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

November 10, 2009

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 05000482/2009004

Dear Mr. Muench:

On September 30, 2009, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Wolf Creek Generating Station. The enclosed integrated inspection report documents the inspection findings, which were discussed on October 14, 2009, with Mr. Matt Sunseri, Vice President of Operations and Plant Manager, and other members of your staff.

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

This report documents eight NRC identified findings of very low safety significance (Green). All of these findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as noncited violations, consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest the violations or the significance of the noncited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 612 E. Lamar Blvd, Suite 400, Arlington, Texas, 76011-4125; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Wolf Creek Generating Station.

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

ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Geoffrey B. Miller, Chief
Project Branch B
Division of Reactor Projects

Docket No. 50-482
License No. NPF-42

Enclosure
Inspection Report 05000482/2009004
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-482
License:	NPF-42
Report:	05000482/2009004
Licensee:	Wolf Creek Operating Corporation
Facility:	Wolf Creek Generating Station
Location:	1550 Oxen Lane SE Burlington, Kansas
Dates:	July 1 through September 30, 2009
Inspectors:	C. M. Long, Senior Resident Inspector C. A. Peabody, Resident Inspector P. A. Jayroe, Project Engineer, Project Branch B G. W. Apger, Operations Engineer S. M. Alferink, Reactor Inspector, Engineering Branch 2 J. P. Adams, Reactor Inspector, Engineering Branch 1, DRS C. M. Ryan, Reactor Inspector, Engineering Branch 1, DRS G. P. Tutak, Reactor Inspector, Engineering Branch 2, DRS
Approved By:	G. B. Miller, Chief, Project Branch B Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000482/2009004, 7/1/2009 – 9/30/2009; Wolf Creek Generating Station, Integrated Resident and Regional Report; Operability Evaluations; Post Maintenance Testing; Plant Modifications; Maintenance Risk Assessments and Emergent Work Control; Fire Protection; Event Followup.

The report covered a 3-month period of inspection by resident inspectors and an announced baseline inspections by regional based inspectors. Seven Green and one Severity Level IV noncited violations of significance were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG 1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. On June 30, 2009, the inspectors identified a noncited violation of Technical Specification 3.8.1 for failure to perform an adequate common cause evaluation within 24 hours to demonstrate no common cause failure mechanism existed between the emergency diesel generators after a through-wall leak was discovered on the essential service water piping. Wolf Creek did not start the opposite train emergency diesel generator and declared that the through-wall flaw was not a common cause failure without any evaluation or supporting statements. Nondestructive testing had not been started at this time. Subsequent evaluation of the flaw per American Society of Mechanical Engineers (ASME) Code Case N513.2 restored operability to the essential service water piping. The licensee entered this issue in their corrective action program as Condition Report 18347.

  The inspectors determined that the failure to demonstrate, per Technical Specifications 3.8.1 Required Actions B.3.1 or B.3.2, that no common cause failure existed for the emergency diesel generators was a performance deficiency. The inspectors determined that this finding was more than minor because it is associated with the equipment performance attribute for the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors evaluated the significance of this finding using Phase 1 of Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At Power Situations," and determined that the finding was of very low safety significance (Green) because the issue was not a design or qualification deficiency confirmed to result in loss of operability or functionality, did not represent a loss of system safety function, an actual loss of safety function of a single train for greater than its technical specification allowed outage time, an actual loss of safety function of a nontechnical specification risk-significant equipment train, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event.

The cause of the finding has a problem identification and resolution crosscutting aspect in the area associated with the corrective action program because Wolf Creek failed to thoroughly evaluate the failure mechanism such that the resolutions address the causes and extent of conditions, as necessary. Specifically Wolf Creek did not properly consider the possibility of common-cause pitting failures which could have impacted the essential service water piping Train A structural integrity thereby affecting its cooling loads, including the Emergency Diesel Generator A [P.1(c)] (Section 1R15).

- Green. The inspectors identified a noncited violation of Technical Specification 3.8.1, Required Action B.4.2.2 on March 24, 2009 when the licensee performed elective maintenance on safety bus relays and removed equipment from service that was required by the technical specification and the NRC Safety Evaluation Report (SER) while in an extended diesel generator outage. The maintenance had the potential to open the normal offsite feeder breaker. This issue has been entered into the corrective action program as Condition Report 15727.

  The inspectors determined that the failure to implement requirements of Technical Specification 3.8.1 and the associated NRC safety evaluation was a performance deficiency. The finding was more than minor because it is associated with the equipment performance attribute for the Mitigating Systems Cornerstone and 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 finding was determined to be of very low safety significance because the issue did not result in the Train B offsite power being inoperable for greater than 24 hours and did not involve external events such as flooding. Additionally, the cause of the finding has a problem identification and resolution crosscutting aspect in the area associated with the corrective action program. Specifically, Wolf Creek did an extent of condition review in response to a previous violation which included Procedure STS IC-208B, but still failed to prohibit performance of STS IC-208B during the 7-day diesel outages [P.1(c)] (Section 1R19).

- Green. On August 22, 2009, the inspectors identified a noncited violation of Technical Specification 3.0.3 in which both trains of Technical Specification 3.3.2 engineered safety features actuation system interlock function 8.a were bypassed with jumper wires in accordance with a plant procedure. Function 8.a is the interlock for reactor trip signal coincident with lo Tave signal. Wolf Creek blocked the signal from the feedwater valves with jumper wires during control rod drive motor-generator testing in Mode 3. The inspectors and the NRR technical specification branch found this to be contrary to the Updated Safety Analysis Report, the technical specifications, the technical specification bases, and the NRC safety evaluations supporting the technical specifications. The licensee entered this issue in their corrective action program as Condition Report 19318.

  The inspectors found that the failure to implement Technical Specification 3.3.2 interlock, function 8.a was a performance deficiency. The inspectors determined that this finding was more than minor because it is associated with the design control attribute of the Mitigating Systems Cornerstone and it affected the cornerstone objective to ensure the availability, reliability, and capability of mitigating systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The inspectors evaluated the significance of this finding using Inspection Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," and screened the finding to Phase 2 because the finding represents a loss of a system's function. The inspectors used Inspection Manual Chapter 0609, Appendix A and screened the finding to the NRC senior reactor analyst for review because there was not an acceptable equipment deficiency in the pre-solved worksheet. The senior reactor analyst determined that the finding is Green because he solved Table 3.10 of the Risk-Informed Inspection Notebook for Wolf Creek Generating Station, Revision 2.1a and found that the loss of feedwater isolation signal for less than 3 days resulted in a 1E-7 (Green) outcome. The inspectors also determined that the cause of the finding has a crosscutting aspect in the human performance area associated with decision making because Wolf Creek failed to make a risk significant decision using a systematic process. This issue was evaluated more than once and those evaluations sought to justify bypassing the interlock rather than seek the full regulatory basis for the interlock [H.1.a] (1R15).

- Green. The inspectors identified a noncited violation of 10 CFR 50 Appendix B, Criterion III, "Design Control," for failing to translate the boric acid design basis into procedures that ensure time sensitive operator actions are completed to achieve the core shutdown margin specified in the core operating limits report. Performance Improvement Request 2005-3461 identified that if the room coolers were started while lake temperature was low, the boric acid solution temperature may decrease below the solubility limit. Corrective actions for heat tracing and room temperature logging took approximately 3 years to implement and stopped short of addressing boric acid system operation when nonsafety power is lost to the heat tracing and the plant must be taken to cold shutdown in accordance with technical specifications. The licensee entered this issue in their corrective action program as Condition Report 20717.

  The failure to translate the design bases into procedures that ensure the function of the safety-related boric acid system upon loss of nonsafety-related heat tracing is a performance deficiency. The inspectors determined that this finding was more than minor because this issue aligned with Inspection Manual Chapter 0612, Appendix E, example 2.f, because the pipe temperature was required to stay above the boric acid solubility limit and the loss of the heat tracing and or room temperature decrease will block the boric acid system. This issue was associated with the equipment performance attribute of the mitigating systems cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events. The inspectors evaluated the significance of this finding using Phase 1 of Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At Power Situations," and determined that the finding screened to phase 2 because the issue was a design or qualification deficiency confirmed to result in loss of operability or functionality The inspectors evaluated the significance of this finding using Phase 2 of Inspection Manual Chapter 0609, Risk Informed Inspection Notebook for Wolf Creek Generating Station, and determined that the finding was of very low safety significance because loss of the boric acid system in Table 3.9 for one year resulted in a 1E-7 CDF when giving recovery credit for the refueling water storage tank. The inspectors determined that this finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Wolf Creek did not take appropriate corrective actions to resolve known deficiencies in the design and operation of the boric acid system for approximately 4 years. The issue was re-evaluated in 2009, and the licensee failed to correct the deficiencies identified in 2005. [P.1.d] (Section 1R18).

- Green. The inspector identified a noncited violation of 10 CFR 50.65(a)(4) for failure to adequately assess and manage the increase in risk during fuse inspection of component cooling water valves supplying cooling loads inside containment. On March 18, 2009, component cooling water Valves EG HV-16 and EG HV-54 were out of service for fuse inspections to verify wiring for fire protection analyses. The inspectors observed that the evolution was not included in the weekly risk assessment and that operations and maintenance personnel did not have guidance or briefings for restoration of the valves. Review of the risk assessment revealed that the impact of de-energizing the valves in the closed position was neglected and that restoration actions credited by the risk analyst were unknown to the control room and craft workers. The issue was entered into the corrective action program as Condition Report 15318.

  The failure to adequately assess and manage risk in accordance with AP 22C-003 and the preplanned risk assessment for the use of local actions to ensure component cooling water cooling to loads inside containment was a performance deficiency. The finding is more than minor because the licensee failed to effectively manage prescribed significant compensatory measures for maintenance activities that could increase the likelihood of initiating events. The finding was of very low safety significance because the magnitude of the calculated risk deficit was less than IE-6 even though risk management actions were not in place. The inspectors also determined that the finding has a human performance crosscutting aspect in the area associated with work control because the risk assessment procedure and clearance order procedure assumed local actions could be accomplished but there was no communication regarding this during the work planning stages [H.3(b)] (Section 1R13).

- Severity Level IV. The inspectors identified a Severity Level IV noncited violation of License Condition 2.C.(5), "Fire Protection," for making changes to the approved fire protection program without the required prior Commission approval. Specifically, the licensee made a change to the Updated Safety Analysis Report that allowed the licensee to violate the requirements of 10 CFR Part 50, Appendix R, Section III.L. Specifically, when the licensee recognized that fire damage could cause a pressurizer power operated relief valve to open long enough to create a void in the reactor vessel, this was documented as acceptable when it was not in compliance with this regulatory requirement. The licensee entered this issue into their corrective action program as Performance Improvement Request 2008-004869.

  This finding was assessed using traditional enforcement since it had the potential for impacting the NRC's ability to perform its regulatory function. This finding is more than minor since the change required prior staff review and approval prior to implementation and it did not receive the required approval. A senior reactor analyst performed a Phase 3 evaluation and determined this performance deficiency was of very low risk significance. In accordance with the guidance in Supplement I of the Enforcement Policy, this issue is considered a Severity Level IV noncited violation because it is of very low risk significance. This finding had a crosscutting aspect in the area of human performance associated with resources because the licensee failed to maintain long-term plant safety by maintaining design margins. Specifically, the licensee's choice to allow reactor vessel head voiding during an alternative shutdown in lieu of restoring the plant to compliance with the requirements of 10 CFR Part 50, Appendix R, Section III.L constituted a reduction in safety margin [H.2(a)] (Section 40A5.3).

Cornerstone: Barrier Integrity

- Green. The inspectors identified a noncited violation of Technical Specification 5.4.1.a, "Procedures," for failure to follow Procedure AP 12-003, "Foreign Material Exclusion." On August 12, 2009, the inspectors conducted a walkdown of the spent fuel pool area and found duct tape attached to various fueling and control rod tools such that duct tape was below the water. This duct tape was not in the foreign material exclusion logs. Spent fuel pool foreign material control is required under Procedure AP 12-003. The licensee entered this issue in their corrective action program as Condition Report 20338.

  The inspectors determined that the failure to log material in accordance with Procedure AP 12-003 was a performance deficiency. This finding is more than minor because it impacted the Barrier Integrity Cornerstone attribute of configuration control and affected the cornerstone objective to maintain functionality of the spent fuel pool system. Using Inspection Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," this finding was determined to be of very low safety significance because the finding only affected the barrier function of the spent fuel pool. The inspectors determined that this finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because although Wolf Creek performed a root cause and extent of condition evaluation for untracked foreign material, the evaluation still failed to find the duct tape in the pool itself. This allowed the tape to continue to be untracked [P.1.c] (Section 1R05).

Cornerstone: Miscellaneous

- Severity Level IV. The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.73, "Licensee Event Report System," with three examples in which the licensee failed to submit licensee event reports within 60 days following discovery of an event meeting the reportability criteria. First, on April 10, 2008, Wolf Creek submitted Licensee Event Report 2008-002-00 under 10 CFR 50.73(a)(2)(i)(B) which is operation prohibited by technical specifications but failed to make a report for a loss of safety function per 10 CFR 50.73(a)(2)(v) for the same event in which both trains of the emergency core cooling system were inoperable on February 13-14, 2008. Second, Wolf Creek filed Licensee Event Report 2008-004-00 on June 6, 2008 under 50.73(a)(2)(iv)(A) for an event that caused automatic start of an emergency diesel during a loss of offsite power on April 16, 2008. No report was made under 50.73(a)(2)(v) for an event or condition that could have prevented a safety function due to the loss of offsite power. Third, on April 10, 2008, Wolf Creek filed Event Notification Report 44131 under 10 CFR 50.72(b)(3)(ii)(B) based on a possible trip of all four containment coolers. The notification was later retracted. The inspectors found insufficient evidence to show that the containment coolers would not trip and concluded the event should have been reported under 10 CFR 50.73(a)(2)(v). All three issues are collectively captured in Condition Report 15318.

The inspectors reviewed this issue in accordance with Inspection Manual Chapter 0612 and the NRC Enforcement Manual. Through this review, the inspectors determined that traditional enforcement was applicable to this issue because the NRC's regulatory ability was affected. Specifically, the NRC relies on the licensee to identify and report conditions or events meeting the criteria specified in regulations in order to perform its regulatory function, and when this is not done, the regulatory function is impacted. The inspectors determined that this finding was not suitable for evaluation using the significance determination process, and as such, was evaluated in accordance with the NRC Enforcement Policy. The finding was reviewed by NRC management, and because the violation was determined to be of very low safety significance, was not repetitive or willful, and was entered into the corrective action program, this violation is being treated as a Severity Level IV noncited violation consistent with the NRC Enforcement Policy. This finding was determined to have a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program in that the licensee failed to appropriately and thoroughly evaluate for reportability aspects all factors and time frames associated with the inoperability of the emergency core cooling system, the offsite power system, and the containment heat removal system [P.1(c)] (Section 4OA3).

REPORT DETAILS

Summary of Plant Status

The plant started the inspection period at 100 percent rated thermal power. On August 19, 2009, Wolf Creek experienced and automatic reactor trip from 100 percent power when a lightning strike caused a loss of offsite power. Wolf Creek restarted on August 24, 2009. On August 28, 2009, Wolf Creek reduced power to 99 percent for the end of core life moderator temperature coefficient surveillance test. On September 30, 2009, Wolf Creek decreased to 97 percent power for heater drain pump repair.

1. REACTOR SAFETY

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

1R01 Adverse Weather Protection (71111.01)

.1 Summer Readiness for Offsite and Alternate ac Power

a. Inspection Scope

On August 10, 2009, the inspectors performed a review of the licensee's preparations for summer weather for selected systems, including conditions that could lead to loss-of-offsite power and conditions that could result from high temperatures. The inspectors reviewed the licensee's procedures affecting these areas and the communications protocols between the transmission system operator and the plant to verify that the appropriate information was being exchanged when issues arose that could affect the offsite power system. Examples of aspects considered in the inspectors' review included:

- The coordination between the transmission system operator and the plant during offnormal or emergency events
- The explanations for the events
- The estimates of when the offsite power system would be returned to a normal state
- The notifications from the transmission system operator to the plant when the offsite power system was returned to normal

These activities constitute completion of one readiness for summer weather affect on offsite and alternate ac power sample as defined in Inspection Procedure (IP) 71111.01-05.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04)

.1 Partial Walkdown

a. Inspection Scope

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

- Motor-Driven auxiliary feedwater Train B, July 7, 2009
- Turbine-Driven auxiliary feedwater, July 7, 2009
- Essential service water Train A, August 9, 2009
- Centrifugal charging pump Train A, September 22, 2009

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 affect the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, Updated Safety Analysis Report (USAR), technical specification requirements, administrative technical specifications, outstanding work orders, condition reports, 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. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of four partial system walkdown sample as defined in IP 71111.04-05.

b. Findings

No findings of significance were identified.

.2 Complete Walkdown

a. Inspection Scope

On September 18, 2009, the inspectors performed a complete system alignment inspection of the main steam system to verify the functional capability of the system. The inspectors selected this system because it was considered both safety significant and risk significant in the licensee's probabilistic risk assessment. The inspectors walked down the system to review mechanical and electrical equipment line ups, electrical power availability, system pressure and temperature indications, as appropriate, component labeling, component lubrication, component and equipment cooling, hangers and supports, operability of support systems, and to ensure that ancillary equipment or debris did not interfere with equipment operation. The inspectors reviewed a sample of past and outstanding work orders to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the corrective action program database to ensure that system equipment-alignment problems were being identified and appropriately resolved. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one complete system walkdown sample as defined in IP 71111.04-05.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Fire Inspection Tours

a. Inspection Scope

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

- 4kV Switchgear rooms, control building 2000' elevation, July 9, 2009
- Diesel generator rooms, diesel building 2000' elevation, July 9, 2009
- Turbine-Driven auxiliary feedwater room, August 11, 2009
- Spent fuel pool 2047' elevation, August 12, 2009

The inspectors reviewed areas to assess if licensee personnel 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 with later additional insights, their potential to affect equipment that 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. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of four quarterly fire protection inspection samples as defined by IP 71111.05-05.

b. Findings

.2 Introduction. On August 12, 2009, the inspectors identified a Green noncited violation of Technical Specification 5.4.1.a, Procedures, for failure to follow AP 12-003, "Foreign Material Exclusion," after a root cause assessment on foreign material exclusion.

Description. On August 12, 2009, the inspectors conducted a walkdown of the spent fuel pool area and found duct tape below the water. Numerous pieces of duct tape, including that on fueling and control rod manipulation tools, were not in the logs. The inspectors reviewed Procedure AP 12-003, "Foreign Material Exclusion," Revision 7. The area surrounding the spent fuel pool is posted as a foreign material exclusion area, a contaminated area, and a hot particle area. Procedure AP 12-003 requires the highest level of foreign material accountability, or Level 1, for the spent fuel pool. Level 1 requires several actions: All materials in the area are to be described; all materials are logged in and out; logs specify how material was removed; logs identify the person writing on the log itself; and the pages of the log itself are tracked. The inspectors reviewed the spent fuel pool area logs and concluded the logs were inadequate. Although Wolf Creek logged some duct tape, numerous pieces of duct tape on fuel and control rod tools were not logged.

The inspectors reviewed the spent fuel pool area material tracking practices since the completion of a root cause evaluation and extent of condition review in response to previous NRC finding 05000482/2009002-03. The inspectors found that Wolf Creek performed an extent of condition review to examine the bottom of the cask pit and the spent fuel racks, but failed to identify the duct tape on the tools. The inspectors did not find any documentation stating that the tape was acceptable for use underwater in an acidic environment. Although the tape markings are used for refueling operations, the inspectors found no documentation that would lead Wolf Creek to identify the missing tape. On August 12, 2009, Wolf Creek initiated Condition Report 19110, but this report only asked how to handle the submerged tape and did not identify the failure to log the material. The issue was appropriately captured in the corrective action program with Condition Report 20338.

Analysis. The inspectors determined that the failure to track foreign material in accordance with Procedure AP 12-003 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 the finding was not the result of any willful violation of NRC requirements or Wolf Creek procedures. This finding is more than minor because it impacted the Barrier Integrity Cornerstone attribute of configuration control and affected the cornerstone objective to maintain functionality of the spent fuel pool system. Using Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," this finding was determined to be of very low safety significance because the finding only affected the barrier function of the spent fuel pool and did not result in actual clogging of the system. The inspectors determined that this finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because although Wolf Creek performed a root cause and extent of condition evaluation for untracked foreign material, the evaluation still failed to find the duct tape in the pool itself. This allowed the tape to continue to be untracked [P.1.c].

Enforcement. Technical Specification 5.4.1.a requires the implementation of written procedures described in Regulatory Guide 1.33, Revision 2, Appendix A, including procedures for performing maintenance that can affect the performance of safety-related equipment. Procedure AP 12-003, "Foreign Material Exclusion," Revision 6, requires foreign material accountability for the spent fuel pool. Contrary to the above, prior to August 12, 2009, the licensee failed account for foreign material in the spent fuel pool. Because this violation was determined to be of very low safety significance and was placed in the corrective action program as Condition Report 20338, this violation is being treated as a noncited violation in accordance with Section VI.A.1 of the Enforcement Policy: NCV 05000482/2009004-01, "Failure to Log Foreign Material in Spent Fuel Pool after Extent of Condition Evaluation."

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors reviewed the USAR, the flooding analysis, and plant procedures to assess seasonal susceptibilities involving internal flooding; reviewed the USAR and corrective action program to determine if licensee personnel identified and corrected flooding problems; inspected underground bunkers/manholes to verify the adequacy of sump pumps, level alarm circuits, cable splices subject to submergence, and drainage for bunkers/manholes; verified that operator actions for coping with flooding can reasonably achieve the desired outcomes; and walked down the one area listed below to verify the adequacy of equipment seals located below the flood line, floor and wall penetration seals, watertight door seals, common drain lines and sumps, sump pumps, level alarms, and control circuits, and temporary or removable flood barriers. Specific documents reviewed during this inspection are listed in the attachment.

- September 17, 2009, Essential service water Manhole MHE-2B for cable splice inspections.

These activities constitute completion of one flood protection measures inspection sample as defined by IP 71111.06-05.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Quarterly Inspection

a. Inspection Scope

On September 14, 2009, the inspectors observed a crew of licensed operators in the plant's simulator to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- Licensed operator performance
- Crew's clarity and formality of communications
- Crew's ability to take timely actions in the conservative direction
- Crew's prioritization, interpretation, and verification of annunciator alarms
- Crew's correct use and implementation of abnormal and emergency procedures
- Control board manipulations
- Oversight and direction from supervisors
- Crew's ability to identify and implement appropriate technical specification actions and emergency plan actions and notifications

The inspectors compared the crew's performance in these areas to pre-established operator action expectations and successful critical task completion requirements. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one quarterly licensed-operator requalification program sample as defined in IP 71111.11.

b. Findings

No findings of significance were identified.

.2 Annual Inspection (71111.11B)

The licensed operator prequalification program involves two training cycles that are conducted over a 2-year period. In the first cycle, the annual cycle, the operators are administered an operating test consisting of job performance measures and simulator scenarios. In the second part of the training cycle, the biennial cycle, operators are administered an operating test and a comprehensive written examination.

a. Inspection Scope

The inspector conducted an in office review of the annual prequalification training program operating test results for 2009. The licensee examined fifty operators (twenty-one reactor operators and twenty-nine senior reactor operators) during this prequalification cycle. In addition, nine operating crews were examined on the facility's simulator. All of the operating crews passed the simulator scenarios and all operators passed the operating tests.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed licensee personnel'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:

- Performance of Procedure TMP 09-014, July 15, 2009
- Failure of flow indicator BBFI-425, July 16, 2009
- Component cooling water valves' fuse inspections, March 16, 2009
- Week of August 24, 2009, planned work risk assessment

The inspectors selected these activities based on potential risk-significance relative to the Reactor Safety Cornerstones. As applicable for each activity, the inspectors verified that licensee personnel performed risk assessments as required by 10 CFR 50.65(a)(4) and that the assessments were accurate and complete. When licensee personnel performed emergent work, the inspectors verified that the licensee personnel promptly assessed and managed plant risk. 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 the technical specification requirements and inspected portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of four maintenance risk assessments and emergent work control inspection samples as defined by IP 71111.13-05.

b. Findings

Introduction. The inspector identified a noncited violation of 10 CFR 50.65(a)(4) for failure to adequately assess and manage the increase in risk during fuse inspection of component cooling water valves supplying cooling loads inside containment.

Description. On March 18, 2008, component cooling water Valves EG HV-16 and EG HV-54 were out of service for fuse inspections to verify wiring for fire protection analyses. The inspectors observed that the evolution was not included in the weekly risk assessment. The inspectors noted that operations and maintenance personnel did not have guidance or briefings for restoration of the valves. Review of the risk assessment by Wolf Creek after inspector questioning revealed that the impact of de-energizing the valves in the close position was neglected and that restoration actions credited by the risk analyst were unknown to the control room and craft workers. Specifically, Condition Report 0015318 states that loss of reactor coolant pump thermal barriers was possible, but it did not state that it was possible to also lose the reactor coolant pump seal water cooling heat exchanger, reactor coolant pump radial bearing cooling, reactor coolant pump motor cooling, the letdown heat exchanger, and the excess letdown heat exchanger. Due to these additional loads, this would also be a reactor trip initiator initiating event.

A local clearance order was assumed to provide local control and restoration instructions but Wolf Creek later found that these actions were not possible because the clearance order contained no restoration instructions. Inspectors reviewed the work package for the fuse inspections and found that the pre-job briefing did not contain any instructions to craftsmen for rapid restoration. Procedure AP 22C-003, Revision 13 Attachment A required risk management actions due to the potential for a trip initiator and the potential to interrupt the thermal barrier heat exchangers for the reactor coolant pump seals. The control room and maintenance personnel were not aware of the restoration actions assumed in the risk assessment. Only the risk engineers were aware of the restoration actions. Wolf Creek's evaluation of the issue found that the maintenance planning group, the risk assessment engineers, and operations were not procedurally required to discuss the restoration actions when changes were made during maintenance planning in the prior weeks.

Analysis. The failure to adequately assess and manage risk in accordance with AP 22C-003 and the preplanned risk assessment for the use of local actions to ensure component cooling water cooling to loads inside containment 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 the finding was not the result of any willful violation of NRC requirements or Wolf Creek procedures. The finding is more than minor because the licensee failed to effectively manage prescribed significant compensatory measures for maintenance activities that could increase the likelihood of initiating events. The finding was of very low safety significance because the magnitude of the calculated risk deficit was less than 1 x 10-6 even though risk management actions were not in place. The inspectors also determined that the finding has a human performance crosscutting aspect in the area associated with work control because the risk assessment procedure and clearance order procedure assumed local actions could be accomplished but there was no communication regarding this during the work planning stages [H.3(b)]

Enforcement. 10 CFR 50(a)(4), requires, in part, that before performing maintenance activities (including but not limited to surveillance, post maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. Procedure AP 22C-003, Revision 13, and the resulting weekly risk assessment implement this regulation. Contrary to the above, on March 18, 2009, the licensee did not effectively manage the increase in risk resulting from a maintenance activity. Specifically, on March 18, 2009, during fuse inspections of component cooling water Valves EG HV-16 and EG HV-54, the licensee failed to adequately assess and manage the increase in risk that resulted from the maintenance activity. Restoration actions credited in Wolf Creek's weekly risk assessment were determined to be not possible to implement. The licensee entered this issue into their corrective action program as Condition Report 15318. Because the licensee has entered the issue into their corrective action program and the finding is of very low safety significance, this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 5000482/2009004-02, "Inability to perform manual actions for risk assessment."

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- Essential service water piping through wall leakage, separate occurrences on June 30, July 28, and August 19, 2009
- Diesel generator common cause failure evaluation on June 30, 2009
- Performance of procedure SYS SB-122 on August 22, 2009
- Nonconservative core flux technical specification on August 5, 2009

The inspectors selected these potential operability issues based on the risk-significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that technical specification operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the technical specifications and USAR to the licensee's evaluations, to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors also reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of four operability evaluations inspection samples as defined in IP-1111.15-05

b. Findings

.1 Introduction. The inspectors identified a Green noncited violation of Technical Specification 3.8.1 for failure to perform an adequate common cause evaluation within 24 hours to demonstrate no common cause failure mechanism existed between the operable and inoperable emergency diesel generators.

Description. At 11:15 a.m., on June 30, 2009, Wolf Creek auxiliary building watch discovered a through-wall leak in the essential service water Train B Piping EF-138-HBC-30 just upstream of valve EF-HV-0038. The piping was leaking through two adjacent pinholes at the bottom of the pipe spaced approximately 0.4 inch apart. This condition was recognized as a limiting condition of operations per Condition A of Technical Requirements Manual 3.4.17, "Structural Integrity," which requires the structural integrity of all ASME Class I, II, and III piping to be maintained. The required action directed operators to declare the essential service water Train B inoperable. Thus Wolf Creek entered Condition A of Technical Specification 3.7.8 "Essential Service Water," for one train of essential service water inoperable. This condition has a required action of restoring the essential service water train to operable status within 72 hours, but it also requires simultaneous entry into Condition B of Technical Specification 3.8.1, "AC Sources Operating," for the emergency diesel generator made inoperable by the essential service water system. There are four required actions associated with Technical Specification 3.8.1, Condition B. First, Required Action B.1, the control room operators are to verify correct breaker alignment and indicated power availability for each offsite power circuit within 1 hour and every 8 hours thereafter. Second, Required Action B.2 requires that features supported by the inoperable diesel generator be declared inoperable when its required redundant feature is inoperable within 4 hours. Third, Required Action B.3.1 requires Wolf Creek to determine that the operable diesel generator is not inoperable due to a common cause failure. Alternatively, Required Action B.3.2 directs Wolf Creek to verify the operable diesel generator starts from standby conditions and achieves steady state voltage and frequency, within 24 hours. Fourth, Required Action B.4.1 directs the restoration the diesel generator to operable status within 72 hours. Wolf Creek properly carried out Required Actions B.1, B.2, and B.4.1 required by Technical Specification 3.8.1, Condition B.

At 12:02 p.m., 47 minutes after the leak was discovered, the control room logs state that Technical Specification 3.8.1, Action B.3.1, is being exited because "Emergency Diesel Generator B inoperable due to ESW being inoperable not a common cause failure." The inspectors interviewed operations personnel on the adequacy of such a justification. Operations provided the inspectors with a completed copy of Procedure SYS KJ-200, "Inoperable Emergency Diesel." Procedure Step 6.1.5 states: "If the absence of any potential common cause failure can be demonstrated . . . then document the evaluation on the cover sheet." However, the cover sheet had only one sentence which matched the log entry verbatim. At the time of this determination, ultrasonic testing to determine flaw size and pipe wall thicknesses had yet to be performed. The results of that testing were the basis for an ASME N513.2 code case which eventually restored operability. During later interviews regarding the control room log entries, Wolf Creek stated that nonlicensed operators did not find any other through wall leaks on essential service water Train B, and therefore Train B was operable. The inspectors found that this type of visual evaluation did not meet the reasonable assurance standard specified in RIS 2005-20. Visual examinations can not identify below minimum wall thickness piping or piping flaws under insulation. The inspectors concluded the licensee's evaluation lacked a valid technical basis for determination that a common cause failure mechanism did not exist on the opposite train emergency diesel generator.

The ASME N513.2 code case was issued and essential service water/emergency diesel generator operability restored at 9:40 p.m. that night. The code case verified the structural integrity of the piping despite the current through-wall flaw; however, it specified that due to the potential common cause nature of pitting flaws, five additional locations had to be ultrasonic tested to verify that minimum wall thickness was met. Although none of the additional locations indicated any below minimum-wall flaws in the essential service water piping, an expanded ultrasonic test of the leak area revealed two additional pits that were below the minimum wall thickness acceptance criteria. Separate evaluations were performed for those flaws and all three were permanently repaired per the ASME code on July 23, 2009.

Analysis: The inspectors determined that the failure to demonstrate operability of Emergency Diesel Generator B per Technical Specification 3.8.1, Required Action B.3.1 or B.3.2 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 the 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 equipment performance 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). Specifically, this issue relates to the availability and reliability examples of the equipment performance attribute because a latent common mode failure mechanism was not correctly evaluated. The inspectors evaluated the significance of this finding using Phase 1 of Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At Power Situations," and determined that the finding was of very low safety significance (Green) because the issue was not a design or qualification deficiency confirmed to result in loss of operability or functionality, did not represent a loss of system safety function, an actual loss of safety function of a single train for greater than its technical specification allowed outage time, an actual loss of safety function of a nontechnical specification risk-significant equipment train, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The cause of the finding has a problem identification and resolution crosscutting aspect in the area associated with the corrective action program because Wolf Creek failed to thoroughly evaluate the failure mechanism such that the resolutions address the causes and extent of conditions, as necessary. Specifically, Wolf Creek did not properly consider the possibility of common-cause pitting failures which could have impacted the essential service water Train A piping structural integrity thereby affecting its cooling loads, including Emergency Diesel Generator A (P.1(c)).

Enforcement: Technical Specification 3.8.1 Required Actions B.3.1 and B.3.2 require, with one diesel generator inoperable, to determine that the operable diesel generator is not inoperable due to common cause failure or else perform SR 3.8.1.2 [run the diesel generator]. Contrary to this requirement, on June 30, 2009, the licensee failed to demonstrate that Emergency Diesel Generator A was operable by evaluation of common cause failure or by performing SR 3.8.1.2 while emergency diesel generator B was inoperable due to essential service water piping corrosion. Specifically, the control room logs exited Required Action B.3.1 stating that “EDG B inoperable due to ESW being inoperable not a common cause failure.” No further evaluation was provided. Because the finding is of very low safety significance and has been entered into the corrective action program as Condition Report 18347, this violation is being treated as an noncited violation, consistent with Section VI.A of the NRC Enforcement Policy:
NCV 05000482/2009004-03: “Inadequate Evaluation of Emergency Diesel Generator for Common Cause Failure in the Supporting Essential Service Water System.”

.2 Introduction. On August 22, 2009, the inspectors identified a violation of Technical Specification 3.0.3 in which both trains of a Technical Specification 3.3.2 interlock in the engineered safety features actuation system were bypassed with jumper wires in accordance with plant procedure.

Description. On August 22, 2009, the inspectors observed that both trains of Technical Specification 3.3.2, function 8.a, P-4, were bypassed while in Mode 3. The inspectors found that Wolf Creek installed jumper wires on both trains in accordance with Procedure SYS SB-122, "Enabling/Disabling P-4/Lo Tave FWIS [feed water isolation signal]." The inspectors found that Wolf Creek has installed the jumper wires on both trains in the past to support reactor trip breaker and control rod drop testing in Mode 3. The jumpers defeated the function of both trains of reset switches on the main control board such that a P4/FWIS cannot be sent to close feedwater valves and trip the main feedwater pumps.

The inspectors reviewed the technical specification bases for the engineered safety features actuation system interlocks and function 8.a. The bases and USAR state that the functions of the interlock are to: 1) trip the main turbine, 2) isolate main feed water coincident with lo Tavg, 3) allow manual block of the automatic re-actuation of safety injection after a manual reset of safety injection, 4) allow arming of the steam dump valves and transfer the steam dump from the load rejection Tavg controller to the plant trip controller, 5) prevents opening of the main feed water isolation valves if they were closed on safety injection or steam generator hi-hi water level. The inspectors found that this was consistent with the standard improved technical specifications for Westinghouse plants and the Wolf Creek USAR, Table 7.3-15, "NSSS Interlocks for Engineered Safety Feature Actuation System." Under License Amendment 123, Wolf Creek converted to improved standard technical specifications in December 1999. The P-4 interlock description has not changed since 1999. The licensee submittals acknowledged that the functions of P-4 were not part of a design basis analysis, but were retained in the technical specifications to limit reactor coolant system cooldown following a reactor trip. Technical Specification 3.3.2 states that "The ESFAS [engineered safety features actuation signal] instrumentation for each Function in Table 3.3.2 shall be OPERABLE According to Table 3.3.2-1." Function 8 of Table 3.2.-1 covers interlocks and specifically interlock 8.a, P-4, is required to be Operable in Modes 1, 2, and 3. The inspectors found that function 8.a is required in Modes 1, 2, and 3. The inspectors consulted with the Office of Nuclear Reactor Regulation's technical specification branch and found that statements in the bases provide a summary of the technical specification and do not override requirements. The sentence in the bases that states: "This Function must be OPERABLE in MODES 1, 2, and 3 when the reactor may be critical or approaching criticality," clarifies why it is required in Modes 1, 2, and 3 and does not permit P-4 to be inoperable if the reactor is not approaching criticality. Operators are trained to anticipate criticality such as during control rod-drive motor-generator testing during August 22-23, 2009.

During interviews, Wolf Creek stated that it was necessary to bypass the P4/FWIS in order to perform rod-drive motor-generator set testing that cycled the reactor trip breakers. Wolf Creek contended that the P-4/FWIS was not necessary to assure compliance with the plant safety analysis. Lastly, Wolf Creek stated that during Mode 3 after refueling outages, it was necessary to install jumpers and bypass the P-4/FWIS for rod-drop testing because operation of the main feedwater system in automatic level control was more desirable than having an operator manually control steam generator levels with auxiliary feedwater. The inspectors agreed that this interlock is not assumed in Chapter 15 of the USAR, but the inspectors found that the Wolf Creek technical specification bases state that “ESFAS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii)” which is identical to the generic standard specifications approved by the NRC. The inspectors found that there are several technical specification systems such as steam generator atmospheric relief valves, the condensate storage tank, and pressurizer power operated relief valves that are not in Chapter 15 of the USAR but are required to be operable under technical specifications per 10 CFR 50.36. Thus, the inspectors found that the interlock’s absence in Chapter 15 of the USAR does not mean it is not required by the technical specification. Wolf Creek previously evaluated this condition in Performance Improvement Request 2001-0041 which concluded this P-4/FWIS was not required to be operable in any Mode because it is not credited in Chapter 15 of the USAR. Wolf Creek also used other plants with NRC approved safety evaluations to justify the use of Procedure SYS SB-122 rather than requesting a license amendment. The inspectors found that these conclusions are incorrect.

The inspectors found that control room operators did not log the inoperability of P-4 until after inspector questioning, and afterward, operators incorrectly applied Technical Specification 3.3.2, Condition F, which allowed 60 hours to return one train of the interlock to service. With both trains of P4 bypassed, Technical Specification 3.0.3 applied and Wolf Creek had 13 hours to be in Mode 4. The P-4 interlock was inoperable for approximately 20 hours from August 22-23, 2009. Wolf Creek missed the transition to Mode 4.

Analysis. The inspectors found that the failure to evaluate implement Technical Specification 3.3.2 interlock, function 8.a was a performance deficiency. The inspectors determined that this finding was more than minor because it is associated with the design control attribute of the Mitigating Systems Cornerstone and it affected the cornerstone objective to ensure the availability, reliability, and capability of mitigating systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The inspectors evaluated the significance of this finding using Inspection Manual Chapter 0609.04, “Phase 1 - Initial Screening and Characterization of Findings,” and screened the finding to Phase 2 because the finding represents a loss of a system’s function. The inspectors used Inspection Manual Chapter 0609, Appendix A and screened the finding to the NRC senior reactor analyst for review because there was not an acceptable equipment deficiency in the pre-solved worksheet. The senior reactor analyst determined that the finding is Green because he solved Table 3.10 of the Risk-Informed Inspection Notebook for Wolf Creek Generating Station, Revision 2.1a and found that the loss of feedwater isolation signal for less than 3 days resulted in a 1E-7 (Green) outcome. The inspectors also determined that the cause of the finding has a crosscutting aspect in the human performance area associated with decision making because Wolf Creek failed to make a risk significant decision using a systematic process. This issue was evaluated more than once and those evaluations sought to justify bypassing the interlock rather than seek the full regulatory basis for the interlock. [H.1.a]

Enforcement. Wolf Creek Technical Specification, Table 3.3.2.1, function 8 includes engineered safety features actuation system interlocks. Function 8.a, the P-4 interlock, requires two trains to be operable in Modes 1, 2, and 3. Function 8.a does not provide a required action for both trains of engineered safety features actuation system interlocks inoperable. Wolf Creek Technical Specification 3.0.3 requires the plant to be in Mode 4 within 13 hours if there is no required action specified for a limiting condition of operation that cannot be met. Contrary to the above, from August 22 to August 23, 2009, Wolf Creek failed to change modes from Mode 3 to Mode 4 when both trains of engineered safety features actuation system interlock function 8.a, P-4, were inoperable for greater than 13 hours. Specifically, from August 22 to 23, 2009, Wolf Creek failed to change modes from Mode 3 to Mode 4 when both trains were removed from service for approximately 20 hours. Because this violation was determined to be of very low safety significance and was placed in the corrective action program as Condition Report 19318, this violation is being treated as a noncited violation in accordance with Section VI.A.1 of the Enforcement Policy: NCV 05000482/2009004-04, "Failure to Implement Engineered Safety Features Actuation System Technical Specification Results in Missed Mode Change."

1R17 Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors reviewed the effectiveness of the licensee's implementation of evaluations performed in accordance with 10 CFR 50.59, "Changes, Tests, and Experiments," and changes, tests, experiments, or methodology changes that the licensee determined did not require 10 CFR 50.59 evaluations. The inspection procedure requires the review of 6 to 12 licensee evaluations required by 10 CFR 50.59, 12 to 25 changes, tests, or experiments that were screened out by the licensee and 5 to 15 permanent plant modifications.

The inspectors reviewed 9 evaluations required by 10 CFR 50.59. These included:

- 2006-001, Radiological Consequences of a Fuel Handling Accident, Revision 0
- 2008-0006, Wolf Creek Generating Station (WCGS) Simplified Head Assembly (SHA) Drop Analysis, Revision 0
- 2008-0008, Use of Dedicated Operator for SI Pump B Room cooler Replacement, Revision 0
- 2005-004, WCGS Rod Withdrawal at Power Event Safety Analysis, Revision 0
- 2008-001, Evaluations of Voids in the ECCS Suction Piping, Revision 0
- 2008-002, Evaluations of Voids in the ECCS Discharge Piping, Revision 0
- 2006-002, Power Operation, Revision 54

- 2008-0003, Use of Dedicated Operator for SI Pump A Room Cooler Replacement , Revision 0

- 2008-0004, MSFIS Controls Replacement , Revision 0

The inspectors reviewed 17 changes, tests, and experiments that were screened out by licensee personnel. These included:

- CP 12731, RCP No. 1 Seal Housing Stud Preload Evaluation, Revision 0

- CP 12746, Torque of Piping Flanges Between EDG Heat Exchangers, Revision 1

- CP 12820, Containment Room Cooler SGN01D, Revision 1

- CP 12876, Main Steam Atmospheric Relief Valve Aux (Pilot) Plug and Main Plug Machining Dimensions, Revision 0

- CP 12979, Updating the RCS pressure and temperature limits, PORV lift setting for the LTOP system, and the PTLR, Revision 0

- CP 13089, EF-138-HBC-30 Essential Service Water Pipe Pit Encapsulation, Revision 1

- CP 11987, EKJ03A/B Replacement Heat Exchangers, Revision 6

- CP 12758, Coating Degradation and Isolated Pitting of Containment Incore Instrumentation Sump Layer, Revision 3

- CP 12240, Over Torque on Valve GTHZ0008, Revision 0

- CP 12273, Shrinkage Effect at the Pressurizer Spray Nozzle on TBB03 Due to Weld-Overlay, Revision 3

- CP 12489, SGK05A Tube Sheet and Channel Cover Degradation Evaluation, Revision 0

- CP 12341, Region 19 Fuel Assembly and Core Component Configuration Changes, Revision 0

- CP 12154, Relocate CVT Level transmitter BGLT0185, Revision 3

- CP 12175, PFSSD MOV Hot Short Mod: BGHV8111, BNLCV0112E, EMHV8803B, Revision 0

- CP 12639, 9 Volt Power Supply for SP067 & SP010 , Revision 0

- CP 12782, NE107187 DG NE01 Generator Differential relay, Revision 0

- TMP 08-022 , SI Accumulator C Boron Concentration Adjustment, Revision 0

The inspectors reviewed 7 permanent plant modifications. These included:

- CP 11987, EKJ03A/B Replacement Heat Exchangers, Revision 6

- CP 11379, Replacement for Obsolete Rad Monitoring Transducer, Revision 2

- CP 13089, EF-138-HBC-30" Essential Service Water Pipe Pit Encapsulation, Revision 1

- CP 12673, Installation of Vents in the Bonnets of EJ8958A and EJ8958B, Revision 1

- CP 9488, Governor Replacement on Emergency Diesel Generators, Revision 7

- CP 11608, MSIV and MFIV Actuator Replacement Electrical Work, Revision 10

- CP 11897, Transformer XNB02 Tap Change, Revision 2

The inspectors verified that when changes, tests, or experiments were made, that evaluations were performed in accordance with 10 CFR 50.59 and that licensee personnel had appropriately concluded that the change, test or experiment can be accomplished without obtaining a license amendment. The inspectors also verified that safety issues related to the changes, tests, or experiments were resolved. The inspectors reviewed changes, tests, and experiments that licensee personnel determined did not require evaluations and verified that the licensee personnel's conclusions were correct and consistent with 10 CFR 50.59. The inspectors also verified that procedures, design, and licensing basis documentation used to support the changes were accurate after the changes had been made and that preparers and reviewers of the evaluations and screens were qualified and certified in accordance with licensee procedures.

During the portion of the inspection dealing with modifications, the inspectors verified that supporting design and license basis documentation had been updated accordingly and was still consistent with the new design. The inspectors verified that procedures, training plans and other design basis features had been adequately accounted for and updated. Additional documents reviewed during this inspection are listed in the attachment.

The inspectors verified that the licensee was identifying permanent plant modification issues and problems related to 10 CFR 50.59 applicability determinations, screenings and evaluations, and had entered them in the corrective action program. The inspectors selected several samples to evaluate the appropriateness of the corrective actions program. No program concerns were identified with corrective action documents reviewed.

These activities constitute completion of one sample as defined in IP 71111.17-05

b. Findings

No findings of significance were identified.

1R18 Plant Modifications (71111.18)

a. Inspection Scope

The inspectors reviewed the following temporary/permanent modifications to verify that the safety functions of important safety systems were not degraded:

- Emergency diesel Generator B oil collection, August 13, 2009
- Heat tracing for the boric acid system, March 26, 2009

The inspectors reviewed the temporary modification and the associated safety evaluation screening against the system design bases documentation, including the USAR and the technical specifications, and verified that the modification did not adversely affect the system operability/availability. The inspectors also verified that the installation and restoration were consistent with the modification documents and that configuration control was adequate. Additionally, the inspectors verified that the temporary modification was identified on control room drawings, appropriate tags were placed on the affected equipment, and licensee personnel evaluated the combined effects on mitigating systems and the integrity of radiological barriers.

These activities constitute completion of two samples for temporary plant modifications as defined in IP 71111.18-05.

b. Findings

Introduction. The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, for failing to translate the boric acid design basis into time sensitive operator actions to ensure the core operating limits report shutdown margin can be achieved with the boric acid flow path.

Description. On March 27, 2009, the inspectors walked down the safety injection pump Room A and noted a temporary modification of heat tracing installed on boric acid piping. The heat tracing was plugged into a nonsafety-related wall outlet for power. From the boric acid tanks, the highly concentrated boric acid piping travels to the safety injection pump Room A and then to the centrifugal charging pump suctions. The inspectors reviewed the temporary modification documentation and found that Wolf Creek had written Performance Improvement Request 2005-3461 in December 2005, stating that this piping carried boric acid. Performance Improvement Request 2005-3461 identified that, if the room coolers were started while lake temperature was low, the room temperature may decrease below the solubility limit. It also identified that compensatory actions may be needed. Corrective actions for heat tracing and instructions to operators took approximately 3 years to implement, and stopped short of addressing boric acid system operation when nonsafety power is lost to the heat tracing and the plant must be taken to cold shutdown in accordance with technical specifications or plant conditions. Achieving cold shutdown using only safety-related components is consistent with Section 9.3 of the USAR. Control room operators had no procedural guidance to ensure that boration would be performed prior to the room and piping cooling to below the boric acid precipitation temperature and blocking the piping. The core operating limits report requires a cold shutdown margin of 1300 percent milirho (pcm). The inspectors found that the procedural path to borating to cold shutdown conditions would likely take longer than the time for the piping to cool to the boric acid precipitation temperature. Wolf Creek performed an informal room heat loss calculation, but neglected forced cooling by the room cooler, particularly with low lake temperature. The other boric acid source is the refueling water storage tank which is not protected from external event such as tornados. Therefore, the refueling water storage tank is not available in all safe shutdown scenarios.

Wolf Creek also performed an informal simulator evaluation with licensed operators. The scenario involved a loss of offsite power without the refueling water storage tank available. The inspectors noted that the operators in the informal evaluation took less time to arrive at the key boration steps in emergency procedures than the operators did during an actual loss of offsite power event of August 19, 2009. The inspectors also noted the August 19 event was less complicated than the simulator scenario, and the simulator evaluation also did not involve emergency action level declarations or loss of large portions of other equipment due to external events, such as a tornado. The inspectors determined that these factors would add considerable time to that demonstrated by the informal simulator evaluation. The inspectors concluded that the licensee had failed to demonstrate that boration could be accomplished prior to boric acid precipitation following a loss of nonsafety-related electrical power.

The inspectors also reviewed Procedure SYS BG-206, “Boric Acid System Operation,” and found that the solubility limit for a 7680 parts per million boric acid solution is 63 degrees Fahrenheit. The inspectors found log entries from March 27, 2008, and February 8, 2009, in which room temperature decreased to 67 and 58 degrees and could have challenged the boric acid system by blocking the piping with precipitated boron. However, the inspectors found that the refueling water storage tank was operable and could have performed the reactivity control function in certain scenarios that do not involve tornados or external events. Using these factors, inspectors concluded that Wolf Creek had less time to accomplish more lengthy tasks in order to perform boration to cold shutdown conditions.

The inspectors reviewed the corrective action history for heat tracing Temporary Modification 07-012-BG. The inspectors reviewed Condition Report 2005-3461 and found that it was continued under Condition Report 2007-2472. Condition Report 2007-2472 created Corrective Action 4222 which was to plan and install heat tracing under a temporary modification. The temporary modification installation work order began on October 29, 2008. Condition Report 2007-2472 also had a corrective action to issue guidance to nonlicensed operators taking temperature readings in the safety injection pump Room A. These updated logs were implemented on December 19, 2008, and instructed operators that the boric acid piping may become inoperable due to precipitation if room temperature dropped below 67 degrees Fahrenheit. There was no guidance to operators in the control room regarding this time sensitive manual action.

<u>Analysis.</u> The failure to translate the design bases into procedures that ensure the function of the safety-related boric acid system upon loss of nonsafety-related heat

tracing is a performance deficiency. The inspectors determined that this finding was more than minor because this issue aligned with Inspection Manual Chapter 0612, Appendix E, example 2.f, because the pipe temperature was required to stay above the boric acid solubility limit and the loss of the heat tracing and or room temperature decrease will block the boric acid system. This issue was associated with the equipment performance attribute of the mitigating systems cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events. The inspectors evaluated the significance of this finding using Phase 1 of Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At Power Situations," and determined that the finding screened to phase 2 because the issue was a design or qualification deficiency confirmed to result in loss of operability or functionality The inspectors evaluated the significance of this finding using Phase 2 of Inspection Manual Chapter 0609, Risk Informed Inspection Notebook for Wolf Creek Generating Station, and determined that the finding was of very low safety significance because loss of the boric acid system in Table 3.9 for one year resulted in a 1E-7 CDF when giving recovery credit for the refueling water storage tank. The inspectors determined that this finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Wolf Creek did not take appropriate corrective actions to resolve known deficiencies in the design and operation of the boric acid system for approximately 4 years. The issue was re-evaluated in 2009 and failed to correct the deficiencies identified in 2005 [P.1.d].

Enforcement. Part 50 of Title 10 of the Code of Federal Regulations, Appendix B, Criterion III, "Design Control," requires, in part, that the design basis is correctly translated into specifications, drawings and procedures. Achieving cold shutdown using only safety-related components is consistent with Section 9.3 of the USAR. Contrary to the above, since December 16, 2005, Wolf Creek has failed to ensure that the boric acid system could perform its design function as specified in USAR, Section 9.3. Specifically, Wolf Creek failed to ensure that time-sensitive operator actions to ensure the core operating limits report specified shutdown margin can be achieved prior to boric acid precipitates blocking the flow path. Because this violation is of very low safety significance and has been entered into Wolf Creek's corrective action program as condition report 20717, this violation is being treated as an noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000482/2009004-05, "Use of Nonsafety-Related Power to Ensure Operability of Safety-Related Boric Acid System."

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the following postmaintenance activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- Emergency Diesel Generator B run after compression fitting lube oil leak repaired on August 17, 2009

- Turbine-Driven auxiliary feedwater pump run after trip and throttle valve maintenance on September 9, 2009

- Component cooling water train swaps after modification to valves on August 14, 2009

- Testing after repair to Emergency Diesel Generator A on December 5, 2008

- Replacement of Flow Transmitter BG FK-121 on August 28, 2009

- Limitorque and gearbox overhaul of essential service water Valve EF HV-31 on August 31, 2009

- Essential service water Valve EF HV-42 after maintenance on August 12, 2009

- Safety Bus NB02 Channel 4 under-voltage relay power supply replacement on March 24, 2009

The inspectors selected these activities based upon the structure, system, or component's (SSC) ability to affect risk. The inspectors evaluated these activities for the following:

- The effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed

- Acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate

The inspectors evaluated the activities against the technical specifications, the USAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with postmaintenance tests to determine whether the licensee was identifying problems and entering them in the corrective action program and that the problems were being corrected commensurate with their importance to safety. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of eight postmaintenance testing inspection samples as defined in IP 71111.19-05.

b. Findings

Introduction. The inspectors identified a Green noncited violation of Technical Specification 3.8.1.B.4 in which the licensee removed equipment from service that was required by technical specifications and the NRC safety evaluation.

Description. On March 24, 2009, the licensee entered Technical Specification 3.8.1, Required Action B.4.2.2. This action allowed an emergency diesel generator to be inoperable for up to 7 days. On March 24, 2009, at 4:20 p.m., the inspectors noted that

Wolf Creek performed Procedure STS IC-208B, “4kV Loss of Voltage and Degraded Voltage TADOT NB02 Bus – Separation Group 4,” Revision 2A, to determine the ‘as-found’ conditions of the Channel 4 under voltage power supply. Operators entered Technical Specification 3.3.5, Condition A.1 and exited 19 minutes later. The power supply voltage ripple passed Procedure STS IC-208B, but Wolf Creek elected to replace it. Again on March 24, 2009, at 4:54 p.m., Wolf Creek entered Technical Specification 3.3.5, Condition A.1, to replace the subject Channel 4 power supply. Condition A.1 required the out-of-service channel to be placed in trip within 6 hours. Wolf Creek exited Technical Specification 3.3.5 at 9:09 p.m., on March 24. The removal of Channel 4 from service resulted in a higher probability of loss of power to the safety bus because the coincidence logic changed from two out of four to one out of three. The inspectors found that this logic was an input to the NB02 normal offsite power feeder breaker described in the offsite power surveillance procedure, STS NB-005, “Breaker Alignment Verification,” Revision 18.

The inspectors reviewed Technical Specification Bases 3.8.1.B.4 which prohibits elective maintenance within the switchyard that would challenge offsite power while in the 7-day emergency diesel generator extended outage. The inspectors also reviewed the NRC Safety Evaluation Report (SER) for the 7-day emergency diesel generator allowed outage time (Technical Specification 3.8.1.B.4.2.2) and found that Section 4.6.c, states: “The <u>offsite power supply</u> [emphasis added] and switchyard conditions are conducive to an extend[ed] DG [completion time], which includes ensuring that switchyard access is restricted and no elective maintenance within the switchyard is performed that would challenge the offsite power availability.” Additionally, Condition D of the technical specification bases states that no equipment or systems assumed to be available for the extended emergency diesel generator completion time are removed from service, which includes auxiliary feedwater, component cooling water, essential service water and their support systems. The support equipment protections are also mirrored in Section 4.0 of the NRC safety evaluation for Amendment 163. However, Wolf Creek removed one channel of under voltage protection for offsite power to Bus NB02 (Train B) which is a support system for the above equipment. The inspectors found that Procedure STS IC-208B permits the testing of degraded voltage relays while the diesel is out of service. These relays control the opening logic for the normal offsite power feed to the safety bus NB02. Additionally, Procedure AP 22C-003, “Operational Risk Assessment Program,” Revision 13, prohibits elective maintenance within the switchyard that would challenge offsite power during Technical Specification 3.8.1.B.4.2.2. Normally the safety bus NB02 cabinets are protected equipment (no work allowed) but because this work was planned in advance for the diesel outage, the work was permitted. In consultation with the Office of Nuclear Reactor Regulation, the inspectors concluded that Procedure STS IC-208B and power supply replacement was inappropriate during the 7-day diesel outages because it increased the probability of the loss of offsite power to safety equipment that could not be powered by the diesel. Wolf Creek appropriately restricted access to the portion of the switchyard outside the protected area but did not appropriately restrict work for offsite power inside the protected area. The inspectors determined that challenges to offsite power can originate with elective maintenance inside the protected area. The inspectors found that Wolf Creek assessed risk under 10 CFR 50.65 a(4) for this evolution, resulting in elevated risk within the Green band during the 7-day diesel outage. The inspectors also found that Wolf Creek appropriately protected component cooling water, emergency service water, instrument busses, dc busses, emergency core cooling, the Train A diesel, and control room ventilation.

The inspectors reviewed corrective actions from NCV 05000482/2008002-02 previously identified by inspectors when Wolf Creek made one of the offsite power sources inoperable during a 7-day diesel outage. The licensee reviewed Procedure STS IC-208B but did not revise it because the load shedder and emergency load sequencer procedure tests one channel at a time. No other expanded explanation was articulated in Condition Report 2008-0489. Condition Report 15727 was initiated for the March 24, 2009, maintenance, and the issue has since been corrected by Wolf Creek.

Analysis. The inspectors determined that the failure to implement requirements of Technical Specification 3.8.1 and the associated NRC safety evaluation 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 the finding was not the result of any willful violation of NRC requirements or Wolf Creek procedures. The finding was more than minor because it is associated with the equipment performance attribute for the Mitigating Systems Cornerstone and 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). Specifically, this issue relates to the availability and reliability examples of the equipment performance attribute because an offsite power source was at greater risk of being lost. The finding was determined to be of very low safety significance because the issue did not result in the Train B offsite power being inoperable for greater than 24 hours and did not involve external events such as flooding. Additionally, the cause of the finding has a problem identification and resolution crosscutting aspect in the area associated with the corrective action program. Specifically, Wolf Creek did an extent of condition review in response to a previous violation which included Procedure STS IC-208B, but still failed to prohibit performance of Procedure STS IC-208B during 7-day diesel outages [P.1(c)].

Enforcement. Technical Specification 3.8.1, Required Action B.4.2.2, permits one diesel generator to be inoperable for 7 days provided the limitations articulated in the NRC SER for License Amendment 163 are met. The NRC SER for License Amendment 163 requires that the offsite power supply and switchyard conditions be conducive to an extended diesel generator completion time, which includes ensuring that switchyard access is restricted and no elective maintenance within the switchyard is performed that would challenge the offsite power availability. Contrary to the above, on March 24, 2009, Wolf Creek performed elective maintenance which challenged offsite power availability while emergency diesel generator B was in the 7-day extended completion time. Specifically the licensee performed maintenance on the safety bus NB02 degraded and undervoltage voltage relay Channel 4 power supply while the emergency diesel generator Train B was in an extended outage. Because the finding is of very low safety significance and has been entered into the corrective action program as Condition Report 15727, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000482/2009004-06, "Performing Prohibited Elective Maintenance on Safety Bus NB02 Channel 4 during Emergency Diesel Generator Maintenance."

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the outage safety plan and contingency plans for the Wolf Creek outage conducted from August 19 to August 24, 2009, to confirm that licensee personnel 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 forced outage, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below.

- Configuration management, including maintenance of defense indepth, is commensurate with the outage safety plan for key safety functions and compliance with the applicable technical specifications when taking equipment out of service.
- Clearance activities, including confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing.
- Status and configuration of electrical systems to ensure that technical specifications and outage safety-plan requirements were met, and controls over switchyard activities.
- Monitoring of decay heat removal processes, systems, and components.
- Controls over activities that could affect reactivity.
- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the drywell (primary containment) to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing.
- Licensee identification and resolution of problems related to the August 19, 2009, forced outage activities.

Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one refueling outage and other outage inspection sample as defined in IP 71111.20-05.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the USAR, procedure requirements, and technical specifications to ensure that the four surveillance activities listed below demonstrated that the SSCs tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed test data to verify that the significant surveillance test attributes were adequate to address the following:

- Preconditioning
- Evaluation of testing impact on the plant
- Acceptance criteria
- Test equipment
- Procedures
- Jumper/lifted lead controls
- Test data
- Testing frequency and method demonstrated technical specification operability
- Test equipment removal
- Restoration of plant systems
- Fulfillment of ASME code requirements
- Updating of performance indicator data
- Engineering evaluations, root causes, and bases for returning tested systems, structures, and components not meeting the test acceptance criteria were correct
- Reference setting data
- Annunciators and alarms setpoints

The inspectors also verified that licensee personnel identified and implemented any needed corrective actions associated with the surveillance testing.

- 4kV loss of voltage and degraded voltage TADOT NB02 bus, July 14, 2009
- Essential service water Pump A inservice test, August 13, 2009
- End of life moderator temperature coefficient measurement, August 28, 2009

- August 12, 2009, missed surveillance for over power deltaT and over temperature deltaT

Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of four surveillance testing inspection samples as defined in IP 71111.22-05.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Data Submission Issue

a. Inspection Scope

The inspectors performed a review of the data submitted by the licensee for the 2nd Quarter 2009 performance indicators for any obvious inconsistencies prior to its public release in accordance with Inspection Manual Chapter 0608, "Performance Indicator Program."

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

b. Findings

No findings of significance were identified.

.2 Unplanned Scrams with Complications

a. Inspection Scope

The inspectors sampled licensee submittals for the unplanned scrams with complications performance indicator for the period from the 1st quarter 2008 through the 2nd quarter 2009. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, was used. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports, and NRC integrated inspection reports for the period of January 1, 2008, through June 30, 2009, 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. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of one unplanned scrams with complications sample as defined by IP 71151-05.

b. Findings

Wolf Creek will submit a Frequently Asked Question to determine if the April 19, 2009, unplanned scram should also be counted as a scram with complications.

.3 Safety System Functional Failures

a. Inspection Scope

The inspectors sampled licensee submittals for the safety system functional failures performance indicator for the period from the 1st quarter 2008 through the 2nd quarter 2009. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, and NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73" definitions and guidance were used. The inspectors reviewed the licensee's operator narrative logs, operability assessments, maintenance rule records, maintenance work orders, issue reports, event reports and NRC Integrated Inspection reports for the period of January 1, 2008, through June 30, 2009, 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 three were identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of one safety system functional failures sample as defined by IP 71151-05.

b. Findings

The inspectors identified one violation of 10 CFR 50.73(a)(2)(v) with three examples. This section of the rule is the NEI 99-02 definition of a safety system functional failure. The enforcement aspects of this violation are discussed in Section 4OA3 of this report.

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 Identification and Resolution of Problems

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. The inspectors reviewed attributes that included: the complete and accurate identification of the problem; the timely correction, commensurate with the safety significance; the evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent of condition reviews, and previous occurrences reviews; and the classification, prioritization, focus, and timeliness of corrective actions. Minor issues entered into the licensee's corrective action program because 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 followup, the inspectors performed a daily screening of items entered into the licensee's corrective action program. The inspectors accomplished this through review of the station's daily corrective action documents.

The inspectors performed these daily reviews as part of their 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 Followup Inspection

a. Inspection Scope

During a review of items entered in the licensee's corrective action program, the inspectors recognized a corrective action item documenting an experimental test to resolve the condition of the reactor coolant pump thermal barriers identified in cited violation: NOV 05000482/2009002-07.

These activities constitute completion of one in depth problem identification and resolution sample as defined in IP 71152-05.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153)

.1 Loss of Offsite Power and Reactor Trip on August 19, 2009

a. Inspection Scope

On August 19, 2009, inspectors responded to a reactor trip and a loss of offsite power when the 345 kV La Cygne line was struck by lightning. The inspectors verified that the emergency diesel generators started and supplied loads. The inspectors monitored control room activities and equipment until normal offsite power feeds were re-aligned to the safety busses. The inspectors walked down portions of the plant to ensure safety systems were functioning.

These activities constitute completion of one event response sample as defined in IP 71153-05.

b. Findings

No findings of significance were identified. This event was reviewed in detail by an NRC special inspection team. The results of the special inspection will be documented in NRC Inspection Report 2009-007.

.2 Failure to Report Conditions that Could Have Prevented Fulfillment of a Safety Function

a. Inspection Scope

The inspectors implemented IP 71151 consistent with Section 4OA1 of this report. The inspectors also utilized IP 71153 to review licensee event reports. The findings are documented below in accordance with Inspection Manual Chapter 0612.

b. Findings

Introduction. The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.73, with three examples in which the licensee failed to submit licensee event reports within 60 days following discovery of events or conditions meeting the reportability criteria.

Description. First, on April 10, 2008, the licensee submitted LER 2008-002 under 10 CFR 50.73(a)(2)(i)(B) which is operation prohibited by technical specifications. For 11 hours from February 13-14, 2008, Wolf Creek did not have an operable emergency core cooling system because no high head charging pumps were operable. Wolf Creek was in Technical Specification 3.0.3 during this time. Wolf Creek received enforcement discretion to remain at power. Charging Pump B was required to be declared inoperable because emergency diesel generator B was inoperable, and charging Pump A was inoperable because it did not have an operable room cooler. On June 25, 2009, the inspectors identified that Wolf Creek failed to report this event as a safety system functional failure under 10 CFR 50.73(a)(2)(v) for the emergency core cooling system being inoperable. The inspectors discussed this with Wolf Creek and Condition Report 00018156 was initiated. On July 30, 2009, the licensee completed the evaluation of this condition report and concluded that the loss of high head charging was not reportable, however no evaluation demonstrated operability of the charging pumps.

The inspectors reviewed this issue under the safety system functional failures performance indicator. NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 5, defines a safety-system functional failure as those events meeting 10 CFR 50.73(a)(2)(v) and requires evaluation of conditions reported under other paragraphs of 50.73 for safety-system functional failures. Wolf Creek did not perform a review. Wolf Creek subsequently drafted a position paper which relied on the statements made in the Letter WO 08-0006, “Request for Notice of Enforcement Discretion from Technical Specification 3.8.1, ‘AC Sources – Operating,’” which contained an attachment that provided information documenting Wolf Creek’s verbal request for the Enforcement Discretion. The attachment contained the risk mitigation manual actions for not shutting down the unit, a discussion of the calculated incremental core damage probability used to justify enforcement discretion, and a qualitative statement regarding the adjacent pumps’ room coolers. Wolf Creek also stated that it considered the centrifugal charging pump to be functional. The manual actions did not involve the failed room cooler. Wolf Creek also cited LER 2008-002-00 which contained the same discussion of the risk assessment, the functionality of the charging Pump A, and the adjacent pumps’ room coolers. The inspectors did not find an evaluation demonstrating the operability of charging Pump A or B and hence the emergency core cooling system.

The inspectors consulted NUREG 1022, “Event Reporting Guidelines 10 CFR 50.72 and 50.73,” Revision 2. NUREG 1022 Section 3.2.7, reportability under 50.73(a)(2)(v), states that operability under Generic Letter 91-18 is the correct standard to apply. Generic Letter 91-18 has been superseded by Regulatory Issue Summary 2005-20 which does not permit the use of risk assessment to justify operability. The inspectors found that Wolf Creek was incorrect in concluding that the application of functional under the risk assessment was equivalent to the words of “safety function” under 50.73(a)(2)(v). Another position paper drafted by Wolf Creek stated that centrifugal charging Pump B was operable although it was not supported by an operable emergency diesel generator. The inspectors disagreed with this application of the definition of the technical specification of operability and this application of Technical Specifications 3.8.1, 3.0.2, and 3.0.6 which require equipment to be supported by emergency power to perform the safety function. The inspectors consulted with NRR, who agreed with the inspectors’ use of the rule and NUREG 1022. The issue was again placed into the corrective action program as Condition Report 19914.

In the second example, Wolf Creek filed LER 2008-004-00 on June 6, 2008. LER 2008 004-00 was filed under 50.73(a)(2)(iv)(A) for an event that caused automatic start of an emergency diesel during a loss of offsite power on April 16, 2008. No report was made under 50.73(a)(2)(v) for an event or condition that could have prevented a safety function due to the loss of offsite power. Inspectors reviewed NUREG 1022, Section 3.2.7 and found that:

> "Both offsite electrical power (transmission lines) and onsite emergency power (usually diesel generators) are considered to be separate functions by GDC 17. If either offsite power or onsite emergency power is unavailable to the plant, it is

> reportable regardless of whether the other system is available. GDC 17 defines the safety function of each system as providing sufficient capacity and capability, etc., assuming that the other system is not available. Loss of offsite power should be determined at the essential switchgear busses."

This missed licensee event report is specifically captured in Condition Report 19371. Wolf Creek indicated that it plans to update LER 2008-004-00 or make a second licensee event report.

Third, on April 10, 2008, Wolf Creek filed Event Notification Report 44131 per 10 CFR 50.72(b)(3)(ii)(B) based on a possible trip of all four containment coolers. The containment coolers have thermal overload protection such that if a cooler trips in fast speed during normal power operation, that cooler will not restart in slow speed for an accident. Wolf Creek evaluated this concern and issued Event Notification 44131. Wolf Creek later retracted the Event Notification stating: "Further analysis of the main steam line break, if this concern had existed, showed that the calculated post-accident pressure and temperature peak values would not exceed the peak accident values in the USAR. Therefore, an unanalyzed condition did not exist and Wolf Creek is retracting the 50.72(b)(3)(ii)(B) notification."

The inspectors found that Wolf Creek did not analyze the current draw for the motors prior to receipt of a safety injection signal. Wolf Creek assumed that the coolers would not restart and relied on containment, but this is still the loss of a safety function to remove heat from containment. Wolf Creek found that without the coolers, containment pressure exceeds the Analysis of Record but not the design pressure in the USAR. Inspectors found that this was not an appropriate method to consider the coolers' heat removal safety function met. At the end of the report period, Wolf Creek did not have an analysis for the containment cooler motors to determine if they would have tripped prior to receiving an accident signal. Wolf Creek's condition report and reportability evaluation has been open since April 11, 2008. No licensee event report has been submitted. The inspectors found insufficient evidence to show that the containment coolers could accomplish their safety function and that this should have been reported under 10 CFR 50.73(a)(2)(v). This issue is captured in Condition Report 15318.

Analysis. The failure to submit a licensee event report was a performance deficiency. The inspectors reviewed this issue in accordance with Inspection Manual Chapter 0612 and the NRC Enforcement Manual. Through this review, the inspectors determined that traditional enforcement was applicable to this issue because the NRC's regulatory ability was affected. Specifically, the NRC relies on the licensee to identify and report conditions or events meeting the criteria specified in regulations in order to perform its regulatory function, and when this is not done, the regulatory function is impacted. The inspectors determined that this finding was not suitable for evaluation using the significance determination process, and as such, was evaluated in accordance with the NRC Enforcement Policy. The finding was reviewed by NRC management, and because the violation was determined to be of very low safety significance, was not repetitive or willful, and was entered into the corrective action program, this violation is being treated as a Severity Level IV noncited violation consistent with the NRC Enforcement Policy. This finding was determined to have a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program in that the licensee failed to appropriately and thoroughly evaluate for reportability aspects all factors and time frames associated with the inoperability of the emergency core cooling system, the offsite power system, and the containment heat removal system [P.1(c)] (4OA3)

Enforcement. Title 10 CFR 50.73(a)(1) requires, in part, that licensees shall submit a licensee event report for any event of the type described in this paragraph within 60 days after the discovery of the event. Title 10 CFR 50.73(a)(2)(v) requires, in part, that events or conditions that could have prevented the fulfillment of the safety function of structures or systems that are needed to shutdown the reactor and maintain it in a safe shutdown condition, remove residual heat, control the release of radioactive material, or mitigate the consequences of an accident. Contrary to the above, in 2008, Wolf Creek failed to submit a licensee event report within 60 days for three separate events that could have prevented the fulfillment of the safety function of structures or systems that are needed to shutdown the reactor and maintain it in a safe shutdown condition, remove residual heat, control the release of radioactive material, or mitigate the consequences of an accident. Specifically, emergency core cooling, offsite power, and containment cooling could have been or were actually lost on February 13-14, 2008, April 16, 2008, and April 10, 2008, respectively, and Wolf Creek did not submit an LER within 60 days. Wolf Creek did not have sufficient analyses to demonstrate that these three events were not reportable. In accordance with the NRC's Enforcement Policy, the finding was reviewed by NRC management and because the violation was of very low safety significance, was not repetitive or willful, and was entered into the corrective action program, this violation is being treated as a Severity Level IV noncited violation, consistent with the NRC Enforcement Policy: NCV 05000482/2009004-07, “Failure to Report Conditions that Could Have Prevented Fulfillment of a Safety Function.”

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors performed observations of security force personnel and activities to ensure that the activities were consistent with Wolf Creek 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 inspector 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.

.2 INPO Training Program Accreditation

a. Inspection Scope

The NRC reviewed the concerns raised by the Accreditation Board. The senior resident inspector read WANO (INPO) accreditation report and discussed the issues with the licensee and NRR and determined that there were not any safety significant training deficiencies. The NRC determined that compliance with the regulations is not affected and that the probationary status is not safety significant. No further NRC action is required under Inspection Procedure 41500.

b. Findings

No findings of significance were identified.

.3 (Closed) Unresolved Item 05000482/2008010-03: Changes to the Approved Fire Protection Program May Not Meet NRC Acceptance Criteria

Introduction. The inspectors identified a Severity Level IV noncited violation of License Condition 2.C.(5), "Fire Protection," for making changes to the approved fire protection program without the required prior Commission approval. Specifically, the licensee made a change to the USAR that allowed the licensee to violate the requirements of 10 CFR Part 50, Appendix R, Section III.L.

Description. During the 2005 triennial fire protection inspection, the team identified an apparent violation concerning the failure to ensure that the reactor coolant system would not lose subcooling during an alternative shutdown scenario if a fire caused both pressurizer power operated relief valves to spuriously open. This issue was documented as Apparent Violation 05000482/2005008-02, "Failure to Maintain Reactor Coolant System Subcooling During the Alternative Shutdown."

After the 2005 inspection, the licensee made significant changes to the alternative shutdown methodology implemented by Procedure OFN RP-017, "Control Room Evacuation." The licensee also developed Report E-1F9915, "Design Basis Document for OFN RP-017, Control Room Evacuation," Revision 0, and Evaluation SA-08-006, "RETRAN-3D Post-Fire Safe Shutdown (PFSSD) Consequence Evaluation for a Postulated Control Room Fire," Revision 0, to demonstrate the adequacy of the revised alternative shutdown procedure. These evaluations predicted that a fire in the control room which led to control room abandonment and caused a single pressurizer power operated relief valve to spuriously open could cause a steam bubble to void approximately 40 percent of the reactor vessel head.

In response to these evaluations, the licensee modified the fire protection program to allow voiding in the core. Specifically, the licensee modified Table 9.5E-1 of the USAR to include the following paragraph:

> Analysis demonstrates that the performance goals of III.L.2 are satisfied. The performance criteria of III.L.1 are also satisfied, with the exception of maintaining reactor process variables within those predicted for a loss of

> normal ac power. This is acceptable, as long as a control room fire will not result in the plant reaching an unrecoverable condition, which could lead to core damage.

During the 2008 triennial fire protection inspection, the team identified an unresolved item related to this change to the fire protection program. The team was concerned that the licensee changed the fire protection program in a manner that could adversely affect the ability to achieve and maintain safe shutdown in the event of a fire without prior NRC approval. This concern was documented as Unresolved Item 05000482/2008010-03, "Changes to the Approved Fire Protection Program May Not Meet NRC Acceptance Criteria."

The licensee stated that their original approved fire protection program was based on the plant "not reaching an unrecoverable condition" during an alternative shutdown, citing Letter SLNRC 84-109, dated August 23, 1984.

The staff reviewed the approved fire protection program, as specified by License Condition 2.C.(5), and concluded the licensee was required to meet the technical requirements of 10 CFR Part 50, Appendix R, Section III.L. As noted in License Condition 2.C.(5), the approved fire protection program is described by the USAR through Revision 17, the Wolf Creek site addendum through Revision 15, and the SER through Supplement 5. In the Wolf Creek SER (NUREG-0881), Supplement 3, the staff concluded that the alternative shutdown capability for the control room met the requirement of 10 CFR Part 50, Appendix R, Section III.L, and was, therefore, acceptable.

The staff also concluded that the standard "not reaching an unrecoverable condition" was not part of the approved fire protection program, nor was the phrase "no unrecoverable condition" used in the context of alternative shutdown in any of the three documents specified in License Condition 2.C.(5). Further, the staff noted that the licensee did not identify this as a deviation from the requirements of 10 CFR Part 50, Appendix R, Section III.L.1, nor did the staff acknowledge any such deviation in their approval of the alternative shutdown approach in the safety evaluation reports.

Section III.L of 10 CFR Part 50, Appendix R specifies:

> During the postfire shutdown, the reactor coolant system process variables shall be maintained within those predicted for a loss of normal ac power, and the fission product boundary integrity shall not be affected; i.e., there shall be no fuel clad damage, rupture of any primary coolant boundary, of rupture of the containment boundary.

The team noted the plant response to a loss of normal ac power was described in the USAR, Chapter 15, Section 15.2.6. The USAR indicated that the plant would maintain reactor coolant system subcooling and no void formation would occur in the reactor vessel head during a loss of normal ac power. Therefore, a change to the fire protection program that allowed voiding in the reactor vessel head during an alternative shutdown would involve a failure to meet the requirements of 10 CFR Part 50, Appendix R, Section III.L.

The staff reviewed the licensee's program change and concluded that this change exceeded the licensee's ability to make changes without prior staff approval, as provided in License Condition 2.C.(5). Specifically, the staff considers a change that allows the licensee to violate a requirement to be a change that adversely affects the ability to achieve and maintain safe shutdown in the event of a fire.

<u>Analysis</u>. Changing the approved fire protection program such that the reactor coolant subcooling process variables would remain within those predicted for a loss of normal ac power without prior Commission approval was a performance deficiency. The team assessed this performance deficiency using traditional enforcement since it had the potential for impacting the NRC's ability to perform its regulatory function. The team determined this performance deficiency was more than minor since the change required prior staff review and approval prior to implementation and it did not receive the required approval.

A senior reactor analyst performed a Phase 3 evaluation to determine the risk significance of this finding since the performance deficiency involved a control room fire that led to control room abandonment. The analyst performed a bounding evaluation to determine an upper limit for the change in core damage frequency.

The analyst assigned a generic fire ignition frequency for the control room (FIF_{CR}), which was slightly higher than the value in Calculation AN-95-029, "Control Room Fire Analysis," Revision 1. The analyst multiplied the fire ignition frequency by a severity factor (SF) and a nonsuppression probability indicating that operators failed to extinguish the fire within 20 minutes assuming a 2 minute detection that required a control room evacuation (NP_{CRE}). The resulting control room evacuation frequency (F_{EVAC}) was:

$$
\begin{aligned}
F_{EVAC} &= FIF_{CR} * SF * NP_{CRE} \\
&= 1.09E\text{-}2/\text{year} * 0.1 * 1.30E\text{-}2 \\
&= 1.42E\text{-}5/\text{year}
\end{aligned}
$$

The control room has a total of 103 cabinets. The analyst determined that a single fire in five of these cabinets could lead to the spurious opening of a pressurizer power-operated relief valve. Therefore, a bounding change in core damage frequency for a control room fire that leads to evacuation and the spurious opening of a pressurizer power-operated relief valve ($F_{EVAC+PORV}$) was determined to be:

$$
\begin{aligned}
F_{EVAC+PORV} &= F_{EVAC} * 5 / 103 \\
&= 1.42E\text{-}5/\text{year} * 5 / 103 \\
&= 6.88E\text{-}7/\text{year}
\end{aligned}
$$

This frequency was considered to be bounding since it assumed:

1) A fire in the applicable cabinets would create a short that caused the pressurizer power-operated relief valve to spuriously open,

2) The conditional core damage probability given a control room fire with evacuation and the spurious opening of a power-operated relief valve was set equal to one, and

3) The performance deficiency accounted for the entire change in core damage frequency (i.e., the baseline core damage frequency for this event was zero).

Since this bounding frequency was less than 1E-6/year, the analyst determined this performance deficiency to have very low risk significance.

This performance deficiency was analogous to Example D.5 in the Enforcement Policy, Supplement 1. Since, the performance deficiency was evaluated as having very low safety significance, the team determined that a Severity Level IV violation was appropriate.

This finding had a crosscutting aspect in the area of human performance associated with resources because the licensee failed to maintain long term plant safety by maintaining design margins. Specifically, the licensee's choice to allow reactor vessel head voiding during an alternative shutdown in lieu of restoring the plant to compliance with the requirements of 10 CFR Part 50, Appendix R, Section III.L constituted a reduction in safety margin (H.2(a)).

Enforcement. License Condition 2.C.(5), "Fire Protection," states, in part:

a) The operating corporation shall maintain in effect all provisions of the approved fire protection program as described in the SNUPPS Final Safety Analysis Report for the facility through Revision 17, the Wolf Creek site addendum through Revision 15, and as approved in the SER through Supplement 5, subject to provisions b & c below.

b) The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

The SER, Section 9.5.1.7 states, in part:

> The staff will condition the operating license to require the applicant to meet the technical requirements of Appendix R to 10 CFR Part 50, or provide equivalent protection.

The SER, Supplement 3, Section 9.5.1.5 states:

> Based on our review, the staff concludes that the alternative shutdown capability for the control room meets the requirements of Appendix R, Section III.L, and is therefore acceptable.

Section III.L of 10 CFR Part 50, Appendix R, specifies:

> During the postfire shutdown, the reactor coolant system process variables shall be maintained within those predicted for a loss of normal ac power, and the fission product boundary integrity shall not be affected; i.e., there shall be no fuel clad damage, rupture of any primary coolant boundary, or rupture of the containment boundary.

The plant response to a loss of normal ac power was described in the USAR, Chapter 15, Section 15.2.6. The USAR indicated that the plant would maintain reactor coolant system subcooling and no void formation would occur in the reactor vessel head during a loss of normal ac power.

Contrary to the above, on September 25, 2008, the licensee made a change to the approved fire protection program that adversely affected the ability to achieve and maintain safe shutdown in the event of a fire without prior approval of the Commission. Specifically, the licensee made a change to Table 9.5E-1 of the USAR that allowed reactor coolant system process variables to exceed those predicted for a loss of normal ac power during an alternative shutdown. This change adversely affected the ability to achieve and maintain safe shutdown in the event of a fire since it allowed the licensee to violate a requirement without an approved deviation.

The licensee entered this issue into their corrective action program as Performance Improvement Request 2008-004869. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program, this violation is being treated as a noncited violation, consistent with the NRC Enforcement Policy: NCV 05000482/2009004-08, "Changes to the Approved Fire Protection Program Without Prior Staff Approval."

.4 <u>(Closed) Apparent Violation 05000482/2005008-02</u>: Failure to Maintain Reactor Coolant System Subcooling During the Alternative Shutdown

The issue documented by this apparent violation is enveloped by Unresolved Item 05000482/2008010-03, "Changes to the Approved Fire Protection Program May Not Meet NRC Acceptance Criteria" and discussed in Section 4OA5.1. This apparent violation is closed.

4OA6 Meetings

<u>Exit Meeting Summary</u>

On July 30, 2009, the inspectors presented the inspection results to Mr. S. A. Henry, Manager, Plant Operations, and other members of the licensee staff. The inspectors stated that they had reviewed proprietary information during the inspection, and verified that all material had been returned to the licensee or destroyed. The licensee acknowledged the inspection results as presented.

The inspector briefed Robert Evenson of the results of the annual licensed operator requalification program inspection on August 5, 2009. The licensee representative acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On September 28, 2009, the inspectors conducted a telephonic exit meeting and presented the results of the staff review of fire protection program changes to Mr. J. Suter, Fire Protection Supervisor, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

On October 14, 2009, the resident inspectors presented the inspection results of the resident inspections to Mr. Matt Sunseri, Vice President 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, none would be included in this report and that the materials were returned to the licensee.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

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

LIST OF ITEMS OPENED AND CLOSED

Opened and Closed

05000482/2009004-01	NCV	Failure to Log Foreign Material in Spent Fuel Pool After Extent of Condition Evaluation (Section 1R05)
05000482/2009004-02	NCV	Inability to Perform Manual Actions for Risk Assessment (Section 1R13)
05000482/2009004-03	NCV	Inadequate Evaluation of Emergency Diesel Generator for Common Cause Failure in the Supporting Essential Service Water System (Section 1R15.1)
05000482/2009004-04	NCV	Failure to Implement Engineered Safety Features Actuation System Technical Specifications Results in Missed Mode Change (Section 1R15.2)
05000482/2009004-05	NCV	Use of Nonsafety-Related Power to Ensure Operability of Safety-Related Boric Acid System (Section 1R17)
05000482/2009004-06	NCV	Performing Prohibited Elective Maintenance on Safety Bus NB02 Channel 4 During Emergency Diesel Generator Maintenance (Section 1R19)
05000482/2009004-07	NCV	Failure to Report Conditions that Could have Presented Fulfillment of a Safety Function (Section4OA3)
05000482/2009004-08	NCV	Changes to the Approved Fire Protection Program Without Prior Staff Approval (Section 4OA5.3)

Closed

05000482/2005008-02	AV	Failure to Maintain Reactor Coolant System Subcooling During the Alternative Shutdown (Section 4OA5.4)
05000482/2008010-03	URI	Changes to the Approved Fire Protection Program May Not Meet NRC Acceptance Criteria (Section 4OA5.3)

LIST OF DOCUMENTS REVIEWED

Section 1RO1: Adverse Weather Protection

DOCUMENTS

NUMBER	TITLE	REVISION
STS NB-005	Breaker Alignment Verification	Revision 18

Section 1RO4: Equipment Alignment

DOCUMENTS

NUMBER	TITLE	REVISION
CKL AL-120	Auxiliary Feedwater Normal Lineup	34
M-12AL01	Piping and Instrumentation Diagram – Auxiliary Feedwater System	10
M-12EF01	Piping and Instrumentation Diagram – Essential Service Water System	21
M-12EF02	Piping and Instrumentation Diagram – Essential Service Water System	25
M-12AB01	Piping and Instrumentation Diagram – Main Steam System	11
M-12AB02	Piping and Instrumentation Diagram – Main Steam System	12

Section 1RO4: Equipment Alignment

DOCUMENTS

NUMBER	TITLE	REVISION
	USAR 15.6-12/13, Steam Generator Tube Rupture with Postulated Stuck-Open Atmospheric Relief Valve	22
	Control Room Logs dated September 16, 2009 at 1:49 a.m.	
M-224A-00037	10” -900 Carbon Steel Flex Wedge Gate Valve with 6:1 B.G. Actuator	G
	USAR Figure 9.3-8-03, Piping and Instrumentation Diagram Chemical & Volume Control System	41

Condition Reports

00019813 00019821 00019825

Work Order

09-3160637-000

Work Request

09-072489

Section 1RO5: Fire Protection

DOCUMENTS

NUMBER	TITLE	REVISION
AP 10-106	Fire Preplans	8
FPPM-009	Control Bldg El.2000’	2
FPPM-014	Diesel Generator Rooms El.2000’	1

Section 1R11: Licensed Operator Requalification Program

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
LR5004004	Shutdown LOCA	009

Section 1R13: Maintenance Risk Assessment and Emergent Work Controls

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
	Week of 3/16/09 – Operational Risk Assessment	

Condition Reports

00016735	00015318	2009-001338

Section 1R15: Operability Evaluations

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
STS SF002	Core Axial Flux difference	9
STS RE-009	Heat Flux Hot Channel Factor Measurement	14
SYS SR-200	Moveable Incore Detector Operation	21
STS RE-012	QPTR Determination	10
STS RE-013C	BEACON SinglePoint AFD Calibration	10
WO 09-318203-002	Engineering Disposition: EF138HBC-30 has a thru wall leak	June 30, 2009
WO 09-318203-009	Engineering Disposition: Minimum Wall Issues with Line EF138HBC-30	July 16, 2009

Section 1R15: Operability Evaluations

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
SWO 09-318982-001	Engineering Disposition: EF150HBC-18 Essential Service Water Pipe Pit Through Wall Leak	July 28, 2009
WO 09-319429-001	Engineering Disposition: EF049HBC-8 Thru Wall Leak Evaluation	August 20, 2009
SYS KJ-200	Inoperable Emergency Diesel	15 / June 30, 2009
NRC GL 93-05	Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation	September 27, 1993
NPF-42	Amendment #101	August 9, 1996
NPF-42	Amendment #123	December 31, 1999
STS IC-232	Channel Operational Test Nuclear Instrumentation System Source Range N-32 Protection Set II	15
	Class IE Environmental Qualification Data Sheet for NI 31 and 32 Source Range Monitors	September 1990

Condition Reports

00018217	00018347	00018611	00018945	00019276
00019282	00019307			

Work Orders

09-318203-001 (Ultrasonic Thickness Report)
09-318268-000 (Ultrasonic Thickness Report)
09-318269-000 (Ultrasonic Thickness Report)
09-318270-000 (Ultrasonic Thickness Report)
09-318271-000 (Ultrasonic Thickness Report)
09-318272-000 (Ultrasonic Thickness Report)
09-318982-000 (Ultrasonic Thickness Report)
09-318982-003 (Ultrasonic Thickness Report)
09-318982-013 (2 Ultrasonic Thickness Reports)

09-318982-014 (3 Ultrasonic Thickness Reports)
09-319473-000 (Ultrasonic Thickness Report)
09-319473-001 (Ultrasonic Thickness Report)
09-319473-002 (Ultrasonic Thickness Report)
09-319473-003 (Ultrasonic Thickness Report)
09-319473-004 (Ultrasonic Thickness Report)
09-319473-006 (Ultrasonic Thickness Report)
09-319473-007 (Ultrasonic Thickness Report)
09-319473-008 (Ultrasonic Thickness Report)
09-319473-009 (Ultrasonic Thickness Report)
09-319473-010 (Ultrasonic Thickness Report)

Section 1R17: Permanent Plant Modifications (71111.17A)

Calculations

Number	Title	Revision
M-628-00131-W01	Control Logic Diagram MSIV PPS-700	0
M-630-0095-W01	Control Logic Diagram MFIV PPS-300	0
XX-E-013	Post-Fire Safe Shutdown Analysis	1
XX-E-016	XNB02 Tap Change Analysis	0
XX-E-006	AC System Analysis	5
AN-06-007	Wolf Creek Generating Station Rod Withdrawal at Power (RWAP) Event Safety Analysis	0
AN-04-015	Radiological Consequences of a Fuel Handling Accident	1
0720517.01-C-001	Wolf Creek Generating Station (WCGS) Simplified Head Assembly (SHA) Drop Analysis	0
EJ-S-008	Installation of Vent Lines on Check Valves EJ8958A, EJ8958B and EJ8958C	0
XX-S-036	Westinghouse Class I Nuclear Valves 6” and Larger Swing Check Valves – EM5093	0

Condition Reports

2006-000363	2006-000577	2006-001070	2006-001447
2006-001858	2006-001923	2006-002412	2006-003067
2006-003135	2006-003235	2006-003241	2007-000070
2007-000235	2007-000416	2007-001115	2007-002153
2007-002251	2007-002329	2007-002401	2007-002459
2007-002727	2007-003578	2007-003767	2007-003782

2007-004696	2008-000028	2008-000083	2008-000662
2008-000826	2008-001445	2008-001727	2008-002157
2008-004744	2008-005500	2008-005550	2008-005808
2009-000409	00014799	00016231	2007-001180
2006-000309	2006-000442	2006-001447	2007-001457
2006-001549	2006-003684		

Drawings

Number	Title	Revision
WIP-E-15000-065-R-1	Electrical Cable, Termination, and Raceway List	5
E-13AB32	Miscellaneous Circuits	7
E-11025	Relay Settings Tabulation and Coordination Curves System NE	13
0405-0003-01	Intercooler Heat Exchanger Analysis Input Data	2

Miscellaneous

Number	Title	Date/Revision
	USA 50.59 Resource Manual	3
NEI 96-07	Guidelines for 10 CFR 50.59 Implementation	1
J-200B-00001	Nutherm Qualification Report Eaton Cutler-Hammer Contact Blocks With Separation Barriers	0
PSA PCR 2006-0002	Maintenance of the Wolf Creek PSA Model	0
PSA 05-0002	WCGS PRA Initiating Event Notebook – 2002 Update	0
M-018B-00001	Instruction Manual for Governor Modification	W03
N/A	Design Change Process Improvements Engineering Initiative Plan	0
N/A	Design Change Process Improvement Initiative: Monthly Progress Report	April 10, 2009

Procedures

Number	Title	Revision
GEN 00-004	Power Operation	54
OFN RP-017	Control Room Evacuation	29
SYS EP-200	Safety Injection Accumulator Operations	30
AP 05-001	Change Package Planning and Implementation	7
AP 05-002	Dispositions and Change Packages	8
AP 05-005	Design, Implementation & Configuration Control of Modifications	13
AP 05F-001	Design Verification	3
AP 26A-003	10 CFR 50.59 Reviews	10

Section 1R18: Plant Modifications

NUMBER	TITLE	DATE
09-005-XX-01	Temporary Modification Order	February 19, 2009
09-008-SG00	Temporary Modification Order	March 5, 2009
09-0019	Essential Required Reading: Responding to an Earthquake with Inoperable Seismic Instrumentation	March 12, 2009
	Change Package No. 011613	

Condition Reports

2009-001278 2009-001194

Work Requests

09-072504 09-072505 09-072506 09-072507 09-072508

Section 1R19: Postmaintenance Testing

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
STS KJ-015B	Manual/Auto Fast Start, Sync & Loading of EDG NE02	27A / August 17, 2009
STS AL-103	Turbine Driven Auxiliary Feedwater Pump Inservice Pump Test	44 / September 9, 2009
NP-1490	4-900' ANSI Trip Throttle Valve	A
103171D	Trip Throttle Valve Electrical Schematic Sheet 1	June 5, 1977
103171D	Trip Throttle Valve Electrical Schematic Sheet 2	November 17, 1980

Work Orders

09-316773-000 09-316773-001

Section 1R20: Refueling and Other Outage Activities

DOCUMENTS

NUMBER	TITLE	REVISION
	Feedwater Isolation Logic Drawing	
Table 7.3.15	USAR – NSSS Interlocks for Engineered Safety Feature Actuation System	13
SYS SB-122	Enabling/Disabling P-4/LO Tavg Fwis	1
Table 7.5-1	Engineered Safety Features - Displays	21
12.2-7	Westinghouse Technology Systems Manual Reactor Protection system – Reactor Trip Signals	0100

SYS SB-122	Enable/Disabling P-4/LO Tavg FWIS	1
7.2-31	USAR – Testing of Reactor Trip Breakers	11

Work Orders

09-314863-002 09-319404-000

Performance Improvement Request

2001-0041

Condition Reports

00019318 00019318

Section 1R22: Surveillance Testing

PROCEDURES

NUMBER	TITLE	REVISION / DATE
STS CR-004	Shift Log for Additional Monitoring	0
STS EF-100B	ESW System Inservice Pump B & ESW B Discharge Check Valve Test	32 / August 13, 2009
STS IC-208B	4 kV Loss of Voltage and Degraded Voltage TADOT NB02 Bus – Separation Group 4	2A /July 14, 2009
STS RE-006	End of Life Core Moderator Temperature Coefficient Measurement	18 / August 28, 2009

Work Order

09-315436-000

Condition Reports

00019069 00019000

Section 4OA1: Performance Indicator Verification

PROCEDURES

NUMBER	TITLE	REVISION
STS IC-203	Channel Operational Test 7300 Process Instrumentation Protection Set III (Blue)	22B
INC C-001	7300 Signal Comparator Card (NAL 1)	6

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION
NEI 99-02	Mitigating Systems Cornerstone	5
LER 2005-004-00	Failure of Auxiliary Building Ventilation Dampers to Close on Safety Injection Signal	
OPR01	Operability/Reportability Detail Report	
LER 2008-007-00	Two Residual Heat Removal Trains Inoperable in Mode 3 due to Check Valve Leakage	
LER 2008-004-00	Loss of Power Event When the Reactor was De-fueled	
LER 2008-008-01/02	Potential for Residual Heat Removal Trains to Be Inoperable During Mode Change	
LER 2008-009-00	Inadequate Compensatory Actions for a Fire Area	
LER 2008-001-00	Containment Cooler Inoperability (Callaway Plant Unit 1)	
AIF 16C-001-02	Maintenance Walkdown Form (Technician)	0
10466-M-761-2076-W05	Interconnecting Wiring Diagram Cabinet 03 SNUPPS Nuclear Power Plant Controls	
2000801894	Adverse Condition- Ameren	

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION
NCV 05000483/2008003-01	Failure to Ensure the Suitability of the Design of the Containment Air Cooler control Circuitry	
	Appendix D, 10 CFR 50.72 Including Statement of Considerations	
	Event Notification Report of June 23, 2008	

Condition Reports

2009-00017786	2009-00017846	2009-00017851	2009-00019914	2009-00019371
2009-001326	2009-00017776	2009-0001261	2009-001326	2009-001004
2008-001307	2009-00018156	200-00018156	2008-000470	2008-001673

Work Orders

09-314726-000	09-317948-000	09-306203-000

Corrective Action Plan

4160	1970	3944	3943

Reportability Evaluation Request

2008-011	2009-012

Section 4OA2: Identification and Resolution of Problems

PROCEDURES

NUMBER	TITLE	REVISION / DATE
TMP 09-014	CCW Flow Balance for Troubleshooting Thermal Barrier Closure	0 / July 15, 2009
SYS EG-201	Transferring Supply of CCW Service Loop and CCW Train Shutdown	36 / July 15, 2009

	Applicability Determination for TMP 09-014	July 14, 2009
	50.59 Screen for TMP 09-014	July 14, 2009
USAR	USAR Section 5.4.1.2.2	0

Work Order

09-316483-000

Corrective Action

00018793

Section 4OA5: Other Activities

PROCEDURES

NUMBER	TITLE	DATE
SLNRC 84-109	Letter to NRC	08/23/1984
USAR CR 2008-009	Updated Final Safety Analysis Report Change Request	09/25/2008
	Evaluation of Proposed Change for USAR CR 2008-009	09/25/2008

Performance Improvement Request

2008-004869