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

November 22, 2013

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

SUBJECT: WOLF CREEK GENERATING STATION – NRC COMPONENT DESIGN BASIS INSPECTION REPORT 05000482/2013008

Dear Mr. Sunseri

On August 29, 2013, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Wolf Creek Generating Station. The NRC inspection team discussed the results of this inspection with Mr. R. Smith, Site Vice President and Chief Nuclear Operations Officer, and other members of your staff. After additional in-office inspection, a final telephonic exit meeting was conducted on October 28, 2013, with Mr. J. Broschak, Vice President, Engineering, and other members of your staff. The inspection team documented the results of this inspection in the enclosed inspection report.

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

The NRC inspection team documented thirteen findings of very low safety significance (Green) in this report. All thirteen of the findings were determined to involve violations of NRC requirements. The NRC is treating these violations as non-cited violations (NCV's) consistent with Section 2.3.2.a of the Enforcement Policy.

If you contest the violations or significance of these non-cited 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 DC 20555 0001; with copies to the Regional Administrator, Region IV; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555 0001; and the NRC resident inspector at the Wolf Creek Generating Station. In addition, if you disagree with the characterization of the cross-cutting aspect assigned to any findings 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 the Wolf Creek Generating Station.

In accordance with Title 10 of the Code of Federal Regulations 2.390, “Public Inspections, Exemptions, Requests for Withholding,” of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC’s Public Document Room or from the Publicly Available Records (PARS) component of the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

Dockets No: 05000482
License No: NPF-42

Enclosure:
Inspection Report 05000482/2013008
w/ Attachment: Supplemental Information

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RI:DRS/EB3	RI:DRS/EB2	SOE:DRS/OB	SRA:DRS/EB1	SRI:DRS/EB1	C:DRS/EB1
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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000482
License:	NPF-42
Report Nos.:	05000482/2013008
Licensee:	Wolf Creek Nuclear Operating Corporation
Facility:	Wolf Creek Generating Station
Location:	1550 Oxen Lane NE Burlington, Kansas
Dates:	August 5-9, 2013 On Site August 12-16, 2013 In Office August 19-30, 2013 On Site September 2 through October 28, 2013 In Office
Team Leader:	R. Kopriva, Senior Reactor Inspector, Engineering Branch 1
Inspectors:	A. Sengupta, Reactor Inspector, Engineering Branch 3, Region II B. Correll, Reactor Inspector, Engineering Branch 2 S. Garchow, Senior Operation Inspector, Operations Branch
Accompanying Personnel:	H. Campbell, Ph.D., Contractor Beckman and Associates H. Leake, Contractor, Beckman and Associates
Approved By:	Thomas R. Farnholtz, Chief Engineering Branch 1

SUMMARY OF FINDINGS

IR 05000482/2013008; August 5, 2013 – October 28, 2013; Wolf Creek Generating Station; Baseline Inspection, NRC Inspection Procedure 71111.21, "Component Design Basis Inspection"

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

Cornerstone: Mitigating Systems

- Green. The team identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," which states, in part, "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings." Specifically, in 2007 the licensee failed to follow Procedure AP 15C-004, "Preparation, Review and Approval of Procedures, Instructions and Forms," when making changes to safety-related emergency diesel generator surveillance testing Procedure OFN NB-042. The technical reviewer failed to identify that the power supply for the communication equipment for the dedicated operator was from non-essential power and would be lost during a loss of offsite power event, losing the communications between the control room and the operator. The licensee has entered this issue into their corrective action program as Condition Report CR-72711.

  The team determined that the failure to follow Procedure AP 15C-004 when making changes to off normal operating Procedure OFN NB-042 was a performance deficiency. This finding was more than minor because it was associated with the Equipment Performance attribute of the Reactor Safety, Mitigating Systems Cornerstone, and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to perform a technical walk-down of the procedure steps to verify the power supply for the communication equipment would not be lost during a loss of power event. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, the inspectors determined the finding was of very low safety significance (Green), because the finding was not a design deficiency and did not result in the loss of operability or functionality. This finding did not have a cross-cutting aspect because the most significant contributor to the performance deficiency did not reflect current licensee performance. (Section 1R21.2.2)

- Green. The team identified a Green, non-cited violation, with three examples, of 10 CFR 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Specifically, on September 12, 2011, the licensee failed to verify or check the adequacy of design Calculation XX-E-006, "AC System Analysis," Revision 6, by 1) not recognizing that the actual switchyard voltage could be lower than the calculated minimum voltage due to loop uncertainties of the switchyard voltmeters, 2) failing to provide a comparison between postulated loading levels and equipment ratings for distribution equipment, in order to verify that overloading conditions would not occur, and 3) not placing limits on the voltages on the Class 1E 480 Vac system which could exceed the allowable maximum equipment voltage rating of 506 Vac. The licensee has entered these issues into their corrective action program as Condition Reports CR-73244, CR-73240, and CR-73206.

  The team determined that the licensee's failure to verify or check the adequacy of design Calculation XX E 006, "AC System Analysis," Revision 6, was a performance deficiency. This finding was more than minor because it was associated with the Design Control attribute of the Reactor Safety, Mitigating Systems Cornerstone, and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to verify or check the adequacy of design Calculation XX-E-006, "AC System Analysis," Revision 6, regarding loop uncertainties of the switchyard voltmeters, equipment loading, and maximum allowed Class 1E 480 voltage. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, the inspectors determined the finding was of very low safety significance (Green), because the finding was not a design deficiency and did not result in the loss of operability or functionality. This finding had a cross-cutting aspect in the area of Human Performance, associated with the Resources component because the licensee failed to ensure that personnel, equipment, procedures, and other resources are adequate to assure nuclear safety by maintaining long term plant safety by maintenance of design margins. [H.2(a)] (Sections 1R21.2.5 and 1R21.2.8.1)

- Green. The team identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Specifically, on May 9, 2003, Calculation XX-E-009, "System NB, NG, PG Undervoltage/Degraded Voltage Relay Setpoints," Revision 1, identified that the degraded voltage relays minimum time delay was 7.5 seconds, and the maximum time delay was 8.5 seconds. During testing of the degraded voltage relays, the calculation states, "In all cases the steady state voltage on NB01 and NB02 recovered within the 7.5 seconds accident criteria. However in some cases the recovery time is marginal." This requirement was not correctly translated into Surveillance Test Procedures STS IC-805A and STS IC-805B which allow a minimum time delay of 7.0 seconds, and a maximum time delay of 9.0 seconds for the degraded voltage relays timeout period during accident conditions. The licensee has entered this issue into their corrective action program as Condition Report CR-72496.

The team determined that the licensee's failure to ensure that the analyzed minimum allowable degraded voltage relay time delay of 7.5 seconds and maximum allowable degraded voltage relay time delay of 8.5 seconds, was incorporated into acceptance criteria for surveillance testing procedures was a performance deficiency. This finding was more than minor because it was associated with the Procedure Quality attribute of the Reactor Safety Initiating Events Cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, it was indeterminate whether the design requirement to prevent spurious actuation of the degraded voltage relays and consequential loss of offsite power would have been met if the time delay had been set at less than 7.5 seconds or greater than 8.5 seconds. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 1, "Initiating Events Screening Questions," the finding was determined to have very low safety significance (Green), because it did not cause a reactor trip and loss of mitigation equipment. This finding did not have a cross-cutting aspect because the most significant contributor to the performance deficiency did not reflect current licensee performance. (Section 1R21.2.6)

- Green. The team identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," which states, in part, "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected." Specifically, in 2006, the licensee implemented corrective actions per Condition Report 2006-2062, to monitor the voltages for the 480 Vac system to ensure that over-voltages would not occur during emergency diesel generator testing. The licensee implemented voltage monitoring for the "B" Train 480 Vac system, but failed to monitor voltages of "A" Train, which had the same vulnerability. The licensee has entered this issue into their corrective action program as Condition Report CR-73209.

  The team determined that the licensee's failure to implement corrective actions into diesel testing Procedure STS KJ-001A was a performance deficiency. This finding was more than minor because it was associated with the Equipment Performance attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to ensure that over-voltages would not occur during the testing of the "A" train emergency diesel generator. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, the inspectors determined the finding was of very low safety significance (Green), because the finding was not a design deficiency and did not result in the loss of operability or functionality. This finding did not have a cross-cutting aspect because the most significant contributor to the performance deficiency did not reflect current licensee performance. (Section 1R21.2.7.1)

- Green. The team identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Specifically, on June 26, 2013, the licensee issued drawing E-11005, "List of Loads Supplied by Emergency Diesel Generator," Revision 39, that identified certain motors with load brake horsepower in excess of the motor nameplate ratings, but failed to verify that the excess horsepower would not result in the motors exceeding their thermal design limits. Additionally, the brake horsepower values on the referenced drawing do not reflect the worst-case condition, which would occur when the diesel generator is operating at maximum allowable frequency and powering the motors. The licensee has entered this issue into their corrective action program as Condition Report CR-72945.

  The team determined that the licensee's failure to evaluate motor loading to confirm margin exists to prevent overheating of the motors was a performance deficiency. This finding was more than minor because it was associated with the Design Control attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, motors serving loads with demands in excess of the motor horsepower ratings were not analyzed to ensure that overheating would not occur. In accordance with Inspection Manual Chapter 0609 Appendix A, Exhibit 2, the inspectors determined the finding was of very low safety significance (Green), because the finding was not a design deficiency and did not result in the loss of operability or functionality. This finding had a cross-cutting aspect in the area of Human Performance, associated with the Resources component, because the licensee failed to ensure that personnel, equipment, procedures, and other resources are adequate to assure nuclear safety by maintaining long term plant safety by maintenance of design margins. [H.2(a)] (Section 1R21.2.7.2)

- Green. The team identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Specifically, on August 27, 2013, the team identified that the licensee had failed to account for flow measurement uncertainties of the Residual Heat Removal System. Technical Specifications require that when operating in Mode 6, the circulating residual heat removal flow is required to be greater than or equal to 1000 gpm for adequate heat removal and to prevent stratification, and Alarm Response Procedure ALR 00-049C, "RHR LOOP 1 FLOW LOW," requires that when operating the residual heat removal pumps at low flows that the flow must be at or above 1700 gpm for pump protection. The failure to account for flow measurement uncertainties could allow flow to actually be below the required technical specification and alarm response limits, without the operator's knowledge. The licensee has entered this issue into their corrective action program as Condition Reports CR-73071 and CR-73231.

The team determined that the failure to account for flow measurement uncertainties when operating Residual Heat Removal pumps at low flows was a performance deficiency. This finding was more than minor because it was associated with the Procedure Quality attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to account for flow measurement uncertainties in the residual heat removal system could allow operation below technical specification and alarm response limits and potentially damage the residual heat removal pumps. In accordance with NRC Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," the finding was determined to have very low safety significance (Green), because the finding did not require a quantitative assessment because adequate mitigating equipment remained available and the finding did not constitute a loss of control as defined in Appendix G. This finding did not have a cross-cutting aspect because the most significant contributor to the performance deficiency did not reflect current licensee performance. (Section 1R21.2.9)

- Green. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," which states, in part, "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, defective materials and equipment, and nonconformance are promptly identified and corrected." Specifically, in April 2011 and November, 2012, the licensee failed to properly categorize Condition Reports CR-37825 and CR-60298 correctly, which resulted in the condition reports not getting an Apparent Cause Evaluation, to promptly identify and correct the cause of the Component Cooling Water Butterfly Valve EGHV0102 loose disc to shaft, failure of the groove pin in the valve, and to investigate the extent of condition for similar valves currently installed in the plant. The licensee has entered this issue into their corrective action program as Condition Report CR-73227.

  The team determined the licensee's failure to follow the Corrective Action Procedure AI 28A-010, "Screening Condition Reports," which improperly categorized Condition Reports CR-37825 and CR-60298, which should have had apparent cause evaluations performed, was a performance deficiency. This finding was more than minor because it adversely affected the Equipment Performance attribute of the Mitigating Systems Cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to perform an apparent cause evaluation resulted in the licensee not identifying a root cause for the valve leakage, preventing reoccurrence, or investigating the extent of condition for other similar valves installed in the plant. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, the inspectors determined the finding was of very low safety significance (Green), because the finding was not a design deficiency and did not result in the loss of operability or functionality. This finding had a cross-cutting aspect in the area of Human Performance, associated with Work Practices. Specifically the licensee defines and effectively communicates expectations regarding procedural compliance and personnel follow procedures. [H.4(b)] (Section 1R21.2.14.1)

- Green. The team identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Specifically, in 1994, the licensee was committed to the requirements specified in Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves," Revision 1, to remove the thermal overload bypass jumpers during maintenance and testing. The licensee failed to translate the requirements into Procedure MGE LT-099, "MOV Diagnostic Testing," and failed to include procedural guidance to remove the thermal overload bypass jumpers when performing maintenance testing that strokes the valve from the control room. Also, the Wolf Creek Updated Safety Analysis Report, Section 8.3.1.1.2, has incomplete information which does not support Regulatory Guide 1.106, in that it does not state that the thermal overload bypass jumpers should be removed when performing maintenance testing that strokes the valve. The licensee has entered this issue into their corrective action program as Condition Reports CR-73120 and CR-73219.

  The team determined that the licensee's failure to provide procedure instructions to remove the thermal over-load bypass jumpers during motor-operated valve diagnostic testing as committed to in Regulatory Guide 1.106, Revision 1, was a performance deficiency. This finding was more than minor because it was associated with the Procedure Quality attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to include procedural guidance to remove the thermal overload bypass jumpers when performing maintenance testing that strokes the valve from the control room, and to include the requirements of Regulator Guide 1.106 in the Updated Safety Analysis Report, Section 8.3.1.1.2, that the bypass jumpers will be removed during testing of the motor-operated valves. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, the inspectors determined the finding was of very low safety significance (Green), because the finding was not a design deficiency and did not result in the loss of operability or functionality. This finding did not have a cross-cutting aspect because the most significant contributor to the performance deficiency did not reflect current licensee performance. (Section 1R21.2.14.2)

- Green. The team identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Specifically, as of September 2011, Wolf Creek Updated Safety Analysis Report, Appendix 9.5 E, required isolation between safe shutdown circuits and non-safe shutdown (associated) circuits, such that "hot shorts, open circuits, or shorts to ground in the associated circuits will not prevent operation of the safe shutdown equipment." On September 29, 2011, the licensee completed study WCNOC-171, "Post-Fire Safe Shutdown Associated Circuits Study," Revision 0, but failed to provide documented verification of the adequacy of electrical protective devices for associated shutdown circuits such that hot shorts or shorts to ground will not prevent operation of the safe shutdown equipment. The licensee has entered this issue into their corrective action program as Condition Report CR-73242.

The team determined that the licensee's failure to provide a documented comparison of upstream and downstream electrical protective devices with maximum short circuit levels, in order to verify the required coordination, was a performance deficiency. This finding was more than minor because it was associated with the Design Control attribute of the Reactor Safety, Mitigating Systems Cornerstone, and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee was unable to provide an analysis to demonstrate that associated shutdown circuits would be isolated from the safe shutdown circuits during fire events. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, the inspectors determined the finding was of very low safety significance (Green), because the finding was not a design deficiency and did not result in the loss of operability or functionality. The finding had a cross-cutting aspect in the area of Human Performance, Resources attribute, because the licensee failed to ensure that personnel, equipment, procedures, and other resources are adequate to assure nuclear safety by maintaining long-term plant safety by maintenance of design margins. [H.2(a)] (Section 1R21.3.3)

- Green. The team identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion XI, "Test Control," which states, in part, "A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design document." Specifically, on August 28, 2013, the team identified that the licensee failed to incorporate minimum pump performance requirements into the corresponding pump surveillances for the Containment Spray and Residual Heat Removal pumps. The acceptance criteria did not adequately overlap with the pump design performance requirements. Further, instrument uncertainty was not adequately evaluated, nor incorporated into the tests. The licensee has entered this issue into their corrective action program as Condition Reports CR-73149 and CR-73070.

  The team determined that the failure to establish and incorporate adequate acceptance criteria into the Containment Spray and Residual Heat Removal pump comprehensive surveillance tests was a performance deficiency. This finding was more than minor because it was associated with the Procedure Quality attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to incorporate adequate acceptance criteria and instrument uncertainties into the safety related surveillances could cause unacceptable pump performance conditions to go undetected. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, the inspectors determined the finding was of very low safety significance (Green), because the finding was not a design deficiency and did not result in the loss of operability or functionality. This finding did not have a cross-cutting aspect because the most significant contributor to the performance deficiency did not reflect current licensee performance. (Section 1R21.3.5)

- Green. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action", which states, in part, "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, defective materials and equipment, and nonconformances, are promptly identified and corrected." Specifically, since May 2011, the licensee had numerous opportunities, but failed to correct calculation GK06W and to adequately assess compensatory actions identified to supplement weaknesses in the calculations for operation of one vital air conditioning unit to cool both trains of Class IE electrical equipment. The licensee has entered this issue into their corrective action program as Condition Report CR73410.

  The team determined the failure to promptly identify and correct the errors in Calculation GK06W and to have adequate compensatory measures in place as required by the calculation was a performance deficiency. This finding was more than minor because it adversely affected the Equipment Performance attribute of the Mitigating Systems Cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, without having an adequate calculation and compensatory measures, the licensee would not be assured that one vital air conditioning unit would be capable of cooling both trains of Class IE electrical equipment. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, the inspectors determined the finding was of very low safety significance (Green), because the finding was not a design deficiency and did not result in the loss of operability or functionality. The finding had a cross-cutting aspect in the area of Human Performance, Resources component, because the licensee failed to ensure that personnel, equipment, procedures, and other resources are available and adequate to assure nuclear safety. Specifically, those resources necessary to provide complete, accurate, and up-to-date design documentations, and equipment are available and adequate to assure nuclear safety. [H.2.(c)]
  (Section 4OA3)

Cornerstone: Barrier Integrity

- Green. The team identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Specifically, on June 23, 2010, the licensee failed to verify that Calculation A-06-W meet all of the criteria identified in the Updated Safety Analysis Report, Section 8.1.4.3. The team determined that the criteria identified in the Updated Safety Analysis Report was not met for several circuits, where the vertical intercept of the magnetic only circuit breaker time-current curve overlaps the penetration conductor damage curve. This indicates that, for a sustained short circuit of a certain magnitude, the thermal limit of the conductor passing through a penetration could be exceeded without tripping of the magnetic-only circuit breaker. The licensee has entered this issue into their corrective action program as Condition Report CR-73124.

The team determined that the licensee's failure to ensure that containment penetrations are properly sized to meet the Updated Safety Analysis Report, Section 8.1.4.3, requirements was a performance deficiency. This finding was more than minor because it was associated with the Configuration Control attribute of the Reactor Safety, Barrier Integrity Cornerstone and adversely affected the cornerstone objective to ensure that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the thermal limit of the penetration conductor could be exceeded without tripping the magnetic-only circuit breaker, jeopardizing the integrity of the electrical penetration. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 3, "Barrier Integrity Screening Questions," the finding was determined to have very low safety significance (Green), because it did not result in an actual open pathway in containment and did not involve hydrogen igniters. This finding did not have a cross-cutting aspect because the most significant contributor to the performance deficiency did not reflect current licensee performance. (Section 1R21.2.8.2)

- Green. The team identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Specifically, on August 28, 2013, the team identified that the licensee had failed to have adequate controls in place to ensure that the bulk average containment temperature would not exceed the Technical Specification limit and design basis limit of 120°F. The licensee did not have: 1) a calculation addressing containment temperature indication uncertainty, 2) there was a lack of temperature sensor and associated circuitry uncertainty, 3) and there was no calculation or justification addressing potential temperature stratification in containment. The licensee has entered these issues into their corrective action program as Condition Reports CR-72639, CR-73118, and CR-73152.

  The team determined that the failure to account for instrument uncertainty on the containment bulk average temperature instrumentation used to determine containment operability was a performance deficiency. This finding was more than minor because it was associated with the Design Control attribute of the Reactor Safety, Barrier Integrity

Cornerstone, and adversely affected the cornerstone objective to ensure that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, by not accounting for the temperature measurement accuracy and stratification, the containment temperature could unknowingly exceed the Technical Specification operability limit. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 3, “Barrier Integrity Screening Questions,” the finding was determined to have a very low safety significance (Green), because it did not result in an actual open pathway in containment and did not involve hydrogen igniters. Operability Evaluation OE GN-13-006 evaluated the containment temperature concerns and concluded that the containment would be operable, but degraded or nonconforming. This finding did not have a cross-cutting aspect because the most significant contributor to the performance deficiency did not reflect current licensee performance.
(Section 1R21.2.11)

REPORT DETAILS

1 REACTOR SAFETY

Inspection of component design basis verifies the initial design and subsequent modifications and provides monitoring of the capability of the selected components and operator actions to perform their design basis functions. As plants age, their design basis 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 Basis Inspection (71111.21)

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

.1 Inspection Scope

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

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

The team performed a margin assessment and detailed review of the selected risk-significant components to verify that the design basis have been correctly implemented and maintained. This design margin assessment considered original design issues, margin reductions because of modifications, and margin reductions identified as a result of material condition issues. Equipment reliability issues were also considered in the selection of components for detailed review. These included items such as failed performance test results; significant corrective actions; repeated maintenance;

10 CFR 50.65(a)1 status; operable, but degraded conditions; NRC resident inspector input of problem equipment; system health reports; industry operating experience; and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense in-depth margins.

The inspection procedure requires a review of 15 to 25 total samples that include risk-significant and low design margin components, containment related components, and operating experience issues. The sample selection for this inspection was seventeen components, two of which were associated with containment; eight operating experience items; and four event based activities associated with the components. The selected inspection and associated operating experience items supported risk significant functions including the following:

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

- New Auxiliary Feed Pump and Diesel Generator Unit
- Emergency Diesel Generator Sequencers
- NK 11 Battery
- 120 Vac System InverterNN11
- Startup Transformer XMR01
- 4.16 kV Bus NB001
- Emergency Diesel Generator NE001 (Electrical aspects)

b. Components that affect large early release frequency (LERF): The team reviewed components required to perform functions that mitigate or prevent an unmonitored release of radiation. As such, the team selected the following components:

- Electrical Containment Penetration Thermal Capabilities
- Containment Temperature Instrumentation

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

- Residual Heat Removal Pump PEJ01B
- Containment Spray Pump PEN01B
- Containment Coolers
- Component Cooling Water Butterfly Valve EGHV102
- New Diesel Fire Pump 1FPPB
- Residual Heat Removal System Isolation Valve EJ 8702A
- Essential Cooler (SGK05B)
- Essential Service Water Pump PEF01AA

.2 Results of Detailed Reviews for Components

.2.1 New Auxiliary Feed Pump and Supporting Diesel Generator

a. Inspection Scope

The team reviewed the system description, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the new auxiliary feed pump and supporting diesel generator. The team also performed walkdowns and conducted interviews with system engineering personnel to verify the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Electrical design basis and engineering change package documents.
- Proposed component maintenance activities and corrective action program reports to verify the monitoring of potential degradation.
- Emergency response procedure guidance and operating conditions to assess feasibility of operator actions.
- Physical walkdown of diesel generator unit and pump/motor unit to verify environmental and physical condition of the equipment.

b. Findings

No findings of significance were identified.

.2.2 Load Shed and Emergency Load Sequencer (LSELS)

a. Inspection Scope

The team reviewed the Updated Safety Analysis Report (USAR), design basis documents, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with emergency diesel generator sequencers to ensure design basis requirements and specifications were met. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Logic circuit design to verify load shed and load sequencing signals are in accordance with design.
- Preventive Maintenance activities for the distribution bus and circuit breakers were verified to maintain the system according to manufacturer recommendations.

- Separation criteria to ensure the direct current (dc) bus meets required separation criteria between Class 1E and Non-class 1E loads.

- Emergency diesel generator procedures to determine operability while paralleled to offsite power.

b. Findings

Failure to follow procedure when making changes to Off Normal Operating Procedure OFN NB-042

Introduction. The team identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because the licensee failed to follow Procedure AP 15C-004, Revision 32, "Preparation, Review and Approval of Procedures, Instructions and Forms," when making changes to safety-related emergency diesel generator operating procedures for dedicated operator instructions. Specifically, the licensee's technical reviewer failed to identify the power supply to the communication equipment for the dedicated operator was from a non-essential power supply, and would be lost during a loss of offsite power event for which the dedicated operator is credited.

Description. The emergency diesel generators are periodically connected to offsite power when required to supply electrical load for periodic surveillance testing or post-maintenance testing. When paralleled to offsite power the speed and voltage regulators are operating in the "Droop" mode of operation, and certain non-essential trips are inserted in the circuitry for diesel and generator protection during testing. When the generator receives an emergency start signal from the safety buses NB01 and NB02, these non-essential trips are bypassed and the speed and voltage regulators are switched to emergency (isochronous) mode. When operating in parallel with offsite power during testing (droop mode), a grid disturbance or loss of offsite power could cause a diesel generator lockout, or overload condition of the generator, or prevent the generator speed and voltage regulators from switching to the emergency mode of operation because the generator maintains the emergency bus energized, preventing the bus undervoltage relays from transferring the regulators to the emergency mode.

To address this concern the licensee created Procedure OFN NB-042, "Loss of Offsite Power to NB01 (NB02) With Emergency Diesel Generator Paralleled." The procedure provided detailed instructions for dedicated operators to place the generator in the emergency mode upon a loss of offsite power. The dedicated operator is required to be in continuous communication with the control room to receive direction to take the required procedural actions of OFN NB-042. While making procedure changes to the off-normal procedure, the technical reviewer failed to follow Procedure AP 15C-004, "Preparation, Review and Approval of Procedures, Instructions and Forms," Revision 32, Attachment E, Step E.1.2. This step requires a technical reviewer to ensure a new procedure is adequate for its purpose, is accurate, and meets the requirements for usability. Specifically, Step E.1.2 states "Is there a high level of assurance that the procedure will work to guide the procedure user in managing the condition for which it was written?" The technical reviewer failed to identify that the power supply for the dedicated operator communications was powered from a non-safety related supply. This non-safety related power supply would be lost during a loss of offsite power which is the plant transient that the dedicated operator is credited, resulting in the loss of communications between the dedicated operator and the control room.

The team determined that the technical reviewer failed to follow Procedure AP 15C-004, “Preparation, Review and Approval of Procedures, Instructions and Forms,” Revision 32, in that he failed to ensure that the new procedure was adequate for its purpose, was accurate, and met the requirements for usability. The licensee initiated essential reading for the operating crews, and to direct the crews to declare the emergency diesel generators inoperable while in parallel with offsite power until the issue is resolved. The licensee entered the issue into their corrective action program as Condition Report CR-72711.

Analysis. The team determined that the failure to follow Procedure AP 15C-004 when making changes to off normal operating Procedure OFN NB-042 was a performance deficiency. This finding was more than minor because it was associated with the Equipment Performance attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to perform a technical walk-down of the procedure steps to verify the power supply for the communication equipment would not be lost during a loss of power event. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, the inspectors determined the finding was of very low safety significance (Green), because the finding was not a design deficiency and did not result in the loss of operability or functionality. This finding did not have a cross-cutting aspect because the most significant contributor to the performance deficiency did not reflect current licensee performance.

Enforcement. The team identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” which states, in part, “Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.” Contrary to the above, the licensee failed to ensure activities affecting quality were prescribed by documented instructions, procedures, or drawings, and failed to accomplish the activities in accordance with these instructions, procedures, or drawings. Specifically, in 2007 the licensee failed to follow Procedure AP 15C-004, “Preparation, Review and Approval of Procedures, Instructions and Forms,” when making changes to safety-related emergency diesel generator surveillance testing Procedure OFN NB-042. The technical reviewer failed to identify that the power supply for the communication equipment for the dedicated operator was from non-essential power, and would be lost during a loss of offsite power event, losing the communications between the control room and the operator. This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy, because it was of very low safety significance (Green), and was entered into the licensee’s corrective action program as Condition Report CR-72711. (NCV 05000482/2013008-01, “Failure to Follow Procedure When Making Changes to Off-Normal Operating Procedure OFN NB-042.”)

.2.3 Safety Related Battery NK011

a. Inspection Scope

The team reviewed the USAR, design basis documents, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the NK 11 Battery to ensure design basis requirements and specifications were met. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Circuit breaker short circuit calculations, sizing calculations, coordination studies, voltage drop calculations, and circuit breaker maintenance activities.

- Duty cycle and capacity testing procedures, pilot cell selection criteria, and vendor technical manual to ensure the battery is maintained in accordance with industry standards and vendor recommendations.

- Preventive Maintenance activities for the distribution bus and circuit breakers were verified to maintain the system according to manufacturer recommendations.

- Battery and battery rack installation drawings to verify the component is installed in accordance with vendor drawings.

b. Findings

No findings of significance were identified.

.2.4 120 VAC System Inverter NN11 and Transformer XNN05

a. Inspection Scope

The team reviewed the USAR, design basis documents, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the 120 Volts alternating current (Vac) System Inverter NN11 and Transformer XNN05 sequencers to ensure design basis requirements and specifications were met. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Circuit breaker short circuit calculations, sizing calculations, coordination studies, voltage drop calculations, and circuit breaker maintenance activities were appropriate for the design of the system.

- Input and output operating voltage characteristics to verify the inverter and transformer can perform their design function through all input voltage ranges of offsite power.

- Performance history for past three years for the inverter.

- Technical specifications and bases documents to verify the licensee is appropriately applying limiting condition for operation allowed outage times.

b. Findings

No findings of significance were identified.

.2.5 Startup Transformer XMR01

a. Inspection Scope

The team reviewed the USAR, design basis documents, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with Startup Transformer XMR01 to ensure design basis requirements and specifications were met. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Voltage calculations and operating procedures to determine whether transformer taps and administrative controls for switchyard voltage were adequate to assure the availability of offsite power during accident conditions.

- Loading calculations to determine whether the capacity of the transformer is adequate to supply worst-case accident loads.

- Corrective action histories to determine whether there had been any adverse operating trends.

- Visual inspection to assess material condition, the presence of hazards, and consistency of installed equipment with design documentation and analyses.

b. Findings

Failure to Verify or Check the Adequacy of Design Calculations

Introduction. The team identified a Green, non-cited Violation, with three examples, of 10 CFR 50, Appendix B, Criterion III, “Design Control.” Specifically, on September 12, 2011, the licensee failed to verify or check the adequacy of design Calculation XX-E-006, “AC System Analysis,” Revision 6, by 1) not recognizing that the actual switchyard voltage could be lower than the calculated minimum voltage due to loop uncertainties of the switchyard voltmeters, 2) failing to provide a comparison between postulated loading levels and equipment ratings for distribution equipment, in order to verify that overloading conditions would not occur, and 3) not placing limits on the voltages on the Class 1E 480 Vac system which could exceed the allowable maximum equipment voltage rating of 506 Vac.

Description. The team was reviewing design Calculation XX-E-006, "AC System Analysis," Revision 6, when the team identified several concerns with the calculation, and with the validation of certain criteria specified in the calculation.

Example 1. The analyzed minimum switchyard voltage of 97 percent is the same as the allowable measured value, without consideration for voltmeter loop uncertainties. Under normal operating conditions, allowable minimum switchyard voltage is 97 percent of 345 kV. Voltmeters are used by the transmission grid operator to monitor real-time levels. Westar Energy, Inc., Transmission Operations Procedure 0400, "Wolf Creek 345 kV Bus Voltage," dated March 30, 2010, directs the transmission grid operator to notify the station when the voltage is below an as-indicated level of 97 percent. Procedure OFN AF-025, "Unit Limitations," requires that, upon receiving this notification, the offsite circuits be evaluated for operability. However, due to voltmeter loop uncertainties, the actual voltage may be lower than the as-indicated level. These uncertainties have not been taken into account in the analysis of the effect of minimum switchyard voltage on plant equipment. In Calculation XX-E-006, Section 5.0, various steady-state modeling cases are described that use a minimum switchyard voltage of 97 percent without consideration for the instrument uncertainties. The licensee has initiated Condition Report CR-73244 with recommended actions to quantify the loop uncertainties and adjust the allowable voltage values accordingly.

Example 2. Wolf Creek USAR, Section 8.2.1, states that transformers and cables associated with the offsite power system "have been sized to carry their anticipated loads continuously." However, there is no documented analysis in Calculation XX-E-006 or elsewhere that compares calculated loading levels for various scenarios with equipment capabilities. In manual WCQPM, "Wolf Creek Quality Program Manual," Revision 9, the licensee has committed to American National Standards Institute (ANSI) N45.2.11-1974, "Quality Assurance Requirements for the Design of Nuclear Power Plants." ANSI N45.2.11 states, "Measures shall be applied to verify the adequacy of design. Design verification is the process of reviewing, confirming, or substantiating the design by one or more methods to provide assurance that the design meets the specified design inputs…. The results of design verification efforts shall be clearly documented." Contrary to this requirement, verification of the load-carrying capability of the alternating current distribution system equipment was not documented. The licensee has initiated Condition Report CR-73240 with a recommended action to revise Calculation XX-E-006 to include a comparison of calculated loading levels with equipment ratings for the electrical distribution equipment.

Example 3. Calculation XX-E-006 analyzes maximum alternating current system voltages to ensure that damaging over-voltages will not occur. In 2011, when Calculation XX-E-006, Revision 6, was completed, it documented an acceptance criterion of ≤ 506 Volts for buses on the 480 Vac systems. At the maximum allowable switchyard voltage of 105 percent of 345 kV and light loading, the calculation concluded that bus voltages could reach as high as 523 Volts, but the calculation did not adequately justify why this condition was considered acceptable. The high switchyard voltage condition would occur only when the main generator is not synchronized to the transmission network and capable of regulating switchyard voltage. At the time Calculation XX-E-006 was issued in 2011, the licensee failed to enter this potential over-voltage condition into their corrective action program, provide a justification for the differing voltages identified in the calculation, or to establish adequate procedural controls to maintain voltage within the 506 Vac limit. Although the voltage is checked twice daily during operator rounds, no provision was made for monitoring the voltage during the rest of the day, instrument uncertainties of the voltmeters were not documented nor taken into account, and there was no procedure to address the required actions when the limit is exceeded. To date, no actual over-voltage condition had been identified. The licensee has initiated Condition Report CR-73206 with a recommended action to perform a study that would determine a solution to the over-voltage concerns.

Analysis. The team determined that the licensee's failure to verify or check the adequacy of design Calculation XX E 006, "AC System Analysis," Revision 6, was a performance deficiency. This finding was more than minor because it was associated with the Design Control attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to verify or check the adequacy of design Calculation XX-E-006, "AC System Analysis," Revision 6 regarding loop uncertainties of the switchyard voltmeters, equipment loading, and maximum allowed Class 1E 480 voltage. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, the inspectors determined the finding was of very low safety significance (Green), because the finding was not a design deficiency and did not result in the loss of operability or functionality. This finding had a cross-cutting aspect in the area of Human Performance, associated with the Resources component because the licensee failed to ensure that personnel, equipment, procedures, and other resources are adequate to assure nuclear safety by maintaining long term plant safety by maintenance of design margins. [H.2(a)]

Enforcement. The team identified a Green, non-cited violation, with three examples, of 10 CFR 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Contrary to the above, the licensee failed to verify or check the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Specifically, on September 12, 2011, the licensee failed to verify or check the adequacy of design Calculation XX-E-006, "AC System Analysis," Revision 6, by 1) not recognizing that the actual switchyard voltage could be lower than the calculated minimum voltage due to loop uncertainties of the switchyard voltmeters, 2) failing to provide a comparison between postulated loading levels and equipment ratings for distribution equipment, in order to verify that overloading conditions would not occur, and 3) not placing limits on the voltages on the Class 1E 480 Vac system which could exceed the allowable maximum equipment voltage rating of 506 Vac. This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy, because it was of very low safety significance (Green), and was entered into the licensee's corrective action program as Condition Reports CR-73244, CR-73240, and CR-73206. (NCV 05000482/2013008-02, "Failure to Verify or Check the Adequacy of Design Calculations.")

.2.6 4.16 KV Bus NB001

a. Inspection Scope

The team reviewed the USAR, design basis documents, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with 4.16 kV Bus NB001 to ensure design basis requirements and specifications were met. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Calculations for electrical distribution system loading, steady-state and transient voltages, and maximum short-circuit levels.
- Protective device settings and circuit breaker ratings to confirm adequate selective protection and coordination of connected equipment during worst-case short circuit conditions.
- Degraded voltage and loss of voltage relay protection schemes that initiate automatic transfers from the offsite power supply to the diesel generator.
- Corrective action histories to determine whether there had been any adverse operating trends.
- Visual inspection to assess material condition, the presence of hazards, and consistency of installed equipment with design documentation and analyses.

b. Findings

Failure to Ensure that Degraded Voltage Relay Minimum Allowable Time Delay Value is Bounded by Analyzed Value

Introduction. The team identified a Green, non-cited Violation of 10 CFR 50, Appendix B, Criterion III, "Design Control." The licensee failed to recognize that the specified minimum allowable value of 7.0 seconds and maximum allowable value of 9.0 seconds, identified in Surveillance Test Procedures STS IC-805A, "Channel Calibration of NB01 Grid Degraded Voltage, Time Delay Trip," and STS IC-805B, "Channel Calibration of NB02 Grid Degraded Voltage, Time Delay Trip," were outside of the degraded voltage relay minimum analyzed values of 7.5 seconds and 8.5 seconds, identified in Calculation XX-E-009, “System NB, NG, PG Undervoltage/Degraded Voltage Relay Setpoints,” Revision 1.

Description. Surveillance Test Procedures STS IC-805A, “Channel Calibration of NB01 Grid Degraded Voltage, Time Delay Trip,” and STS IC-805B, “Channel Calibration of NB02 Grid Degraded Voltage, Time Delay Trip,” allow a minimum time delay of 7.0 seconds for the degraded voltage relays timeout period during accident conditions. However, Calculation XX-E-006, “AC System Analysis,” Revision 6, uses Calculation XX-E-009 as an input reference, which specifies the minimum delay time of 7.5 seconds. The design consideration is that voltage dips, resulting from automatic motor starting during a Loss of Cooling Accident (LOCA), recover prior to the relay timeout period in order to prevent spurious actuation of the degraded voltage relays and consequential loss of offsite power events. Vulnerability to consequential losses of offsite power is prohibited by General Design Criterion 17 which states: “Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit….” Wolf Creek USAR, Section 8.1.4.1.1, states: “The loss of the nuclear unit… will not result in the loss of offsite power to the Class IE busses.” This requirement would not be met if the degraded voltage time delay was not long enough to ride through transient voltage dips caused by automatic motor starting during LOCA events. Calculation XX-E-006 states, “In all cases the steady state voltage on NB01 and NB02 recovered within the 7.5 seconds accident criteria. However, in some cases the recovery time is marginal.” Based on this statement, it is indeterminate whether the design requirement would have been met if the time delay was at the 7.0 seconds minimum allowable value in the surveillance test procedures. However, the licensee provided calibration data that indicates that the as-found time delay values during the last three years have all been longer than 7.5 seconds. The licensee has initiated Condition Report CR-72496 with a recommended action to revise the minimum allowable time delay value in the surveillance test procedures such that it is bounded by the value analyzed in Calculations XX-E-006 and XX-E-009.

Analysis. The team determined that the licensee’s failure to ensure that the analyzed minimum allowable degraded voltage relay time delay of 7.5 seconds, and maximum allowable degraded voltage relay time delay of 8.5 seconds, was incorporated into acceptance criteria for surveillance testing procedures was a performance deficiency. This finding was more than minor because it was associated with the Procedure Quality attribute of the Reactor Safety, Initiating Events Cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, it was indeterminate whether the design requirement to prevent spurious actuation of the degraded voltage relays and consequential loss of offsite power would have been met if the time delay had been set at less than 7.5 seconds or greater than 8.5 seconds. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 1, “Initiating Events Screening Questions,” the finding was determined to have very low safety significance (Green), because it did not cause a reactor trip and loss of mitigation equipment. This finding did not have a cross-cutting aspect because the most significant contributor to the performance deficiency did not reflect current licensee performance.

Enforcement. The team identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Contrary to the above, the licensee failed to ensure measures were established to assure that applicable regulatory requirements were correctly translated into procedures and instructions. Specifically, on May 9, 2003, Calculation XX-E-009, "System NB, NG, PG Undervoltage/Degraded Voltage Relay Setpoints," Revision 1, identified that the degraded voltage relays minimum time delay was 7.5 seconds, and the maximum time delay was 8.5 seconds. During testing of the degraded voltage relays, the calculation states, "In all cases the steady state voltage on NB01 and NB02 recovered within the 7.5 seconds accident criteria. However in some cases the recovery time is marginal." This requirement was not correctly translated into Surveillance Test Procedures STS IC-805A and STS IC-805B which allow a minimum time delay of 7.0 seconds, and a maximum time delay of 9.0 seconds for the degraded voltage relays timeout period during accident conditions. This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy, because it was of very low safety significance (Green), and was entered into the licensee's corrective action program as Condition Report CR-72496. (NCV 05000482/2013008-03, "Failure to Ensure that Degraded Voltage Relay Minimum Allowable Time Delay Value is Bounded by Analyzed Value.")

.2.7 Emergency Diesel Generator NE001

a. Inspection Scope

The team reviewed the USAR, design basis documents, current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the electrical aspects of emergency diesel generator NE001 to ensure design basis requirements and specifications were met. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Loading calculations to determine whether the capacity of the emergency diesel generator is adequate to supply worst case accident loads.

- Voltage and frequency calculations and operating procedures to determine whether steady-state limits are adequate to assure the adequacy of power to load equipment.

- Performance of the replacement governor to determine whether it adequately controls frequency during transient motor starting scenarios.

- Corrective action histories to determine whether there had been any adverse operating trends.

- Visual inspection to assess material condition, the presence of hazards, and consistency of installed equipment with design documentation and analyses.

b. Findings

1. Failure to Prevent Over-voltages on the 480 Vac System During Emergency Diesel Generator Testing

Introduction. The team identified a Green, non-cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," which states, in part, "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected." Specifically, in 2006 the licensee failed to implement voltage monitoring for the 480 Vac 'A' Train system to ensure that overvoltages would not occur during emergency diesel generator testing, as recommended in Condition Report CR-2006-2062, "Potential Over-voltages on Equipment During Diesel Test."

Description. During emergency diesel generator testing, with the diesel generator paralleled to offsite power, voltage levels downstream of 480 Volts bus NG001 could exceed the equipment ratings, and the procedures currently in place do not require monitoring to ensure that the limits are not exceeded. In 2006, the licensee recognized this vulnerability and initiated Condition Report CR-2006-2062. However, the corrective actions only addressed the 'B' Train emergency diesel generator and failed to address the 'A' Train emergency diesel generator, which has the same vulnerability. In order to provide adequate voltage to load equipment during low voltage conditions, certain transformers between the 4160 Volt and 480 Volt systems have their taps set to boost the downstream voltage, such that the ratio of downstream voltage to upstream voltage is 480/4000. This condition applies to transformer XNG01 on 'A' Train emergency diesel generator, which feeds load center NG001 and downstream motor control centers. The buses downstream of these transformers are susceptible to overvoltage conditions when the upstream voltage is high. The corrective actions of the 2006 condition report implemented voltage monitoring for 'B' Train load center NG002, but failed to address the 'A' Train emergency diesel generator load center NG001. The licensee has initiated condition report CR-73209 with a recommended action to revise surveillance test Procedure STS KJ-001A, "Integrated D/G and Safeguards Actuation Test – Train A," Revision 50, to require voltage monitoring, in order to ensure that voltages on the 480 Vac system are maintained within acceptable limits for load center NG001, and its downstream motor control centers.

Analysis. The team determined that the licensee's failure to implement corrective actions into diesel testing Procedure STS KJ-001A was a performance deficiency. This finding was more than minor because it was associated with the Equipment Performance attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to ensure that over-voltages would not occur during the testing of the "A" train emergency diesel generator. In accordance with

NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, the inspectors determined the finding was of very low safety significance (Green), because the finding was not a design deficiency and did not result in the loss of operability or functionality. This finding did not have a cross-cutting aspect because the most significant contributor to the performance deficiency did not reflect current licensee performance.

Enforcement. The team identified a Green, non-cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," which states, in part, "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected." Contrary to the above, the licensee failed to establish measures to assure that conditions adverse to quality were corrected. Specifically, in 2006, the licensee implemented corrective actions per Condition Report 2006-2062, to monitor the voltages for the 480 Vac system to ensure that over-voltages would not occur during emergency diesel generator testing. The licensee implemented voltage monitoring for the "B" Train 480 Vac system, but failed to monitor voltages of "A" Train, which had the same vulnerability. This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy, because it was of very low safety significance (Green), and was entered into the licensee's corrective action program as Condition Report CR-73209. (NCV 05000482/2013008-04, "Failure to Prevent Over-Voltages on the 480 Vac System During Emergency Diesel Generator Testing.")

2. Failure to Ensure Motors are Operated Within their Thermal Limits

Introduction. The team identified a Green, non-cited Violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," which states, in part, "The design control measures shall provide for verifying or checking the adequacy of design… ." Specifically, on June 26, 2013, the licensee issued drawing E-11005, "List of Loads Supplied by Emergency Diesel Generator," Revision 39, that identified certain motors with load brake horsepower in excess of the motor nameplate ratings, but failed to verify that this would not result in the motors exceeding their thermal design limits.

Description. In the Wolf Creek USAR, Section 8.3.1.1.8, there is a discussion regarding motor loading, which was intended to confirm that margin exists to prevent overheating of certain motors in the plant. Contrary to this objective, the USAR also states: "For additional components with brake horsepower exceeding the nameplate rating of the motor at pump runout conditions, refer to Figure 8.3-2 [drawing E-11005, "List of Loads Supplied by Emergency Diesel Generators"]." The referenced drawing does not analyze or justify the overload condition, nor is there any other analysis that justifies this condition. Additionally, the brake horsepower values on the referenced drawing do not reflect the worst-case condition, which would occur when the diesel generator is operating at maximum allowable frequency and powering the motors. In manual WCQPM, "Wolf Creek Quality Program Manual," Revision 9, the licensee is committed to ANSI N45.2.11-1974, "Quality Assurance Requirements for the Design of Nuclear Power Plants." ANSI N45.2.11 states, "Measures shall be applied to verify the adequacy of design. Design verification is the process of reviewing, confirming, or substantiating the design by one or more methods to provide assurance that the design meets the specified design inputs…. The results of design verification efforts shall be clearly documented." Contrary to this requirement, verification of the thermal capability of motors serving loads in excess of the motor horsepower rating had not been documented. The licensee initiated Condition Report CR-72945 with recommended actions to formally document the operating conditions for the motors and revise the USAR to clarify the wording.

Analysis. The team determined that the licensee's failure to evaluate motor loading to confirm margin exists to prevent overheating of the motors was a performance deficiency. This finding was more than minor because it was associated with the Design Control attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, motors serving loads with demands in excess of the motor horsepower ratings were not analyzed to ensure that overheating would not occur. In accordance with Inspection Manual Chapter 0609 Appendix A, Exhibit 2, the inspectors determined the finding was of very low safety significance (Green), because the finding was not a design deficiency and did not result in the loss of operability or functionality. This finding had a cross-cutting aspect in the area of Human Performance, associated with the Resources component, because the licensee failed to ensure that personnel, equipment, procedures, and other resources are adequate to assure nuclear safety by maintaining long term plant safety by maintenance of design margins. [H.2(a)]

Enforcement. The team identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Contrary to the above, the licensee failed to ensure measures were established to assure that applicable regulatory requirements were correctly translated into procedures and instructions. Specifically, on June 26, 2013, the licensee issued drawing E-11005, "List of Loads Supplied by Emergency Diesel Generator," Revision 39, that identified certain motors with load brake horsepower in excess of the motor nameplate ratings, but failed to verify that the excess horsepower would not result in the motors exceeding their thermal design limits. Additionally, the brake horsepower values on the referenced drawing do not reflect the worst-case condition, which would occur when the diesel generator is operating at maximum allowable frequency and powering the motors. This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy, because it was of very low safety significance (Green), and was entered into the licensee's corrective action program as Condition Report CR-72945. (NCV 05000482/2013008-05, "Failure to Ensure Motors are Operated Within Their Thermal Limits.")

.2.8 Motor Control Center NG01A

a. Inspection Scope

The team reviewed the USAR, design basis documents, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with Motor Control Center NG01A to ensure design basis requirements and specifications were met. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Calculations for electrical distribution system loading, minimum and maximum voltages, and maximum short-circuit levels.

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

- Corrective action histories to determine whether there had been any adverse operating trends.

- Visual inspection to assess material condition, the presence of hazards, and consistency of installed equipment with design documentation and analyses.

b. Findings

1. Failure to Verify or Check the Adequacy of Design

The team identified a Green, non-cited violation, with three examples, of 10 CFR 50, Appendix B, Criterion III, “Design Control.” The failure to prevent over-voltages on the 480 Vac system during high switchyard voltage conditions was Example 3 of the violation which is located in Section 1R21.2.5 of this report.

2. Failure to Fully Implement Electrical Protection Criteria for Containment Penetrations

Introduction. The team identified a Green, non-cited Violation of 10 CFR 50, Appendix B, Criterion III, “Design Control,” which states, in part, “The design control measures shall provide for verifying or checking the adequacy of design… .” Specifically, on June 23, 2010, the licensee failed to verify or check that Calculation A-06-W, “Thermal Capability of Electrical Penetration Assemblies (EPA) Versus Dual Short Circuit Protection to satisfy Regulatory Guide 1.63,” Revision 6, complied with the criteria found in Section 8.1.4.3 of the Wolf Creek USAR. This section states that electrical containment penetrations for motor-operated valves with thermal overloads bypassed “are sized such that their thermal limits are above… the vertical intercept of the magnetic-only circuit breakers.” Upon review of the calculation, it was determined that some of the penetration thermal limits intersect the vertical intercepts of the associated magnetic-only circuit breaker time-current curves.

Description. In Section 8.1.4.3 of the Wolf Creek USAR, which discusses containment penetration protection for motor-operated valve circuits with bypassed thermal overloads, it states, in part: "the penetrations are sized such that their thermal limits are above the… vertical intercept of the magnetic only circuit breakers." Penetration protection is analyzed in Calculation A-06-W, but this calculation failed to demonstrate that the USAR criterion was met. Upon further inspection, it was determined that the criterion was not met for several circuits. For these circuits, the vertical intercept of the magnetic only circuit breaker time-current curve overlaps the penetration conductor damage curve. This indicates that, for a sustained short circuit of a certain magnitude, the thermal limit of the penetration conductor could be exceeded without tripping of the magnetic-only circuit breaker. Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants," Revision 2, to which the licensee is committed, states, "The electric penetration assembly should be designed to withstand, without loss of mechanical integrity, the maximum short-circuit current vs. time conditions that could occur given single random failures of circuit overload protection devices." The licensee has implemented this requirement by providing two overload protection devices for each penetration conductor that could be exposed to damaging overcurrents. The deficiency involves a small range of currents, for one of the two devices, for which the required level of protection would not be assured. The licensee initiated Condition Report CR-73124 with recommended actions to update calculation A-06-W and revise USAR, Section 8.1.4.3, to address this condition.

Analysis. The team determined that the licensee's failure to ensure that containment penetrations are properly sized to meet the USAR, Section 8.1.4.3, requirements was a performance deficiency. This finding was more than minor because it was associated with the Configuration Control attribute of the Reactor Safety, Barrier Integrity Cornerstone and adversely affected the cornerstone objective to ensure that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the thermal limit of the penetration conductor could be exceeded without tripping the magnetic-only circuit breaker, jeopardizing the integrity of the electrical penetration. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 3, "Barrier Integrity Screening Questions," the finding was determined to have very low safety significance (Green), because it did not result in an actual open pathway in containment and did not involve hydrogen igniters. This finding did not have a cross-cutting aspect because the most significant contributor to the performance deficiency did not reflect current licensee performance.

Enforcement. The team identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Contrary to the above, the licensee failed to provide design control measures for verifying or checking the adequacy of design. Specifically, on June 23, 2010, the licensee failed to verify that Calculation A-06-W met all of the criteria identified in the USAR, Section 8.1.4.3. The team determined that the criteria identified in the USAR was not met for several circuits, where the vertical intercept of the magnetic-only circuit breaker time-current curve overlaps the penetration conductor damage curve. This indicates that, for a sustained short circuit of a certain magnitude, the thermal limit of the conductor passing through a penetration could be exceeded without tripping of the magnetic-only circuit breaker. This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy, because it was of very low safety significance (Green) and was entered into the licensee's corrective action program as Condition Report CR-73124. (NCV 05000482/2013008-06, "Failure to Fully Implement Electrical Protection Criteria for Containment Penetrations.")

.2.9 Residual Heat Removal Pump PEJ01B

a. Inspection Scope

The team reviewed the USAR, design basis documents, selected drawings, and condition reports associated with Residual Heat Removal Pump PEJ01B, to ensure design basis requirements and specifications were met. The team also performed walkdowns and conducted interviews with system and design engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Piping and instrumentation diagrams.
- Inservice Testing (IST) Quarterly and Comprehensive pump performance surveillance test results.
- System operating instructions.
- Specifications of flow instrumentation used in both surveillances and low flow pump operation.
- Technical specifications and associated bases document.

b. Findings

Failure to Account for Flow Measurement Uncertainty when Operating the Residual Heat Removal Pumps in the Low Flow Regime

Introduction. The team identified a Green, non-cited Violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," which states, in part, "The design control measures shall provide for verifying or checking the adequacy of design… ." Specifically, on August 27, 2013, the licensee failed to factor flow measurement uncertainty into two plant evolutions, (a) Technical Specification low flow verification and (b) alarm procedures for pump protection during low flow operation.

<u>Description</u>. During refueling operations (Mode 6), Technical Specification 3.9.5 requires that the residual heat removal pumps circulate reactor coolant flow above 1,000 gallons per minute (gpm). This flow ensures adequate heat removal and also prevents thermal and boron stratification in the reactor core. The corresponding Technical Specification Surveillance SR 3.9.5.1 states that the flow rate must be greater than or equal to 1000 gpm. Procedure STS CR-002, "Shift Logs for Modes 4, 5, and 6" are used to record and demonstrate verification of this requirement. Step A.13 of STS CR-002 records the flow and any value greater than 1000 gpm would be judged acceptable. The procedure does not address or provide any guidance or requirements regarding flow measurement uncertainty. Without incorporating any flow measurement uncertainty into the procedure or requirements, it would be possible operate the residual heat removal pumps at an actual flow rate less than what is required by technical specifications or procedures.

Also, Information Bulletin 88-04, ""Potential Safety-Related Pump Loss" notified licensees of a mechanism which could potentially lead to pump damage. In response to the bulletin, licensees were required to contact their pump vendors to obtain direction on how to avoid and protect against conditions that could damage the pumps. For the residual heat removal pumps, the vendor recommendations, as provided in alarm Procedure ALR 00-049C, "RHR LOOP 1 FLOW LOW", were as follows:

- Pump flow less than 500 gpm is prohibited.

- Residual Heat Removal Pump "A" should not be operated with not less than or equal to 1700 gpm for greater than 2.25 hours at any one time and is limited to less than or equal to 100 hours per month.

These requirements also applied to Residual Heat Removal Pump "B". Without consideration of flow measurement uncertainty, pump operation below the technical specification limit of 1000 gpm or alarm procedure limit of 1700 gpm could occur, without the operator's knowledge, which would then violate compliance with Technical Specification 3.9.5 or alarm Procedure ALR 00-049C.

The team concluded that the licensee had not accounted for flow measurement uncertainties in the residual heat removal system; therefore, they did not have adequate controls in place to ensure that, (a) when operating in Mode 6, the circulating residual heat removal flow would be greater than or equal to 1000 gpm, and (b) when operating the residual heat removal pumps at low flows that the flow must be at or above 1700 gpm for pump protection. Condition Report CR-73071 documents the concern for flow measurement uncertainty when performing flow surveillance SR 3.9.5.1. Condition Report CR-73231 documents the need to evaluate flow uncertainty when operating the residual heat removal pumps in the low flow region. The recommended action for these condition reports is to provide an evaluation of the measurement uncertainty.

<u>Analysis</u>. The team determined that the failure to account for flow measurement uncertainties when operating Residual Heat Removal pumps at low flows was a performance deficiency. This finding was more than minor because it was associated with the Procedure Quality attribute of the Reactor Safety, Mitigating Systems

Cornerstone, and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to account for flow measurement uncertainties in the residual heat removal system could allow operation below technical specification and alarm response limits and potentially damage the residual heat removal pumps. In accordance with NRC Inspection Manual Chapter 0609, Appendix G, “Shutdown Operations Significance Determination Process,” the finding was determined to have very low safety significance (Green), because the finding did not require a quantitative assessment because adequate mitigating equipment remained available and the finding did not constitute a loss of control as defined in Appendix G. This finding did not have a cross-cutting aspect because the most significant contributor to the performance deficiency did not reflect current licensee performance.

Enforcement. The team identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion III, “Design Control,” which states, in part, “Measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.” Contrary to the above, the licensee failed to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Specifically, on August 27, 2013, the team identified that the licensee had failed to account for flow measurement uncertainties of the Residual Heat Removal System. Technical Specifications require that when operating in Mode 6, the circulating residual heat removal flow is required to be greater than or equal to 1000 gpm to for adequate heat removal and to prevent stratification, and Alarm Response Procedure ALR 00-049C, “RHR LOOP 1 FLOW LOW” requires that when operating the residual heat removal pumps at low flows that the flow must be at or above 1700 gpm for pump protection. The failure to account for flow measurement uncertainties could allow flow to actually be below the required technical specification and alarm response limits, without the operator’s knowledge. This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy, because it was of very low safety significance (Green), and was entered into the licensee’s corrective action program as Condition Reports CR-73071 and CR-73231. (NCV 05000482/2013008-07, “Failure to Account for Flow Measurement Uncertainty when Operating the Residual Heat Removal Pumps in the Low Flow Regime.”)

.2.10 Containment Spray Pump PEN01B

a. Inspection Scope

The team reviewed the USAR, design basis documents, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with Containment Spray Pump PEN01B, to ensure design basis requirements and specifications were met. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Corrective action program documents and system health reports.

- Piping and instrumentation diagrams.

- IST Quarterly and Comprehensive pump performance surveillance test results.

- Flow instrumentation documentation, specifically the correlation of valve opening to flow used in the IST tests.

- Work orders and corrective action program documents.

b. Findings

No findings of significance were identified.

.2.11 Containment Temperature Instrumentation

a. Inspection Scope

The team reviewed portions of the USAR, design basis documents, selected drawings, maintenance and test procedures, and condition reports associated with Containment Temperature Instrumentation, to ensure design basis requirements and specifications were met. The team also conducted interviews with system engineering personnel to assess the licensee's capability to perform its desired design basis function. Specifically, the team reviewed:

- Original design specifications and vendor documentation for the containment Resistance Temperature Detectors.

- Containment elevation and locations of the containment Resistance Temperature Detectors.

- Calibration and functional procedures of the containment Resistance Temperature Detectors and associated process circuitry.

- Station compliance with Regulatory Guide 1.97, Revision 2.

- Surveillance procedures, containment temperature monitoring.

b. Findings

Failure to Account for Containment Temperature Measurement Uncertainties

Introduction. The team identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion III, “Design Control”, for the licensee’s failure to account for instrument uncertainty when determining the containment bulk temperature. Technical Specification 3.6.5 requires that containment temperature be maintained below 120°F to ensure compliance with the licensee’s safety-analysis which, without allowances for instrument uncertainty, was not assured.

Description. The team reviewed the bases for Technical Specification 3.6.5, which states, in part, “The limiting design basis accident for the maximum peak containment air temperature is a steam line break. The initial containment average air temperature assumed in the design basis analysis is 120°F.” The Updated Safety Analysis Report, Table 6.2.1-5, “Containment and Reactor Coolant System Initial Conditions for Containment Analysis” also lists the containment atmosphere temperature at 120°F. With both the technical specification limit and the design basis calculations equal to the same value, 120° F, there is no margin provided between analysis and normal operating conditions. Further, Operations Procedure STS CR-001, “Shift Logs for Modes 1, 2, and 3,” which requires operations to record the containment average temperature on a daily basis, does not provide any guidance or allowance for temperature instrument uncertainty. The licensee’s controlled documents do not specify any tolerance values associated with instrument measurement uncertainty.

The team concluded that the licensee did not have adequate controls in place to ensure that the bulk average containment temperature would not exceed the technical specification limit of 120°F. Several condition reports were entered into the licensee’s corrective action program to address this concern including (a) Condition Report CR-73152 which identified that Procedure STN IC-213, “Channel Calibration Containment Temperatures,” at Step 8.6.3 required verification that the difference between the four containment temperature indications differ by no more than 6.9°F, which did not include or account for any instrument tolerance specified, (b) Condition Report CR-72639 identified that there was no calculation addressing containment temperature indication uncertainty, and (c) Condition Report CR-73118, which not only addressed the lack of temperature sensor and associated circuitry uncertainty, but additionally addressed the likelihood of temperature stratification in the containment. The four temperature sensors used to determine the global bulk average containment temperature are all located at the intake of the containment air coolers at a height of only 40 percent of the maximum interior containment height. In response to Condition Report CR-73118, Operability Evaluation OE GN-13-006 was performed to evaluate these concerns. As part of the operability evaluation, preliminary calculations were performed which accounted for both potential containment temperature stratification and temperature sensor indication uncertainties. The conclusion of the operability evaluation was that the subject structures, systems, and components (containment temperature sensors and Reactor building), were “Operable but Degraded or Nonconforming” and that reasonable assurance exists that the structures, systems, and components associated with the operability evaluation remained capable of performing their safety functions.

Analysis. The team determined that the failure to account for instrument uncertainty on the containment bulk average temperature instrumentation, used to determine containment operability, was a performance deficiency. This finding was more than minor because it was associated with the Design Control attribute of the Reactor Safety, Barrier Integrity Cornerstone and adversely affected the cornerstone objective to ensure that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, by not accounting for the temperature measurement accuracy and stratification, the containment temperature could unknowingly exceed the Technical Specification operability limit. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 3, "Barrier Integrity Screening Questions," the finding was determined to have very low safety significance (Green), because it did not result in an actual open pathway in containment and did not involve hydrogen igniters. Operability Evaluation OE GN-13-006 evaluated the containment temperature concerns and concluded that the containment would be operable, but degraded or nonconforming. This finding did not have a cross-cutting aspect because the most significant contributor to the performance deficiency did not reflect current licensee performance.

Enforcement. The team identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Contrary to the above, the licensee failed to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Specifically, on August 28, 2013, the team identified that the licensee had failed to have adequate controls in place to ensure that the bulk average containment temperature would not exceed the Technical Specification limit and design basis limit of 120°F. The licensee did not have: 1) a calculation addressing containment temperature indication uncertainty, 2) there was a lack of temperature sensor and associated circuitry uncertainty, 3) and there was no calculation or justification addressing potential temperature stratification in containment. This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy, because it was of very low safety significance (Green), and was entered into the licensee's corrective action program as Condition Reports CR-72639, CR-73118, and CR-73152.
(NCV 05000482/2013008-08, "Failure to Account for Containment Temperature Measurement Uncertainty").

.2.12 Non-Safety Related Auxiliary Feedwater Pump PAP01

a. Inspection Scope

The team reviewed portions of DCP-014189, "Non-Safety Related Auxiliary Feedwater Pump PAP01." This pump was designed and installed at Wolf Creek to improve the

Mitigating Systems Performance Index (MSPI) for the Auxiliary Feedwater System. The team reviewed the USAR, design basis documents, selected drawings, maintenance and test procedures, and condition reports associated with Non-Safety Relate Auxiliary Feedwater Pump PAP01, to ensure design basis requirements and specifications were met. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- USAR, Rev.25 for Non-Safety Relate Auxiliary Feedwater Pump PAP01.
- Piping and instrumentation diagrams.
- System design criteria.
- Calculations demonstrating the capability of non-safety related auxiliary feedwater pump to satisfy design basis accidents; these included PAP01 pump curves adjusted for eventual degradation coupled with system hydraulic resistance.
- Initial pump startup testing documentation.
- System operating instructions.

b. Findings

No findings of significance were identified.

.2.13 Containment Coolers

a. Inspection Scope

The team reviewed the design basis documents, the current system health report, selected drawings, and condition reports associated with the containment coolers, to ensure design basis requirements and specifications were met. The team also conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- System design criteria provided in the USAR.
- Calculations which provide system performance requirements.
- Piping and Instrumentation diagrams.
- System performance surveillance tests.
- Corrective action program documents and system health reports.

- Corrective action program reports to verify the monitoring and correction of potential degradation, operability evaluations and Root/Apparent Cause evaluations.

b. Findings

No findings were identified.

.2.14 Component Cooling Water Butterfly Valve EGHV0102

a. Inspection Scope

The team reviewed the USAR, design basis documents, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with Component Cooling Water Butterfly Valve EGHV102, to ensure design basis requirements and specifications were met. The team also performed walk downs and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Calculations.
- Corrective action program documents and system health reports.
- Corrective action program reports to verify the monitoring and correction of potential degradation, operability evaluations and Root/Apparent Cause evaluations.
- System design criteria and system health reports.
- Piping and instrumentation diagrams.
- System operating instructions.
- System functional tests.
- Technical specifications and bases document.
- Vendor documentation.
- Work orders and corrective action program documents.

b. Findings

1. Failure to Properly Assess Problems with Component Cooling Water Valve EGHV102

Introduction. The team identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action”, involving the licensee’s failure to promptly assess and correct a condition adverse to quality. Specifically, the licensee failed to assess the cause of the loose valve disc, and broken groove pin in the high risk, safety-related, Component Cooling Water Butterfly Valve EGHV102.

Description. The Component Cooling Water Butterfly Valve, EGHV0102, is an 18-inch Fisher butterfly valve. This valve is a normally closed motor-operated valve that must open to provide a flow path from the component cooling water system to the residual heat removal system heat exchanger. When the valve is opened, it provides component cooling water flow to the residual heat removal heat exchanger to perform its function of cooling containment sump water during the recirculation phase of safety injection following a loss of coolant accident (USAR 6.3.2). This valve opens automatically upon receipt of a safety injection signal (SIS) coincident with a low refueling water storage tank level signal. Additionally, this valve is opened/throttled to support the residual heat removal function of decay heat removal after the reactor coolant system temperature and pressure have been reduced during normal operating conditions (USAR 9.2.2). This valve must close to isolate the component cooling water system from the residual heat removal heat exchanger. In the event of a residual heat removal heat exchanger tube failure during operation of the heat exchanger in the reactor coolant system cooling mode, this valve would be closed to isolate the reactor coolant system (USAR 9.2.2).

According to calculations EGM-0-23 and EGM-0-24, the limiting component on this valve is the groove pin. Therefore, the groove pin would be the first part in the valve to break if the valve were over-torqued.

In January 2011, the licensee identified leakage past the seat in the Train ‘B’ Component Cooling Water Valve EGHV0102. The licensee entered this into their corrective action program as Condition Report CR-32813. In April, 2011, the licensee commenced work on Valve EGHV0102 to address the leakage, when they identified that the valve disc to shaft connection was loose. Condition Report CR-37825 was initiated to address the loose disc to shaft. The licensee elected to replace the valve, which was completed on May 2, 2012.

The old, leaking valve had been in storage for over a year and half when the licensee took it out of storage for refurbishment. On November 26, 2012, during disassembly of the valve, the groove pins that hold the valve disc to the valve shaft, were found broken in several pieces. The pin pieces had to be drilled and die tapped to remove them from the connection by the mechanics. The groove pins had not been sheared off, and when discussing the disassembly of the valve with the mechanic, the mechanic stated that he could not rotate the shaft without disc movement and that the connection was loose. Condition Report CR-60208 was initiated to address this concern. The inspector reviewed the surveillance history of this valve, and found that valve had passed all of its surveillance requirements, and that the valve was capable of opening and closing (i.e., stroking). The licensee had reviewed several event reports within the industry, including some NRC information notices, pertaining to broken pins connecting valve discs to shafts. Specifically, the licensee had reviewed NRC Information Notices 2005-2525 and 2008-11, which identified events concerning problems with valve discs.

The inspector reviewed the licensee’s corrective action Procedure AI 28A-010, “Screening Condition Reports.” Attachment A, Section A.2.1, of the procedure states in part “Level 3 Condition Report Evaluation, Apparent Cause Evaluation, should be the evaluation type selected by the Screening Review Team when: 1. the risks of a condition or event were understood and manageable and the consequences were tolerable but clearly undesirable, and we want to learn from the condition or event to improve performance and minimize the likelihood it might happen again." The Screening Review Team failed to categorize Condition Reports CR-37825 and CR-60298 as Level 3 condition reports, mandating an evaluation when: 1) the Component Cooling Water Valve EGHV0102 disc to shaft was found loose, 2.) the failure of the groove pin in the valve when it was disassembled, and 3.) to evaluate the extent of condition for similar valves currently installed in the plant.

Analysis. The team determined the licensee's failure follow the Corrective Action Procedure AI 28A-010, "Screening Condition Reports," which improperly categorized Condition Reports CR-37825 and CR-60298, which should have had apparent cause evaluations performed, was a performance deficiency. This finding was more than minor because it adversely affected the Equipment Performance attribute of the Mitigating Systems Cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to perform an apparent cause evaluation resulted in the licensee not identifying a root cause for the valve leakage, preventing reoccurrence, or investigating the extent of condition for other similar valves installed in the plant. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, the inspectors determined the finding was of very low safety significance (Green), because the finding was not a design deficiency and did not result in the loss of operability or functionality. This finding had a cross-cutting aspect in the area of Human Performance, associated with Work Practices. Specifically the licensee defines and effectively communicates expectations regarding procedural compliance and personnel follow procedures. [H.4(b)]

Enforcement. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," which states, in part, "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, defective materials and equipment, and nonconformance are promptly identified and corrected." Contrary to the above, the licensee failed to promptly identify and correct a condition adverse to quality. Specifically, in April 2011 and November, 2012, the licensee failed to properly categorize Condition Reports CR-37825 and CR-60298 correctly, which resulted in the condition reports not getting an Apparent Cause Evaluation, to promptly identify and correct the cause of the Component Cooling Water Butterfly Valve EGHV0102 loose disc to shaft, failure of the groove pin in the valve, and to investigate the extent of condition for similar valves currently installed in the plant. This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy, because it was of very low safety significance (Green), and was entered into the licensee's corrective action program as Condition Report CR-73227. (NCV 05000482/2013008-09, "Failure to Properly Assess Problems with Component Cooling Water Valve EGHV102").

2. Failure to Provide Procedure Instructions to Remove Thermal Overload Bypass Jumpers

Introduction. The team identified a Green, non-cited Violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements are correctly translated into specification, drawings, procedures, and instructions." Specifically, in 1994 the licensee was committed to the requirements specified in Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves," Revision 1, to remove the thermal overload bypass jumpers. The licensee failed to translate the requirements into motor-operated valve diagnostic testing Procedure MGE LT-099, "MOV Diagnostic Testing." Also, the Wolf Creek USAR, Section 8.3.1.1.2, has incomplete information which does not support Regulatory Guide 1.106, in that it does not state that the jumpers should be removed during testing.

Description. The licensee committed to meeting one or more of the methods described in Regulatory Guide 1.106, Revision 1, which describes three methods acceptable to the NRC for addressing motor-operated valve thermal overload protection schemes. The regulatory position described in the regulatory guide, which the licensee intended to meet, was to continuously bypass thermal overload protection devices and to place the overload devices in force (bypass jumper removed) only when the valve motors are undergoing periodic or maintenance testing. The regulatory guide discusses the function of the overload protective devices which must be placed in force when the valve motors are undergoing periodic testing.

The licensee's current practice is to remove the thermal overload bypass jumpers when performing local motor testing. During performance of Procedure MGE LT-011, "Limitorque Deadman & Functional Testing," the test requires a "deadman" control device to be installed into the control circuitry to allow maintenance technicians to locally stroke the valve. The team identified that maintenance Procedure MGE LT-099, "MOV Diagnostic Testing," Revision 0 through the current Revision 11, does not require the removal of the overload bypass jumpers when stroking the valve from the control room. The Regulatory Guide 1.106, Revision 1, position was that the thermal overloads be placed in force (bypass jumper removed) during any stroking of the motor-operated valve for surveillance testing and routine maintenance, including preventive maintenance and post-maintenance testing, to protect the motor from incremental damage that may be caused during the testing. The licensee has written Condition Report CR-73120 to address the testing procedure issue to restore conformance with the regulatory guide.

The team also identified incomplete USAR wording that does not fully incorporate the regulatory guide position that the licensee is committed to. The USAR, Section 8.3.1.1.2, "Class 1E AC System," states that "Prior to core loading, the thermal overload relay trip contacts for all Class 1E valves are permanently bypassed with jumpers." This USAR section failed to include the regulatory guide requirement that the bypass jumpers will be removed during testing of motor-operated valves. The licensee has written Condition Report CR-73219 to address this concern.

Analysis. The team determined that the licensee's failure to provide procedure instructions to remove the thermal over-load bypass jumpers during motor operated valve diagnostic testing as committed to in Regulatory Guide 1.106, Revision 1, was a performance deficiency. This finding was more than minor because it was associated with the Procedure Quality attribute of the Reactor Safety, Mitigating Systems Cornerstone, and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to include procedural guidance to remove the thermal overload bypass jumpers when performing maintenance testing that strokes the valve from the control room, and to include the requirements of Regulator Guide 1.106 in USAR, Section 8.3.1.1.2, that the bypass jumpers will be removed during testing of the motor-operated valves. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, the inspectors determined the finding was of very low safety significance (Green), because the finding was not a design deficiency and did not result in the loss of operability or functionality. This finding did not have a cross-cutting aspect because the most significant contributor to the performance deficiency did not reflect current licensee performance.

Enforcement. The team identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Contrary to the above, the licensee failed to establish measures to assure that applicable regulatory requirements are correctly translated into specification, drawings, procedures, and instructions. Specifically, in 1994, the licensee was committed to the requirements specified in Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves," Revision 1, to remove the thermal overload bypass jumpers during maintenance and testing. The licensee failed to translate the requirements into Procedure MGE LT-099, "MOV Diagnostic Testing," and failed to include procedural guidance to remove the thermal overload bypass jumpers when performing maintenance testing that strokes the valve from the control room. Also, the Wolf Creek USAR, Section 8.3.1.1.2, has incomplete information which does not support Regulatory Guide 1.106, in that it does not state that the thermal overload bypass jumpers should be removed when performing maintenance testing that strokes the valve. This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy, because it was of very low safety significance (Green), and was entered into the licensee's corrective action program as Condition Reports CR-73120. (NCV 05000482/2013008-10, "Failure to Provide Procedure Instructions to Remove Thermal Overload Bypass Jumpers.")

.2.15 New Diesel Driven Fire Pump 1FP01PB

a. Inspection Scope

The team reviewed the USAR, design basis documents, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the new Diesel Driven Fire Pump 1FP01PB to ensure design basis requirements and specifications were met. The team also performed walk downs and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Calculations.
- Corrective action program documents and system health reports.
- Corrective action program reports to verify the monitoring and correction of potential degradation, operability evaluations and Root/Apparent Cause evaluations.
- System design criteria and system health reports.
- Piping and instrumentation diagrams.
- System operating instructions.
- System functional tests.
- Technical specifications and bases document.
- Vendor documentation.
- Work orders and corrective action program documents.

b. Findings

No findings of significance were identified.

.2.16 Residual Heat Removal Pump Suction Isolation Valve BBPV8702A

a. Inspection Scope

The team reviewed the USAR, design basis documents, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with Residual Heat Removal Pump Suction Isolation Valve BBPV8702A to ensure design basis requirements and specifications were met. The team also performed walk downs and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Calculations.
- Corrective action program documents and system health reports.
- Corrective action program reports to verify the monitoring and correction of potential degradation, operability evaluations and Root/Apparent Cause evaluations.
- System design criteria and system health reports.
- Piping and instrumentation diagrams.
- System operating instructions.
- System quarterly functional tests.
- Technical specifications and bases document.
- Vendor documentation.
- Work orders and corrective action program documents

b. Findings

No findings of significance were identified.

.2.17 Essential Service Water Pump PEF01AA

a. Inspection Scope

The team reviewed the USAR, design basis documents, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with Essential Service Water Pump PEF01AA to ensure design basis requirements and specifications were met. The team also performed walk downs and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Calculations.
- Corrective action program documents and system health reports.
- Corrective action program reports to verify the monitoring and correction of potential degradation, operability evaluations and Root/Apparent Cause evaluations.
- System design criteria and system health reports.
- Piping and instrumentation diagrams.

- System operating instructions.
- System quarterly functional tests.
- Technical specifications and bases document.
- Vendor documentation.
- Work orders and corrective action program documents.

b. Findings

No findings of significance were identified.

.3 Results of Reviews for Operating Experience

.3.1 Inspection of NRC Information Notice 2013-05 "Battery Expected Life and Its Impact on Surveillance Requirements"

a. Inspection Scope

The team reviewed the licensee's evaluation of Information Notice 2013-05, "Battery Expected Life and Its Impact on Surveillance Requirements," to verify the licensee perform an applicability review and took appropriate corrective actions, if appropriate, to address the concerns. The team verified that the licensee's review adequately addressed the issues in the information notice.

b. Findings

No findings of significance were identified.

.3.2 Inspection of NRC Information Notice 2012-11 "Age-Related Capacitor Degradation"

a. Inspection Scope

The team reviewed the licensee's evaluation of Information Notice 2012-11, "Age-Related Capacitor Degradation," to verify the licensee performed an applicability review and took appropriate corrective actions, if appropriate, to address the concerns. The team verified that the licensee's review adequately addressed the issues in the information notice.

b. Findings

No findings of significance were identified.

.3.3 Inspection of NRC Information Notice 1988-45 "Problems in Protective Relay and Circuit Breaker Coordination"

a. Inspection Scope

The team reviewed the licensee's evaluation of Information Notice 1988-45, "Problems in Protective Relay and Circuit Breaker Coordination," to verify that the review adequately addressed the industry operating experiences discussed in the information notice. The information notice discusses events in which electrical faults caused unnecessary de-energization of significant portions of the electrical distribution system due to lack of coordination of the protective devices. It states the following: "Coordination is the selection and/or setting of protective devices so as to sequentially isolate only that portion of the system where the abnormality occurs. To achieve this isolation, it is necessary to set protective devices so that only the device nearest the fault opens and isolates the faulted circuit from the system." The team reviewed the licensee's analyses to determine whether this level of coordination exists.

b. Findings

Failure to Verify Adequacy of Electrical Protective Devices to Isolate Fire-Damaged Associated Circuits

Introduction. The team identified a Green, non-cited Violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," which states, in part, "The design control measures shall provide for verifying or checking the adequacy of design… ." Specifically, on September 29, 2011, when study WCNOC-171, "Post-Fire Safe Shutdown Associated Circuits Study," Revision 0 was completed, the licensee failed to provide documented verification of the adequacy of electrical protective devices for associated circuits such that hot shorts or shorts to ground will not prevent operation of the safe shutdown equipment. Wolf Creek USAR, Appendix 9.5E requires isolation between safe shutdown circuits and associated circuits, such that "hot shorts, open circuits, or shorts to ground in the associated circuits will not prevent operation of the safe shutdown equipment." The licensee failed to perform a comprehensive analysis that compares all of the relevant upstream and downstream protective device time-current curves and maximum short circuit levels to verify proper coordination exists.

Description. The team reviewed Wolf Creek USAR, Appendix 9.5 E, which describes compliance with 10 CFR 50, Appendix R, which requires isolation between safe shutdown circuits and associated circuits, such that "hot shorts, open circuits, or shorts to ground in the associated circuits will not prevent operation of the safe shutdown equipment." The team requested documentation to confirm that the licensee was complying with the USAR requirement. The licensee was unable to provide any comprehensive analysis that compares all of the relevant upstream and downstream protective device time current curves and maximum short circuit levels in order to verify that the required coordination exists. The concern pertains to electrical panels, motor control centers, load centers, and switchgear that feed both safe shutdown and non-safe shutdown components. If the non-safe shutdown (i.e., "associated") circuits are damaged by a fire, analysis should demonstrate that the individual electrical protective device to the damaged circuit shall open to clear the fault, but none of the upstream electrical protective devices shall open and de-energize the safe shutdown circuits. In manual WCQPM, “Wolf Creek Quality Program Manual,” Revision 9, the licensee is committed to the American National Standards Institute (ANSI) N45.2.11-1974, “Quality Assurance Requirements for the Design of Nuclear Power Plants.” ANSI N45.2.11 states, “Measures shall be applied to verify the adequacy of design. Design verification is the process of reviewing, confirming, or substantiating the design by one or more methods to provide assurance that the design meets the specified design inputs. The results of design verification efforts shall be clearly documented.” Contrary to this requirement, verification of the electrical protective device coordination for some of the distribution system equipment that powers safe shutdown components had not been documented. The licensee has initiated Condition Report CR-73242, which identifies that a comprehensive associated circuit electrical protective device coordination analysis does not exist.

Analysis. The team determined that the licensee’s failure to provide a documented comparison of upstream and downstream electrical protective devices with maximum short circuit levels, in order to verify the required coordination, was a performance deficiency. This finding was more than minor because it was associated with the Design Control attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee was unable to provide an analysis to demonstrate that associated shutdown circuits would be isolated from the safe shutdown circuits during fire events. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, the inspectors determined the finding was of very low safety significance (Green), because the finding was not a design deficiency and did not result in the loss of operability or functionality. The finding had a cross-cutting aspect in the area of Human Performance, Resources attribute because the licensee failed to ensure that personnel, equipment, procedures, and other resources are adequate to assure nuclear safety by maintaining long term plant safety by maintenance of design margins. [H.2(a)]

Enforcement. The team identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion III, “Design Control,” which states, in part, “Measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.” Contrary to the above, the licensee failed to provide documented verification of the adequacy of electrical protective devices for associated circuits such that hot shorts or shorts to ground would not prevent operation of the safe shutdown equipment. Specifically, as of September 2011, Wolf Creek USAR, Appendix 9.5 E, required isolation between safe shutdown circuits and non-safe shutdown (associated) circuits, such that “hot shorts, open circuits, or shorts to ground in the associated circuits will not prevent operation of the safe shutdown equipment.” On September 29, 2011, the licensee completed study WCNOC-171, “Post-Fire Safe Shutdown Associated Circuits Study,” Revision 0, but failed to provide documented verification of the adequacy of electrical protective devices for associated shutdown circuits such that hot shorts or shorts to ground will not prevent operation of the safe shutdown equipment. This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy, because it was of very low safety significance (Green), and was entered into the licensee's corrective action program as Condition Report CR-73242. (NCV 05000482/2013008-11, "Failure to Verify Adequacy of Electrical Protective Devices to Isolate Fire-Damaged Associated Circuits.")

.3.4 Inspection of NRC Information Notice 2010-09 "Blown Circuit Breaker Fuses"

a. Inspection Scope

The team reviewed the licensee's evaluation of Information Notice 2010-09, "Importance of Understanding Circuit Breaker Control Power Indications," to verify that the review adequately addressed the industry operating experiences discussed in the information notice. The information notice discusses an event in which a circuit breaker did not open to clear a fault because the breaker did not have control power to its trip circuit. The control panel indicating lights on the front panel of the breaker had not been illuminated for approximately one year, thus signaling an issue with the control circuit. The licensee initiated Condition Report CR-25404 to review their programs and practices with respect to this event and concluded that their testing and maintenance programs would result in the initiation of a more timely repair under similar circumstances. The team verified that the licensee's review adequately addressed the issues in the information notice.

b. Findings

No findings of significance were identified.

.3.5 Inspection of NRC Information Notice 1997-90 "Use of Non-Conservative Acceptance Criteria in Safety Related Surveillance Tests"

a. Inspection Scope

The team reviewed the licensee's evaluation of NRC Information Notice 1997-90, "Use of Non-Conservative Acceptance Criteria in Safety Related Surveillance Tests," to verify that applicable design bases performance requirements had been translated into surveillance tests.

b. Findings

Failure to Translate Design Basis Performance Requirements into Pump Surveillance Tests

Introduction. The team identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion XI, "Test Control," for the licensee's failure to establish a test program which incorporated test requirements and acceptance limits contained in applicable design documents. Specifically, the licensee failed to incorporate minimum safety-related pump performance requirements into the corresponding safety-related pump surveillances. This violation is based on reviews of surveillance test results for two safety-related pumps, the Residual Heat Removal and Containment Spray pumps.

Description. The team reviewed Residual Heat Removal and Containment Spray pump, safety related, quarterly and full flow IST surveillance tests. Reviewing Information Notice 1997-90, the team placed the majority of the focus of the review on the full flow comprehensive tests.

For the Containment Spray Pump, the team determined that the acceptance criteria for the comprehensive pump test STS EN-101B for Containment Spray Pump "PEN01B" could allow the pump performance to degrade below what is assumed in the calculations for containment pressure-temperature response to a loss of coolant accident (LOCA) and steam line break (SLB). Therefore, the acceptance criteria was non-conservative. It was also determined that one of the supporting calculations, EN-32, for the system resistance of the spray piping, used some outdated and non-conservative inputs. Further review of the containment spray system also revealed that the flow through the Containment Spray Pumps PEN01A/B was evaluated based on the number of turns of throttle valves ENV0124/126. The team questioned relationship of the number of turns on the throttle valves ENV0124/126 as it related to the pump linear flow curve. The licensee had no evaluation or justification to correlate the number of turns on throttle valves ENV0124/126 to the amount the valve was open to calculate the flow. The licensee also failed to provide an uncertainty analysis regarding the containment spray pump flow test. The licensee has entered these concerns into their corrective action program as Condition Report CR-73149.

For Residual Heat Removal Pump "B", the licensee was using a general 2.0 percent uncertainty for the full scale reading of the flow instruments, and the team found that the flow element uncertainty had not been included. When combining uncertainties using the square root of the sum of the squares (SRSS), of all the variables included in taking a flow reading, the accuracy comes out to be as high as plus or minus 2.86 percent of the full scale reading of the instrument. The variables that the licensee needed to consider for inclusion of the system flow uncertainties were the flow element, flow transmitter, square root card, and flow indicator. Also, in Residual Heat Removal "A" full flow pump test, STS CV-211, it lists the minimum acceptable pressure (pounds per square inch differential - psid) as 160.2 psid, which should be 167.9 psid, and the maximum acceptable pressure is listed as 183.3 psid and should be 175.7 psid. Additionally, in regards to flow, the currently listed values in the surveillance lists 3600 gpm as the minimum acceptable flow which should be 3516.6 gpm, and the maximum acceptable flow is listed as 3700 gpm, and should be 3783.4 gpm. The surveillance test confirmed that the residual heat removal pump can satisfy its design flow requirement but the currently assigned boundaries are not accurately defined in the surveillance which has the potential to procedurally pass the surveillance test with unacceptable pump performance. The licensee has entered these concerns into their corrective action program as Condition Report CR-73070

Analysis. The team determined that the failure to establish and incorporate adequate acceptance criteria into the Containment Spray and Residual Heat Removal pump comprehensive surveillance tests was a performance deficiency. This finding was more than minor because it was associated with the Procedure Quality attribute of the Reactor Safety, Mitigating Systems Cornerstone, and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to incorporate adequate acceptance criteria and instrument uncertainties into the safety-related surveillances could cause unacceptable pump performance conditions to go undetected. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, the inspectors determined the finding was of very low safety significance (Green), because the finding was not a design deficiency and did not result in the loss of operability or functionality. This finding did not have a cross-cutting aspect because the most significant contributor to the performance deficiency did not reflect current licensee performance.

Enforcement. The team identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion XI, "Test Control," which states, in part, "A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design document." Contrary to the above, the licensee failed to establish a test program that assured that all testing required to demonstrate that structures, systems, and components will perform satisfactorily. Specifically, on August 28, 2013 the team identified that the licensee failed to incorporate minimum pump performance requirements into the corresponding pump surveillances for the Containment Spray and Residual Heat Removal pumps. The acceptance criteria did not adequately overlap with the pump design performance requirements. Further, instrument uncertainty was not adequately evaluated, nor incorporated into the tests. This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy, because it was of very low safety significance (Green), and was entered into the licensee's corrective action program as Condition Reports CR-73149 and CR-73070. (NCV 05000482/2013008-12, "Failure to Translate Design Basis Performance Requirements into Pump Surveillance Tests.")

.3.6 Inspection of NRC Information Notice 2009-04 "Age-Related Constant Support Degradation"

a. Inspection Scope

The team reviewed the licensee's evaluation of Inspection of NRC Information Notice 2009-04 "Age-Related Constant Support Degradation," to verify the licensee perform an applicability review and took appropriate corrective actions, if appropriate, to address the concerns. The team verified that the licensee's review adequately addressed the issues in the information notice.

b. Findings

No findings of significance were identified.

.3.7 Inspection of NRC Information Notice 2010-01 “Pipe Support Anchors Installed Improperly”

a. Inspection Scope

The team reviewed the licensee’s evaluation of Inspection of NRC Inspection of NRC Information Notice 2010-01, “Pipe Support Anchors Installed Improperly,” to verify the licensee performed an applicability review and took appropriate corrective actions, if appropriate, to address the concerns. The team verified that the licensee’s review adequately addressed the issues in the information notice.

b. Findings

No findings of significance were identified.

.3.8 Inspection of NRC Information Notice 1984-69 “Operation of Emergency Diesel Generators”

a. Inspection Scope

The team reviewed the licensee’s evaluation of Inspection of NRC Information Notice 2009-04, “Operation of Emergency Diesel Generators,” to verify that the licensee performed an applicability review and took appropriate corrective actions, if appropriate, to address the concerns. The team conducted interviews with two licensed shift managers, two licensed senior reactor operators, and two licensed reactor operators to assess the training and use of operating experience of the Operations Department with respect to this information notice. The team verified that the licensed operators had sufficient knowledge to adequately address the issues in the information notice. The team verified that the licensee’s review adequately addressed the issues in the information notice.

b. Findings

No findings of significance were identified.

.4 Results of Reviews for Operator Actions

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

a. Inspection Scope

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

The selected operator actions were:

- Diagnose a steam line break event and terminate Auxiliary Feedwater flow to the faulted steam generator following a steam line break outside containment: The team observed licensed operator crews perform a simulator scenario consisting of a large steam line break outside containment in accordance with Procedure E-0, “Reactor Trip or Safety Injection.” including foldout page Item 3, “Faulted S/G Isolation Criteria.”

- Open logic cabinet doors for instrumentation cooling following a loss of all AC: The team observed a licensed operator simulate opening cabinet doors in the control room following a loss of all AC power in accordance with EMG C-0, “Loss of all AC Power.”

- Diagnose that a steam generator tube rupture has occurred and identify the ruptured steam generator: The team observed licensed operator crews perform a simulator scenario consisting of a steam generator tube rupture Operator Action #3.

- Startup the Non-Safety Auxilary Feedwater Pump: The team observed a non-licensed operator perform in-plant job performance measures simulating the startup of the Non-Safety Auxilary Feedwater Pump and diesel. This was performed in accordance with Procedure AP-122, "Non-Safety Auxilary Feedwater Pump.”

b. Findings

No findings of significance were identified.

4 OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

The team reviewed various controlled documents (i.e., procedures, drawings, instructions, calculations, and licensee basis documents), which are discussed in previous sections of this report. The attributes of the licensee’s corrective action program, should include: the complete and accurate identification of the problem; the timely correction, commensurate with the safety significance; the evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent of condition reviews, previous occurrences reviews, and the classification, prioritization, focus, and timeliness of corrective actions.

During the inspection, the team identified multiple examples of document control errors associated with instructions, procedures, drawings, and calculations, including changes to these documents that had not been captured by the licensee's corrective action program. Specifically, on August 26, 2013, the team identified numerous inconsistencies in controlled documents. Collectively, the failure to properly control the accuracy of controlled documents could become a programmatic concern.

The document errors ranged from transposing the identification of a specific train (i.e., "A" train instead of "B" train) to more significant issues (i.e., USAR, Section 8.3.1.1.8, states that the brake horsepower of the centrifugal charging pumps is 600 BHP, with a nameplate rating of 480 BHP, and the current revision of Drawing E-11005, "List of Loads Supplied by Emergency Diesel Generator," Revision 39, indicated that the brake horsepower of centrifugal charging pump was 680 BHP). The team found that none of the documents with the identified errors had been used in response to any events or plant perturbations. This observation could become significant, because if left uncorrected, there is a potential that the errors could lead to a more significant safety concern.

4OA3 Followup of Events and Notices of Enforcement Discretion (71153)

Failure to fully identify and assess for Class 1E Vital Air Conditioning Unit: URI 2012004-01

a. Inspection Scope

From Inspection Report 05000482/2012004, the inspectors had identified an unresolved item (URI) involving the licensing basis and cooling capability of the safety-related air conditioning units and the ability to cool both trains of safety-related switchgear, batteries, battery chargers, and inverters with a single train of cooling. The above concerns needed to be addressed before a decision of the combined effect of these concerns could be made. Pending further evaluation of the above issues by the licensee, the issue was tracked as Unresolved Item (URI) 05000482/2012004-01, "Determine Licensing Basis and Capability of One Vital Air Conditioning Unit to Cool Both Trains of Class IE Electrical Equipment."

The team performed a follow-up inspection on URI 05000482/2012004-01. In preparation to address the URI, the team reviewed the USAR, design basis documents, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the Vital Essential Chiller SGK05 A/B Air Conditioning Units, to ensure design basis specification requirements were met. The team also performed walk downs and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Calculations.
- Corrective action program documents and system health reports.

- Corrective action program reports to verify the monitoring and correction of potential degradation, operability evaluations and Root/Apparent Cause evaluations.

- System design criteria and system health reports.

- Piping and instrumentation diagrams.

- System operating instructions.

- System quarterly functional tests.

- Technical specifications and bases document.

- Vendor documentation.

- Work orders and corrective action program documents.

The team concluded that the licensee had satisfactorily responded to the concerns identified in the URI. Therefore the URI is being closed. During the inspection follow-up of the URI, the team identified a performance deficiency with respect of the inadequate corrective actions associated with calculations and compensatory measures.

b. Findings

Failure to Fully Establish Design Control Measures for Vital Essential Chiller SGK05 A/B Air Conditioning Units

Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action", involving inadequate calculations and condition reports supporting one Vital Essential Chiller SGK05 A/B Air Conditioning Units to cool both trains of Class 1E Electrical Equipment. Specifically, the licensee failed to establish measures to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. The licensee failed to correct calculations and assess the adequacy of compensatory measures to be put into place.

Description. Wolf Creek is designed with two vital switchgear air conditioning units. Each air conditioning unit cools one vital switchgear room, DC battery rooms, and vital DC switchgear. On August 3, 2010, Wolf Creek experienced a trip of the Train "A" vital chiller air conditioning unit, SGK05A. Wolf Creek entered Technical Requirements Manual, "Limiting Conditions of Operation" 3.7.23. The licensee used Calculation GK06W, Revision 1, to support continued operation with only one vital chiller air conditioning unit operational. The calculation identified the vital switchgear room temperatures with only one vital switchgear cooler operable. During May 2011, the NRC inspectors found inadequate assumptions and heat loads used in Calculation GK-06W, such that one vital switchgear chiller would not provide sufficiently cooling to the specific areas the chillers are designed to cool. The licensee entered this issue into their corrective actions program as Condition Reports CR-27276, CR-28252, and CR-31452.

As part of the licensee's corrective actions, Calculation GK06W was revised. The inspectors noted discrepancies in the final room temperature assumptions in Revision 1 of Calculation GK06W. The licensee revised Calculation GK06W twice, Revisions 2 and 3. Between May 29, 2012, and August 30, 2012, the inspectors identified several more concerns with Calculation GK06 W, Revision 3, and problems associated with the compensatory measures the licensee had put in place to support operation with only one vital chiller air conditioning unit available. The licensee was not using safety-related power to temporary fans used to increase air movement in strategic locations and had to deal with oil degradation in the vital chiller air conditioning unit compressors. The licensee issued several more condition reports, including Condition Reports CR-53672, CR-53710, and CR-55265.

During August 2013, the inspection team reviewed the licensee's efforts in responding to the URI. The licensee had contracted with a vender to completely revise the calculations used to support the operation of the vital chiller SGK05 A/B air conditioning units. These included new Calculations GKM-11, 12, 13, 14, 15, and GKM06, Revision 4. The calculations support the use of one vital chiller air conditioning unit to provide sufficient cooling to the required spaces, based on the use of certain compensatory measures (i.e., the use of safety-related fans, which considered seismic concerns, and safety-related power to support operation of the temporary fans), and the proper placement of the fans.

Analysis. The team determined the failure to promptly identify and correct the errors in Calculation GK06W and to have adequate compensatory measures in place as required by the calculation was a performance deficiency. This finding was more than minor because it adversely affected the Equipment Performance attribute of the Mitigating Systems Cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, without having an adequate calculation and compensatory measures, the licensee would not be assured that one vital air conditioning unit would be capable of cooling both trains of Class IE electrical equipment. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, the inspectors determined the finding was of very low safety significance (Green), because the finding was not a design deficiency and did not result in the loss of operability or functionality. The finding had a cross-cutting aspect in the area of Human Performance, Resources component, because the licensee failed to ensure that personnel, equipment, procedures, and other resources are available and adequate to assure nuclear safety. Specifically, those resources necessary to provide complete, accurate, and up-to-date design documentations, and equipment are available and adequate to assure nuclear safety. [H.2.(c)]

Enforcement. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action", which states, in part, "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, defective materials and equipment, and nonconformances, are promptly identified and corrected." Contrary to the above, the licensee failed to assure that conditions adverse to quality, such as failures, deficiencies, and equipment, and nonconformances, are promptly identified and corrected. Specifically, since May 2011, the licensee had numerous opportunities, but failed to adequately correct Calculation GK06W and to adequately assess compensatory actions identified to supplement weaknesses in the calculations for operation of one vital air conditioning unit to cool both trains of Class IE electrical equipment. This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy, because it was of very low safety significance (Green), and was entered into the licensee's corrective action program as Condition Report CR-73410. (NCV 05000482/2013008-13, "Failure to Fully Establish Design Control Measures for Vital Essential Chiller SGK05 A/B Air Conditioning Units.")

4OA6 Meetings, Including Exit

On August 29, 2013, the team leader presented the preliminary inspection results to Mr. R. Smith, Site Vice President and Chief Nuclear Operations Officer, and other members of the licensee's staff. After additional in-office inspection, a final telephonic exit meeting was conducted on October 28, 2013, with Mr. J. Broschak, Vice President, Engineering, and other members of your staff. The licensee acknowledged the findings during each meeting. While some proprietary information was reviewed during this inspection, no proprietary information was included in this report.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel
D. Alford, Engineer, PRA
T. Baban, Manager, Systems Engineering
B. Berland, Civil Engineer, Design Engineering
M. Blow, Manager, Operations
P. Bradateanu, Electrical Engineer, Design Engineering
R. Braddy, Manager, Design Engineering
J. Broschak, Vice President, Engineering
W. Camp, Operations
B. Carlson, Mechanical Engineer, Design Engineering
R. Clemens, Vice President, Strategic Projects
L. Comfort, Civil Engineer, Design Engineering
G. Curten, Engineer, Design Engineering
D. Dandreo, Project Manager, Design Engineering
J. Ernest, Mechanical Engineer, Design Engineering
K. Fredrichson, Licensing
J. Fritton, Owners Representative
R. Hobby III, Licensing, Regulatory Affairs
J. Harris, Residual Heat Removal System Engineer, System Engineering
S. Henry, Manager, Operations
P. Herrman, Manager, Programs Engineering
V. Kanal, Engineer, Design Engineering
J. Keim, Engineering Supervisor, Engineering Programs
R. Kelly, Inservice Testing Coordinator, Design Engineering
W. Ketchum, PRA
T. Jensen, Manager, Chemistry
S. Leonard, Corrective Action Contractor, Quality
D. Long, Mechanical Engineer, Design Engineering
D. Mand, Manager, Design Engineering
E. NcIntire, Manager, Human Resources
L. McNabb, Electrical Engineer, Design Engineering
D. Meredith, Mechanical Engineer, Design Engineering
W. Muilenburg, Licensing Supervisor, Regulatory Affairs
G. Pendergrass, Manager, Engineering Performance
L. Ratzlaff, Manager, Maintenance
E. Ray, Manager, Training
R. Reitman, Containment Spray System Engineer, Systems Engineering
L. Rockers, Licensing, Regulatory Affairs
L. Sawyer, Supervisor Corrective Actions, Quality
R. Smith, Site Vice President and Chief Nuclear Operations Officer
A. Steuve, Containment Instrumentation System Engineer, Systems Engineering
P. Wagner, Engineer, Program Engineering
M. Westman, Manager, Regulatory Affairs
B. Williams, Electrical Engineer, Design Engineering
C. Williams, Electrical Engineer, Design Engineering
S. Yunk, Operations.

<u>NRC personnel</u>
C. Peabody, Senior Resident Inspector
R. Stroble, Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

<u>Opened and Closed</u>

05000482/2013008-01	NCV	Failure to Follow Procedure When Making Changes to Off-Normal Operating Procedure OFN NB-042 (Section 1R21.2.2)
05000482/2013008-02	NCV	Failure to Verify or Check the Adequacy of Design Calculations (Section 1R21.2.5 and 1R21.2.8.1)
05000482/2013008-03	NCV	Failure to Ensure that Degraded Voltage Relay Minimum Allowable Time Delay Value is Bounded by Analyzed Value (Section 1R21.2.6)
05000482/2013008-04	NCV	Failure to Prevent Over voltages on the 480 Vac System During Emergency Diesel Generator Testing (Section 1R21.2.7.1)
05000482/2013008-05	NCV	Failure to Ensure Motors are Operated Within their Thermal Limits (Section 1R21.2.7.2)
05000482/2013008-06	NCV	Failure to Fully Implement Electrical Protection Criteria for Containment Penetrations (Section 1R21.2.8.2)
05000482/2013008-07	NCV	Failure to Account for Flow Measurement Uncertainty when Operating the Residual Heat Removal Pumps in the Low Flow Regime (Section 1R21.2.9)
05000482/2013008-08	NCV	Failure to Account for Containment Temperature Measurement Uncertainty (Section 1R21.2.11)
05000482/2013008-09	NCV	Failure to Properly Assess Problems with Component Cooling Water Valve EGHV102 (Section 1R21.2.14.1)
05000482/2013008-10	NCV	Failure to Provide Procedure Instructions to Remove Thermal Overload Bypass Jumpers (Section 1R21.2.14.2)
05000482/2013008-11	NCV	Failure to Verify Adequacy of Electrical Protective Devices to Isolate Fire-Damaged Associated Circuits (Section 1R21.3.3)
05000482/2013008-12	NCV	Failure to Translate Design Basis Performance Requirements into Pump Surveillance Tests (Section 1R21.3.5)
05000482/2013008-13	NCV	Failure to Fully Establish Design Control Measures for Vital Essential Chillers SGK05 A/B Air Conditioning Units. (Section 4OA4)

LIST OF DOCUMENTS REVIEWED

Calculations

NUMBER	TITLE	REVISION
06227-TR-01	Containment Fan Cooler Response to a Simultaneous LOCA and LOOP Event	4
25360-000-MOC-AN-00001	Emergency Diesel Generator Frequency Variation Impact on Motor Operated Mechanical Equipment Performance	0
69446-M-003	Non-Safety Auxiliary Feedwater Pump Analysis and System Resistance Curve	1
A-06-W	Thermal capability of electrical penetration assemblies (EPA) versus dual short circuit protection to satisfy Reg. Guide 1.63	6
AN-05-016	Wolf Creek MSLB Containment Pressure and Temperature Response Analysis with GOTHIC for the MSIV/MFIV Replacement Project	0
E-11005A	Emergency Diesel Generator Loading Data	3
EF-10	Essential Service Water System Flow Requirements	1
EF-10W	Essential Service Water Flows at 90F-Normal Mode	1
EF-35	ESW Pump Head Requirement	2
E-G-M-23	Determination of Structural Weak Link Capacity of Motor Operated Butterfly valve	1
E-H-8	System NB Protective Relays	5
E-J-M-005	EJHV8701A/B Motor Operated Valve Boundary Conditions Determination	4
E-J-M-006	Motor Operated Valves Limitorque Set up requirements for Valves	11
E-J-M-24	Required Torque Calculation for EGHV Valves	1
EN-32	Containment Spray System Head Curves, (Updated with CN001)	0

Calculations

NUMBER	TITLE	REVISION
EN-M-020	Containment Spray Pumps A/B – Full Flow Test Line Operation – Hydraulic & Thermal Analysis	0
GK-06-W	SGK05A/B Class 1E Electrical Equipment Rooms A/C Units, Single Unit Operation Capability	4
GK-361	Control Building HVAC Nodal Points for the Flow Diagram	1
G-K-M-012	Cooling and Heating Load for Control Building Class 1E Electrical Equipment Areas during Normal Conditions-Train B	0
G-K-M-013	Accident Condition for Room Temperature in Control Building Rooms Not Shared By safety Related Cooling	0
G-K-M-015	Cooling and Heating Load for Control Building Class 1E Electrical Equipment Areas during Accident Conditions-Train B	0
H-10	System NE Relay Settings	7
H-11	System MR Protective Relay Settings	2
K-20-02.1-F	ESW Pump Anchor Bolts	0
K-C-0911-1	Support # K-EF-11-R001-091	2
NG-E-004	Class 1E 120 VAC Power Distribution System	1
NK-E-001	125 VDC Class 1E Battery System Sizing, Voltage Drop and Short Circuit Studies	003
NK-E-001, CN004	125 VDC Class 1E Battery System Sizing, Voltage Drop and Short Circuit Studies	003
NN-E-001	Class 1E NN Inverter Loading	00
NN-E-002	Class 1E 120 Vac Instrument Power distribution system (NN system) voltage drop and short circuit protection	0
SA-91-016	The Impact of the Diesel Generator Degraded Frequency on the Performance of the ECCS Pumps	0 (CN003)

Calculations

NUMBER	TITLE	REVISION
WIP-E-11035-000-A-1	Relay Setting Tabulation and Coordination Curve System AP	01
XX-E-004	AC Motor Operated Valve Minimum Terminal Voltage	13
XX-E-004	AC Motor Operated Valve Minimum Terminal Voltage	13
XX-E-006	AC System Analysis	6
XX-E-009	System NB, NG, PG Undervoltage/ Degraded Voltage Relay Setpoints	1
XX-E-009-001-CN003	Calculation Change Notice: System NB, NG, PG Undervoltage/ Degraded Voltage Relay Setpoints	1
XX-E-012	Safety-Related MCC Control Circuit Allowable Wire Lengths	3

Procedures

NUMBER	TITLE	REVISION/DATE
	Westar Energy NUC-001-2 R8 Coordination	NUC-001 R8
0101	Westar Energy, Inc. Transmission Operation Procedure: Division of Responsibility of the Wolf Creek Substation	May 2, 2013
0303	Westar Energy, Inc. Transmission Operating Directive: Planned Outage of the Wolf Creek No. 7 Transformer	October 24, 2012
0305	Westar Energy, Inc. Transmission Operating Directive: Loss of Power to the Wolf Creek Generating Station	October 5, 2007
0306	Westar Energy, Inc. Transmission Operating Directive: Outage of Wolf Creek 345 kV Breaker 345-	December 19, 2011

Procedures

NUMBER	TITLE	REVISION/DATE
	40 or 345-60	
0400	Westar Energy, Inc. Transmission Operations Procedure: Wolf Creek 345 kV Bus Voltage	March 30, 2010
0414	Westar Energy, Inc. Transmission Operations Procedure: Monitoring Wolf Creek Contingency Study 345 kV Bus Voltage	June 20, 2013
23M-050	Monitoring Performance to Criticality and Goals	3
AI 14-006	Severe Weather	13A
AI 14-006	Severe Weather	9A
AI 21-016	Operator Time Critical Actions and Validation	8
AI 23D-0003	MOV Trending ad Verification Programs	2
AI 28A-010	Screening Condition Reports	16
AI 28A-010	Screening Condition Reports	15
ALR 00-049C	RHR LOOP 1 Flow LO	13A
AP 05D-001	Calculations	15
AP 05D-001	Calculations	13
AP 15C-004	Preparation, Review and Approval of Procedures, Instructions and Forms	32
AP 21-004	Time Critical Action Program	3C
AP 21C-001	Wolf Creek Substation	14
AP 28A-100	Condition Reports	12
AP 28A-100	Condition Report	20A

Procedures

NUMBER	TITLE	REVISION/DATE
AP 28A-100	Condition Reports	14, 16
AP 29B-002	ASME Code Testing of Pumps & Valves	10
AP-122	Non-Safety Auxilary Feedwater Pump	1
AP-21G-001	Control of Locked Component Status	60
BD-EMG-C-12	Loca Outside Containment	4A
EMG C-0	Loss of All AC Power	31
EMG C-0	Loss of All AC Power	31A
EMG E-0	Reactor Trip or Safety Injection	26
EMG E-0	Reactor Trip or SI	31
EMG E-1	Loss of Reactor or Secondary Coolant	21
EMG E-1	Loss of Reactor or Secondary Coolant	21
EMG ES-03	SI Termination	21A
EMG ES-11	Post LOCA Cooldown and Depressurization	20
EMG ES-11	Post LOCA Cooldown and Depressurization	20
EMG FR-H1	Response to Loss of Secondary Heat Sink	29A
EMG-C-12	Loca Outside Containment	15
FP-209	Fire Pump Performance	20
GEN 00-004	Power Operation	72
GEN 00-008	RCS Level Less Than Reactor Vessel Flange Operations	24

Procedures

NUMBER	TITLE	REVISION/DATE
INC C-1000	Calibration of Miscellaneous Components	7A
M-10EF	System Description ESW Pump	9
M-10GK	System Description, Control Building Ventilation System	5
M-10KC	Fire Protection System	5
MGE EOOP-11	Molded Case Circuit Breaker and Ground Fault Sensor Testing	29B
MGE LT-011	Limitorque Deadman & Functional Testing	5
MGE LT-099	MOV Diagnostic Testing	11
MGE LT-099	MOV Diagnostic Testing	0
MPE GK-003	Control Room and Class 1E AC Units Preventive Maintenance Activity	4
MPE NE-003	Governor Adjustments For Emergency Diesel Generator NE01	11
MPE-E009Q-01	13.8 kV and 4.16 kV Switchgear Inspection and Testing	30
OFN AF-025	Unit Limitations	39
OFN BB-031	Shutdown Loca	26
OFN EJ-015	Loss of RHR Cooling	21
OFN KC-216	Fire Response	38
OFN NB-030	Loss of AC Emergency Bus NB01 (NB02)	30
OFN NB-034	Loss of AC Power – Shutdown Conditions	22
OFN NB-042	Loss of Offsite Power to NB01 (NB02) With EDG Paralleled	7

Procedures

NUMBER	TITLE	REVISION/DATE
OFN NB-34	Loss of All AC Power – Shutdown Conditions	22
OFN SG-003	Natural Events	25
OFN SG-003	Natural Events	18
STN AP-101	NSAFP Recirc Test	3
STN AP-101	NSAFP Recirc Test	3
STN AP-103	NSAFP Post Installation/Overhaul Testing	2A
STN AP-103	NSAFP Post Installation/Overhaul Testing	2A
STN AP-104	NSAFP Full Flow Test for R19	0
STN AP-104	NSAFP Full Flow Test for R19	0
STN EF-100A	ESW Pump A Reference Pump Curve Determination	20
STN FP-211	Diesel Fire Pump 1FP01PB Monthly Operation and Fuel Level Check	31
STN FP-211	Diesel Fire Pump Monthly Operation and Fuel Level Check	20
STN IC-213	Channel Calibration of Containment Temperatures	5A
STN IC-245	Calibration of RHR/SIS Hot Leg Recirc Flow Loop	3A
STN IC-252B	Calibration of RHR Pump B Mini Flow Valve Control Switch	8A
STN IC-424A	Calibration of Train A RHR Heat Exchanger By-Pass Flow Instrumentation (EJ LPF-0618)	9
STN IC-424B	Calibration of Train B RHR Heat Exchanger By-Pass Flow Instrumentation (EJ LPF-0619)	14A
STN PE-038	Containment Cooler Performance Test	13
STN-KAT-001	Technical Support Diesel Generator	10

Procedures

NUMBER	TITLE	REVISION/DATE
STS CR-001	SHIFT LOGS FOR MODES 1, 2, and 3	84A
STS CR-002	SHIFT LOGS FOR MODES 4, 5, AND 6	66
STS CR-01	Shift Logs	84A
STS EN-101B	Containment Spray Pump B Comprehensive Test	10
STS IC-802A	4KV Loss of Voltage and Loss of Offsite Power Channel Calibration Train A	8A
STS IC-902A	Actuation Logic Test Train A RHR Suction Isolation Valves	3A
STS IC-930A	LOCA and Shutdown Sequencer Time Interval Verification Train A	6A
STS KJ-005A	Manual/Auto Start, Sync & Loading of EDG NE01	59
STS MT-024A	Functional Test of 480 and 120 Volt Molded Case Circuit Breakers	12
STS MT-024B	Functional Test of 480 and 120 Volt Molded Case Circuit Breakers	12
STS MT-024C	Functional Test of 480 and 120 Volt Molded Case Circuit Breakers	12
STS MT-075	SGK05B Heat Exchanger Inspection	5
STS PE-007	Periodic Verification of MOV (8702A), 7/2013 RHR Isolation Valve	4
STS-IC-530D	Channel Calibration Wide Range Temperature and Pressure Instrument Protection	24
STS-PE-019B	RHR Suction Valve Leak Test	20
SYS AP-122	Non-Safety Aux Feed Pump Operation	1
SYS EJ-110	RHR System Fill and Vent Including Initial RCS Fill	63

Procedures

NUMBER	TITLE	REVISION/DATE
SYS EJ-120	Startup of a RHR Train	63B
SYS FP-290	Temporary Fire Pump Operations	14
SYS FP293	Fire Pump Manual Operation	26A
SYS GK-200	Inoperable Class 1E AC Unit	24A
TMP 12-017	Diesel Fire Pump Installation Test	3
WCQPM	Wolf Creek Quality Program Manual	9

Drawings

NUMBER	TITLE	REVISION
2E-2494	Outline Drawing (ESW)	D
2E-2494	2 STG VCT	D
6.2.4-1	Containment Penetration	5
E-025-0007	CCW Inlet to RHR Heat Exchanger B ISO, Sheet 225	W14
E-025-0007	CCW Inlet to RHR Heat Exchanger B ISO, Sheet 225	W14
E-050A-00001	Layout for NK Batteries	W07
E-050A-00002	Layout for NK Batteries	W08
E-050A-00003	Layout for NK Batteries	W08
E-051-0018-01	Qualification of Battery Chargers	1
E-11005	List of Loads Supplied by Emergency Diesel Generator	39
E-11005	List of Loads Supplied by Emergency Diesel Generator	39

Procedures

NUMBER	TITLE	REVISION/DATE
E-11023	Relay Setting Tabulation and Coordination Curves System NB	7
E-11024	Relay Setting Tabulation & Coordination Curves Systems NG/PG	3
E-11025	Relay Setting Tabulation and Coordination Curves System NE	14
E-11032	Substation and Plant Transformer Tap Settings	5
E-11MR01	Startup Transformer Single Line Metering and Relaying Diagram	07
E-11NB01	Lower Medium Voltage System Class 1E 4.16 kV Single Line Meter and Relay Diagram	5
E-11NE01	Standby Generation System Meter and Relay Diagram	10
E-11NG01	Low Voltage System Class 1E 480V Single Line Meter and Relay Diagram	11
E-11NG20	Low Voltage System Class 1E Motor Control Center Summary	285
E-13EG01A	Component Cooling Water Pump A	08
E-13EG01B	Component Cooling Water Pump C	04
E-13EG07A	Component Colling Water Supply to RHR Heat E (EGHV0102)	02
E-13EG07A	Schematic Diagram Component Cooling Water Supply to RHR Heat Exchanger	2
E-13EJ05A	RHR Loop 1 Inlet Isolation Valve	4
E-13EN01	Containment Spray Pumps	05
E-13NN01	Class 1E Instrument AC Schematic	04
E-EF-11-A001/011	Hanger	0

Procedures

NUMBER	TITLE	REVISION/DATE
E-EF-11-R001/91	Pipe Support	5
KD-7496	One Line Diagram	41
KL1909	Load Shedding & Emergency Load Sequencing System (LSELS)	E
M-088-089	Ingersall Rand Company, Vendor Certified Performance Curve Data for PEN01B	February 9, 1978
M-11GK08R0	System Flow Diagram Control Building HVAC	0
M-11GK09R0	System Flow Diagram Control Building HVAC	0
M-11GK10R0	System Flow Diagram Control Building HVAC	0
M-11GK11R0	System Flow Diagram Control Building HVAC	0
M-11GK11R0	System Flow Diagram Control Building HVAC	0
M-12 EJ01	Piping and Instrumentation Diagram Residual Heat Removal System	47
M-12AL01	Piping & Instrumentation Diagram Auxiliary Feedwater System	23
M-12AP01	Piping & Instrumentation Diagram Condensate Storage and Transfer System	12
M-12EG02	Piping and Instrumentation Diagram Component Cooling Water System	21
M-12EG02	Piping and Instrumentation Diagram Component Cooling Water System	21
M-12EJ01	P&ID Diagram Residual Heat Removal System	47
M-12EN01	P&ID Diagram Containment Spray System	12
M-12GN01	P&ID Diagram Containment Cooling System	24
M-189-50EF-01-01	Essential Service Water A Train Supply	1

Procedures

NUMBER	TITLE	REVISION/DATE
M-1H3311	Heating, Ventilating and Air Condition Control Building El. 2000-0	4
M-1H3311	Heating, Ventilating and Air Condition Control Building El. 2000-0	4
M-1H3411	Heating, Ventilating and Air Condition Control Building El. 2016-0	2
M-1H3411	Heating, Ventilating and Air Condition Control Building El. 2016-0	2
M-1H3511	Heating, Ventilating and Air Condition Control Building El. 2032-0	0
M-1H3511	Heating, Ventilating and Air Condition Control Building El. 2032-0	0
M-1H3711	Heating, Ventilating and Air Condition Control Building El. 2073-0	0
M-1H3711	Heating, Ventilating and Air Condition Control Building El. 2073-0	0
M-1HX001	Heat Exchanger Tube Sheet Map document Index & General Notes	73
M-236-00073	18 inch Type 9220 Valve Assembly	W05
M-236-00073	18 inch Type 9220 Valve Assembly	W05
M-236-00084	18 inch Type 9220 Valve Assembly	W09
M-236-00084	18 inch Type 9220 Valve Assembly	W09
M5-VB-00003	System Flow Diagram Control Building HVAC	0
M5-VB-00004	System Flow Diagram Control Building HVAC	0
M5-VB-00005	System Flow Diagram Control Building HVAC	0
M-761-02029-W06	Interconnecting Wiring Diagram Cabinet Snupps Nuclear Power Plant Controls	1W
M-771-00270	Specification for Orifice Plates, Specification Sheet 0380	6/18/03

Procedures

NUMBER	TITLE	REVISION/DATE
M-IG028	Equipment Locations Reactor & Auxiliary Building Section C	06
M-IG029	Equipment Locations Reactor & Auxiliary Building Section D	06
M-K50111	ESW Pumphouse Piping Plan Hanger Location	6
M-K80111	Hanger Details Small Pipes ESW Pumphouse	9
M-K90111	Hanger Location Small Pipe ESW Pumphouse	10
M-KC0911	ESW Pumphouse Piping Sections	22
M-KCO111	ESW Pumphouse Piping Plan	26
M-KG080	ESW System Pumphouse Equipment Location Plan	10
N-1190-3	MK 52 Orifice Plates	2/14/83
NI387-637A-A1	SGK05A/B Air Conditioner Refrigeration Schedule	7

Design Basis Document

NUMBER	TITLE	REVISION
E-00NB	System Description for Lower Medium Voltage System – 4.16 kV	8
E-00NG	System Description for Low Voltage System – 480 V	5
E-10MR	System Description for Startup Transformer	1
E-10NE	System Description for Standby Generation System	1
E-10NF	System Description for Load Shedding and Emergency Load Sequencing	1
M-000	Mechanical Design Criteria for Chillers	

Design Basis Document

NUMBER	TITLE	REVISION
M-000	Mechanical Design Criteria for Chillers	
M-089	ESW System Pumphouse Equipment Location Plan	10
M-089, 10881,10884, 10736	Design Specification for ESW Pumps	
M-10EJ	Residual Heat Removal System	05
M-10EN	Containment Spray System (System Description)	65
M-10GN	Containment Cooling System, (System Description)	07
M-10GR	Containment Atmospheric Control	02
M-10GS	Containment Hydrogen Control, (System Description)	03
M-236	Design Spec for CCW Butterfly Valve	16
M-622.1A-VDS-1.08	SGK05 A/B Cooling Coil Compressor Vendor Data Sheet	W02
Regulatory Guide 1.106	Thermal Overload Protection for Electric Motors on Motor-Operated Valves	1

Condition Reports (CR-…)

02973	27198	47708	58535	63293
03699	27276	47965	60208	66211
06347	27912	52846	60570	66371
14853	28187	53709	60602	68818
21039	28252	53709	60659	68818
22933	28252	53710	60659	71447
24997	28474	53791	61036	72122
25083	31452	53796	62542	72434
25084	39230	54095	62544	72775
25085	43710	55032	62546	OAR 84-0200
25404	47654	56939	62553	OAR 86-0029

Condition Reports Generated During the Inspection (CR-…)

72363	72711	73073	73137	73227

Condition Reports Generated During the Inspection (CR-…)

72365	72740	73118	73139	73231
72390	72775	73119	73148	73240
72449	72927	73120	73150	73242
72496	72944	73120	73152	73263
72634	72945	73123	73206	73271
72639	73070	73124	73209	73333
72664	73071	73134	73210	73351
72670	73072	73135	73219	73740

Work Orders

02-233286-02	07-300672-000	10-352474-000	11-345007-000	12-353259-000
03-253596-000	07-300672-001	11-330609-000	11-345041-000	12-355289-000
03-253617-000	08-303007-000	11-337844-002	11-346762-060	12-356200-000
03-255133-000	08-308122-000	11-338609-000	11-346762-164	12-357413-000
04-268801-000	09-316577-000	11-339421-000	11-347036-000	12-361300-000
05-270489-000	09-318427-000	11-340725-000	11-347036-002	12-361925-000
06-285855-000	10-323426-000	11-340753-000	11-347036-003	12-361956-000
06-286028-000	10-324715-000	11-343331-000	11-347036-004	13-365857-000
06-286028-015	10-324715-000	11-343922-000	11-348007-000	13-366170-000
06-286028-051	10-333811-000	11-343954-000	12-353036-000	13-366479-010
06-288007-000	10-335412-000	11-345007-000	12-353072-000	13-366576-001
06-291297-004				

Miscellaneous

NUMBER	TITLE	REVISION/DATE
	Health Report, CCW Butterfly Valve 4/1/2013-6/30/2013	
	Health Report, New Fire Pump 1/1/2013-3/31/2013	
	Health Report, Chillers 4/1/2013-6/30/2013	
	Health Report, ESW Pump 4/1/2013-6/30/2013	
	MOV Risk Ranking Worksheet for RCS Hot Leg to RHR Pump A Suction for Valve 8702A	
	MOV Risk Ranking Worksheet,	
	NERC Interface Coordination Agreement for the Wolf Creek Substation	
	Project Plan, Class 1E Air Conditioning Units Improvement Plan, August 27, 2013	
	Valve Set up and Operation, WCGS Standing Order 1	44
	Vendor Manual, Installation, Operation, Maintenance of Vertical Turbine Pumps for New Temporary Diesel Fire Pump	
	Westar Energy NUC-001-2 R8 Coordination	NUC-001 R8

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NUMBER	TITLE	REVISION/DATE
011988	50.59 Screening, Containment Spray Pump Full Flow Testing Line	7L April 23, 2008
014189	Non-Safety Auxiliary Feedwater Pump Installation	0
031C49A	Oil Analysis, Chillers, 3/7/13, Tribology Lab	
10446-E-012.2-0010-05	Seismic Support for ESW Pump A	
10466-J-104-0258-02	Seismic Test Procedure for 9N39 ESFAS and 9N40 LSELS Actuation System	A
10466-J-558B-1	Purchase Order, Bechtel to Weed Instrument Co Inc., (Containment RTDs)	0
10881-M-089-K027-06	Seismic Support for ESW Pump A	
12OW103	Operator Work Around, SGK05A/B	
13-PRE-01	Simulator Scenario	July 29, 2013
2010-013	USAR Change Request regarding substation switching with a transmission line out of service	April 14, 2010
201870-1	Containment Spray Pump Test Loop Throttling Valve	April 27, 2006
50.59 Screen	MGE LT-099	0
69466.3.20.006	PAP01 Post Modification Testing Summary, (Burns & McDonnell Document)	0 June12, 2013
AI 23-008	Program and Component Health Report, ESW Pump	4
AIF26A-006-01	MSPI Failure Determination Evaluation	0
BP # 6 & 7	Black & Veatch Engineering Study: Reconfiguration of substation	0
BP #3 & 5	Black & Veatch Engineering Study: Generator and	0

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NUMBER	TITLE	REVISION/DATE
	substation protection upgrade	
BP #9A	Black & Veatch Engineering Study: Generator breaker	0
CCP 126666	MOV Motor Rotors-Magnesium to Aluminum (50.59 Screen)	
Change Notice	OFN NB-042 Revision 0	November 28, 2007
Change Notice	STS KJ-005A Revision 48	April 3, 2007
Change Notice	STS KJ-005A Revision 49	December 4, 2007
DCP 014512	Safety-related Power for Temporary Fans Between SGK05A, B Areas	
DCP M-089-K27	Seismic Analysis for ESW Support	
DCP13923	Diesel Fire Pump Replacement (50.59 Screen)	0
DCP13923	Diesel Fire Pump Replacement (Applicability Determination)	0
E-050A-00011	Lucent Technologies Lineage 2000 Round Cell Battery	W02
EER 90-KJ-16	DG Kilowatt Number and Power Factor 0.8	0
EER 92-EJ-03	Lowering the Setpoint From 2,500 to 1700 gpm for F-618 and F-619 LO ALARM	May 14, 1992
GK-12-011	Operability Evaluation of Chillers	7
IEN 92-16	Loss of Flow From the Residual Heat Removal Pump During Refueling Cavity Draindown	May 7, 1992
IEN 92-16	Loss of Flow From the Residual Heat Removal Pump During Refueling Cavity Draindown	May 13, 1992
ITIP No. 01902	Station Evaluation of NRC IEN 92-16, Loss of Flow From the RHR Pump During Refueling Cavity	May 13, 1992

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NUMBER	TITLE	REVISION/DATE
	Draindown	
Letter, Bechtel to NRC	BN-TOP-3, Performance and Sizing of Dry Pressure Containments, (Containment Temperature Sensitivity Studies)	September 30, 1977
Letter, Daniel International to Wolf Creek	IEEE-323 Tracking, Review and Closure	February 16, 1984
LO1732413	OFN SG-003, Natural Events	011
LR1005002	Outage Modifications	12
M-089-OK083 RLSA06899	Test Report 37 KVH 2 Stages, ESW Pump A	
M-620-A	Technical Specification for Spare Cooling Coils for the Containment Coolers for the WCGS	02
M-724-00409 W13	Instruction Manual for Gate and Check Valves, 3/18/2009	
N9885-1	WEED Instrument Co Inc., (Certification for Containment RTDs, Tag No. OGN-TE-60, -61, -62, -63)	May 25, 1982
NRC Letter to Wolf Creek	Revise Technical Specification 3.8.1, “AC Sources – Operating”, (One percent DG frequency)	April 11, 2013
NRC Letter to Wolf Creek,	Wolf Creek Generating Station – Issuance of Amendment Re: Replacement of Main Steam and Main Feedwater Isolation Valves (TAC NO. MD4840)	March 21, 2008
NRCB 88-04	NRC Bulletin 88-04, Potential Safety-Related Pump Loss	May 5, 1988
NUREG-0830	Safety Evaluation Report Related to the Operation of Callaway Plant, Unit No. 1	October 1981 and Supplements 1 through 4
NUREG-0881	Safety Evaluation Report Related to the Operation of Wolf Creek Generating Station No. 1	April 1982 and Supplements 1

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NUMBER	TITLE	REVISION/DATE
		through 6
OE GN-13-006	Operability Evaluation GNT10060 CTMT COOLER TEMP,…, And Z2, Reactor Building	0
OE-BB-12-005	Operability Evaluation, RHR Isolation Valve	0
OFN EJ-015	CL Recirculation O Ring Mode 3 with Accumulator Isolated Mode 4,5,6	7
OP1306102	JITT – Non Safety Auxiliary Feedwater Pump	0
Operations Essential Reading 2013-0063	Unreliable communications with local EDG Operator	
PIR 2005-2525	Review of NRC Information Notice 2005-23, Vibration Induced Degradation of Butterfly Valve	
PIR 95-066B	Review of information in the Limitorque Letter ITTP#02933	
Reg. Gd. 1.97	Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident	2
SLNRC 81-002	Transcript of FSAR Review Meeting	January 20, 1981
SPDS Read-Out	Containment Cooling Data, (09:30}	August 7, 2013
STN EF-100A	ESW Pump A Reference Pump Curve Determination, 3/30/13	
STS EF-100A	ESW System Inservice Pump A and ESW System A Discharge Check Valve Test, June 2013	
STS EJ-320	Placing RHR Suction in Safety Injection Standby Conditions	41
STS KJ-001A	Integrated D/G & Safeguards Actuation Test Train A	April 3, 2013

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NUMBER	TITLE	REVISION/DATE
STS MT-075	SGK05B Condenser Heat Exchanger Tube Inspection,6/5/2013	
STS PE-007	Periodic Verification of MOV (8702A), 7/2013 RHR Isolation Valve	
STS PE-009	Control Room Ventilation System Flow Rate and Combined Pressure Drop Tests B Train, 5/19/2012	
STS PE-016B	Train B Class 1E Electrical System A/C System Flow Rate Verification, 2/27/2013	
STS PE-019B	RHR Suction Valve Leak Test,	4/12/2013
SY1505600	Main Condensate System, (Training)	13
SYS AP-122	Non-Safety Aux Feed Pump Operation	1
SYS FP-290	Temporary Fire Pump Operation Test	January 16, 2012
SYS FP-293	Temporary Fire Pump Operation Test	January 17, 2012
SYS1505600	Main Condensate System, (Lesson Plan)	13
TB-11-4 Westinghouse Technical Bulletin	Resistor Capacitor Suppressor Failure	February 28, 2011
TMO12-008-FP	Diesel Fire Pump Test	February 4, 2012
TP-09	Technical Position, IST Program Plan for ESW Pump	September 1, 2005
TP-09, (part of IST Program Plan)	IST Program Plan Wolf Creek Generating Station	September 1, 2005
V5007756	Oil Analysis, Chillers, Herguth Lab	June 11, 2012
VA11077	Nupic Audit of Numerical Applications Quality Program	

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NUMBER	TITLE	REVISION/DATE
WCNOC-171	Post-Fire Safe Shutdown Associated Circuits Study	0
WCRE-08	WCGS Fuse List	14
WCRE-19	Third 10-Year Interval Inservice Testing Basis Document	July 26, 2007
WCRE-19	Third 10-Year Interval Inservice Testing Basis Document, (PEN01B)	00
WCRE-19	Third 10 Year Interval Inservice Testing Basis Documentation for valve EGHV0102	
WCRE-19	Third 10 Year Interval Inservice Testing Basis Documentation for Pump PEF01A	0
WM 13-0014	WCNOC Commitments to NRC Regarding Actions Necessary to increase safety margins at Wolf Creek Generating Station through ongoing actions to resolve SCCIs	June 27, 2013
WM 13-0014	Docket#50-482 WCNOC Commitment to NRC Regarding Necessary to Increase Safety Margins at WC	
WM 89-0264, {Wolf Creek Letter to NRC)	Revision to Technical Specifications 3.4.1.4.2, 3.5.4, 4.9.8.1, and 4.9.8.2 – Residual Heat Removal Flow Rate and Safety Injection Pump Availability	November 30, 1989