UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

May 22, 2006

Rick A. Muench, President and
Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, KS 66839

SUBJECT: WOLF CREEK GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 05000482/2006002

Dear Mr. Muench:

On April 7, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Wolf Creek Generating Station. The enclosed integrated report documents the inspection findings which were discussed on April 14, 2006, with you and other members of your staff.

The inspections 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. Inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents three NRC-identified and one self-revealing finding of very low safety significance (Green). Three of these findings were determined to involve violations of NRC requirements. The NRC is treating these violations as noncited violations consistent with Section VI.A.1 of the NRC Enforcement Policy because of the very low safety significance and because the findings were entered into your corrective action program. If you contest these noncited violations, you should provide a response within 30 days 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, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Wolf Creek Generating Station.

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

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

William B. Jones, Chief
Project Branch B
Division of Reactor Projects

Docket: 50-482
License: NPF-42

Enclosure:
NRC Inspection Report 05000482/2006002
w/attachment: Supplemental Information

cc w/enclosure:
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RIV:SRI:DRP/B	RI:DRP/B	C:DRS/EB2	C:DRS/EB1
SDCochrum:sa	TBRhoades	LJSmith	JAClark
WBJones for	***WBJones for***	***/RA/***	***/RA/***
4/18/06	4/18/06	5/16/06	5/22/06

C:DRS/OB	C:DRS/PS	C:DRP/PBB	
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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-482
License:	NPF-42
Report:	5000482/2006002
Licensee:	Wolf Creek Nuclear Operating Corporation
	Wolf Creek Generating Station
Location:	1550 Oxen Lane NE Burlington, Kansas
Dates:	January 1 through April 7, 2006
Inspectors:	S. Cochrum, Senior Resident Inspector T. Rhoades, Resident Inspector F. Brush, Senior Resident Inspector R. Kopriva, Senior Project Engineer J. Drake, Operations Engineer B. Tindell, Reactor Inspector K. Clayton, Operators Engineer P. Elkmann, Emergency Preparedness Inspector B. Henderson, Reactor Inspector R. Taylor, Reactor Inspector L. Ellershaw, PE, Consultant
Approved By:	William B. Jones, Chief, Project Branch B

SUMMARY OF FINDINGS

IR 05000482/2006002; 01/01/06 - 04/07/65; Wolf Creek Generating Station; Fire Protection, Operator Performance During Nonroutine Plant Evolutions and Events, Operability Evaluations, and Postmaintenance Testing.

This report covered a 3-month period of inspection by resident and regional inspectors. The inspection identified three Green noncited violations and one Green finding. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 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 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A self-revealing Green finding was identified for the failure to provide adequate instructions for the operation of Limitorque motor-operated valves. The instructions were inadequate because they failed to provide guidance on declutching Limitorque motor-operated valves, such that the valve operators are not damaged. The inadequate guidance resulted in the degraded operation of a Limitorque motor-operated valve in the circulating water system. During maintenance activities on November 30, 2005, a Limitorque motor-operated valve would not stay declutched without an operator hanging onto the declutch lever. The declutch mechanism had become misaligned from previous improper manual operation of the Limitorque operator. The inability of the operators to promptly close the valve resulted in lowering the condenser vacuum which approached the turbine trip/reactor trip setpoint before the valve was closed. This finding had crosscutting aspects of human performance. The licensee had not provided adequate instructions for manual operation of the Limitorque motor-operated valve, which subsequently resulted in damage to the declutch mechanism.

  The failure to provide adequate instructions for the operation of Limitorque motor-operated valves was a performance deficiency. This finding is more than minor because it affected the initiating events cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions and affected the cornerstone attribute of procedural quality because an inadequate procedure increased the probability of an initiating event. Using the Phase 1 worksheets in Manual Chapter 0609, “Significance Determination Process,” the issue was determined to have very low safety significance because the finding did not contribute to both the likelihood of a

reactor trip and loss of mitigation equipment (power conversion system would have remained available), nor increase the likelihood of fire or flooding. (Section 1R14)

Cornerstone: Mitigating Systems

- Green. An NRC identified Green noncited violation of Facility Operating License Condition 2.C.5, Fire Protection, was identified for inadequate fire detection in the emergency diesel generator rooms. The infrared detectors' view of some combustibles in the rooms was blocked by temporary scaffolding and permanent plant equipment, which could delay the detection of fires and the fire brigade response. Wolf Creek Fire Hazard Analysis E-19905 Sections D.1.7.1 and D.2.7.1 state that the diesel generator rooms early warning fire detection is by infrared detectors, which will readily detect the type of fire caused by the burning of fuel and lube oils. Wolf Creek Updated Safety Analysis Report Section 9.5.1.2.3 states that these detectors respond directly to infrared radiation emanating from a flickering flame. However, with solid objects in between the detectors and the combustibles, the infrared light from the flame would not be sensed by the infrared detectors. The control room would still be alerted to the fire, but only if the fire spread to a part of the room visible to the infrared detectors or the heat from the fire reached thermal fire detectors also installed in the room.

  The failure to provide adequate fire detection in the emergency diesel generator rooms was a performance deficiency. The inspectors determined that the inadequate fire detection in the diesel generator rooms was more than minor because it potentially affected diesel generator availability due to fire under the mitigating systems cornerstone. The inspectors used Inspection Manual Chapter 0609, Appendix F, Fire Protection Significance Determination Process, to determine the significance of the finding. The finding is of very low safety significance because a postulated fire in a diesel room would still be detected and extinguished before it affected any other safe shutdown equipment. The inspectors assigned a low degradation rating to the finding in the significance determination process because the fire detection would have nearly the same level of effectiveness and reliability with the degradation. Therefore, the significance determination process screens the finding as very low safety significance. (Section 1R05)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a noncited violation of Technical Specification 5.4.1 for failure to follow Administrative Procedure AP28-011, "Resolving Deficiencies Impacting SSC's [structures, systems and components]," Revision 1. The inspectors identified a faulty evaluation of a containment spray system degraded condition. The degraded condition was caused by the potential for a 5 cubic foot void in both trains of the containment spray system. The licensee identified the condition and performed their evaluation in response to industry operating experience regarding voiding in safety-related fluid

systems. The evaluation was faulty in its interpretation of the information provided in NUREG/CR-2792. Once aware of the faulty evaluation, the licensee failed to adhere to Procedure AP 28-011 in the following ways: (1) they failed to document the deficiency as soon as possible; (2) they failed to inform the shift manager immediately; (3) they failed to provide reasonable assurance of operability in a time frame commensurate with safety; and (4) they failed to provide a valid reasonable assurance of operability prior to completion of a prompt operability evaluation. This finding had crosscutting aspects associated with problem identification and resolution based on the fact that both the original evaluation of the industry operating experience and the engineering judgement used to provide reasonable assurance of operability were inadequate.

The failure to implement Procedure AP 28-011 following identification of a degraded condition was a performance deficiency. This finding is more than minor because, if left uncorrected, the failure to follow Procedure AP 28A-011 would become a more significant safety concern. Based on the results of a significance determination process Phase 1 evaluation, this finding was determined to have very low safety significance since the licensee was ultimately able to demonstrate operability of the affected equipment. (Section 1R15)

Cornerstone: Occupational Radiation Safety

- Green. The inspectors identified a noncited violation of Technical Specification 5.4.1(a) for the failure to follow licensee Procedure SYS SR-200, “Movable Incore Detector Operation,” Revision 18. Contrary to this procedure, during troubleshooting activities on the incore detector drive system, an incore detector was moved with personnel in the area. This issue was determined to have crosscutting aspects regarding human performance.

  The failure to follow the procedure for incore detector system operation was a performance deficiency. The finding is more than minor because it is associated with the occupational radiation safety cornerstone attribute regarding programs and processes and affected the cornerstone objective of ensuring the adequate protection of worker health and safety from exposure to radiation in that not following station procedure could increase personnel exposure. Using the occupational radiation safety determination process to analyze the significance of the finding, the inspectors concluded the issue was of very low safety significance because the inspection finding was not related to ALARA, did not involve an overexposure, and there was no substantial potential for overexposure. (Section 1R19)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

The plant started the inspection period at 100 percent rated thermal power and remained at or near this power level for the entire report period.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness for Seasonal Susceptibilities

a. Inspection Scope

The inspectors completed a review of the licensee's readiness of seasonal susceptibilities involving winter weather. The inspectors: (1) reviewed plant procedures, the Updated Safety Analysis Report (USAR), and Technical Specifications to ensure that operator actions defined in adverse weather procedures maintained the readiness of essential systems; (2) walked down portions of the two systems listed below to ensure that adverse weather protection features were sufficient to support operability, including the ability to perform safe shutdown functions; (3) evaluated operator staffing levels to ensure the licensee would maintain the readiness of essential systems required by plant procedures; and (4) reviewed the corrective action program to determine if the licensee identified and corrected problems related to adverse weather conditions.

- February 24, 2006, licensee verification of cold weather preparations for essential service water warming line flow after a routine surveillance revealed a potential low flow condition.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.2 Readiness For Impending Adverse Weather Conditions

a. Inspection Scope

On March 7, 2006, the inspectors completed a review of the licensee's readiness for impending adverse weather, specifically with respect to thunderstorms. The inspectors: (1) reviewed plant procedures, the USAR, and Technical Specifications to ensure that operator actions defined in adverse weather procedures maintained the readiness of essential systems; (2) walked down portions of the facility grounds to identify hazards

that could potentially be blown by heavy winds; (3) reviewed plant modifications, procedure revisions, and operator workarounds to determine if recent facility changes challenged plant operation.

- March 7, 2006, licensee preparations for severe thunderstorm weather

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R02 Evaluations of Changes, Tests, or Experiments (71111.02)

a. Inspection Scope

The inspectors reviewed the effectiveness of the licensee's implementation of changes to the facility structures, systems, and components (SSCs); risk-significant normal and emergency operating procedures; test programs; and the USAR in accordance with 10 CFR 50.59. The inspectors utilized Inspection Procedure 71111.02 for this inspection.

The minimum sample size for this procedure is five evaluations and 10 screenings. The inspectors were unable to review the minimum sample of 10 CFR 50.59 evaluations because only two had been performed by the licensee since the last NRC inspection of this area at the Wolf Creek Generating Station. All 10 CFR 50.59 evaluations performed by this licensee since 2003 have now been reviewed by the NRC. The evaluations were reviewed to verify that licensee personnel had appropriately considered the conditions under which the licensee may make changes to the facility or procedures or conduct tests or experiments without prior NRC approval. To ensure that the exclusion of a full evaluation was consistent with the requirements of 10 CFR 50.59, the inspectors reviewed 16 licensee-performed 10 CFR 50.59 screenings in which licensee personnel determined that evaluations were not required. Procedures, evaluations, and screenings reviewed are listed in the attachment to this report

The inspectors reviewed and evaluated a sample of recent licensee performance improvement requests (PIRs) to determine whether the licensee had identified problems related to 50.59 evaluations, entered them into the corrective action program, and resolved technical concerns and regulatory requirements.

The inspectors completed two of the five required licensee safety evaluations and 6 more than the 10 required changes, tests, or experiment screenings.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdowns

a. Inspection Scope

The inspectors: (1) walked down portions of the two risk important systems listed below and reviewed plant procedures and documents to verify that critical portions of the selected systems were correctly aligned and (2) compared deficiencies identified during the walkdown to the licensee's USAR and corrective action program to ensure problems were being identified and corrected.

- Turbine-driven auxiliary feedwater pump train during a motor-driven auxiliary feedwater pump Train A outage, January 13, 2006
- Residual heat removal system Train A during a Train B outage, January 18, 2006
- Primary containment, March 3, 2006
- Component cooling water Train B during a Train A outage, March 2, 2006
- Chemical and volume control system, Train B, April 4, 2006

The inspectors completed five samples.

.2 Complete Walkdown

The inspectors: (1) reviewed plant procedures, drawings, the USAR, Technical Specifications, and vendor manuals to determine the correct alignment of the systems listed below; (2) reviewed outstanding design issues, operator workarounds, and corrective action program documents to determine if open issues affected the functionality of the system; and (3) verified that the licensee was identifying and resolving equipment alignment problems.

- Containment spray system, March 31, 2006

The inspectors completed one sample.

b. Findings

No findings of significance were identified

1R05 Fire Protection (71111.05)

.1 Fire Protection Tours

a. Inspection Scope

The inspectors walked down the six plant areas listed below to assess the material condition of active and passive fire protection features, their operational lineup, and their operational effectiveness. The inspectors: (1) verified that transient combustibles and hot work activities were controlled in accordance with plant procedures; (2) observed the condition of fire detection devices to verify they remained functional; (3) observed fire suppression systems to verify they remained functional; (4) verified that fire extinguishers and hose stations were provided at their designated locations and that they were in a satisfactory condition; (5) verified that passive fire protection features (electrical raceway barriers, fire doors, fire dampers, steel fire proofing, penetration seals, and oil collection systems) were in a satisfactory material condition; (6) verified that adequate compensatory measures were established for degraded or inoperable fire protection features; and (7) reviewed the corrective action program to determine if the licensee identified and corrected fire protection problems.

- January 18, 2006, auxiliary feedwater system rooms
- January 25 2006, auxiliary building 2000 foot elevation hallways
- February 2, 2006, spent fuel pool cooling pump Room A
- February 2, 2006, spent fuel pool cooling pump Room B
- February 15, 2006, emergency diesel generator rooms
- February 15, 2006, battery and associated switchgear rooms
- February 15, 2006, engineered safety feature switchgear rooms
- February 27, 2006, turbine building 2000 foot elevation

The inspectors completed eight samples.

b. Findings

Introduction: A Green noncited violation (NCV) of Facility Operating License Condition 2.C.5, fire protection, was identified for inadequate fire detection in the emergency diesel generator rooms. Specifically, the infrared detectors' view of some combustibles in the rooms was blocked by temporary scaffolding and permanent plant equipment, which could delay the detection of fires and the fire brigade response.

Description: During a walkdown, it was noted that temporary scaffolding was installed in the line of sight between infrared detectors in the emergency diesel generator rooms and the fuel oil tank, as well as other combustibles. The licensee placed this issue into their corrective action program as PIR 2006-0448 and, during their investigation, discovered that some permanent plant equipment was also installed in the line of sight between some combustibles and the detectors.

Wolf Creek Fire Hazard Analysis E-19905 Sections D.1.7.1 and D.2.7.1 state that the diesel generator room's early warning fire detection is by infrared detectors, which will

readily detect the type of fire caused by the burning of fuel and lube oils. Wolf Creek USAR Section 9.5.1.2.3 states that these detectors respond directly to infrared radiation emanating from a flickering flame.

However, with solid objects in between the detectors and the combustibles, the infrared light from the flame would not be sensed by the infrared detectors. The control room would still be alerted to the fire, but only if the fire spread to a part of the room visible to the infrared detectors or the heat from the fire reached thermal fire detectors also installed in the room. The diesel generator rooms also contain automatic sprinkler systems that were unaffected by this condition that would extinguish a large fire. Therefore, the condition discovered by the inspectors would have delayed the fire brigade response to a fire, but in the event of a serious fire, the control room would still have been alerted and the fire extinguished automatically by sprinklers or by the fire brigade at a later time.

Analysis: The failure to provide adequate fire detection in the emergency diesel generator rooms was a performance deficiency. The inspectors determined that the inadequate fire detection in the diesel generator rooms was more than minor because it potentially affected diesel generator availability due to fire under the mitigating systems cornerstone. The inspectors used Inspection Manual Chapter 0609, Appendix F, Fire Protection Significance Determination Process (SDP), to determine the significance of the finding. The finding is of very low safety significance because a postulated fire in a diesel room would still be detected and extinguished before it affected any other safe shutdown equipment. The inspectors assigned a low degradation rating to the finding in the SDP because the fire detection would have nearly the same level of effectiveness and reliability with the degradation. Therefore, the SDP screens the finding as very low safety significance (Green).

Enforcement: Wolf Creek Generating Station, Unit 1 Facility Operating License Condition 2.C.5, fire protection, states, “The Operating Corporation shall maintain in effect all provisions of the approved fire protection program as described in the SNUPPS Final Safety Analysis Report for the facility through Revision 17, the Wolf Creek site addendum through Revision 15, and as approved in the SER through Supplement 5, subject to provisions b & c below.” Wolf Creek USAR Appendix 9.5A, “Design Comparison to Regulatory Positions of Regulatory Guide 1.120", Section F.9, “Diesel Generator Areas,” states in the Wolf Creek comparison to branch technical Position APCSB 9.5-1, Appendix A, “Fire detectors are installed in the ceiling of the room.” Wolf Creek USAR Appendix 9.5B, Revision 19, states, “The USAR FHA has been superseded by the following documents: E-1F9905, Fire Hazard Analysis.” Analysis E-1F9905, Revision 0, states in Sections D.1.7.1 and D.2.7.1 for the diesel generator rooms, “Early warning fire detection is by infrared detectors, which will readily detect the type of fire caused by the burning of fuel and lube oils.” Contrary to the above, the licensee failed to provide the diesel generator rooms with an infrared detector system that would detect a fire at some of the combustibles in the room. The issue is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as NCV 05000482/2006002-01: Failure to provide adequate fire detection in the diesel generator rooms. This issue is in the licensee’s corrective action program as PIR 2006-0448.

1R06 Flood Protection Measures (71111.06)

Semiannual Internal Flooding

a. Inspection Scope

For the area listed below, the inspectors: (1) reviewed the USAR, the flooding analysis, and plant procedures to assess seasonal susceptibilities involving internal flooding; (2) reviewed the corrective action program to determine if the licensee identified and corrected flooding problems; (3) inspected underground bunkers/manholes to verify the adequacy of: (a) sump pumps, (b) level alarm circuits, (c) cable splices subject to submergence, and (d) drainage for bunkers/manholes; (4) verified that operator actions for coping with flooding can reasonably achieve the desired outcomes; and (5) walked down the areas listed below to verify the adequacy of: (a) equipment seals located below the flood line, (b) floor and wall penetration seals, (c) watertight door seals, (d) common drain lines and sumps, (e) sump pumps, level alarms, and control circuits, and (f) temporary or removable flood barriers.

C Containment building 2000' elevation, February 27, 2006

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Biennial Review

a. Inspection Scope

The following inspection activities were performed using Inspection Procedure 71111.11, "Licensed Operator Requalification Program," and 10 CFR 55.46, "Simulation Facilities," as acceptance criteria.

The inspector reviewed the simulator annual performance test book for 2005, in which all annual tests were conducted on November 11, 2005, using ANS/ANSI 3.5 -1998, "Nuclear Power Plant Simulators for Use in Operator Training and Examination," as committed to by the licensee in their "Continued Assurance of Simulator Fidelity," Procedure AI-30C-001. The inspector also discussed facility operating events with the resident staff. In addition, the core performance test documents were reviewed because the licensee had previously taken credit for reactivity and control manipulations using the simulator for several Personal Qualification Statement–Licensee 398 forms during the initial license examination in 2004. While simulator use for reactivity and control manipulation is permitted by 10 CFR 55.46, the simulator must meet the appropriate standards of fidelity, as required by 10 CFR 55.46©)(2). Documents reviewed during the inspection are listed in the back of this report. The purpose of this review was to

determine if the simulator was capable of supporting initial examinations, supporting reactivity and control manipulations, and supporting requalification training required for all licensed operators on shift. Although the licensee sent a letter to the inspector certifying that the simulator would not be used for reactivity or control manipulations for the 2006 examination, the inspector reviewed the criteria in 10 CFR 55.46©)(2) against the test documents and the Cycle 15 test data from the plant. The simulator was using the Cycle 15 core load for the current training cycle.

The inspector interviewed two instructors, three reactor operators, and four senior reactor operators for feedback regarding the fidelity of the simulator, the simulator discrepancy reporting system effectiveness, and training on differences between the simulator and the plant. In addition, one previous fidelity issue, documented in a previous resident inspector report (entered in the corrective action program as PIR 2004-2016), was reviewed for closure in the PIR system, as well as the simulator work package system. The associated lesson plan for the postmodification training was reviewed for adequacy and roster completeness for all licensed operators at the plant.

The inspector reviewed several program documents that describe the overall simulator program and how management groups such as the simulator review board coordinate discrepancy priorities and their subsequent repair decisions, such as cost versus training impact and major model upgrades in order to enhance training on the emergency operating procedures. These items were reviewed in order to satisfy the requirements of 10 CFR 55.46(d) for continued assurance of simulator fidelity through problem identification and resolution, proper reporting, root cause evaluations, and a planned schedule for implementing timely corrective actions with proper content.

The licensee ran three transient tests, one steady state test, and two scenarios on the simulator with data capture enabled in order to verify that the data collected was an accurate representation of the annual test data run in November 2005. Also verified was reasonable model performance based on the current design of the plant and the specified standard reference plant data used for comparison. These tests were: (1) manual reactor trip-transient test one; (2) dual feed pump trip-transient test two; (3) maximum rate power ramp from 100 to 75 percent power, then back to 100 percent power-transient test seven; (4) steady state test at 100 percent power, (5) design basis loss-of-coolant accident with a corresponding loss of offsite power; and (6) grid instability conditions with subsequent loss-of-offsite power. In addition, the inspector verified that the licensee's training programs included grid instability topics as described in Generic Letter 2006-02, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power," dated February 1, 2006.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.2 Resident Inspector Quarterly Review

a. Inspection Scope

On March 24, 2006, the inspectors observed testing of senior reactor operators and reactor operators to identify deficiencies and discrepancies in the training, to assess operator performance, and to assess the evaluator's critique. The scenario involved a large break loss of coolant accident during power operations. The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed the three maintenance activities listed below to: (1) verify the appropriate handling of SSC performance or condition problems; (2) verify the appropriate handling of degraded SSC functional performance; (3) evaluate the role of work practices and common cause problems; and (4) evaluate the handling of SSC issues reviewed under the requirements of the Maintenance Rule, 10 CFR Part 50, Appendix B, and Technical Specifications.

- Emergency diesel Generators A and B lube oil keep warm pump failures, January 23, 2006
- Containment sump pump float level indication failure, February 27, 2006
- Essential service water Pump B flow degradation, March 3, 2006

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope

Risk Assessment and Management of Risk

The inspectors reviewed the assessment activities listed below to verify:
(1) performance of risk assessments when required by 10 CFR 50.65 (a)(4) and licensee procedures prior to changes in plant configuration for maintenance activities and plant operations; (2) the accuracy, adequacy, and completeness of the information

considered in the risk assessment; (3) that the licensee recognizes, and/or enters as applicable, the appropriate licensee-established risk category according to the risk assessment results and licensee procedures; and (4) that the licensee identified and corrected problems related to maintenance risk assessments.

- Work week 105, January 23-29, 2006
- Work week 108, February 13-19, 2006
- Work week 112, March 13-19, 2006

Emergent Work Control

The inspectors: (1) verified that the licensee performed actions to minimize the probability of initiating events and maintained the functional capability of mitigating systems and barrier integrity systems; (2) verified that emergent work-related activities such as troubleshooting, work planning/scheduling, establishing plant conditions, aligning equipment, tagging, temporary modifications, and equipment restoration did not place the plant in an unacceptable configuration; and (3) reviewed the USAR and corrective action program to determine if the licensee identified and corrected risk assessment and emergent work control problems.

- Troubleshooting of dual position indication of main steam isolation Valve C, March 6, 2006

The inspectors completed four samples.

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Nonroutine Plant Evolutions and Events (71111.14)

a. Inspection Scope

For the nonroutine event listed below, the inspectors: (1) reviewed operator logs, plant computer data, and/or strip charts for the below evolutions to evaluate operator performance in coping with nonroutine events and transients; (2) verified that the operator response was in accordance with the response required by plant procedures and training; and (3) verified that the licensee has identified and implemented appropriate corrective actions associated with personnel performance problems that occurred during the nonroutine evolutions:

- On November 30, 2005, the inspectors observed the site response to reduced circulating water flow and lowering of main condenser vacuum. The licensee determined that site operators were unable to quickly shut the discharge valve for circulating Pump B after securing it for maintenance. This resulted in backflow from the remaining pumps reducing flow to the main condenser. The discharge valve was shut after approximately 10 minutes and plant conditions returned to normal.

- On February 6, 2006, the inspectors reviewed the response of the control room operators to a failure of one power range indication. Main control board NI Channel 4 pegged high at 120 percent. There were no alarms, bistable responses, or control rod motion. After checking other redundant indications, the operators concluded that the indication was the result of a malfunction and entered OFN SB-008, "Instrument Malfunctions," Revision 21. Troubleshooting resulted in identification of a failed isolation amplifier that provides the input to several nonsafety-related functions for NI Channel 4.

- On February 7, 2006, the inspectors observed manual control of letdown heat exchanger outlet temperature during maintenance and testing of temperature control Valve BG TV-130. An operator was stationed in the field and communication was established during the operation to ensure the temperature was controlled to prevent damage to the chemical and volume control system demineralizers.

- On March 6, 2006, the inspectors reviewed operator response to lowering volume control tank level indication. Following maintenance on chemical volume control system resin Bed B, volume control tank level started decreasing at 0.5 gallons per minute. After approximately 6 hours of troubleshooting, the licensee determined that the level decrease was caused by a liquid radiation waste valve not being fully seated. The valve was subsequently shut and the volume control tank level returned to normal.

The inspectors completed four samples.

b. Findings

Introduction: A self-revealing Green finding was identified for the failure to provide adequate instructions for the operation of Limitorque motor-operated valves. The inadequate instructions resulted in the degraded operation of a Limitorque motor-operated valve in the circulating water system. The degraded condition was identified during a maintenance activity where the valve could not be shut without an operator hanging on the declutch lever to maintain the valve declutched so a second operator could close the valve. This resulted in an extended period in closing the valve and a lowering of the condenser vacuum and could have resulted in a turbine/reactor trip.

Description: On November 30, 2005, while securing the circulating water Pump B for planned maintenance, operators' attempted to manually close the circulating Pump B discharge valve, but noted the valve drifted further open instead of closing as expected. This allowed the circulating water discharge from the operational pumps to flow back through the secured pump discharge and bypass the condenser. This resulted in a drop of condenser vacuum, drop in service water pressure, and an electric fire pump start on low header pressure. The main generator load dropped from 1235 Mwe to1220 Mwe due to the decrease in condenser vacuum. When the declutch lever was engaged, the operators found the valve declutch lever did not stay in the "manual" position as expected while the handwheel was turned during manual operation of the valve.

Circulating discharge Valve B required its declutch lever to be held in the engaged position to manually stroke the valve. When questioned by the inspectors, the operators stated that several attempts were made to declutch the valve, but they were required to hold down the declutch lever with as much force as they were able to generate to facilitate closing the valve. The operators concluded that the added length of time required to hold the declutch lever engaged to close the valve caused the circulating water discharge from the operational pumps to flow back through the secured pump, resulting in the plant transient.

The inspectors reviewed Wolf Creek Generating Station Standing Order 1, "Valve Setup and Operation," Revision 36, which contains Limitorque motor-operated valve guidelines. The inspectors determined that the standing order did not contain adequate instructions for proper operation of the declutch mechanism. Once engaged, the declutch lever should remain in the declutched position to permit manual operation of the valve. The valve should remain in the declutched position until the electric motor is engaged. The inspectors reviewed Procedure MGE LT-100, "Limitorque Valve Operator General Notes and Details," Revision 5, which stated that using mechanical leverage on the declutch lever or forcing the declutch lever can result in valve operator damage. Procedure MGE LT-100 also stated that it shall be used as a general introduction for all Limitorque procedures; however, the procedure was not referenced in the standing order.

Analysis: The failure to provide adequate instructions for the operation of Limitorque motor-operated valves was considered a performance deficiency. This finding is more than minor because it affected the initiating events cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions and affected the cornerstone attribute of procedural quality because an inadequate procedure increased the probability of a initiating event. The Wolf Creek SDP worksheets do not credit the power conversion system for mitigation; therefore, the inspectors considered the condition for transient initiator only. Using the Phase 1 worksheets in Manual Chapter 0609, "Significance Determination Process," the issue was determined to have very low safety significance because the finding did not contribute to both the likelihood of a reactor trip and loss of mitigation equipment (would not result in the loss of the power conversion system), nor increase the likelihood of fire or flooding. This finding had crosscutting aspects of human performance. The licensee had not provided adequate instructions for the manual operation of Limitorque motor-operated valves that resulted in numerous declutch problems.

Enforcement: No violation of regulatory requirements occurred. Because the finding was associated with nonsafety-related secondary plant equipment and the licensee entered the finding into the correction action program as PIR 2005-3322, this issue is being treated as a finding (FIN 05000482/2006002-02, inadequate procedure for the operation of Limitorque motor-operated valves.)

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors: (1) reviewed plants status documents such as operator shift logs, emergent work documentation, deferred modifications, and standing orders to determine if an operability evaluation was warranted for degraded components; (2) referred to the USAR and design basis documents to review the technical adequacy of licensee operability evaluations; (3) evaluated compensatory measures associated with operability evaluations; (4) determined degraded component impact on any Technical Specifications; (5) used the SDP to evaluate the risk significance of degraded or inoperable equipment; and (6) verified that the licensee has identified and implemented appropriate corrective actions associated with degraded components.

- Essential service water warming line flow
- Auxiliary shutdown panel indication inaccuracies
- Emergency diesel Generator B fuel rack misalignment
- Containment spray system voiding

The inspectors completed four samples.

b. Findings

Introduction: The inspectors identified a self-revealing Green noncited violation of Technical Specification 5.4.1 regarding the failure to follow administrative Procedure AP 28A-011, "Resolving Deficiencies Impacting SSC's," Revision 20.

Description: On February 8, 2006, the inspectors identified an inadequate evaluation of a deficiency impacting the operability of the licensee's containment spray trains. In September 2004, the licensee initiated PIR 2004-2502 to evaluate the applicability of industry operating experience related to emergency core cooling system suction pipe voiding. While evaluating the operating experience for the licensee's systems, a potential void of approximately 5 cubic feet was identified in both containment spray trains between the sump suction isolation valves and suction check valves. In PIR 2004-2502, the information presented in NUREG/CR-2792, "An Assessment of Residual Heat Removal and Containment Spray Pump Performance Under Air and Debris Ingesting Conditions," was used to justify operability of the containment spray pumps. Citing studies performed under NUREG/CR-2792, the evaluation in this PIR asserts that containment spray pump performance degradation will be minor when air ingestion levels are between 3 and 15 percent. This is not what the NUREG states. The NUREG actually states that for air ingestion greater than 2 percent, pump performance degradation varies substantially depending on design and operating conditions. Additionally, it states that for air ingestion greater than about 15 percent, most centrifugal pumps are fully degraded. The PIR evaluation stated that the 5 cubic feet of air equals less than 8 percent by volume in the suction piping. This calculation is potentially nonconservative because it assumes that the air volume is uniformly distributed over the entire length of suction piping and it assumes a length of piping longer than the actual suction piping. It is likely that the air would be transported

through the pipe in a more concentrated group of bubbles, subjecting the pump to a higher air percentage. Therefore, the information in NUREG/CR-2792 actually raises some doubt about the operability of the containment spray pumps in this condition. On February 8, 2006, the inspectors discussed this issue with the system engineer and a licensing representative.

According to licensee Procedure AP 28-011, "Resolving Deficiencies Impacting SSC's," deficiencies and potential deficiencies are to be documented as soon as possible after discovery. Additionally, Procedure AP 28-011 requires that the on-duty shift manager shall be notified immediately.

About 1 hour after the discussion with the system engineer, the inspectors discussed these operability concerns with the shift manager. The shift manager was not aware of the concern. The shift manager made a control room log entry to document the discussion at 4:40 p.m. on February 8, 2006. The shift manager then conducted a conference call with operations and engineering personnel to discuss the operability concerns. At 6:40 p.m. on February 8, 2006, the shift manager documented in the control room logs the basis for his engineering judgement that the potential deficiency did not adversely effect the operability of the containment spray pumps. A formal operability evaluation was then requested.

According to licensee Procedure AP 28-011, if uncertainty exists as to the potential impact on the capability of the SSC to perform its specified safety function, then additional time should be taken to assess the potential impact. This additional time shall be commensurate with the safety significance of the condition. Technical Specification action statements are used as guidelines for the safety significance of the condition. In this case, inoperability of both containment spray trains would require the licensee to take the actions listed in Technical Specification 3.0.3. This states, in part, that "Action shall be initiated within 1 hour to place the unit . . . in Mode 3 within 7 hours."

Contrary to the procedural requirements of Procedure AP 28-011, the licensee did not immediately notify the shift manager of a deficiency with a potential impact on an SSC's capability to perform its safety function. Additionally, approximately 3 hours passed between discovery of the deficiency and the exercise of engineering judgement to provide reasonable expectation of operability. Since the potential action statement time requirement was 1 hour to initiate action to place the unit in Mode 3, 3 hours was not commensurate with the safety significance of the condition.

The engineering judgement was based on the following: "Industry experience at Palo Verde shows that a void volume of 100 cubic feet did not damage their [Palo Verde's] single stage pumps." While a 100 cubic foot void was identified in the Palo Verde recirculation suction lines, their pumps did not run in the recirculation mode while this condition existed. Therefore, this event provides no data that can be used to support engineering judgement regarding the operability of the licensee's containment spray pumps. Additionally, the containment spray pump instruction manual cautions that, "Before starting or while operating the pump, the casing and suction line must be completely filled with the liquid being pumped." The manual goes on to state that this

liquid is relied upon for lubrication and the pump may seize if operated without it. The manual also cautions never to operate the containment spray pumps without the system being completely primed.

Procedure AP 28-011 requires that, during the operability determination process, a reasonable expectation must exist that the SSC is operable and that the prompt determination process will support that expectation. Contrary to this requirement, "reasonable expectation" was not established because the engineering judgement used was invalid.

Analysis: The failure to implement Procedure AP 28-011 following identification of a degraded condition was a performance deficiency. This finding is more than minor because, if left uncorrected, the failure to follow Procedure AP 28-011 would become a more significant safety concern. Based on the results of an SDP Phase 1 evaluation, this finding was determined to have very low safety significance since the licensee was ultimately able to demonstrate operability of the affected equipment. This finding had crosscutting aspects associated with problem identification and resolution. This assessment was based on the fact that both the original evaluation of the industry operating experience and the engineering judgement used to provide reasonable assurance of operability were inadequate.

Enforcement: Technical Specification 5.4.1(a) requires written procedures be implemented as recommended by Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Appendix A recommends procedures for equipment control. Administrative Procedure AP28-011, "Resolving Deficiencies Impacting SSC's," Revision 1, required plant personnel to immediately document the deficiency and inform the shift manager of the deficiency or potential deficiency. Additionally, Procedure AP 28-011, required the shift manager to document a basis for reasonable assurance of operability in a time frame commensurate with the safety significance of the system affected. Contrary to these requirements, the licensee failed to do the following:

- Immediately notify the shift manager
- Document the deficiency as soon as possible
- Document a reasonable assurance in a time frame commensurate with safety
- Declare affected equipment inoperable in the absence of a reasonable assurance of operability

Because of the very low safety significance and the licensee's action to place the issue in their corrective action program (CR 2006-000075), this violation is being treated as an NCV in accordance with Section VI.A.1 of the Enforcement Policy (NCV 05000482/2006002-03, failure to follow administrative procedure for operability determination).

1R17 Permanent Plant Modifications (71111.17)

.1 Annual Resident Review

a. Inspection Scope

The inspectors reviewed key affected parameters associated with energy needs, materials/replacement components, timing, heat removal, control signals, equipment protection from hazards, operations, flowpaths, pressure boundary, ventilation boundary, structural, process medium properties, licensing basis, and failure modes for the modification listed below. The inspectors verified that: (1) modification preparation, staging, and implementation does not impair emergency/abnormal operating procedure actions, key safety functions, or operator response to loss of key safety functions; (2) postmodification testing will maintain the plant in a safe configuration during testing by verifying that unintended system interactions will not occur, SSC performance characteristics still meet the design basis, the appropriateness of modification design assumptions, and the modification test acceptance criteria has been met; and (3) the licensee has identified and implemented appropriate corrective actions associated with permanent plant modifications.

- Residual heat removal Pump B room cooler replacement, February 28, 2006

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.2 Biannual Review

a. Inspection Scope

The inspection procedure requires inspection of a minimum sample size of five plant modifications. The inspectors reviewed 11 permanent plant modification packages and associated documentation, such as implementation reviews, safety evaluation applicability determinations, and three screenings to verify that they were performed in accordance with regulatory requirements and plant procedures. The inspectors also reviewed the procedures governing plant modifications to evaluate the effectiveness of the program for implementing modifications to risk-significant SSC, such that these changes did not adversely affect the design and licensing basis of the facility. Procedures and permanent plant modifications reviewed are listed in the attachment to this report. Further, the inspectors interviewed the cognizant design and system engineers for the identified modifications as to their understanding of the modification packages and process.

The inspectors evaluated the effectiveness of the licensee's corrective action process to identify and correct problems concerning the performance of permanent plant modifications by reviewing a sample of related PIRs. The reviewed PIRs are identified in the attachment.

The inspectors completed 10 of the 5 required samples.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors selected the six below listed postmaintenance test activities of risk significant systems or components. For each item, the inspectors: (1) reviewed the applicable licensing basis and/or design-basis documents to determine the safety functions; (2) evaluated the safety functions that may have been affected by the maintenance activity; and (3) reviewed the test procedure to ensure it adequately tested the safety function that may have been affected. The inspectors either witnessed or reviewed test data to verify that acceptance criteria were met, plant impacts were evaluated, test equipment was calibrated, procedures were followed, jumpers were properly controlled, the test data results were complete and accurate, the test equipment was removed, the system was properly realigned, and deficiencies during testing were documented. The inspectors also reviewed the USAR and corrective action program to determine if the licensee identified and corrected problems related to postmaintenance testing.

- Motor-driven auxiliary feedwater Pump A following planned maintenance, January 13, 2006
- Main steam isolation Valve C dual indication troubleshooting, March 6, 2006
- Flux mapping system drive unit replacement, March 13, 2006
- Auxiliary feedwater Pump B and essential service water suction valve, February 23, 2006
- Turbine-driven auxiliary feedwater pump speed control troubleshooting, March 15, 2006

The inspectors completed five samples.

b. Findings

Introduction: A self-revealing Green NCV of Technical Specification 5.4.1(a) for the failure to follow Procedure SYS SR-200, "Movable Incore Detector Operation,"

Revision 18 was identified. Contrary to this procedure, during troubleshooting activities on the incore detector drive system, an incore detector was moved with personnel in the area.

Description: On February 8, 2006, instrumentation and control personnel and a health physics technician were in the containment building to support troubleshooting and postmaintenance testing of an incore detector drive system. The detector had been stuck in the core for approximately 3 days after implementing drive system modifications. The incore detector was to be withdrawn to allow performance of required monthly and quarterly core flux map surveillances.

This work was preceded by a prejob brief on February 6, 2006. The briefing included the reactor engineer, radiation protection, and instrumentation and control personnel. Operations department personnel did not participate in the brief. The prejob briefing discussions included: the scope and sequence of the work; maintaining the work as low as is reasonably achievable (ALARA); and Procedure APF 25A-100-03, High Risk Work at Power Checklist, requirements. The inspectors noted that the high risk work at power checklist considered the troubleshooting work but did not address personnel observation of the incore detector withdrawal. The troubleshooting plan provided that, to verify proper incore detector drive system operation after troubleshooting, the individuals in the containment building were to observe the incore detector drive system as it started to withdraw the detector. The observation point was about 20 feet from the drive unit.

Prior to the withdrawal, the location of the detector within the system was unknown. Shortly after initiation of the withdrawal, the detector passed the seal table. The radiation level rapidly increased in the area. The area radiation monitor in the seal table area measured a peak of about 10 Rem/hr and a control room alarm was received. The health physics technician in the containment building observed the increasing dose rate and evacuated the group to a low dose area until the detector was withdrawn to the storage location. The peak dose rate measured by the personal dosimetry worn by workers was 232 mRem/hr.

Based on statements made during the postjob debrief, there was confusion among the group inside the containment building about the way the detector was to be withdrawn. One person understood that the detector drive system would be momentarily driven in the withdrawal direction to verify proper operation, then stopped while the group moved to a low dose area for the remainder of the withdrawal. Others expected the continuous withdrawal of the incore detector.

The licensee has established Procedure SYS SR-200, "Movable Incore Detector Operation," Revision 18, for control of the incore detector system. Prerequisite 5.5 of this procedure requires that the shift manager, or his designee, and health physics personnel shall be advised of impending flux mapping so that any personnel in the seal table or detector drive area are removed from the area prior to conducting flux mapping. Neither the shift manager, nor a designee, was aware of the impending incore detector operation.

Analysis: The failure to follow the procedure for incore detector system operation was a performance deficiency. The finding is more than minor because it is associated with the occupational radiation safety cornerstone attribute regarding programs and processes and in that not following station procedure could increase personnel exposure affected the cornerstone objective of ensuring the adequate protection of the worker health and safety from exposure to radiation. Using the occupational radiation safety determination process to analyze the significance of the finding, the inspectors concluded the issue was of very low safety significance because the inspection finding was not related to ALARA, did not involve an overexposure, and there was no substantial potential for overexposure.

This issue was determined to have crosscutting aspects regarding human performance due to the failure to follow the procedure for control of personnel during operation of the incore detector drive system.

Enforcement: Technical Specification 5.4.1(a) requires written procedures to be established, implemented, and maintained as recommended by Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Appendix A recommends procedures for operation of the in-core system. The detector withdrawal was controlled by licensee Procedure SYS SR-200, "Movable Incore Detector Operation," Revision 18, and requires that the shift manager or a designee be aware of impending flux mapping so any personnel in the seal table or detector drive area are removed from the area prior to system operation. Contrary to the requirements of this procedure, neither the shift manager nor a designee were advised of the incore detector system operation with personnel in the area of the detector drive unit. Because of the very low safety significance and the licensee's action to place this issue in their corrective action program (PIRs 2006-0292, -0292, and -0327), this violation is being treated as an NCV in accordance with Section VI.A.1 of the Enforcement Policy (NCV 50-482/2006002-04, Failure to Follow Station Procedure for Operation of the Incore Detector Drive System).

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the USAR, procedure requirements, and Technical Specifications to ensure that the five surveillance activities demonstrated that the SSCs tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed test data to verify that the following significant surveillance test attributes were adequate: (1) preconditioning; (2) evaluation of testing impact on the plant; (3) acceptance criteria; (4) test equipment; (5) procedures; (6) jumper/lifted lead controls; (7) test data; (8) testing frequency and method demonstrated Technical Specification operability; (9) test equipment removal; (10) restoration of plant systems; (11) fulfillment of ASME Code requirements; (12) updating of performance indicator data; (13) engineering evaluations, root causes, and bases for returning tested SSCs not meeting the test acceptance criteria were correct; (14) reference setting data; and (15) annunciators and alarms setpoints. The inspectors also verified that the licensee identified and implemented any needed corrective actions associated with the surveillance testing:

- January 19, 2006, STS KJ-005B, “Manual/Auto Start, Synchronization & Loading of Emergency D/G NE02,” Revision 43

- January 25, 2006, STS BB-005, “RCS Water Inventory Balance Using Excel,” Revision 6

- January 9, 2006, STS BG-002, “ECCS Valve Check and System Vent,” Revision 16

- February, 27, 2006, STS EJ-201A, “Train A RHR System Inservice Valve Test,” Revision 4

- January 13, 2006, STS AL-101A, “MDAFW Pump A Inservice Pump Test,” Revision 34

- February 24, 2006, STN EF-020B, “ESW Train B Warming Line Verification,” Revision 3

- February 2, 2006, STS PE-125, “LLRT Valve Lineup for Penetration 25,” Revision 4, and STS BL-205, “Reactor Make-up Water System Inservice Valve Test,” Revision 7

The inspectors completed seven samples.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the USAR, plant drawings, procedure requirements, and Technical Specifications to ensure that the two temporary modifications were properly implemented. The inspectors: (1) verified that the modification did not have an affect on system operability/availability, (2) verified that the installation was consistent with the modification documents, (3) ensured that the postinstallation test results were satisfactory and that the impact of the temporary modification on permanently installed SSCs were supported by the test, (4) verified that the modifications were identified on control room drawings and that appropriate identification tags were placed on the affected drawings, and (5) verified that appropriate safety evaluations were completed. The inspectors verified that the licensee identified and implemented any needed corrective actions associated with temporary modifications:

- TMO 06-005-SA, main steam isolation Valve C monitoring equipment, March 3, 2006

The inspectors completed one sample.

b. Findings

No findings of significance was identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

.1 Review of Revisions 3, 5, and 8

a. Inspection Scope

The inspector performed in-office reviews of:

- Revision 8 to the Wolf Creek radiological emergency response plan
- Revision 3 to emergency plan implementing Procedure EPP 06-005, "Emergency Classification"
- Revision 5 to APF 06-002-01, "Emergency Action Levels"

These revisions implemented definitions and emergency action levels described in NRC Bulletin 2005-002.

These revisions were compared to their previous revisions; to the criteria of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1; to Nuclear Energy Institute 99-01, "Methodology for Development of Emergency Action Levels," Revision 2; and to the requirements of 10 CFR 50.47(b) and 50.54(q) to determine if the licensee adequately implemented 10 CFR 50.54(q).

The inspector completed three samples during this inspection.

b. Findings

No findings of significance were identified.

.2 Review of Revision 6

a. Inspection Scope

The inspector performed an in-office review of Revision 6 to Wolf Creek emergency plan implementing Procedure Form APF 06-002-001, "Emergency Action Levels," submitted March 6, 2006. This revision implemented an emergency action level as described in NRC Bulletin 2005-002, "Emergency Preparedness and Response Actions for Security-Based Events."

The revision was compared to its previous revision; to the criteria of NUREG-0654, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” Revision 1; to Nuclear Energy Institute 99-01, “Methodology for Development of Emergency Action Levels,” Revision 2; to NRC Bulletin 2005-002; and to the requirements of 10 CFR 50.47(b) and 50.54(q) to determine if the licensee adequately implemented 10 CFR 50.54(q). This review was not documented in a safety evaluation report and did not constitute approval of licensee changes; therefore, these changes are subject to future inspection.

The inspector completed one sample during this inspection.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

Cornerstone: Reactor Safety

The inspectors performed a review of performance indicator data. The inspectors reviewed the licensee’s data submittal using Nuclear Energy Institute 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 2. Licensee performance indicator data were also reviewed against the requirements of Procedure AP 26A-007, "NRC Performance Indicators," Revision 4, and "Performance Improvement and Learning Desktop Instruction, NRC Performance Indicator Program Owner Guidance,” Revision 2. The inspectors reviewed various licensee indicator input information to determine the accuracy and completeness of the performance indicator:

- Safety System Unavailability: High pressure safety injection system centrifugal charging pumps and safety injection pumps, April 2005 through March 2006, completed April 6, 2006

- Safety System Unavailability: Auxiliary feedwater system, April 2005 through March 2006, completed April 6, 2006

Cornerstone: Emergency Preparedness

The inspectors sampled submittals for the performance indicators listed below for the period April 1, 2004, through September 30, 2005. The definitions and guidance of Nuclear Energy Institute 99-02, “Regulatory Assessment Indicator Guideline,” Revisions 2 and 3, were used to verify the licensee’s basis for reporting each data element in order to verify the accuracy of performance indicator data reported during the assessment period.

- Drill and exercise performance
- Emergency response organization participation
- Alert and notification system reliability

The inspectors reviewed a 100 percent sample of drill and exercise scenarios, licensed operator simulator training sessions, notification forms, and attendance and critique records associated with training sessions, drills, and exercises conducted during the verification period. The inspectors reviewed the qualification, training, and drill participation records for a sample of 10 emergency responders. The inspectors reviewed alert and notification system maintenance records and procedures and a 100 percent sample of siren test results. The inspectors also interviewed licensee personnel that were responsible for collecting and evaluating the performance indicator data.

Cornerstone: Barrier Integrity

Reactor Coolant System Specific Activity

On January 20, 2006, the inspector completed verification that the percent of the Technical Specification limit for dose equivalent Iodine-131 was the same or lower than the maximum value reported each month for the period January 1, 2004, through September 30, 2005. The inspectors also observed a chemistry technician obtain and analyze a reactor coolant system sample.

Reactor Coolant System Leak Rate

On January 24, 2006, the inspector completed verification that the licensee reported to the NRC the maximum identified reactor coolant system leakage and the percentage of the Technical Specification limit each month for the period January 1, 2004, through September 30, 2005.

The inspectors completed seven samples during the inspection.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Identification and Resolutions of Problems

a. Inspection Scope

The inspectors performed a daily screening of items entered into the licensee's corrective action program. This assessment was accomplished by reviewing work requests, work orders, and PIRs, and attending corrective action review and work control meetings. The inspectors: (1) verified that equipment, human performance, and program issues were being identified by the licensee at an appropriate threshold and

that the issues were entered into the corrective action program; (2) verified that corrective actions were commensurate with the significance of the issue; and (3) identified conditions that might warrant additional followup through other baseline inspection procedures.

b. Findings

No findings of significance were identified.

.2 Selected Issue Follow-up

a. Inspection Scope

In addition to the routine review, the inspectors selected the below listed issue for a more in-depth review. The inspectors considered the following during the review of the licensee's actions: (1) complete and accurate identification of the problem in a timely manner; (2) evaluation and disposition of operability/reportability issues; (3) consideration of extent of condition, generic implications, common cause, and previous occurrences; (4) classification and prioritization of the resolution of the problem; (5) identification of root and contributing causes of the problem; (6) identification of corrective actions; and (7) completion of corrective actions in a timely manner.

PIR 2005-2161, Room Cooler Coil Support Structure Damaged During Modification

PIR 2005-2161 was initiated to address damage to the room cooler for the motor-driven auxiliary feedwater pump in Room A. This damage was to the cooling coil support structure and occurred either during the removal of the old coil or installation of the new coil. This PIR was considered to be duplicate to PIR 2005-2182 and, therefore, closed. PIR 2005 2182 contains a more detailed consideration of the issue and evaluates the need for changes to the design change package for future room cooler replacements. Based on the inspectors review, PIR 2005-2182 satisfactorily addressed the subject of PIR 2005-2161.

PIR 2005-3322, Circulating Water Pump B Discharge Valve Closing Resulting in Plant Transient

The licensee initially entered this issue into the corrective action program as a significance Level IV condition PIR, which is defined as a condition that did not meet the level of a noncompliance and did not require extent of condition review nor apparent cause determination. The inspectors reviewed Procedure AP 28A-001, “Performance Improvement Request,” Revision 25, which defined a noncompliance in part as a procedure implementation or personnel actions that result in faults, failures, malfunctions, or deficiencies. Following inspector discussions with the licensee, this issue was upgraded in the corrective action program 14 days later to a higher significance level requiring an extent of condition and apparent cause evaluation. The inspectors reviewed the extent of condition which failed to address whether other plant equipment was impacted by improper operation of Limitorque valve declutch levers or if

the declutch levers are required to be held down on other valves. The inspectors also reviewed an apparent cause evaluation which failed to address the cause of why the inappropriate action occurred. Based on the inspectors review of operating experience, the most probable cause of the declutch lever failing to remain engaged when in its manual position is due to the tripper fingers being out of adjustment from improper lever manipulation during valve operations.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

On February 2, 2006, the inspectors debriefed the inspection results of the licensed operator requalification program with Mr. Westman, Training Manager, and other members of the licensee's staff. A telephone exit was held with Mr. Westman on February 14, 2006. The licensee acknowledged the findings presented in both the briefing and the final exit meeting.

On February 22, 2006, the inspector conducted a telephonic exit meeting to present the emergency preparedness inspection results to Mr. T. East, Manager, Emergency Planning, who acknowledged the findings. The inspector confirmed that proprietary information was not provided or examined during the inspection.

On March 16, 2006, the inspector conducted a telephonic exit meeting to present the emergency preparedness inspection results to Mr. T. East, Manager, Emergency Planning, who acknowledged the findings. The inspector confirmed that proprietary information was not provided or examined during the inspection.

On March 31, 2006, the inspectors presented the inspection results of biannual permanent plant modification inspection to Mr. T. Garrett, Vice President, Engineering, and other members of his staff who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

On April 14, 2006, the resident inspectors presented the inspection results of the resident baseline inspections to Mr. R. Muench, President and Chief Executive Officer, and other members of the licensee's management staff. The licensee acknowledged the findings presented. The inspectors verified that no proprietary information was provided during the inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

T. J. Garrett, Vice President Engineering
S. E. Hedges, Vice President Operations and Plant Manager
R. A. Muench, President and Chief Executive Officer
K. Scherich, Director Engineering
M. Sunseri, Vice President Oversight

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000482/2006002-01	NCV	Failure to provide adequate fire detection in the diesel generator rooms (Section 1R05)
05000482/2006002-02	FIN	Inadequate procedure for the operation of Limitorque motor-operated valves (Section 1R14)
05000482/2006002-03	NCV	Failure to follow administrative procedure for operability determination (Section 1R15)
05000482/2006002-04	NCV	Failure to follow a station procedure for the operation of the incore detector drive system (Section 1R19)

LIST OF DOCUMENTS REVIEWED

In addition to the documents referred to in the inspection report, the following documents were selected and reviewed by the inspectors to accomplish the objectives and scope of the inspection and to support any findings:

Performance Improvement Requests

97-4040	2002-0272	2005-2161
97-3288	2003-0118	2005-2166
2006-0292	2006-0294	
2006-0327	2005-2182	

Work Orders

03-253088-015	97-125181-009	04-263375-003
00-215842-001	97-125181-003	
02-233939-011	04-263375-003	
97-125181-037	04-263375-014	

1R04 Equipment Alignment

Procedures and Instructions

STS BG-002, “ECCS Valve Check and System Vent,” Revision 16
CKL EJ-120, “RHR Normal System Lineup,” Revision 34
CKL AL-120, “Auxiliary Feedwater Normal System Lineup,” Revision 33B
CKL BG-120, “Chemical and Volume Control System Normal Valve Lineup,” Revision 36A.

Plant Drawings

M-13AL05 Revision 4
M-13AL02 Revision 3
M-13AL01 Revision 10
M-13EJ01 Revision 13
M-13EJ06 Revision 0
M-13BN01 Revision 1
M-13BG02 Revision 5
M-13EM02 Revision 6
M-13EM06 Revision 0
M-13EJ02 Revision 8

Miscellaneous
"Determination of Allowable Gas Venting Volumes for the Callaway Plant Emergency Core Cooling And Auxiliary Feedwater System," R-4152-00-1, Revision 0

1R05 Fire Protection

AP 10-106, "Fire Preplans," Revision 3

1R11 Licensed Operator Requalification Program

Open simulator discrepancy reports from January 2004 through December 2005

Closed simulator discrepancy reports from January 2004 through December 2005

Training performance indicators procedure, revised November 5, 2005

Simulator facility review board (SFRB) meeting minutes for last two quarters of 2005

Simulator annual performance test book for 2005

Simulator steady state testing procedure, AI 30C-005

Simulator transient testing procedure, AI 30C-006

Plant modification summary from January 2004 through December 2006

Simulator differences list

Completed SBT test package for loss of offsite power

Loop tolerance Procedures MAP0001 and BBL0459

List of all current exceptions to ANS/ANSI 3.5 - 1998, "Nuclear Power Plant Simulators for Use in Operator Training and Examination"

Simulator Mod (work) packages (SMP) closed for training disk 14 (dated May 3 to December 20, 2005)

Core physics testing packages for simulator, Cycle 15

Nuclear parameters and operations package (NUPOP) book, Cycle 15

Simulator work package for atmospheric relief valve fidelity issue in 2004, PIR 2004-016

PIR 2004-2016

Training package on atmospheric relief valve fidelity issue from 2004, LR3332401

Reactor coolant pump fidelity simulator work package

Simulator work package report for enhancements

Operator licensing tracking system active operator licenses (R4 OLTS report)

Current operator license list from Wolf Creek

Grid stability training lector notes, LR 1410502

Grid stability scenario LR 5001013

"Unit Limitations" (grid stability entry conditions), Procedure OFN-AF-025

1R12 Maintenance Effectiveness

Miscellaneous

Emergency diesel generator analysis reports
Maintenance rule expert panel meeting minutes for emergency diesel generators, KJ
Operability log for 2005 for emergency diesel generators, KJ
Operator logs
System health report for emergency diesel generators, KJ

PIRs
2005-3072, -3075, -3085, -3184, and -3408

Work Orders
05-278249-001, 05-278265-000, 05-278652-000, 05-278874-000, and 05-279453-000

1R14 Personnel Performance During Nonroutine Plant Evolutions
OFN SB-008, “Instrument Malfunctions,” Revision 21
TMP 06-001, “Bypassing BG TV-130,” Revision 0
WR 06-054205

1R22 Surveillance Testing

Procedures and Instructions

STS BG-002, “ECCS Valve Check and System Vent,” Revision 16

STS EJ-100A, “RHR System Inservice Pump A Test,” Revision 29

STS IC-208B, “4KV Loss of Voltage and Degraded Voltage TADOT NB02 Bus, Separation Group 4,” Revision 1

System Drawings
M-13EJ01 Revision 13
M-13EJ06 Revision 0

M-13BN01 Revision 1
M-13BG02 Revision 5
M-13EM02 Revision 6
M-13EM06 Revision 0
M-13EJ02 Revision 8

Miscellaneous

"Determination of Allowable Gas Venting Volumes for the Callaway Plant Emergency Core Cooling And Auxiliary Feedwater System," R-4152-00-1, Revision 0

Interoffice Correspondence ES05-0009, "Post Outage Local Leak Rate Testing (LLRT) Report - Fourteenth Refueling Outage"

WO 05-271944-000, 05-271944-001, 05-271944-002, 05-271944-004, and 05-271944-005

WR 05-048998

Dose Equivalent Iodine Sample Results for January 2004 through September 2005

CHA RC-004, "Gamma Isotopic, Total Curie Content and Dose Equivalent Iodine Determination," Revision 10

CHA RC-005, "Determination of Gas Activity," Revision 7

CHS SJ-143, "Sample Station Sampling Instructions," Revision 21

Technical Specifications