the time of reactor vessel breach. Since this finding is not related to ISLOCA or SGTR, and the dominant core damage scenarios for this finding leave the RCS at high pressure, the analysts concluded that LERF is not a significant contributor to the risk associated with this finding.