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

August 7, 2008

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 05000482/2008003

Dear Muench:

On June 28, 2008, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Wolf Creek Generating Station. The enclosed report documents the inspection results, which were discussed on July 9, 2008, with Mr. Steve Hedges and other members of your staff.

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

Based on the results of this inspection, seven NRC-identified and self-revealing findings of very low safety significance (Green) are documented in this report. Five of these findings were determined to involve violations of NRC requirements. Additionally, a licensee-identified violation of very low safety significance is listed in this report. However, because of the very low safety significance and because the findings were entered into your corrective action program, the NRC is treating these violations as noncited violations consistent with Section VI.A of the NRC Enforcement Policy.

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 made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Vincent G. Gaddy, Chief
Project Branch B
Division of Reactor Projects

Docket: 50-482
License: NPF-42

Enclosure:
NRC Inspection Report 05000482/2008003
w/Attachment: Supplemental Information

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SUNSI Review Completed: VGG ADAMS: ☑ Yes ☐ No Initials: VGG
☑Publicly Available ☐ Non-Publicly Available ☐ Sensitive ☑ Non-Sensitive
R:_REACTORS_WC\2008\WC2008-003RP-SDC.doc ADAMS ML082210141

RIV:SRI:DRP/B	C:DRS/PSB1	C:DRS/EB1	C:DRS/PSB2
SDCochrum	MPShannon	RLBywater	GEWerner
/RA VGaddy for/	***/RA JRLarsen for/***	***/RA/***	***/RA/***
8/ /2008	7/30/2008	7/28/2008	7/31/2008
C:DRS/OB	C:DRS/EB2	C:DRS/EB1	C:DRP/B
RELantz	NFO'Keefe	RLBywater	VGGaddy
/RA FCOKeefe for/	***/RA/***	***/RA/***	***/RA/***
7/29/2008	7/29/2008	7/29/2008	08/07/2008

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U. S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-482
License:	NPF-42
Report:	05000482/2008003
Licensee:	Wolf Creek Operating Corporation
Facility:	Wolf Creek Generating Station
Location:	1500 Oxen Lane SE Burlington, Kansas
Dates:	April 7 through June 28, 2008
Inspectors:	S. D. Cochrum, Senior Resident Inspector C. M. Long, Resident Inspector P. J. Elkmann, Senior Emergency Preparedness Inspector L. E. Ellershaw, Senior Reactor inspector K. D. Clayton, Senior Reactor Inspector J. F. Drake, Senior Reactor Inspector
Approved By:	V. G. Gaddy, Chief, Project Branch B Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000482/2008003; 4/07/08 - 6/28/08; Wolf Creek Generating Station; Fire Protection, Maintenance Effectiveness, Operability Evaluations, Outage Activities, Event Follow-up and Other Activities.

This report covered a 3-month period of inspection by resident inspectors and regional specialists. The inspection identified seven Green findings, five of which are noncited violations. 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's review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A self-revealing finding was identified for an inadequate maintenance procedure that resulted in operators manually tripping the plant due a loss of all condensate pumps. In order to support planned maintenance for a different service transformer, the licensee aligned power to all three condensate pumps from Service Transformer PB003. Because the work order used to perform work on Service Transformer PB003 two weeks earlier was inadequate, two phases of the bus cables were installed using the incorrect configuration and subsequently failed, causing the transformer and condensate pumps to trip. The licensee entered this issue into their corrective action program as Condition Report 2008-0908.

  The finding was more than minor because it is associated with the procedure quality attribute of the initiating events cornerstone and it affected the cornerstone objective to limit the likelihood of those events that upset plant stability. This finding also affected the mitigating systems cornerstone and the inspectors evaluated the significance of this finding using a Phase 2 analysis. Further assessment using a Phase 3 analysis by a senior reactor analyst determined the finding to be of very low safety significance. The performance deficiency was analyzed as a reactor trip with a loss of normal feedwater with a 13 day exposure time. This finding has human performance crosscutting aspects in the area associated with resources component because the licensee failed to provide a complete and thorough maintenance procedure to assure nuclear safety [H.2(c)].

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation of Technical Specification 5.4.1.d for failing to control combustible materials in an area of the plant that contained safety related equipment. During a walkdown on May 1, 2008, inspectors discovered that a filled temporary propane cylinder for a generator did not have a transient combustible materials permit because of the licensee's longstanding inappropriate practice of exempting propane cylinders from permit controls. The licensee entered this issue into their corrective action program as Condition Report 2008-2571.

The inadequate control of transient combustibles in containment was more than minor because, if left uncorrected, it would become a more significant-safety concern and could potentially affect residual heat removal availability due to fire under the mitigating systems cornerstone. Using the fire protection findings significance determination process, the finding was determined to be of very low safety significance because the finding represented a moderate degradation and only affected the ability to achieve and maintain cold shutdown. The finding also had crosscutting aspects in the problem identification and resolution area associated with corrective actions because the licensee failed to take appropriate corrective actions for a previous NRC identified deficiency in the exempted use of transient combustibles [P.1(d)].

- <u>Green</u>. The inspectors identified a noncited violation of 10 CFR 50.65(a)(1) for failure to establish goals for the safety-related room coolers and monitor room cooler performance against those goals. In 2005, residual heat removal pump Room Cooler SGL10A had accumulated enough unavailability time to move the room cooler function to a(1) status. While dealing with a room cooler replacement schedule that was delayed several times, the licensee decided that a(1) goals did not need to be established until all room coolers were replaced. The inspectors considered this to be a case where no technically justified goals were established. The licensee entered this issue into their corrective action program as Condition Report 2008-3145 which included action to expedite room cooler procurement and replacement.

  This finding is more than minor because it is consistent with Inspection Manual Chapter 0612, Appendix E, Example 7.a. Specifically, The licensee failed to establish a(1) goals and monitor performance against those goals for the a(1) GL-5 function for 3 years. The inspectors evaluated the significance of this finding and determined that the finding is of very low safety significance because the support function (GL-5) to cool pump rooms does not result in a total loss of any safety function as identified by the licensee probability risk assessment that contributes to external event initiated core damage accident sequences (i.e., initiated by a seismic, flooding, or severe weather event). The finding has a crosscutting aspects in the problem identification and resolution area associated with corrective action program because the licensee failed to take appropriate corrective actions to address this safety issue and the adverse room cooler trends in a timely manner, commensurate with safety significance and complexity [P.1(d)].

- <u>Green</u>. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V for the licensee's failure to specify inspection criteria in its containment sump inspection procedure. During a Mode 4 containment walkdown on May 9, 2008, the inspector identified a gap in containment Sump A that was not previously identified. Operators subsequently declared containment Sump A inoperable and because Train B of the residual heat removal system was already inoperable for maintenance, the licensee entered Technical Specification 3.0.3. The licensee entered this issue into their corrective action program as Condition Report 2008-2219 which included action to repair the gap in the sump.

The finding was more than minor because it affected the procedure quality and human performance attributes of the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that to responds to initiating events and prevent undesirable consequences. The inspectors determined that the finding was of very low safety significance because the deficiency did not result in the complete loss of operability or functionality and did not represent a risk significant external event such as flooding. The finding has human performance crosscutting aspects in the area associated with resources, in that Procedure STS-EJ-002 was not adequate to assure nuclear safety because it did not include complete, accurate and up-to-date specifications or acceptance criteria for the sump [H.2(c)].

- Green. A self-revealing finding was identified for an inadequate switchyard maintenance work instruction which resulted in the loss of offsite power. On April 7, 2008, while the plant was defueled for a refueling outage and safety-related 4 kV Bus NB01 was secured for maintenance, offsite power was lost to the safety-related 4 kV Bus NB02 when switchyard workers tripped the incorrect “breaker failure” trip relay while testing the Rose Hill 345kV offsite switchyard breakers. The work orders only provided generic instructions. The incorrect closed trip relay made up the logic for the startup transformer protection circuit and extended the trip signal to all 345kV offsite breakers, resulting in the loss of power. The licensee entered this issue into their corrective action program as Condition Report 2008-1457.

  This finding is greater than minor because it is associated with the equipment performance attribute of the mitigating systems cornerstone and affected the objective to ensure availability and reliability of systems that respond to initiating events to prevent undesirable consequences. Using the shutdown findings significance determination process, the finding was determined to be of very low safety significance because the finding did not increase the likelihood of a loss of reactor coolant system inventory, degrade the ability to terminate a leak path or add reactor coolant system inventory when needed during shutdown operations. This finding had human performance crosscutting aspects in the area of resources because the work instructions did not provide complete and accurate information to ensure safety [H.2(c)].

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for failure to implement engineering procedures and approve a third party calculation prior to use. Specifically, the calculation review failed to identify the incorrect design inputs to the net positive suction head calculations on two occasions for residual heat removal and containment spray pumps. The licensee entered this issue into their corrective action program as Condition Report 2008-1305 which included action to correct the calculations.

  This finding was more than minor because they were similar to nonminor Example 3.j from NRC Inspection Manual Chapter 0612, Appendix E, "Examples of Minor Issues," in that, there was a reasonable doubt on the operability of the residual heat removal and containment spray pumps; and if left uncorrected, could result in a more significant-safety concern. The finding is of very low safety significance because it was a qualification deficiency confirmed not to result in loss-of-operability in accordance with NRC Manual Chapter Part 9900, Technical Guidance, "Operability Determination Process for Operability and Functional Assessments." The finding had a problem identification and resolution area crosscutting aspects in the corrective action program component, because the site failed to perform a thorough evaluation of vendor calculations to ensure conditions adverse to quality are identified and resolved [P.1(c)].

Cornerstone: Barrier Integrity

- Green. The inspectors identified a noncited violation of Technical Specification 5.4.1.a in which the licensee raised the winch load setpoint for its fuel transfer system to avoid trips without knowing the cause. Because of repeated trips of the fuel handling system winch during core reload, the load setpoints and slow speed zones were inappropriately and nonconservatively changed. When questioned by the inspector, the licensee was unable to produce modification documentation that justified these software changes. The licensee entered this issue into their corrective action program as Condition Reports 2008-0254 and 2008-0255 which included action to reset the setpoints.

  The finding was more than minor because it is associated with the human performance attribute of the barrier integrity cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding) protect the public from radionuclide releases caused by accidents or events. Specifically, this issue relates to the procedure adherence example of the human performance attribute because the design process was bypassed to mask fuel cart problems. The finding was of very low safety significance because the issue did not result in fuel handling errors that caused damage to fuel clad integrity or a dropped fuel assembly. The cause of the finding has human performance crosscutting aspects in the area associated with decision making. Specifically, the licensee did not ensure safety by making safety or risk-significant decisions by using a procedural or systematic process when faced with the unexpected and repeated fuel transfer cart winch trips [H.1(a)].

B. Licensee-Identified Violations

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

REPORT DETAILS

Summary of Plant Status

The plant started the inspection period shutdown for Refueling Outage 16. The licensee completed the refueling outage and synchronized the generator to the grid on May 14, 2008. The licensee returned to full power operations on May 18, 2008. Wolf Creek remained at full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness

1R01 Adverse Weather Protection (71111.01)

.1 Readiness for Impending Adverse Weather Conditions

a. Inspection Scope

The inspectors completed a review of the licensee's readiness for impending adverse weather involving severe thunderstorms and heavy rains. The inspectors: (1) reviewed plant procedures, the Updated Safety Analysis Report (USAR), and Technical Specifications (TS) to ensure that operator actions defined in adverse weather procedures maintained the readiness of essential systems; (2) walked down portions of the systems listed below to ensure that adverse weather protection features were sufficient to support operability, including the ability to perform safe shutdown functions; (3) reviewed maintenance records to determine that applicable surveillance requirements were current before the anticipated weather developed; and (4) reviewed plant modifications, procedure revisions, and operator work arounds to determine if recent facility changes challenged plant operation.

- June 5, 2008, readiness for severe thunderstorms

Documents reviewed are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.2 Readiness of Offsite and Onsite Alternate Alternating Current (AC) Power Sources

a. Inspection Scope

The inspectors completed a review of the licensee's readiness of offsite and onsite AC power sources. The inspectors: (1) reviewed plant procedures, the USAR, and TSs to ensure that operator actions defined in procedures maintained the readiness of essential systems; (2) walked down portions of the systems to ensure that protection features were sufficient to support operability, including the ability to perform safe shutdown functions; (3) reviewed maintenance records to determine that applicable surveillance requirements were current, (4) reviewed plant modifications, procedure revisions, and operator work arounds to determine if recent facility changes challenged plant operation; and (5) interviewed plant personnel regarding coordination with system operations.

- May 30, 2008, offsite and onsite alternate AC power

Documents reviewed are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

Quarterly Partial System Walkdowns

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk-significant systems:

- June 4, 2008, Train B essential service water (ESW) during planned maintenance on Train A ESW

- June 18, 2008, Train A ESW during planned maintenance on Train B ESW

The inspectors selected these systems based on their risk-significance relative to the reactor safety cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could impact the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, USAR, TS requirements, administrative TSs, outstanding work orders (WOs), condition reports (CRs), and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program with the appropriate significance characterization.

Documents reviewed are listed in the attachment.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

The inspectors conducted fire protection walkdowns which were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- April 14, 2008, residual heat removal (RHR) heat exchanger Rooms A & B
- April 15, 2008, containment 2047' elevation
- April 28, 2008, control building 2032' elevation, upper cable spreading room
- June 4, 2008, Train A ESW pump house

The inspectors reviewed areas to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, maintained passive fire protection features in good material condition, and had implemented adequate compensatory measures for out of service, degraded or inoperable fire protection equipment, systems, or features in accordance with the licensee's fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's individual plant examination of external events (IPEEE) with later additional insights, their potential to impact equipment which could initiate or mitigate a plant transient, or their impact on the plant's ability to respond to a security event. Using the documents listed in the attachment, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed, that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee's corrective action program.

The inspectors completed four samples

b. Findings

Introduction. A Green noncited violation (NCV) TS 5.4.1.d was identified for failing to control combustible materials inside containment in an area that contained safety-related equipment.

Description. During a walkdown on May 1, 2008, the inspectors noted that a temporary propane cylinder for a generator contained a large amount of propane. The inspectors identified that the propane cylinder did not have a transient combustible materials permit. The inspectors asked about the combustible permits and/or fire impairments associated with this area. Operators informed the inspectors that there were no active permits or impairments for this propane cylinder. The operators further stated that no such actions would be necessary because the generator and its propane cylinder are exempt from permit controls. The inspectors were concerned that a large amount of combustibles was located inside containment. The inspectors reviewed Procedure AP 10-102, "Control of Combustible Materials," Revision 13. Section 7.10 of this procedure stated that the, "propane cylinder. . . is exempt from the transient combustible permit requirements of this procedure." Section 6.2.1 also states, in part, that a transient combustible materials permit is required if 2 gallons of flammable liquid or 14 pounds of flammable gas (not connected with hot work) are used. Using industry standards, the inspectors estimated the propane to be approximately 33.5 pounds of liquid and calculated the combustible loading to be approximately 772,000 Btu. Procedure MPMC151Q-01, "CTMT Equipment Hatch Maintenance and Operation," Revision 5, for the generator also referenced AP 10-102 for the exemption. Neither had an evaluation for 33.5 pounds of propane.

If all the liquid propane was released as a gas, it would occupy approximately 307 cubic feet at 100 percent concentration. With a lower explosive limit of 2 percent, the gas cloud could expand another 50 times and still be within the flammable range. This quantity of liquid propane could, under ideal conditions, create an ignitable vapor cloud of 15,300 cubic feet or enough to occupy the lower levels of containment. Propane gas is approximately 52 percent heavier than air.

The inspectors spoke with Wolf Creek fire protection and licensing personnel and expressed that there seemed to be an inadequacy in their fire protection program. These personnel stated that their exemption of the propane was a long-standing policy of the station and fire protection plan. The licensee stated they considered the practice routine for outages and that because the generator was not in operation, that there was not an ignition hazard. They disagreed that there was any problem with the fire protection program or that any violation of NRC requirements had occurred. The inspectors contacted NRC regional fire protection specialists. The specialists informed the inspectors that Wolf Creek's position was contrary to industry standards and practice. The specialists stated that industry standards also consider heat of combustion or fire load, and a potential fire hazard and combustible characteristics of a material (i.e., an explosion). The inspectors determined that the licensee's interpretation that the propane cylinder should be exempt from permit requirements was inappropriate. Various hot work activities were conducted inside containment during the outage, including welding in the instrument tunnel sump at the lowest elevation.

<u>Analysis</u>. The inspectors determined that the inadequate control of transient combustibles in containment was more than minor because, it affected the Mitigating System Cornerstone objective to ensure the availability, reliability, and operability of equipment used to mitigate the affect of external events (fire) since it could potentially affect RHR availability due to fire. Traditional enforcement does not apply since there were no actual safety consequences or potential for impacting the NRC's regulatory function, and this finding was not the result of any willful violation of NRC requirements or Wolf Creek procedures. The inspectors determined that this finding was of very low safety significance using Phase 1 of Inspection Manual Chapter (IMC) 0609, Appendix F, "Fire Protection Significance Determination Process," because this finding was assigned a high degradation factor and the issue screened to Green because it only affected the ability to achieve and maintain cold shutdown. During the time of concern, Wolf Creek was in Modes 5, 6, and defueled. The inspectors also determined that this finding had crosscutting aspects in the problem identification and resolution area associated with corrective actions because the licensee failed to take appropriate corrective actions for a previous NRC-identified deficiency in the exempted use of Class A transient combustibles [P.1(d)].

Enforcement. Technical Specification 5.4.1.d requires that written procedures be established, implemented, and maintained covering activities related to fire protection program implementation. Administrative Procedure AP 10-102, “Control of Combustible Materials,” Revision 12, which is part of the approved fire protection program, states, in part, that a transient combustible material permit is required for transient combustibles if the quantity exceeds 14 pounds of combustible gas. Contrary to the above, from March 21 to May 5, 2008, approximately 33.5 pounds of propane was in containment without implementing a transient combustible material permit or establishing appropriate compensatory measures. This issue and the corrective actions are being tracked by the licensee in CR 2008-002571. Because this finding is of very low safety significance and has been entered into the corrective action program, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy:
NCV 05000482/2008003-01, Failure to Implement Transient Combustible Control Permit Requirements for a Propane Tank.

1R08 Inservice Inspection Activities (71111.08)

02.01 Inspection Activities Other Than Steam Generator Tube Inspection, Pressurized Water Reactor Vessel Upper Head Penetration Inspections, and Boric Acid Corrosion Control

a. Inspection Scope

The inspection procedure requires review of two or three types of nondestructive examination (NDE) activities and, if performed, one-to-three welds on the reactor coolant system pressure boundary. Also, review is required of one or two examinations with relevant indications that have been accepted by the licensee for continued service.

The inspectors directly observed the following NDEs:

System	**Weld Identification**	**Exam Type**
Reactor Coolant	Pressurizer Relief Valve C Nozzle Weld Overlay	UT
Containment Spray	Weld MW 7022	UT
Containment Spray	Welds 7044 and 7046	RT
Reactor Coolant	Reactor Pressure Vessel Outlet Nozzle DM weld 1-RV-301-121-C	VT-2

The inspectors reviewed records for the following NDEs:

System	**Identification**	**Exam Type**
Reactor Coolant	RPV Outlet Nozzle DM Welds1-RV-301-121-A, -B, -C, -D	VT-2

System	Identification	Exam Type
Reactor Coolant	RPV Outlet Nozzle DM Welds 1-RV-301-121-A, -B, -C, –D	UT RF 14 (April 2005)
Reactor Coolant	Reactor Pressure Vessel Inlet Nozzle DM Welds 1-RV-302-121-A, -B, -C, -D	UT RF 14 (April 2005)
Reactor Coolant	Pressurizer Surge Line DM Weld Overlay	UT

During the review and observation of each examination, the inspectors verified that activities were performed in accordance with American Society of Mechanical Engineers (ASME) boiler and pressure vessel code requirements and applicable procedures. Indications were compared with previous examinations and dispositioned in accordance with ASME Code and approved procedures. The qualifications of all NDE technicians performing the inspections were verified to be current.

None of the above observed or reviewed NDE examinations identified any relevant indications and cognizant licensee personnel stated that no relevant indications were accepted by the licensee for continued service.

One example of welding on the high pressure coolant injection system and two examples of welding on the main steam system were examined through direct observation and record review as follows:

System	Component/Weld Identification
High Pressure Coolant Injection	Vent Valve Assembly 258
Main Steam	MSIV 17, Weld MW 7081
Main Steam	MSIV 17, Weld FO28A

The inspectors verified, by review, that the welding procedure specifications and the welders had been properly qualified in accordance with ASME Code, Section IX, requirements. The inspectors also verified, through observation and record review, that essential variables for the gas tungsten arc welding process (machine and manual) were identified, recorded in the procedure qualification record, and formed the bases for qualification of the welding procedure specifications.

Since the WO for the replacement of the main steam isolation valves specified postweld heat treatment, the inspectors reviewed the certified material test reports representing the welding materials being used on the main steam isolation valves. This review determined that the welding materials had been qualified in the as-welded condition as well as the postweld heat treated condition.

The inspectors completed one sample under Section 02.01.

b. Findings

No findings of significance were identified.

02.02 Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

The licensee performed the required visual inspection of pressure-retaining components above the reactor pressure vessel (RPV) head. The results of this inspection confirmed that there was no evidence of leaks or boron deposits on the surface of the RPV head or related insulation. The personnel performing the visual inspection were certified as Level II and Level III VT-2 examiners. The inspectors performed an independent partial visual inspection of the RPV flange area and the related insulation and did not observe any evidence of boron deposits.

The inspectors completed one sample under Section 02.02.

b. Findings

No findings of significance were identified.

02.03 Boric Acid Corrosion Control Inspection Activities

a. Inspection Scope

The inspectors evaluated the implementation of the licensee’s boric acid corrosion control program for monitoring degradation of those systems that could be adversely affected by boric acid corrosion.

The inspection procedure required review of a sample of boric acid corrosion control walkdown visual examination activities through either direct observation or record review. The inspectors reviewed the documentation associated with the licensee’s boric acid corrosion control walkdown as specified in Procedure STN PE-040D, “RCS-Pressure Boundary Integrity-Walkdown,” Revision 2. Visual records of the components and equipment were also reviewed by the inspectors. Additionally, the inspectors independently performed examinations of piping and components containing boric acid during a walkdown of the containment and auxiliary building. The inspection procedure required verification that visual inspections emphasize locations where boric acid leaks can cause degradation of safety-significant components. The inspectors verified through direct observations and by program/record review that the licensee's boric acid corrosion control inspection efforts are directed towards locations where boric acid leaks can cause degradation of safety-related components. On those components where boric acid was identified, the engineering evaluations gave assurance that the ASME Code wall thickness limits were properly maintained. The evaluations also confirmed that the corrective actions performed for evidence of boric acid leaks were consistent with requirements of the ASME Code.

The inspection procedure required both a review of one to three engineering evaluations performed for boric acid leaks found on reactor coolant system piping and components, and one to three corrective actions performed for identified boric acid leaks. The inspectors reviewed three engineering evaluations: (1) BBHV8141D, reactor coolant Pump D seal water outlet isolation valve; (2) RBB06, incore seal table; and (3) BBHV8000A, pressurizer power operated relief inlet isolation Valve PCV-455A. The inspectors reviewed two corrective action plans for two engineering evaluations where boric acid leakage was confirmed: (1) BBHV8000A, pressurizer power operated relief inlet isolation Valve PCV-455A and (2) RBB06, incore seal table. The evaluations appropriately addressed the causes and corrective actions, and were generally consistent with industry standards.

The inspectors completed one sample under Section 02.03.

b. Findings

No findings of significance were identified.

02.04 Steam Generator Tube Inspection Activities

a. Inspection Scope

The inspection procedure specified performance of an assessment of in situ screening criteria to assure consistency between assumed NDE flaw sizing accuracy and data from the Electric Power Research Institute (EPRI) examination technique specification sheets. It further specified assessment of appropriateness of tubes selected for in situ pressure testing, observation of in situ pressure testing, and review of in situ pressure test results.

At the time of this inspection, no conditions had been identified that warranted in situ pressure testing. The inspectors did, however, review the licensee's Report SG SGDA 06-039, Wolf Creek Refueling Outage 15 condition monitoring and operational assessment dated October 2006, and compared the in situ test screening parameters to the guidelines contained in the EPRI document "In Situ Pressure Test Guidelines," Revision 2. This review determined that the remaining screening parameters were consistent with the EPRI guidelines.

In addition, the inspectors reviewed both the licensee site-validated and qualified acquisition and analysis technique sheets used during this refueling outage and the qualifying EPRI examination technique specification sheets to verify that the essential variables regarding flaw sizing accuracy, tubing, equipment, technique, and analysis had been identified and qualified through demonstration. The inspector-reviewed acquisition technique and analysis technique sheets are identified in the attachment.

The inspection procedure specified comparing the estimated size and number of tube flaws detected during the current outage against the previous outage operational assessment predictions to assess the licensee's prediction capability. The inspectors compared the previous outage operational assessment predictions contained in Report SG-SGDA-06-039, with the flaws identified thus far during the current steam generator tube inspection effort. Compared to the projected damage mechanisms identified by the licensee, the number of identified indications fell within the range of prediction and was quite consistent with predictions.

The inspection procedure specified confirmation that the steam generator tube eddy current test scope and expansion criteria meet TS requirements, EPRI guidelines, and commitments made to the NRC. The inspectors evaluated the recommended steam generator tube eddy current test scope established by TS requirements. The inspectors compared the recommended test scope to the actual test scope and found that the licensee had accounted for all known flaws and had, as a minimum, established a test scope that met TS requirements, EPRI guidelines, and commitments made to the NRC. The scope of the licensee's eddy current examinations of tubes in both steam generators included:

- Bobbin examination full length of tubing (TEH-TEC) from both hot and cold legs
- Penetrant testing (PT) examination of cold leg top of tubesheet
- PT examination of hot leg tubesheet
- PT hot leg full-depth tubesheet
- PT Row 1 and 2, U-bend
- PT hot leg tube support plate and freespan (special interest)
- PT cold leg tube support plate and freespan (special interest)

During the steam generator tube inspections, indications were identified in the portion of the tubes located in the bottom 1 inch of the hot leg side of the tubesheet. Two tubes in Steam Generator B and one tube in Steam Generator C contained circumferential indications that exceeded the alternate repair criteria of more than 94 degrees in the bottom 1 inch of the tubesheet. This necessitated expanding the scope of inspection to include 20 percent of the tubes in the bottom 1 inch of the hot leg side of the tubesheet in Steam Generators A and D.

The results, as known to the inspectors at the conclusion of this inspection, are as follows:

The initial analysis for Steam Generator A tubes indicated four potential indications in the bottom 1 inch of the hot leg side of the tubesheet. Based on that, the licensee made a decision to expand the tube inspection scope of Steam Generator A to 100 percent. Subsequent to the implementation of that decision, the lead analyst completed review of the four potential indications and determined that they were not considered to be reportable degradation. A decision was made to halt further tube inspections in Steam Generator A. The delayed review resulted in an increase of tubes inspected over the

20 percent expanded scope to an additional 10.5 percent tube inspection; thus a total of 30.5 percent of steam generator tubes were inspected in the bottom 1 inch of the hot leg side of the tubesheet with no tubes requiring plugging in Steam Generator A.

With respect to Steam Generator B, as mentioned above four circumferential indications were found that exceeded the alternate repair criteria of more than 94 degrees in the bottom 1 inch of the hot leg side of the tubesheet and consequently were plugged. Because of anti-vibration bar (AVB) wear on 13 additional tubes, a total of 17 tubes were planned for plugging in Steam Generator B.

With respect to Steam Generator C, one circumferential indication was found that exceeded the alternate repair criteria of more than 94 degrees in the bottom 1 inch of the hot leg side of the tubesheet and consequently was plugged. Because of AVB wear on seven additional tubes, a total of eight tubes were planned for plugging in Steam Generator C.

With respect to Steam Generator D, one circumferential indication exceeding the alternate repair criteria of more than 94 degrees was identified in the bottom 1 inch of the hot leg side of the tubesheet of the 20 degrees expanded sample. The licensee, at that point, made a further scope expansion from 20 to 100 percent of the tube portions located within the bottom 1 inch of the hot leg side of the tubesheet. Two additional circumferential indications exceeding the alternate repair criteria of more than 94 degrees were found, for a total of three tubes in this steam generator that were planned to be plugged. No AVB wear was found on this steam generator.

The inspection procedure specified that, if new degradation mechanisms were identified, the licensee would verify the analysis fully enveloped the problem of the extended conditions including operating concerns and that appropriate corrective actions were taken before plant startup. To date, the only new degradation mechanism identified by the eddy current examination results in the bottom 1 inch of the tubesheet was the above discussed degradation.

The inspection procedure required confirmation that the licensee inspected all areas of potential degradation, especially areas that were known to represent potential eddy current test challenges (e.g., top of tubesheet, tube support plates, and U-bends). The inspectors confirmed that all known areas of potential degradation were included in the scope of inspection and were being inspected.

The inspection procedure further required verification that repair processes being used were approved in the TSs. At the time of this inspection, it was estimated that a total of approximately 25 tubes would be plugged. The inspectors verified that the mechanical expansion plugging process to be used was an NRC-approved repair process. At the completion of the inspection, the inspectors were informed that a total of 28 tubes were plugged.

The inspection procedure also required confirmation of adherence to the TS plugging limit, unless alternate repair criteria had been approved. The inspection procedure further requires determination whether depth sizing repair criteria were being applied for indications other than wear or axial primary water stress corrosion cracking in dented tube support plate intersections. The inspectors determined that the TS plugging limits were being adhered to (i.e., 40 percent maximum through-wall indication).

If steam generator leakage greater than three gallons per day was identified during operations or during post shutdown visual inspections of the tubesheet face, the inspection procedure required verification that the licensee had identified a reasonable cause based on inspection results and that corrective actions were taken or planned to address the cause for the leakage. The inspectors did not conduct any assessment because this condition did not exist.

The inspection procedure required confirmation that the eddy current test probes and equipment were qualified for the expected types of tube degradation and an assessment of the site-specific qualification of one or more techniques. The inspectors observed portions of eddy current tests performed on the tubes in Steam Generators B and C. During these examinations, the inspectors verified that: (1) the probes appropriate for identifying the expected types of indications were being used, (2) probe position location verification was performed, (3) calibration requirements were adhered to, and (4) probe travel speed was in accordance with procedural requirements. The inspectors performed a review of site-specific qualifications of the techniques being used. These are identified in the attachment.

If loose parts or foreign material on the secondary side were identified, the inspection procedure specified confirmation that the licensee had taken or planned appropriate repairs of affected steam generator tubes and that they inspected the secondary side to either remove the accessible foreign objects or perform an evaluation of the potential effects of inaccessible object migration and tube fretting damage. At the time of this inspection, four objects were found: 1) two screws (Size 2 with 0.085 inch diameter and 0.75 inch length), 2) a probe head from the camera (1.3 inches wide by 0.213 inches thick by 0.486 inches long), and 3) a piece of inspection tooling (approximately 0.025 inches in diameter). The inspection tooling piece was removed but the remaining items could not be recovered. The required chemical and mechanical effects of these remaining pieces were analyzed with the conclusion of negligible effects on the respective steam generators.

Finally, the inspection procedure specified review of one-to-five samples of eddy current test data if questions arose regarding the adequacy of eddy current test data analyses. The inspectors did not identify any results where eddy current test data analyses adequacy was questionable.

The inspectors completed one sample under Section 02.04.

b. Findings

No findings of significance were identified.

02.05 Identification and Resolution of Problems

a. Inspection scope.

The inspection procedure required review of a sample of problems associated with inservice inspections documented by the licensee in the corrective action program for appropriateness of the corrective actions.

The inspectors reviewed 30 CRs which dealt with inservice inspection activities and found the corrective actions were appropriate. The specific CRs reviewed are listed in the documents reviewed section. In the area of inservice inspection activities, the inspectors concluded that the licensee had an appropriate threshold for entering issues into the corrective action program and had procedures that direct a root cause evaluation when necessary. The licensee also effectively reviewed and applied industry operating experience in this area.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

The inspectors observed testing and training 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 training scenario involved:

- June 26, 2008, increasing steam generator tube leak resulting in a Notice of Unusual Event (NOUE)

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

Routine Quarterly Evaluations (71111.12Q)

a. Inspection Scope

The inspectors reviewed the maintenance activities listed below to: (1) verify the appropriate handling of structures, systems, and components (SSCs) performance or condition problems; (2) verify the appropriate handling of degraded SSCs functional performance; (3) evaluate the role of work practices and common cause problems; and (4) evaluate the handling of SSCs issues reviewed under the requirements of the maintenance rule, 10 CFR Part 50, Appendix B, and TSs.

- June 19, 2008, KJ01B, Emergency Diesel Generator (EDG) B output breaker trip
- June 27, 2008, auxiliary building room cooling Function GL-5
- June 29, 2008, SGN01B, containment cooler Fan B tripped in fast speed

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed three samples.

b. Findings

Introduction. A Green NCV of 10 CFR 50.65(a)(1) was identified by the inspectors for failure to establish a(1) goals for the safety-related room coolers and monitor room cooler performance against those goals.

Description. The room coolers use ESW to cool the safety-related pump rooms. The maintenance rule function for these coolers is to remove a calculated amount of heat from each pump room and this is considered the GL-5 maintenance rule function. The licensee's maintenance rule program required that room cooler unavailability time be assigned to the associated pump room and count towards the pump's unavailability goal per 10 CFR 50.65 a(2). If the pump's unavailability goal is exceeded, both the room cooler (GL-5 function) and pump maintenance rule function are affected.

On May 5, 2005, the Train A RHR pump accumulated enough unavailability time to exceed the 10 CFR 50.65 a(2) goal due to a 0.5 gpm through-wall leak on Room Cooler SGL10A. Unavailability time was previously incurred for RHR A due to preventive maintenance. The licensee wrote Performance Improvement Request (PIR) 2005-2507 on August 31, 2005, to document that the maintenance rule expert panel moved the room cooler function to a(1) status. PIR 2005-2507, Action Item 4, required the expert panel to establish a(1) monitoring goals with a monitoring duration by June 30, 2006.

Wolf Creek performed a 10 CFR 50.65 a(3) review on April 27, 2007, to determine if the room cooler performance was disproportionate to its established goals. Wolf Creek determined that its current cooler replacement schedule was appropriate. The April 27, 2007, expert panel meeting minutes, in part, state that a(1) goals had not been established because all of the room coolers had not been replaced and after all room coolers are replaced, that a(1) goals and monitoring will be implemented in the future. Inspectors questioned this practice of only monitoring for performance after corrective action rather than before and after corrective action. Thus, no technically justified goals were established.

The inspectors reviewed the performance history and noted that previous causal determinations had concluded that tube wall thinning and pit corrosion were/are causing leaks. The plant had a history of room cooler tube leaks, H-bend leaks, and O-ring failures. A prior 2004 corrective action to reduce the maximum allowable heat exchanger tube flaw depth from 100 to 80 percent was unsuccessful in preventing through-wall leaks. PIR 2005-2507, Action Item 2, had a deadline for Room Cooler SGL09A (Safety Injection A pump room) to be replaced by June 30, 2006, however, this replacement was deferred until January 29, 2008. On October 17, 2007, Room Cooler SGL09A experienced a room cooler leak. SGL09A was actually replaced on January 28, 2008.

SGL10A (RHR A pump room) was deferred from June 30, 2006, until December 15, 2006, because Wolf Creek installed Room Cooler SGL10B (RHR B pump room) backwards and, thus, caused the licensee problems. Wolf Creek believed that room cooler replacement would take longer than the associated pump's TS allowed outage time and thus the work needed to be performed during a plant shutdown. Room Cooler SGL10A was replaced on October 22, 2006.

PIR 2005-2507, Action Item 2 had a deadline for Room Cooler SGL12A (Centrifugal Charging Pump A pump room) to be replaced by June 30, 2006, however its replacement was deferred to April 2, 2007, due to time needed to replace the cooler. On October 6, 2006, SGL12A experienced an O-ring leak which was a mechanical joint leakage. On February 13, 2008, Room Cooler SGL12A experienced a through-wall leak. Room Cooler SGL12A was replaced on April 25, 2008, during Refueling Outage 16. These leaks were not counted against any a(1) goal.

The inspectors determined that the replacement plan did not implement maintenance activities, which would improve the availability of the systems. This was contrary to the guidance in NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3, which states that while waiting to implement modifications, increased preventive maintenance may be necessary to ensure the affected function will remain reliable.

Contrary to PIR 2005-2507, Action Item 4 (due June 30, 2006), no 10 CFR 50.65 (a)(1) monitoring goals and duration were established until April 27, 2008. On April 27, 2008, Wolf Creek established the goal of zero through-wall leaks from April 23, 2008, to the end of Refueling Outage 17 in October 2009. If this goal is met, Wolf Creek will return the room cooler function to 10 CFR 50.65 a(2) status. The inspectors questioned the process of considering the GL-5 function as a(1) status for 3 years of corrective actions with no a(1) monitoring goals in the intervening time. After inspector questioning in February 2008, Wolf Creek has expedited room cooler procurement and replacement. The NRC Enforcement Manual, Section 7.11.1.a.1(a), specifies that failure to establish a(1) performance goals is a violation of 10 CFR 50.65. The NRC inspectors also found that Wolf Creek Procedure AP 23M-001, "WCGS Maintenance Rule Program," Revision 6, Step 6.1.9, requires a(1) goals be established when performance under a(2) cannot be demonstrated. Procedure AP 23M-001 then defers to the corrective action program where inspectors found that the issue was not adequately addressed.

Analysis. Failure to establish monitoring goals in response to this system's unreliable performance and classification as Maintenance Rule (a)(1) was a performance deficiency. This finding is more than minor because it is consistent with IMC 612, Appendix E, Example 7.a. Specifically, Wolf Creek failed to establish a(1) goals and monitor performance against those goals for the a(1) GL-5 function for 3 years. The inspectors evaluated the significance of this finding using Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," and determined that this finding is Green. Specifically, the support function (GL-5) to cool pump rooms does not result in a total loss of any safety function as identified by the licensee's probability risk assessment that contributes to external event initiated core damage accident sequences (i.e., initiated by a seismic, flooding, or severe weather event). The inspectors determined that this finding has a crosscutting aspect in the problem identification and resolution area associated with corrective action program because Wolf Creek failed to take appropriate corrective actions to address this safety issue and the adverse room cooler trends in a timely manner, commensurate with safety significance and complexity [P.1(d)].

Enforcement. Title 10 CFR 50.65 requires, in part, when performance of SSCs cannot be demonstrated per Paragraph a(2), that performance goals and corrective action shall be established under Paragraph a(1). On May 5, 2005, Wolf Creek safety-related room cooler Function GL-5 exceeded its a(2) monitoring goals. Contrary to the above, between May 5, 2005, and April 27, 2008, Wolf Creek failed to establish a(1) goals for the safety-related room cooler Function GL-5. The inspectors identified that an a(1) goal was not established for approximately 3 years while Wolf Creek continued to experience room cooler failures. This was a violation of 10 CFR 50.65(a)(1). Because this violation was of very low safety significance and was entered into the licensee's corrective action program under CR 2008-3145, this violation is being treated as an NCV in accordance with the NRC Enforcement policy: NCV 05000482/2008003-02, Failure to Establish Goals and Monitor for a(1) Emergency Core Cooling System (ECCS) Room Coolers.

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

a. Inspection Scope

Risk Assessment and Management of Risk

The inspectors reviewed the licensee's evaluation and management of plant risk for the maintenance and emergent work activities affecting risk-significant and safety-related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- Refueling Outage 16 risk assessment
- May 14, 2008, weekly T-0 risk assessment profile for power ascension
- June 3, 2008, weekly T-0 risk assessment profile for EDG A outage

These activities were selected based on their potential risk-significance relative to the reactor safety cornerstones. As applicable for each activity, the inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4) and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed TS requirements and walked down portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met.

Emergent Work Control

For the emergent work activities listed below, 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 corrective action program to determine if the licensee identified and corrected risk assessment and emergent work control problems.

- June 18, 2008, repairs to body to union nut leak on letdown throttle Valve BGV-003

The inspectors completed four samples.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors: (1) reviewed plant 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 TSs; (5) used the significance determination process 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.

- April 25, 2008, corrosion of ESW piping in 1974’ elevation of the control building
- May 5, 2008, permanent removal of RHR heat exchanger piping insulation
- May 9, 2008, gap in containment Sump A filter surface
- June 2, 2008, containment spray header contact with the containment polar crane
- June 27, 2008, undersized containment sump flow orifices

The inspectors completed five samples.

b. Findings

Introduction. The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion V, “Procedures, Instructions, and Drawings,” for Wolf Creek’s failure to specify acceptance criteria in its containment sump inspection procedure which led to unidentified gaps in containment Sump A.

Description. During a Mode 4 containment walkdown on May 9, 2008, the inspectors identified a gap in containment Sump A not previously identified by Wolf Creek. Based on previous Engineering Disposition 12684, the gap acceptance criteria was 0.045 inch. The gap that the inspectors identified was 1/8-inch wide by1/2-inch tall on one of the upper sump strainers. After raising the issue to the control room, Wolf Creek declared containment Sump A inoperable and entered TS 3.5.3. RHR Train B was already inoperable for maintenance. Wolf Creek subsequently entered TS 3.0.3, and repaired the sump.

Wolf Creek Procedure STS EJ-003, "Containment Sump Inspection," Revision 14, Step 8.1, contains no guidance on filter screen gap acceptance criteria, other than "verify no evidence of structural distress." Wolf Creek last implemented STS EJ-003 during their May 7, 2008, walkdown prior to ascending from Mode 5 to Mode 4. The inspectors considered this a missed opportunity as Wolf Creek should have identified these deficiencies prior to entering Mode 4. Although the inspectors could not determine with complete certainty that the sump screen gap existed at the time of Wolf Creek's walkdown on May 7, Wolf Creek was not able to identify any work activity performed in the recirculation sump area since that time.

Wolf Creek performed a subsequent engineering evaluation to determine the impact of the sump gap. Wolf Creek determined that this screen bypass flowpath and postulated debris, degraded but did not cause a loss of operability for the ECCS or the containment spray system. The inspectors reviewed Wolf Creek's operability determination and the applicable USAR sections to ensure that operability was justified and that potentially affected ECCS components and containment spray remained available and capable of performing their respective design functions. Wolf Creek's evaluation took credit for Number 7 and 8 gage solid stainless steel support wiring beneath the perforated stainless steel sheet metal which is the filter surface. This support wiring 'zig-zags' around the circumference of each strainer module. This previously uncredited support wire provided additional filtration that appreciably narrowed the identified gap, but it was still greater than 0.045 inch. The ingestion of such debris was evaluated by Wolf Creek. The containment spray nozzle openings are 7/16-inch and would not be affected. However, debris would affect the graphite pump seals for the safety injection pumps. Wolf Creek found that degradation of the seals during long-term core cooling would progressively occur and lead to seal leakage, but the pumps would still supply enough flow to assure core cooling. Wolf Creek also credited a Westinghouse analysis to demonstrate that ingested debris would not block narrow passages in the fuel assemblies. The inspectors reviewed these evaluations and found them to be acceptable.

Analysis. The failure to identify containment sump gaps greater than the acceptance criteria prior to Mode 4 which had the potential to impact operability is a performance deficiency. The inspectors determined that this finding was more than minor because it affected the procedure quality and human performance attributes of the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events and prevent undesirable consequences. The inspectors determined that this finding was of very low safety significance, Green, using the significance determination process Phase 1 screening worksheet for mitigating systems. Specifically, the deficiency did not result in the loss of operability or functionality and did not represent a risk significant external event such as flooding. The inspectors determined that the cause of this finding has a human performance crosscutting aspect in the area associated with resources. Specifically, Wolf Creek did not ensure that Procedure STS EJ-002 was adequate to ensure nuclear safety including complete, accurate, and up-to-date specifications or acceptance criteria for the sump [H.2(c)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, "Procedures, Instructions, and Drawings," requires, in part, that procedures shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, on May 7, 2008, Procedure STS EJ-003, "Containment Sump Inspection," Revision 14, was utilized to inspect containment Sump A and allowed a gap greater than the acceptance criteria to pass inspection because it failed to specify any acceptance criteria. However, because of the very low safety significance and because the issue was entered into Wolf Creek's corrective action program as CR 2008-002219, this finding is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000482/2008003-03, Inadequate Containment Sump Inspection Procedure.

1R18 Plant Modifications (71111.18)

Temporary Modification Review

a. Inspection Scope

The inspectors reviewed plant drawings, procedure requirements, and TSs to ensure that the below temporary modification was 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 SSC's 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.

- March 27, 2008, TMO 08-008-KE, refuel crane festoon cable tape

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors selected the 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.

- April 12, 2008, containment sump valves after limitorque maintenance
- April 19, 2008, safety injection pump motor replacement
- May 19, 2008, main steam and feedwater isolation valve replacement
- June 5, 2008, EDG A run following maintenance
- June 18, 2008, EDG B run following maintenance

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed five samples.

b. Findings

No findings of significance were identified.

1R20 Outage Activities (71111.20)

Refueling Outage Activities

a. Inspection Scope

The inspectors reviewed the outage safety plan and contingency plans for Wolf Creek Refueling Outage 16 that started on March 22, 2008, and ended on May 14, 2008, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense in-depth. During the refueling outage, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below.

- Licensee configuration management, including maintenance of defense in-depth commensurate with the outage safety plan for key safety functions and compliance with the applicable TSs when taking equipment out of service.

- Implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing.

- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication and accounting for instrument error.

- Controls over the status and configuration of electrical systems to ensure that TSs and outage safety plan requirements were met, and controls over switchyard activities.

- Monitoring of decay heat removal processes, systems, and components.

- Controls to ensure that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system.

- Reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss.

- Controls over activities that could affect reactivity.

- Maintenance of secondary containment as required by TSs.

- Refueling activities, including fuel handling and sipping to detect fuel assembly leakage.

- Licensee identification and resolution of problems related to refueling outage activities.

- Initial containment entry including normally inaccessible areas behind the bio-shield wall.

- Reactor vessel head de-tensioning and removal.

- Removal of reactor upper internals.

- Core offload (observed from containment and control room).

- Outage risk assessment techniques including assessment of decay heat removal/spent fuel pool cooling, availability of offsite power, reactivity control and control of containment.

- Core reload (observed from control room and fuel handling platform).

- Reactor vessel head set (observed from outage control center).

- Reactor coolant system fill and vent including reduced inventory or midloop conditions including adherence to commitments in response to Generic Letter 88-17.

- Reactor coolant system heatup and establishing bubble in pressurizer.

- Containment closure walkdown including inspection of containment recirculation sumps.

- Under reactor vessel inspection.

- Reactor startup, physics testing, and review of core operating limits report.

- Verification of core reload (by video).

Documents reviewed are listed in the attachment.

The inspectors completed one sample.

b. Findings

Introduction. The inspectors identified a violation of TS 5.4.1.a, in which, Wolf Creek raised the winch load setpoint for its fuel transfer system to avoid trips without knowing the cause.

Description. During core reload, Wolf Creek experienced repeated trips of the fuel handling system winch. It was not until after NRC involvement that it was identified that the winch load setpoints were inappropriately altered. The inspectors reviewed the fuel transfer system vendor Document M-716-00787, “Instruction Manual for Transfer Machine Operations Manual,” Revision W02, and found that it is controlled by Wolf Creek as a design document, and it is accepted by Wolf Creek engineering under Design Change Package Number 07020. Inspectors found that the "KE" fuel transfer system is designated as being “controlled” under Attachment D to Procedure AP 05-005, “Design, Implementation, & Configuration Control of Modifications,” Revision 12. Under M-716-00787, Section E, “Factory Acceptance Test Wolf Creek Transfer Machine and Refueling Machine Upgrades,” Wolf Creek accepted the 250 pound/1 second setpoint to verify “proper functional performance of the transfer machine control circuitry” and “cable load interlocks.” It also accepted the slow speed zone of 590 inches to verify “proper functional performance of the transfer machine control circuitry.”

Contrary to the above, the inspectors found that on April 17, 2008, using WO 08-305599-000, the load setpoints and slow speed zones were inappropriately changed from 250 pounds/1 second and 590 inches, to 300 pounds/2 seconds and 585 inches, respectively. The inspectors found that under M-716-00787, Section-G, “Software Change Log,” no changes to the winch load limits or slow speed zones were referenced. Inspectors were unable to locate, and Wolf Creek was unable to produce, modification documentation that justified these software changes.

The inspectors found troubleshooting WO 08-305599-000 was amended three times for the fuel transfer system. After the discovery that the setpoints were inappropriately changed, the 250 pounds/1 second and 590 inches were loaded into the EEPROM (nonvolatile memory for the programmable logic controller) at 4 a.m. on April 18. Power to the fuel transfer system was cycled and the speed change for the cart was observed at 590 inches. On this basis, Wolf Creek believed that the settings had been correctly re-established, and fuel moves continued.

During subsequent fuel moves, it was reported that the winch load was greater than 270 pounds, sustained, without winch trips and that the slow zones were also greater than normal. This suggested that the correct settings were not loaded into the EEPROM. After fuel reload was complete, the vendor representative performed a “read only” check of the fuel transfer system logic and found that the transfer system did not match the settings on his laptop computer. The vendor downloaded the EEPROM data to the laptop. The vendor representative found that the 300 pound/2 second setting was still in the fuel transfer system, but Wolf Creek engineering could not inform the inspectors what the vendor representative found with regard to the slow speed zone setting. The settings on the vendor’s laptop computer should have reflected the 250 pound/1 second and 590 inch settings. Subsequently, the fuel transfer system engineer was informed of the vendor's findings, and the system engineer connected the laptop computer to the fuel transfer system. The system engineer found the 250 pound/1 second setpoint, but found that the slow speed zone still at 585 inches. The system engineer then wrote the correct setpoints to the EEPROM. The system engineer powered the fuel transfer system off and on twice and found the setpoints to be correct at 11:42 p.m. on April 18.

During interviews, Wolf Creek engineering postulated that the change to the setpoints (to change back to the original 250 pounds/1 second and 590 inch slow zone) at 4 a.m. on April 18 was written to the transfer system's RAM and not the EEPROM. Thus, if power to the fuel transfer system was cycled, the incorrect setpoints would be read into the programmable logic controller RAM from the EEPROM when the system boots. This theory contradicts the postmaintenance testing on April 18 at 4 a.m. because the subject WO states that power was cycled on and off to the transfer machine. Wolf Creek engineering has only been able to state that: "At the time on April 18 at 4 a.m., it was believed that the setpoints were correct." Planning is ongoing with the vendor to reconstruct how the settings were not correctly changed.

Inspectors reviewed the various logs for the timing of fuel movements to determine if fuel was moved after concerns were raised and/or after the NRC contacted the Wolf Creek control room. The inspectors determined that fuel moves were performed after concerns were raised as well as after the NRC contacted Wolf Creek. The inspectors determined that six fuel assemblies were moved after concerns were raised regarding this issue.

The shift manager informed the resident inspectors that the adjustment of the 250 pound/1 second setpoint was "administrative" in nature and that the design setpoints were at 600 pounds instantaneous, and 250 pounds/5 seconds. It was later reported to the control room that, in fact, the 250 pound/1second setpoint was not administrative and was controlled via the vendor technical manual. Subsequently, inspectors reviewed the vendor technical manual for the various setpoints and found no information to support Wolf Creek's initial claim that there were "administrative" and "design" setpoints. The inspectors found no distinction between the two in the vendor technical manual. The inspectors found that all of the subject setpoints were controlled by design because on April 2, 1999, the design of the fuel transfer system was changed which incorporated the vendor technical manual under Design Change Package Number 07020.

Lastly, after refueling, dry inspections of the fuel transfer cart were performed in the fuel building. These inspections found that the anti-tip latch had worn bushings causing it to engage slowly and movement of the cart may have been initiated prior to full disengagement. Small metal burrs were found on the anti-motion latches which also may have contributed to the winch trips. The anti-tip latch is key to ensuring movement of the fuel cart only when the cart is fully horizontal to prevent contact with the fuel transfer tube. The anti-motion latch ensures that the fuel cart is fully engaged in the upender prior to moving the assembly to the vertical position.

<u>Analysis</u>. The inspectors determined that failure to follow the modification process under AP 05-005 for the fuel transfer system is a performance deficiency. Traditional enforcement does not apply since there were no actual safety consequences or potential for impacting the NRC's regulatory function, and this finding was not the result of any willful violation of NRC requirements or Wolf Creek procedures. The inspectors

determined that this finding was more than minor because it is associated with the human performance attribute for the barrier integrity cornerstone; and, it affected the cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding) protect the public from radio nuclide releases caused by accidents or events. Specifically, this issue relates to the procedure adherence example of the human performance attribute because the design process was bypassed to mask fuel cart problems.

The inspectors evaluated the significance of this finding using Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," and determined that this finding was of very low safety significance because the issue did not result in fuel handling errors that caused damage to fuel clad integrity or a dropped fuel assembly. Additionally, the cause of this finding has human performance crosscutting aspects in the area associated with decision making. Specifically, Wolf Creek did not ensure safety by making safety or risk-significant decisions by using any procedural or systematic process when faced with the unexpected and repeated fuel transfer cart winch trips [H.1(a)].

Enforcement. TS 5.4.1.a requires, in part, that procedures shall be established, implemented, and maintained as recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A, Section 1, "Administrative Procedures," requires, in part, that "temporary change activities" be performed in accordance with written procedures appropriate to the circumstances. Wolf Creek Procedure AP 05-005, "Design, Implementation, & Configuration Control of Modifications," Revision 12, implements this requirement. Additionally, Procedure AP 05-005, Attachment D, designates the "KE" fuel transfer system as being controlled.

Contrary to the above, the licensee did not implement design control Procedure AP 05-005 which controls modifications to the fuel transfer system and the controlled vendor technical documentation. Specifically, on April 17, 2008, Wolf Creek did not implement its design change process to ensure that the fuel transfer cart loading setpoints were adequately maintained. Because this issue was of very low safety significance and has been entered into the licensee's corrective action program as CR 2008-000255 and 2008-000254, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000482/2008003-04, Troubleshooting Activities Bypass Design Control for the Fuel Transfer System.

1R22 Surveillance Testing (71111.22)

.1 Routine Surveillance Testing

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- April 21, 2008, comprehensive test of containment spray Pump A
- April 22, 2008, integrated engineered safety features actuation signal Train B testing
- April 23, 2008, comprehensive test of containment spray Pump B
- April 30, 2008, integrated engineered safety features actuation signal Train A testing

The inspectors observed in-plant activities and reviewed procedures and associated records to determine whether: any preconditioning occurred; effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing; acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis; plant equipment calibration was correct, accurate, and properly documented; as-left setpoints were within required ranges; the calibration frequency was in accordance with TSs, the USAR, procedures, and applicable commitments; measuring and test equipment calibration was current; test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied; test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used; test data and results were accurate, complete, within limits, and valid; test equipment was removed after testing; where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable; where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure; where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished; prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test; equipment was returned to a position or status required to support the performance of the safety functions; and all problems identified during the testing were appropriately documented and dispositioned in the corrective action program.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed four samples

b. Findings

No findings of significance were identified.

.2 Inservice Testing Surveillance

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- June 11, 2008, turbine-driven auxiliary feedwater surveillance

The inspectors observed in-plant activities and reviewed procedures and associated records to determine whether: any preconditioning occurred; effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing; acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis; plant equipment calibration was correct, accurate, and properly documented; as-left setpoints were within required ranges; and the calibration frequency were in accordance with TSs, the USAR, procedures, and applicable commitments; measuring and test equipment calibration was current; test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied; test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used; test data and results were accurate, complete, within limits, and valid; test equipment was removed after testing; where applicable for inservice testing activities, testing was performed in accordance with the applicable version of Section XI, ASME Code, and reference values were consistent with the system design basis; where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable; where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure; where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished; prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test; equipment was returned to a position or status required to support the performance of its safety functions; and all problems identified during the testing were appropriately documented and dispositioned in the corrective action program.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.3 Reactor Coolant System Leak Detection Surveillance

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- June 18, 2008, reactor coolant system leakage surveillance calculation due to pressure isolation check valve leakage

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

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

a Inspection Scope

The inspectors performed an in-office review of Revision 11 to Wolf Creek Generating Station Form APF 06-002-01, "Emergency Action Levels," submitted March 18, 2008. This revision added the condition, "Safety injection initiated," as an option for entering Emergency Action Level 3, "Loss of Reactor Coolant Boundary," added the condition, "Annunciator 00-98-E, Seismic Recorder On, In Alarm," to the entry conditions of Box 11-NP4, in Emergency Action Level 11, "Natural Phenomena," and moved the condition, "Annunciator 00-98D, OBE, In Alarm," to Box 11-NP5, restoring Emergency Action Level 11 to its original revision.

The revision was compared to the 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 the criteria of Nuclear Energy Institute (NEI) Report 99-01, "Methodology for Development of Emergency Action Levels," Revision 2, and to the standards in 10 CFR 50.47(b) to determine if the revision adequately implemented the requirements of 10 CFR 50.54(q). This review was not documented in a safety-evaluation report and did not constitute approval of licensee changes to their emergency action levels, therefore, these revisions are subject to future inspection.

The inspectors completed one sample.

b. Findings

No findings of significance were identified

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

Cornerstone: All

The inspectors performed a review of the data submitted by the licensee for the fourth quarter 2007, performance indicators for any obvious inconsistencies prior to its public release in accordance with IMC 0608, "Performance Indicator Program."

Documents reviewed are listed in the attachment.

This review was performed as part of the inspectors' normal plant status activities and, as such, did not constitute a separate inspection sample.

Cornerstone: Barrier Integrity

The inspectors sampled licensee submittals for the two performance indicators listed below for the period, September 30, 2005, through January 1, 2007. The definitions and guidance of NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, 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. The inspectors: (1) reviewed reactor coolant system chemistry sample analyses for dose equivalent Iodine-131 and compared the results to the TS limit; (2) observed a chemistry technician obtain and analyze a reactor coolant system sample; (3) reviewed operating logs and surveillance results for measurements of reactor coolant system identified leakage; and (4) observed a surveillance test that determined reactor coolant system identified leakage.

- Reactor coolant system specific activity
- Reactor coolant system leakage

The inspectors completed two samples during this inspection.

Cornerstone: Initiating Events

The inspectors sampled licensee submittals for the unplanned scrams with complications performance indicator for the period from the first quarter 2007 through the first quarter 2008. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in Revision 5 of the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports and NRC Integrated Inspection reports for the period of March 31, 2007 through March 31, 2008, to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified.

- Unplanned scrams with complications

The inspectors completed one sample.

b. Findings

No findings that were more than minor were identified; however, the inspectors identified that Wolf Creek failed to count the March 17, 2008, scram as complicated. All condensate pumps were lost because both transformers that can provide power to the condensate pump motors were lost. One transformer was lost due to the inadequate maintenance (described in 4OA5) and the other transformer was out of service due to planned maintenance. Since normal feedwater could not be re-established within 30 minutes without performing maintenance, NEI 99-02 considers this a scram with complications. Including the March 17 scram, the performance indicator remains Green and, as such, this deficiency is considered minor. Wolf Creek has captured this issue in CR 2008-002938.

4OA2 Identification and Resolution of Problems (71152)

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, Occupational Radiation Safety, and Physical Protection

.1 Routine Review of Items Entered Into the Corrective Action Program

a. Inspection Scope

As part of the various baseline inspection procedures discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's corrective action program at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Attributes reviewed included: the complete and accurate identification of the problem; that timeliness was commensurate with the safety significance; that evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent of condition reviews, and previous occurrences reviews were proper and adequate; and that the classification, prioritization, focus, and timeliness of corrective actions were commensurate with safety and sufficient to prevent recurrence of the issue. Minor issues entered into the licensee's corrective action program as a result of the inspectors' observations are included in the attached list of documents reviewed.

These routine reviews for the identification and resolution of problems did not constitute any additional inspection samples. Instead, by procedure they were considered an integral part of the inspections performed during the quarter and documented in Section 1 of this report.

b. Findings

No findings of significance were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. This review was accomplished through inspections of the station's daily CR packages.

These daily reviews were performed by procedure as part of the inspectors' daily plant status monitoring activities and, as such, did not constitute any separate inspection samples.

b. Findings

No findings of significance were identified.

.3 Selected Issue Follow-up Inspection

a. Inspection Scope

During a review of items entered in the licensee's corrective action program, the inspectors selected the corrective action report listed below for a more indepth 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.

- May 10, 2008, use of duct tape on insulation for containment cooler piping

The above constitutes completion of one in-depth problem identification and resolution sample.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.4 Semiannual Trend Review

a. Inspection Scope

The inspectors completed a semiannual trend review of repetitive or closely-related issues that were documented in trend reports, problem lists, performance indicators, health reports, quality assurance audits, corrective action documents, corrective maintenance documents, and departmental self-assessments and interviewed selected licensee staff to determine if any adverse trends existed. Additionally, the inspectors reviewed the licensee's trending efforts to identify trends that might indicate the existence of more safety-significant issues. The inspectors' review consisted of the 6-month period from January through June 2008. When warranted, some of the samples expanded beyond those dates to fully assess the issue. The inspectors also reviewed corrective action program items associated with the below listed issues. The inspectors compared and contrasted their results with the results contained in the licensee's quarterly trend reports. Corrective actions associated with a sample of the issues identified in the licensee's trend report were reviewed for adequacy. These areas were chosen based on information gathered by the inspectors during daily plant status reviews over the previous 6 months.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

The inspectors noticed a continuing negative trend in the area of plant status control issues. During the first half of 2008, 21 status control issues were identified with four considered significant or consequential. The licensee had previously identified trends in these areas and has completed several performance improvement actions in this area; however, the current actions have been slow to adequately correct the problems.

4OA3 Follow-up of Events and Notices of Enforcement Discretion (71153)

.1 (Closed) Licensee Event Report (LER) 2008-003-00, Manual Reactor Trip Due to Loss of Steam Generator Level

On March 17, 2008, at approximately 1 p.m. while operating at 100 percent, Wolf Creek was manually tripped due to loss of steam generator level. The 4.16 kV busses PB03 and PB04 were cross-tied and being supplied by transformer XPB03. A secondary power cable from XPB03 faulted to ground causing loss of power to all condensate pumps that tripped the main feedwater pumps on low suction. During previous maintenance, the transformer bushing connector was reassembled improperly resulting in a loose connection that failed.

The inspectors reviewed LER 05000482/2008-003-00 to verify that the cause was identified and that corrective actions were appropriate. See Section 4OA5.2 for enforcement actions taken. This LER is closed.

.2 (Closed) LER 2008-004-00, Loss of Power Event when the Reactor was Defueled

a. Inspection Scope

On April 7, 2008, during Refueling Outage 16, work was ongoing in the switchyard. At 10:17 a.m., a loss of offsite power event was initiated when a relay technician performing breaker failure trip testing on a 345 kV breaker closed the wrong set of trip links resulting in the lost of safety Bus B. Safety Bus A was out of service for maintenance. EDG B automatically started and powered safety Bus B and equipment. As a result of the loss of offsite power, the control room staff declared a NOUE.

The inspectors responded to the control room and reviewed: (1) operator logs, plant computer data, and/or strip charts for the above listed event to evaluate operator performance in coping with nonroutine events and transients; (2) verified that operator actions were 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 event.

The inspectors reviewed LER 05000482/2008-004-00 to verify that the cause was identified and that corrective actions were appropriate. This LER is closed.

Documents reviewed by the inspectors are listed in the attachment.

b. Findings

An Inadequate Switchyard Work Procedure Resulted in a Loss of Offsite Power and NOUE

Introduction. A self-revealing Green finding was identified for an inadequate Wolf Creek switchyard maintenance work instruction which resulted in the loss of offsite power.

Description. On April 7, 2008, offsite power was lost to the NB02 4 kV safety-related bus when switchyard workers tripped the incorrect "breaker failure" trip relay while testing the Rose Hill 345 kV offsite switchyard breakers. The incorrect closed trip relay made up the logic for the startup transformer protection circuit and extended the trip signal to all 345 kV offsite breakers, resulting in the loss of power. The loss of the switchyard bus deenergized the "protected train," 4 kV Train B bus. The EDG automatically started and supplied power to the Train B bus. Offsite power was restored to the Train B bus approximately 8 hours later. The plant was defueled for a refueling outage and Bus NB01 bus was secured for maintenance.

The inspectors reviewed the guidance used by switchyard workers which included WOs, drawings for the switchyard, and the test trip relay protection scheme. The WOs did provide test link asset numbers but did not identify individual power breakers. The inspectors noted that the WOs only provided generic instructions and did not contain any detailed information or any specific step-by-step instructions on how the work was to be conducted. It was also noted that the switchyard workers did not have a copy of the maintenance procedure in hand and were on the phone with another switchyard worker who coordinated/directed the work.

The inspectors noted that Administrative Procedure AP 21C-001, "WCGS/WESTAR Substation," Revision 8, in part, contained steps for the Wolf Creek switchyard coordinator to review and monitor switchyard activities; and prepare a substation work authorization which describes the type of work to be performed and oversight of work needed. This review process was to ensure control of maintenance which could affect the availability of offsite power. Procedure AP 21C-001 also contains guidance that if either NB bus is deenergized, then work should not be performed that could jeopardize power to the in-service NB bus. However, this review did not identify the inadequate instructions provided to the workers nor did it prevent work that could jeopardize power to the in-service NB bus.

Analysis. The performance deficiency associated with this finding involved an inadequate switchyard maintenance procedure. This finding is more than minor because the availability and reliability of a safety-related 4 kV bus was challenged when offsite power was lost. This finding was associated with the equipment performance attribute of the mitigating systems cornerstone and affected the objective to ensure availability and reliability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined this finding to be of very low safety significance (Green) using the significance determination process for shutdown operations. Using the NRC IMC 0609, Appendix G, Phase 1 Screening Worksheet, this finding was determined to be of very low safety significance since this finding did not increase the likelihood of a loss of reactor coolant system inventory, degrade the ability to terminate a leak path or add reactor coolant system inventory when needed. This finding had human performance crosscutting aspects in the area of resources because personnel did not have adequate procedures and work instructions for switchyard maintenance to ensure that the trip relay testing would not create an inadvertent loss of offsite power [H.2(c)].

Enforcement. No violation of regulatory requirements occurred. The inspectors determined that this finding did not represent a noncompliance because it did not involve a safety-related or TS required procedure. Wolf Creek entered this issue into their corrective action program as CR 2008-001457. Finding (FIN) 05000482/2008003-05, Inadequate Switchyard Work Procedure Resulted in a Loss of Offsite Power.

4OA5 Other Activities

.1 (Closed) Unresolved Item (URI) 05000482/2008002-03, Containment Net Positive Suction Head Losses

a. Inspection Scope

A URI was identified when an operability determination dated January 22, 2008, was required to ensure design error discoveries did not create unacceptable reductions in margin-to-net positive suction head requirements for core cooling components associated with the already installed containment recirculation sump strainer modification. The size of the orifice beneath each strainer was not large enough to prevent head loss in excess of the net positive suction head required per the design conditions defined in the purchase specification supplied to the strainer vendor. This resulted in required net positive suction head being less than available. The inspectors reviewed previous design analysis calculations that accepted the vendor containment sump clean head loss calculation and design analysis calculations that corrected the design errors.

Documents reviewed by the inspectors are listed in the attachment.

b. Findings

Introduction. The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for failure to implement engineering procedures and approve a third party calculation prior to use at Wolf Creek. Specifically, the calculation review failed to identify the incorrect design inputs to the net positive suction head calculations on two occasions for RHR and containment spray.

Discussion. On October 5, 2006, Wolf Creek engineering approved Design Change Package 011295 which accepted the associated vendor calculation, TDI-6002-05, Revision 0, for clean strainer head loss as a design analysis calculation for the new containment sump. On January 22, 2008, an operability evaluation documented design errors that created unacceptable reductions in margin-to-net positive suction head requirements for core cooling components associated with the already installed containment recirculation sump strainer modification. Revision 0 of the calculation had omitted the head loss component associated with the as-built orifices located in the strainer support plate. The size of the orifice beneath each strainer was not large enough to prevent head loss in excess of the net positive suction head required per the design conditions defined in the purchase specification supplied to the strainer vendor.

The additional head loss due to the calculation error was 2.28 feet. This resulted in required net positive suction head being less than available.

On February 21 and April 17, 2008, Wolf Creek engineering accepted the associated vendor calculations, TDI-6002-05, Revisions 1 and 2, respectively, which corrected the clean strainer head loss for the as-built orifice size. However, on April 1, 2008, additional concerns were identified by Callaway Plant in which the vendor used nonconservative temperature correction through the orifices.

Wolf Creek, subsequently, replaced the strainer support plate in Refueling Outage 16 (March 21 thru May 14, 2008) with larger orifices to regain head loss margin associated with the incorrect orifice size and non-conservative temperature correction concerns.

As part of the follow-up inspection, the inspectors reviewed previous design analysis calculations that accepted the vendor containment sump clean head loss calculation. On three separate reviews, Wolf Creek engineering accepted the vendor calculation without completely evaluating the calculation as acceptable to Wolf Creek in accordance with plant procedures. Administrative Procedure AP 05D 001, "Calculations," Revision 11, Step 6.11.3, states, in part, that design analysis calculations shall be reviewed and accepted by engineering prior to being used to support plant design or operability. This review shall compare calculations to design inputs, verify assumptions, verify analytical methods, verify accuracy, and ensure compliance with design criteria. Contrary to the above, the licensee's acceptance review of Revision 0 of the calculation failed to identity incorrect design inputs to the as-built orifice size and Revisions 1 and 2 failed to identity the nonconservative temperature correction prior to being accepted.

<u>Analysis</u>. Failure to follow engineering procedures and evaluate a third party calculation prior to use and the failure to properly translate design and licensing basis information into specifications were performance deficiencies. This finding was more than minor because they were similar to non-minor Example 3.j from NRC IMC 0612, Appendix E, "Examples of Minor Issues," in that there was a reasonable doubt on the operability of the RHR and containment spray pumps; and if left uncorrected, could result in a more significant safety concern. Using Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," the issue screened as having very low safety significance because it was a qualification deficiency confirmed not to result in loss of operability in accordance with NRC IMC Part 9900, Technical Guidance, "Operability Determination Process for Operability and Functional Assessments." This finding had crosscutting aspects in the problem identification and resolution area associated with the corrective action program component, because the site failed to perform a thorough evaluation of vendor calculations to ensure conditions adverse to quality are identified and resolved [P.1(c)].

<u>Enforcement</u>. Title 10 CFR Part 50, Appendix B, Criterion III, Design Control," requires in part that the licensee establish measures for the identification and control of design interfaces and for coordination among participating design organizations. These measures shall include the establishment of procedures among participating design organizations for the review, approval, release, distribution, and revision of documents involving design interfaces. It also requires, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in 10 CRF 50.2 and as specified in the license application, for those SSCs to which this appendix applies are correctly translated into specifications, drawings, procedures, and

instructions. Contrary to the above, on October 5, 2006, February 21, 2008, and April 17, 2008, the licensee failed to follow a procedure for the review and approval of a design document; in that Procedure AP 05D-001, "Calculations," Revision 11, step 6.11, states, in part, that design analysis calculations shall be reviewed and this review shall compare calculations to design inputs, verify assumptions, verify analytical methods, verify accuracy, and ensure compliance with design criteria. The licensee acceptance review of Revision 0 of the calculation failed to identity incorrect design inputs to the as-built orifice size and Revisions 1 and 2 failed to identity the nonconservative temperature correction prior to being accepted. Because this violation is of very low safety significance and has been entered into the licensee's corrective action program as CR 2008-001305, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV05000482/2008003-06, Failure to Verify Engineering Design Calculation Prior to Use.

.2 (Closed) URI 05000482/2008002-08, Transformer Trip Resulted in an Unplanned Reactor Trip and Forced Outage

a. Inspection Scope

The inspectors responded to the control room on March 17, 2008, due to a reactor trip from the XPB03 transformer trip, and reviewed: (1) operator logs, plant computer data, and/or strip charts for the above listed event to evaluate operator performance in coping with nonroutine events and transients; (2) verified that operator actions were 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 event. The inspectors observed the reactor shutdown and cooldown.

Documents reviewed by the inspectors are listed in the attachment.

b. Findings

Introduction. A self-revealing finding was identified for an inadequate maintenance procedure that resulted in a reactor trip due a loss of all condensate pumps.

Description. On March 17, 2008, plant operators observed that steam generator water level was lowering and main feed pump speed was decreasing. Based on these indications, Wolf Creek operators manually tripped the plant. Post-trip immediate and follow-up actions were completed without deviation. An auto actuation of auxillary feedwater occurred due to low-low steam generator water levels as expected. No other ECCS or engineered safety features actuations occurred. All plant equipment responded as expected.

Following the trip, control room operators observed that the plant had experienced a loss of the XPB03 13.8 kV to 4.16 kV nonsafety transformer which powers PB003 4.16 kV nonsafety bus. Approximately 12 hours prior to the transformer trip, Wolf Creek had removed from service XPB04 transformer for planned maintenance and cross-connected XPB04 transformer PB004 bus loads to the XPB03 transformer PB003 bus. This arrangement powered all three condensate pumps from PB003 4.16 kV bus. The PB003 bus powers the condensate Pumps A and C and the PB004 powers the condensate Pump B. The XPB03 transformer trip resulted in losing power to all three condensate pumps which tripped the main feed pumps on low suction pressure. The licensee determined the transformer tripped because two phases of the XPB03 transformer 4.16 kV output cables had overheated and failed. Additional investigation into the cable failures discovered that two multi-directional conductor connectors used to terminate two phases of the 1000 MCM 4.16 kV bus cables were installed using the incorrect configuration. The cable connector had been installed using a 1500-2000 MCM configuration which resulted in the conductor connector bottoming out before applying sufficient compression to ensure adequate connection to the cable.

The inspectors reviewed the work history of the XPB03 transformer and found WO 06-291275-000, Revision 0, in which the licensee had performed maintenance on the XPB03 transformer on March 4, 2008, that required removal of the XPB03 transformer 4.16 kV output cables. The WO provided general guidance to disconnect the high/low side of the transformer using attached instructions and Procedure MTE TL-001, "Wiring Termination and Lug/Connector Installation," Revision 11A. The inspectors noted that neither the WO nor Procedure MTE TL-001 contained any guidance or specified the conductor connector configuration and only provided general guidance to disconnect and re-term the cables. It was also noted that this work was performed by first-time performers who had no experience with this type of connector. The inspectors reviewed electrical maintenance training and did not identify any training that would have provided knowledge or skills on multi-directional conductor connectors. Additionally, the inspectors noted electrical maintenance personnel were not trained to use match-marking of removed components to ensure configuration control as mechanical maintenance personnel are trained to do.

Analysis. Failure to develop an adequate procedure for the XPB03 transformer maintenance was a performance deficiency. Traditional enforcement does not apply since there were no actual safety consequences or potential for impacting the NRC's regulatory function, and this finding was not the result of any willful violation of NRC requirements or Wolf Creek procedures. The inspectors determined that this finding was more than minor because it is associated with the procedure quality attribute of the initiating events cornerstone and it affected the cornerstone objective to limit the likelihood of those events that upset plant stability. This finding also affected the procedure quality attribute for the mitigating systems cornerstone and it affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The inspectors evaluated the significance of this finding using Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," and because two cornerstones were affected, a Phase 2 analysis was required. The consequences were assessed using a Phase 3 analysis by the Region IV senior reactor analyst. The consequence of the performance deficiency was a reactor trip with a loss of normal feedwater. This event occurred 13 days following maintenance using the flawed procedure. For this analysis, it is assumed that:

1. The reactor trip event was certain to occur at some point in time following use of the maintenance procedure, but that additional reactor trip events attributable to the deficiency were not likely to occur after the procedure was corrected. Therefore, for this case, the initiating event frequency for a reactor trip with loss of feed is set at 1/yr.

2. No other equipment was determined to be nonfunctional at the time of the event and thus nominal reliability for all other mitigating equipment is assumed. Because the event could have occurred at any time, average test and maintenance is assumed.

3. As a bounding assumption, no recovery is modeled for the normal feedwater system, although in this instance it was likely that such recovery was possible.

4. Risk associated with initiating events other than reactor trips was dismissed for this performance deficiency because the loss of feedwater from a transformer trip was unlikely to occur within 24 hours of such an event.

Consequently, this finding was determined to be of very low safety significance (Green). The inspectors also determined that this finding has human performance crosscutting aspects in the area associated with the resources component because the licensee failed to provide an adequate maintenance procedure to assure nuclear safety [H.2(c)].

Enforcement. No violation of NRC requirements occurred because the transformer maintenance WO was neither TS nor quality related. Wolf Creek entered this issue into their corrective action program as CR 2008-000908. FIN 05000482/2008003-07, Inadequate Transformer Procedure Resulted in an Unplanned Reactor Trip and Forced Outage.

.3 (Closed) Temporary Instruction (TI) 2515/150, Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles, (NRC Order EA-03-009)

NRC first revised Order EA-03-009, issued on February 20, 2004, that requires that low susceptible plants such as Wolf Creek perform at least a bare metal valuation examination every third refueling outage or every 5 years, whichever occurs first. Further, the order requires a nonvisual NDE with ultrasonic, eddy current, liquid penetrant, or a combination of these techniques at least once prior to February 11, 2008, and, thereafter, at least every four refueling outages or every 7 years, whichever comes first. Finally, the order specifies that during each refueling outage, visual inspections are to be performed to identify potential boric acid leaks from pressure-retaining components above the RPV head.

The following represents a summation of previous NRC inspections and applicable NRC inspection reports, and the current inspection effort. Taken together, the inspectors are able to determine that all necessary actions required by TI 2515/150 have been completed and the TI can be closed.

The inspectors reviewed NRC Inspection Report 05000482/2002-02, which documented performance of a bare metal visual examination in accordance with TI 2515/145, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles." The bare metal valuation examination was performed during Refueling Outage 12. TI 2515/145 was initiated to confirm licensee responses to NRC Bulletins 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," and 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity."

In conjunction with that review, the inspectors reviewed approximately 80 high-resolution photographs taken during Refueling Outage 12 that demonstrated performance and basis for acceptance of the bare metal valuation examination during Refueling Outage 12. Additionally, the inspectors reviewed the licensee's letter and Report CT-02-0029, dated May 24, 2002, which had been submitted to the NRC to satisfy the 30-day response required by NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." The report described the visual examination of the reactor vessel head and penetrations by examination personnel certified to a minimum of Level II in the VT-2 method, in which 100 percent of the carbon steel surface area of the head and 100 percent of the interface areas between the head penetrations and the carbon steel on top of the head were visually examined. The report stated that there were no indications of leakage or potential leakage through the reactor vessel head penetrations, and there was no degradation identified on the reactor vessel head.

The NRC, by letter to Wolf Creek Nuclear Operating Corporation, "Wolf Creek Generating Station – Response to NRC Bulletin 2002-02, Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs," dated May 24, 2002, acknowledged that the above 30-day response to NRC Bulletin 2002-01 also satisfied the report requirements for NRC Bulletin 2002-02 and NRC Order EA-03-009.

The inspectors reviewed licensee letter and Report WM 04-0001 to the NRC dated January 22, 2004, which documented the partial visual examination of the RPV head and penetrations during Refueling Outage 13 (October 2003), as required by Section IV.D in Order EA-03-009. The results of this examination, performed by a certified Level III examiner, showed that there were no indications of leakage or potential leakage through the head penetrations, nor were there any material deficiencies identified on the head or penetration boundaries.

With respect to Refueling Outage 14, RPV head inspections performed on April 9, 2005, the inspectors reviewed the inspection record identified as Attachment G, "Containment-Reactor Cavity," to Procedure STN PE-040D, "RCS Pressure Boundary Integrity Walkdown," Revision 2. The inspection record showed that there was no evidence of boron leakage; thus no bare metal valuation examination was required. Since no bare metal valuation examination was performed, no report to the NRC was required.

The inspectors also reviewed Inspection Report 05000482/2006-05, which documents performance of bare metal valuation examination, nonvisual NDE (i.e., ultrasonic and eddy current examinations) of the reactor pressure vessel head and penetration nozzles, and visual inspection of pressure-retaining components above the RPV head. The inspection report also stated that the inspectors had conducted a visual inspection of the head, including portions of the head in which the insulation had been removed, and did not identify any issues. The inspection report stated that no significant differences in the condition of the head were identified between Refueling Outages 14 and 15, and that the inspectors determined that the bare metal valuation and nonvisual NDE of the RPV head met the requirements of NRC Order EA-03-009.

Finally, the inspectors reviewed the licensee's 60-day report for NRC Order EA-03-009 dated December 20, 2006, which stated that the results of the visual inspection of pressure-retaining components above the RPV head showed no evidence of leaks or boron deposits on the surface of the RPV head or related insulation.

Current inspection activities (i.e., Refueling Outage 16) of the RPV vessel head and penetration nozzles are addressed in Paragraph 02.02, Section 1R08, above.

TI 2515/150 has been completed and closed.

.4 (Discussed)Temporary Instruction 2515-172, Reactor Coolant System Dissimilar Metal Butt Welds

Portions of Temporary Instruction TI2515/172, “Reactor Coolant System Dissimilar Metal Butt Welds,” were performed at Wolf Creek Generating Station during Refueling Outage 16 in April 2008.

03.01 Licensee’s Implementation of the MRP-139 Baseline Inspections

The inspectors observed performance and reviewed records of structural weld overlays and NDE activities associated with the licensee’s pressurizer structural weld overlay mitigation effort. The baseline inspections of the pressurizer dissimilar metal butt welds were completed during Refueling Outage 15 (the Fall 2006 refueling outage).

At the present time, the licensee is not planning to take any deviations from the baseline inspection requirements of MRP-139, and all other applicable dissimilar metal butt welds are scheduled in accordance with MRP-139 guidelines.

03.02 Volumetric Examinations

The inspectors reviewed the ultrasonic examination and eddy current examination records of the unmitigated hot leg and cold leg dissimilar metal butt welds (Welds 1-RV-301-121-A, -B, -C, -D and 1-RV-302-121-A, -B, -C, -D), respectively, performed on April 22 and 23, 2005. These examinations were conducted in accordance with ASME Code, Section XI, Supplement VIII [PDI] requirements regarding personnel, procedures, and equipment qualifications. No relevant conditions were identified during the examinations of the hot and cold leg unmitigated dissimilar metal butt welds.

Inspectors directly observed and/or reviewed records of NDE performed on pressurizer weld overlays. This effort is documented in Section 1R08 of this inspection report.

The certification records of ultrasonic examination personnel used in the examination of the unmitigated hot and cold legs dissimilar metal butt welds, and the mitigated pressurizer dissimilar metal butt welds were reviewed. All personnel records showed that they were qualified under the EPRI performance demonstration initiative.

Inspection coverage met requirements of MRP-139. No relevant conditions were identified. No deficiencies were identified during the NDE.

03.03 Weld Overlays

Review of welding activities associated with full structural weld overlays will receive in-office review at a later date.

The licensee submitted and received NRC authorization by letter dated April 3, 2007, for the use of 10 CFR 50.55a, Request 13R-05, “Installation and Examination of Full

Structural Weld Overlays for Repairing/Mitigating Pressurizer Nozzle-to-Safe End Dissimilar Metal Welds and Adjacent Safe End-to-Piping Stainless Steel Welds." Verification of welders' qualifications that performed the full structural weld overlays will be performed during either in-office review at a later date, or during the next NRC inservice inspection.

Deficiencies have not been identified in the completed pressurizer full structural weld overlays.

03.04 Mechanical Stress Improvement

This item is not applicable because the licensee did not employ a mechanical stress improvement process.

03.05 Inservice Inspection Program

The licensee's MRP-139 inservice inspection program will receive in-office review at a later date.

.5 Quarterly Resident Inspectors' Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors conducted the following observations of security force personnel and activities to ensure that the activities were consistent with licensee security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours.

These quarterly resident inspectors' observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status review and inspection activities.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

On April 10, 2008, the senior reactor inservice inspection inspector presented the results of the inservice inspection to Mr. R. A. Muench, President and Chief Executive Officer, and other members of licensee management. Licensee management acknowledged the inspection findings.

On April 15, 2008, the emergency preparedness inspector conducted a telephonic exit meeting to present the results of the in-office inspection of changes to the licensee's emergency action levels to Mr. T. East, Superintendent, Emergency Planning, who acknowledged the findings. The inspector confirmed that no proprietary, sensitive, or personal information had been examined during the inspection.

On July 9, 2008, the resident inspectors presented the inspection results of the resident inspections to Mr. Hedges, Vice President of Oversight, and other members of the licensee's management staff. The licensee acknowledged the findings presented. The inspectors noted that while proprietary information was reviewed the material was returned to the licensee, and none would be included in this report.

4OA7 Licensee-Identified Violations

The following violation of very low significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

- Part 50.54(q) of Title 10 of the Code of Federal Regulations requires, in part, that a power reactor licensee may make changes to their emergency plan without Commission approval only if the changes do not decrease the effectiveness of the plans and the plans, as changed, continue to meet the standards of Part 50.47(b) and the requirements of 10 CFR Part 50, Appendix E. Contrary to this, between May 1995, and March 2007, Wolf Creek reduced the effectiveness of its emergency plan, in that, changes to Emergency Action Level 11, "Natural Phenomena," reduced or eliminated an operator's ability to recognize an earthquake occurring at the NOUE emergency classification level. This was identified in the licensee's corrective action program as CR 2007-004340. This finding is of very low safety significance because it only affects classification at the NOUE emergency classification.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

R. A. Muench, President and Chief Executive Officer
M. W. Sunseri, Vice President Operations and Plant Manager
S. E. Hedges, Vice President Oversight
K. L. Scherich, Director Engineering
T. F. East, Manager, Emergency Planning
P. J. Bedgood, Superintendent, Chemistry/Radiation Protection

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000482/2008003-01	NCV	Failure to Implement Transient Combustible Control Permit Requirements for a Propane Tank (Section 1R05)
05000482/2008003-02	NCV	Failure to Establish Goals and Monitor for a(1) ECCS Room (Section 1R12)
05000482/2008003-03	NCV	Inadequate Containment Sump Inspection Procedure (Section 1R15)
05000482/2008003-04	NCV	Troubleshooting Activities Bypass Design Control for the Fuel Transfer System (Section 1R20)
05000482/2008003-05	FIN	Inadequate Switchyard Work Procedure Resulted in a Loss of Offsite Power (Section 4OA3.2)
05000482/2008003-06	NCV	Failure to Verify Engineering Design Calculation Prior to Use (Section 4OA5.1)
05000482/2008003-07	FIN	Inadequate Transformer Procedure Resulted in an Unplanned Reactor Trip and Forced Outage (Section 4OA5.2)

Closed

05000482/2008002-03	URI	Containment Sump Net Positive Suction Head Losses (Section 4AO5.1)
05000482/2008002-08	URI	Transformer Trip Resulted in an Unplanned Reactor Trip and Forced Outage (Section 4OA5.2)
05000482/2008-003-00	LER	Manual Reactor Trip Due to Loss of Steam Generator Level (Section 4OA3.1)

05000482/2008-004-00	LER	Loss of Power Event when the Reactor was Defueled (Section 4OA3.2)

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:

Section 1R01: Adverse Weather

Procedures

STN EF-020B, ESW Train B Warming Line Verification, Revision 6
SYS EF-205, ESW/CIRC Water Cold Weather Operations, Revision 19
AI 14-006, Severe Weather, Revision 7

Section 1R04: Equipment Alignment

Procedures

CKL EF-120, ESW Valve, Breaker and Switch Lineup, Revision 41
SYS KJ-121, Diesel Generator Lineup for Auto Ops, Revision 39

Work Orders

06-289610-000

Work Requests

07-063628

Miscellaneous

Engineering Disposition, Relocate I/P from the ARVs, ABPV001 Thru 004, Revision 6
Wolf Creek Generating Station USAR, Revision 19

Section 1R05: Fire Protection

Procedures

ALR KC-888, Fire Protection Panel KC-008 Alarm Response, Revision 15

AP 10-106, Fire Preplans, Revision 5

OFN ST-003, Natural Events, Revision 13A

STN FP-815A, Heat Trip Actuation Device Operational Test Zones BZ 503, 016/SZ1-5Z47,1-2Z28, Train A EDG and Engineered Safety Feature Transformer, Revision 3

Condition Reports

2007-002929

Work Requests

07-063647

Work Orders

06-284430-000　　06-284436-000

Drawings

E-OFO221, Fire Detection/Protection System Yard Transformer Area EL. 2000'-0", Revision 5
M-13EA01, Piping Orthographic Service Water System Communication Corridor, Revision 6
M-13EF01, Piping Isometric ESW System Control Bldg. Train A & B, Revision 11

Miscellaneous

Wolf Creek Generating Station Individual Plant Examination Summary Report, September 1992
Post Fire Safe Shutdown Area Analysis, E-1F9910, Revision 2
Fire Hazard Analysis Fire Area H-1, Revision 0
Prefire Plan Auxiliary Building Prefire Plans, Revision 6
Prefire Plan, Fire Protection Water Supply and Hydrant Locations, Revision 0
Fire Hazard Analysis, Fire Area CST & Refueling Water Storage Tank, Revision 0

Section 1R08: Inservice Inspection Activities

Procedures

AI 16F-001, Evaluation of Boric Acid Leakage, Revision 5

AI 16F-002, Boric Acid Leakage Management, Revision 4

AP 12-002, Internal/External System Cleanliness, Revision 4

AP 12-003, Foreign Material Exclusion, Revision 6

AP 16C-006, MPAC Work Request/Work Order Process Controls, Revision 11a

AP 16F-001, Boric Acid Corrosion Control Program, Revision 4

AP 22A-001, Screening, Prioritization, and Pre-approval, Revision 9

AP 29A-003, Steam Generator Management, Revision10

APF 29A-003-01, Steam Generator Management Program Guideline Exception Detail Sheet, Revision 1

EPRI PDI-UT-2, Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds, Revision C

I-ENG-023, Steam Generator Data Analysis Guidelines, Revision 7

LMT UT-95, Ultrasonic Examination of Austenitic Piping Welds, Revision 3

MRS 2.4.2 GEN-35, Eddy Current Inspection of Preservice and Inservice Heat Exchanger Tubing, Revision 13

PCI Procedure, WCP-8, Preheating and Postweld Heat Treatment, Revision 8

PDI ISI-254-SE, Ultrasonic Examination, Revision 2

QCP-20-520, Pressure Test Examination, Revision 6A

STN PE-040D, RCS Pressure Boundary Integrity Walkdown, Revision 2

STN PE-370, Foreign Object Search and Retrieval and Secondary side Inspections, Revision 10

STN-PE-040D, Reactor Coolant System Pressure Boundary Integrity Walkdown, Revision 2

STS PE-022, Steam Generator Tube Inspection, Revision 15

STS PE-040E, RPV Head Bare Metal Inspection, Revision 0

WDI-ET-008, IntraSpect Eddy Current Inspection of Vessel Head Penetration J-welds and Tube OD Surfaces, Revision 8

WDI-STD-146, Eddy Current Procedure, Revision 3

Calculations

CN-NCE-DCPPRSG-12, Feedwater Nozzle and Thermal Sleeve Analysis, Revision 1

Corrective Action Documents

2006-002449	2006-002668	2006-002694	2006-002794
2006-002850	2006-002891	2006-002973	2006-00311
2006-003561	2006-003616	2006-003618	2006-003630
2006-003816	2006-003885	2007-000756	2007-000884
2007-003502	2007-003550	2007-003648	2007-003855
2007-004007	2008-000008	2008-000031	2008-000091
2008-000292	2008-001290	2008-001383	2008-001512
2008-001534	2008-001543		

Drawings

WIP-M-AB01-001-3-1, Piping Isometric, Main Steam system, Reactor Building and Auxiliary Building, Area 5, Revision 00

M-AB01(Q), Piping Isometric, Main Steam system, Reactor Building and Auxiliary Building, Revision 1

E-A0-103.226.014.500, System Medium Operated Gate Valve-MSIV, Main Steam Isolation Valve, Revision 1-1g

WIP-M-13EM01-007-A-1, Piping Isometric, High Pressure Coolant Injection System-Auxiliary Building, Revision 0

Work Orders

06-289464-005, QC to perform BMV of RBB01 outlet nozzle to safe-end welds

02-240956-112, replacement of MSIV ABHV0017

08-303761-002, Installation of ECCS Vent Valve Assemblies, Revision 2

Miscellaneous

MRP-139 ISI Program/Plan, Revision 0

Certification packages of NDE personnel

PCI Weld Data Sketch Sheet, ABHV0017 Replacement, Revision 0

PCI Quality Assurance Traveler, 900877 ABHV0017 Replacement, Revision 0

Certificate of conformance 900877-02 and applicable certified material test reports for welding materials being used on MSIV and FWIV replacements.

PCI approved welders list for MSIV/FWIV replacement

Welding material withdrawal slips associated with traveler 900877-ABHV 0017

SG-SGDA-08-12, Steam Generator Degradation Assessment for Wolf Creek Refueling Outgage 16 Outage, March 2008, Revision 0

MRS-TRC-1790, Omni 200 to TC6700 Digital Tester Equivalency

DDM-96-009, Documentation of Appendix H Compliance and Equivalency, Revision 0

IIT 08-001 for CR 2008-000091, Noncondensable Gas Accumulation in ECCS Piping Technical Specifications

RCMS 2003-061

NRC IE Bulletin 82-02, Degradation of threaded fasteners in the reactor coolant pressure boundary of PWR Plants

Relief Request 13R-05, “Installation and Examination of Full Structural Weld Overlays for Repairing/Mitigating Pressurizer Nozzle-to-Safe End Dissimilar Metal Welds and Adjacent Safe End-to-Piping Stainless Steel Welds”

Alloy 600 Program Review, Revision 3, 9/5/06

TI-2515/150, Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles, Revision 3

WCAP-13254, Addendum to Analytical Report For Kansas Gas and Electric Company Wolf Creek Nuclear Power Plant Reactor Vessel

Wolf Creek Nuclear Power Plant Reactor Vessel Head Visual Examination Plan RF Outage 12

Wolf Creek Nuclear Power Plant Reactor Vessel Head Visual Examination Plan RF Outage 15

Wolf Creek Nuclear Power Plant Reactor Vessel Head Visual Examination Plan Partial Inspection for Penetrations 70, 46, 26, 58, 34, 52, 63

NRC Integrated Inspection Report 05000482/2002-02

NRC Integrated Inspection Report 05000482/2003-006

NRC Integrated Inspection Report 05000482/2006-005

ESH-102, STARS Plants Alloy 600 Program Review, dated September 9, 2006

WCRE-15, Program Plan for Management of Alloy 600 Components and Alloy 82/182 WELDS, Revision 1

Wolf Creek Request and NRC Approval letters for 10CFR 50.55a I3R-05 Relief Request

MRP-139, Primary System Butt Weld Inspection and Evaluation Guidelines, dated Sept 12, 2005

Data and report for Weld Overlay package on “C” Safety Nozzle for Pressurizer in RF 15 Outage

Ultrasonic test results during RF 15 outage for the inlet and outlet reactor nozzles dissimilar metal butt-welds (Reports DM22-1, DM158-1, DM202-1, DM338-1, DM67-1, DM113-1, DM247-1, and DM293-1)

Ultrasonic test results during Refueling Outage 16 outage (current) for the Pressurizer Surge Line dissimilar metal weld overlay (Report WLT-001)

Containment Spray Radiography Reports 3828 and 3831

Refueling Outage 14 Pictures of Bottom Mounted Instrument nozzles

Boric Acid Picture Reviews (100% of documented cases)

Two engineering evaluations for boric acid found on RCS piping and components

Two corrective action work packages performed for boric acid leaks

Boric Acid Corrosion Program 2007 4th quarter inspection/monitoring Report

Refueling Outages 12 and 15 Bare Metal Visual inspection reports and pictures of RPV

Refueling Outage 15 Outage Video of BMV Inspection of RPV

Wolf Creek reports to NRC regarding NRC order EA-03-005 for Reactor Pressure Vessel Head Examinations for Refueling Outages 12, 13, 14, and 15

Wolf Creek Reactor Pressure Vessel Head Inspection during Outage 16

WCRE-16, Inservice Inspection Program Plan for Interval 3, Revision 3

Wolf Creek Request and NRC Approval letter dated December 7, 2006, for Request to Relax Nondestructive Examination of Reactor Pressure Vessel Head Penetration Nozzles in First Revised Order EA-03-009

WCRE-16, Inservice Inspection Program Plan for Interval 3, Revision 3

Steam Generator Inspection Summary and Data Reports for Refueling Outage 15

Steam Generator Inspection Summary and Data Reports for Refueling Outage 16

Wolf Creek Letter ET-98-0019, Response to NRC GL 97-05 Regarding Tube Inspection Techniques

Westinghouse Preliminary Startup Approval Letter with FME in Steam Generators

Steam Generator Data Analysis Desktop Instruction, Revision 4

MRS-TRC-1864, Use of Appendix H Qualified Techniques at Wolf Creek RF16 March 2008

Various Calibration Certificates for all Eddy Current Equipment

SG-SGDA-06-039, Wolf Creek RF15 Condition Monitoring and Operational Assessment October 2006

AP 28D-001-02 Self-Assessment Report SEL 04-038 Steam Generator Program, Revision 4

Acquisition Technique Sheets (ACTS) SAP-05-08, SAP-01-08, SAP-07-08, and SAP-06-08

Analysis Technique Sheets (ANTS) SAP-+PT-08, SAP-BOB-08, SAP-+PTUB-08, and SAP-+PT-08

Welding Procedure Specifications and their Supporting Procedure Qualification Records

WPS-1 MC-GTAW-S6-HT-1, Revision 1 with Supplement Revision 3, and Procedure Qualification Record PQR-627B

WPS-1-MN-SMAW-1, and Procedure Qualification Record PQR 721A, Revision 1

WPS 1 MN-GTAW/SMAW-1 and Procedure Qualification Record PQR 627A and 721A

Section 1R11: Operator Requalification

Procedures

AI 21-100, Operations Guidance and Expectations, Revision 8
AP 21-001, Conduct of OPS, Revision 36A
APF 06-02-001, Emergency Action Levels, Revision 8
EDI 23M-050, Monitoring Performance to Criteria and Goals, Revision 3
EPP 06-06, Protective Action Recommendations, Revision 4

Miscellaneous

Operations Requalification Cycle 07-01, Revision 0

Section 1R12: Maintenance Effectiveness

Performance Improvement Requests

1996-2671	2007-1952	2007-1953	2007-2100
2007-2141			

Work Requests

07-059846	07-060117	07-060141	07-060514
07-061766	07-061883	07-061884	

Work Orders

05-270547-001	05-271470-000	06-287445-000
07-291889-000	07-291903-000	07-292308-000
07-293935-000	07-293935-003	07-294968-000
07-294968-003	07-295395-000	07-295396-000
07-296463-000	07-298545-000	07-301051-001
07-301051-011		

Condition Reports

2007-000860	2007-000879	2007-000897	2007-000943
2007-000988	2007-004154		

Maintenance Rule

Scoping Evaluation for System BB - Reactor Coolant System
Scoping Evaluation for System INS -Reg. Guide 1.97 Instrumentation
Final Scoping Evaluation AB-05
Final Scoping Evaluation GN-01
Final Scoping Evaluation GN-02
Final Scoping Evaluation GN-03
Final Scoping Evaluation GN-04
Final Scoping Evaluation GN-06
Final Scoping Evaluation GN-08
Final Scoping Evaluation KA-01
Final Scoping Evaluation KA-03
Final Scoping Evaluation KA-04
Final Scoping Evaluation KA-06

Miscellaneous

EDI 23M-050 Attachment B, Functional Failure Determination Checklist

M-12KA01, Piping & Instrumentation Diagram Compressed Air System, Revision 27

INC C-1000, Calibration of Miscellaneous Components, Revision 7

STS AB-201A, Main Steam Isolation Bypass Inservice Valve Test, Revision 14

Calculation E-11005, List of Loads Supplied by EDG, Revision 32

BD-EMG ES-04, Natural Circulation Cooldown, Revision 8

Engineering Disposition 116451-10

USAR 1.2.9.6, Compressed Air Systems

EDI 23M-050, Engineering Desktop Instruction Monitoring Performance to Criteria Goals, Revision 3

Calculation AN-99-031, Development of PSA based Reliability Performance Criteria for Maintenance Rule, Revision 0

Section 1R18: Plant Modifications

Procedures

Procedure AP 29B-002, ASME Code Testing of PUMPS and Valves, Revision 6

Miscellaneous

Engineering Permanent Modification Change Package No. 12179, Remote Racking Device – 4.16 kV 1E Switchgear NB001 and NB002, Revision 1

Temporary Modification Order 07-010-RP for 7300 System Cabinets 8 & 9, RP 044

Inservice Testing program Third 10-Year Interval, Containment Spray Pump Full Flow Testing Line, Revision 5

WCOP-02, Revision 14, IST Program Plan

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LIST OF ACRONYMS

AC	alternating current
ASME	American Society of Mechanical Engineers
AVB	anti-vibration bar
CFR	*Code of Federal Regulations*
CR	condition report
ECCS	emergency core cooling system
EDG	emergency diesel generator
EPRI	Electric Power Research Institute
ESW	essential service water
FIN	finding
IMC	inspection manual chapter
LER	licensee event report
NCV	noncited violation
NDE	non destructive examination
NEI	Nuclear Energy Institute
NOUE	Notice of Unusual Event
NRC	Nuclear Regulatory Commission
PIR	performance improvement request
RHR	residual heat removal
RPV	reactor pressure vessel
PT	penetrant testing
SSC	structure, system, and component
TI	Temporary Instruction
TS	Technical Specification
URI	unresolved item
USAR	Updated Safety Analysis Report
WO	work order