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

August 12, 2011

EA-11-149

Matthew Sunseri, 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 AND NOTICE OF VIOLATION 05000482/2011003

Dear Mr. Sunseri:

On June 30, 2011, 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 July 13, 2011, with Mr. Stephen Hedges, Site Vice President, 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.

Based on the results of this inspection, one violation is cited in the enclosed Notice of Violation (Notice) and the circumstances surrounding this violation are described in detail in the enclosed report. The violation involved the failure to implement procedures for opening of main steam isolation valves without causing safety system actuations (EA-11-149). Although determined to be of very low safety significance (Green), this violation is being cited in the Notice because Wolf Creek failed to restore compliance within a reasonable time after the violation was identified in NRC Inspection Report 05000482/2010004, per Section 2.3.2 of the NRC Enforcement Policy. The current Enforcement Policy is included on the NRC's Web site at http://www.nrc.gov/about-nrc/regulatory/enforcement/enforce-pol.html.

Please note you are required to respond to this letter and should follow the instructions specified in the enclosed Notice. If you have additional information that you believe the NRC should consider, you may provide it in your response to the Notice. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

This report also documents nine additional NRC-identified and self-revealing issues that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC determined that violations are associated with eight of these issues. Additionally, two licensee-identified violations, which were determined to be of very low safety significance, are listed in this report. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these findings as noncited violations, consistent with Section 2.3.2 of the NRC Enforcement Policy.

If you contest the violation 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 facility. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV, and the NRC Resident Inspector at the facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if you choose to provide one for cases where a response is not required, will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy or proprietary, information so that it can be made available to the Public without redaction.

Sincerely,

/RA/

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

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

Enclosure:
NRC Inspection Report and Notice of Violation 05000482/2011003
w/Attachment: Supplemental Information

cc w/Enclosure:
Distribution via Listserv

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R:_REACTORS_WC\2011\WC2011003-RP-CML

ADAMS: ☐ No ☑ Yes	☑ SUNSI Review Complete	Reviewer Initials: RWD
	☑ Publicly Available	☑ Non-Sensitive
	☐ Non-publicly Available	☐ Sensitive

SRI:DRP/B	RI:DRP/B	C:DRS/EB1	C:DRS/EB2	DRS/PSB1
CLong	CPeabody	TFarnholtz	NO'Keefe	MHay
/E-GBM/	***/E-GBM/***	***/RA/***	***/JMateychick for/***	***/JLarson for/***
8/12/2011	8/2/2011	8/9/2011	8/9/2011	8/10/2011
C:DRS/OB	C:DRS/TSB	DRS/PSB2	RIV:ACES	GMiller
MHaire	DPowers	GWerner	RKellar	
/RA/	***/RA/***	***/RA/***	***/RA/***	***/RA/***
8/10/2011	8/10/2011	8/9/2011	8/11/2011	8/12/2011

OFFICIAL RECORD COPY T=Telephone E=E-mail F=Fax

NOTICE OF VIOLATION

Wolf Creek Nuclear Operating Corporation
Wolf Creek Generating Station

Docket: 50-482
License No: NPF-42
EA-11-149

During an NRC inspection conducted March 19 through June 30, 2011 a violation of an NRC requirement was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> Technical Specification 5.4.1.a requires that procedures be established, implemented, and maintained covering the activities described in Regulatory Guide 1.33, Revision 2, Appendix A. Regulatory Guide 1.33, Revision 2, Appendix A, Section 3.i requires procedures for the startup, operation and shutdown of the main steam system. Wolf Creek Procedure SYS AB-120, "Main Steam and Steam Dump Startup and Operation," Revision 27, implements these requirements for the main steam system.
>
> Contrary to the above, from March 5, 2010, to March 19, 2011, Wolf Creek Procedure SYS AB-120 had not been maintained to cover activities for the startup, operation and shutdown of the main steam system. Specifically, Procedure SYS AB-120, Revision 27, contained inadequate steps necessary to open a main steam isolation valve without causing a safety injection signal.
>
> This violation is associated with a Green Significance Determination Process finding (EA-11-149).

Pursuant to the provisions of 10 CFR 2.201, Wolf Creek Nuclear Operating Corporation is required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator, Region IV, and a copy to the NRC Senior Resident Inspector at the facility that is the subject of this Notice of Violation (Notice), within 30 days of the date of the letter transmitting this Notice. This reply should be clearly marked as a "Reply to Notice of Violation EA-11-149," and should include for each violation (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an Order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information. If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Dated this 12th day of August 2011.

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000482
License:	NPF-42
Report:	05000482/2011003
Licensee:	Wolf Creek Nuclear Operating Corporation
Facility:	Wolf Creek Generating Station
Location:	1550 Oxen Lane NE Burlington, Kansas
Dates:	April 1 to June 30, 2011
Inspectors:	C. Long, Senior Resident Inspector C. Peabody, Resident Inspector D. Reinert, Acting Resident Inspector J. Drake, Senior Reactor Inspector A. Fairbanks, Reactor Inspector G. Guerra, CHP, Emergency Preparedness Inspector G. Pick, Senior Reactor Inspector D. Strickland, Operations Engineer
Approved By:	G. Miller, Chief, Project Branch B Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000482/2011003, 4/1 – 6/30/2011; Wolf Creek Generating Station, Integrated Resident Report, Adverse Weather Protection, Equipment Alignments, Inservice Inspection Activities, Postmaintenance Testing, Event Follow-up, and Other Activities.

The report covered a 3-month period of inspection by resident inspectors and announced baseline inspections by region-based inspectors. One Green cited violation, eight Green noncited violations, and one finding 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." The cross-cutting aspect is determined using Inspection Manual Chapter 0310, "Components Within the Cross-Cutting Areas." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a noncited violation of Technical Specification 5.4.1.a, "Administrative Procedures," for having no procedure to address onsite debris impacting plant equipment during severe weather. The inspectors walked down external areas of the plant on June 1 and June 9, 2011, prior to the onset of predicted severe thunderstorms and tornadoes. The inspectors found loose debris each time and brought it to the attention of the licensee who secured the materials. The inspectors walked down the transformer yard and tank yard during a thunderstorm on June 16 and found loose debris such as plywood, trash, wood planks, and fiberglass planks. The inspectors brought this to the attention of Wolf Creek and the materials were removed or secured. Wolf Creek initiated several condition reports but they only addressed immediate cleanup. Wolf Creek procedures had no steps for securing potential wind-driven projectiles prior to severe weather. After June 16, Wolf Creek wrote Condition Report 40573 which started a weekly maintenance activity to remove loose materials and added procedure steps to have operations walk down external areas prior to severe weather.

  This finding was more than minor because it impacted the protection against external factors attribute of the Initiating Events Cornerstone, and it affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors evaluated this finding using Inspection Manual Chapter 0609.04, and determined that it was of very low safety significance (Green) for June 16, 2011, because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment would be unavailable since the reactor was shutdown. Inspectors used Manual Chapter 0609 Appendix G, Checklist 4 for the other occurrences because Wolf Creek was in

Modes 4 or 5. The finding again screened to Green because it did not increase the likelihood of a loss of inventory, did not cause the loss of reactor coolant system instrumentation, did not degrade the ability of the licensee to terminate a leak path or add inventory when needed, or degrade the ability to recover residual heat removal if it was lost. This finding has a cross-cutting aspect in the area of problem identification and resolution, specifically the corrective action program attribute because licensee's short-term corrective actions failed to ensure debris was secured or removed prior to severe weather [P.1(d)](Section 1R01).

- Green. The inspectors documented a self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion IX, "Control of Special Processes." Specifically, in October 2009, welders failed to ensure the fillet weld between the train B charging header and the half coupling used to attach two vent valves met the specified weld requirements. This weld failed in January 2011, rendering the train B charging system inoperable. The licensee's extent of condition review identified 12 vent line welds which did not meet ASME code weld size requirements and/or procedural requirements for 2:1 weld taper configuration. Additionally, quality assurance inspectors failed to identify that the 2:1 taper weld requirements specified by procedure, and ASME minimum weld size requirements, were not met in multiple vent line welds. The weld was repaired and built up to the correct 2:1 aspect ratio. This issue was entered into the licensee's corrective action program as Condition Reports 32648, 33686, 33689, and 36438.

  The finding was more than minor because it was associated with the equipment performance attribute of the Initiating Events Cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. The inspectors performed a Phase 1 screening in accordance with Inspection Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," and determined that the finding was of very low safety significance (Green) because the issue did not result in exceeding the technical specification limit for identified reactor coolant system leakage or affect other mitigating systems resulting in a total loss of their safety function. This finding had a cross-cutting aspect in the area of human performance, resources, because the licensee failed to ensure that personnel, specifically welders and quality assurance inspectors, were adequately trained in the procedural requirements and methods for measuring weld dimensions to assure nuclear safety [H.2(b)](Section 1R08).

- Green. The inspectors identified a noncited violation of 10 CFR Part 50 involving the failure of the licensee to ensure that weld preparation was protected from deleterious contamination in that drawers (located in the hot tool room) containing files, grinding wheels, flapper wheels, and cutting wheels, used for the purpose of weld preparation, contained a mixture of both stainless steel tools and carbon steel tools. The failure to separate tools used for stainless steel weld preparation from tools used for carbon steel preparation could result in the contamination of stainless steel welds by carbon steel and affect the material integrity and corrosion resistance. The licensee immediately removed the tools and replaced them with new tools stored separately for use on specific types of metal. This issue was entered into the licensee's corrective action program as Condition Report 36444.

The finding was more than minor because it was associated with the equipment performance attribute of the Initiating Events Cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations, and if left uncorrected the finding would become a more significant safety concern. The inspectors performed a Phase 1 screening in accordance with Inspection Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," and determined that the finding was of very low safety significance (Green) because the issue did not result in exceeding the technical specification limit for identified reactor coolant system leakage or affect other mitigating systems resulting in a total loss of their safety function. This finding had a cross-cutting aspect in the area of human performance, resources, because the licensee did not provide complete, accurate, and up-to-date procedures for the preparation of stainless steel and carbon steel welds [H.2(c)](Section 1R08).

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," involving the failure of the licensee to review the suitability of installing brass fittings and leaving test fittings on pressure, differential pressure, and flow transmitter equalizing block valve drain ports instead of the design specified stainless steel manifold plugs. During a boric acid walkdown, the inspectors identified that drain ports on the equalizing block of two separate reactor coolant system flow transmitters had brass fittings installed instead of the design specified stainless steel fittings. In response to inspector concerns about the brass fittings, the licensee subsequently discovered that a design configuration nonconformance existed by leaving the test fittings on the drain port during plant operation. Licensee Drawing J-17D22 specifies that manifold plugs be installed in the drain ports during plant operation. The licensee immediately replaced the brass caps with stainless steel fittings. This issue was entered into the licensee's corrective action program as Condition Report 36439.

  The finding was more than minor because it was associated with the design control attribute of the Initiating Events Cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. The inspectors performed a Phase 1 screening in accordance with Inspection Manual 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," and determined that the finding was of very low safety significance (Green) because the issue would not result in exceeding the technical specification limit for identified reactor coolant system leakage or affect other mitigating systems resulting in a total loss of their safety function. The inspectors also determined that the finding had a cross-cutting aspect in the area of human performance, resources, because the licensee did not provide adequate training of personnel so that the inappropriately installed fittings could be identified during system walkdowns [H.2(b)](Section 1R08).

- Green. The inspectors identified a cited violation of Technical Specification 5.4.1.a, "Administrative Procedures," involving Wolf Creek's failure to correct Procedure SYS AB-120 for main steam isolation valve operation. Specifically, between March 3, 2010, and March 19, 2011, Wolf Creek experienced repeat cases of safety-system actuations due to Procedure SYS AB-120 containing inadequate steps to establish conditions necessary to open a main steam isolation valve. Corrective actions were previously limited to steam header pressures below 300 psi. Wolf Creek commenced a root cause evaluation of the March 19, 2011, safety injection under Condition Report 34964. Due to Wolf Creek's failure to restore compliance from previous NCV 05000482/2010004-01 within a reasonable time after the violation was identified, this violation is being cited as a Notice of Violation consistent with the Enforcement Policy.

  Failure to correct deficiencies in Procedure SYS AB-120 for steam pressures above 300 psi was a performance deficiency. The inspectors determined that this finding was more than minor because it impacted the equipment performance attribute for the Initiating Events Cornerstone, and it affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, this issue relates to the configuration control attribute for shut down equipment alignment. The inspectors evaluated the significance of this finding using Inspection Manual Chapter 0609.04. Assuming worst case degradation, the finding resulted in exceeding the technical specification limit for reactor coolant system leakage due to the pressurizer power-operated relief valve cycling. Therefore, the inspectors screened the finding to a Phase 2 review by the senior reactor analyst. The senior reactor analyst used the Wolf Creek SPAR model and concluded that the incremental core damage probability was 3.7E-7 (Green). The inspectors found that the cause of the finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program. Specifically, several evaluations failed to have an adequate extent of condition review and did not find that procedures were inadequate for opening a main steam isolation valve above 300 psi [P.1(c)](Section 4OA3.1).

- Green. The inspectors reviewed a self-revealing noncited violation of Technical Specification 5.4.1.a, "Administrative Procedures," for failure to follow procedural requirements to maintain reactor coolant system pressure below 350 psig. Control room operators increased charging flow at too great a rate with the reactor coolant system water-solid which caused the pressurizer power-operated relief valve to cycle three times over several minutes until adjustments to letdown could be made to reduce reactor coolant system pressure. Also, the letdown pressure controller was left in manual when automatic control would have lessened the pressure increase. Wolf Creek wrote Condition Report 35244 to correct the deficiency by changing several procedures for water-solid plant operations.

The failure to maintain pressure below the power-operated relief valve setpoint was a performance deficiency. The performance deficiency was more than minor because it impacted the Initiating Events Cornerstone objective of configuration control to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The significance of the finding was determined using Inspection Manual Chapter 0609, Significance Determination Process, Appendix G, Checklist 2, and determined to be of very low safety significance (Green), because it did not cause the loss of mitigating capability of core heat removal, inventory control, power availability, containment control, or reactivity control. Additionally, the finding also did not cause any low temperature overpressure technical specifications to be exceeded. The inspectors found that the cause of the finding had a cross-cutting aspect in the area of human performance. Specifically, operators had to rely on skill of the craft when procedures should have supplied more instruction for manipulating charging and letdown with a water-solid plant [H.2.c](Section 4OA3.2).

- <u>Green</u>. The inspectors reviewed a self-revealing noncited violation of License Condition 2.C.5 for failure to implement adequate fire watches which affected both trains of vital ac and dc switchgear. The inadequate fire watches occurred during an actual fire which negated the Halon system discharge because internal fire doors were not shut, as required, by the fire watch. The inspectors found problems with fire impairments and watches from 2008 that had not been corrected. Subsequent to the fire, Wolf Creek again briefed and trained its personnel on the requirements for fire watches. This issue is captured in the corrective action program as Condition Report 36719.

  Failure to implement adequate fire impairments such that the fire watches ensured the success of the Halon system was a performance deficiency. The performance deficiency was more than minor because it impacted the Initiating Events Cornerstone and its objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the protection against external factors attribute was impacted by the fire impairment. To determine significance, the inspectors used Inspection Manual Chapter 0609.04 to screen the finding to Inspection Manual Chapter 0609, Appendix F, because the fire protection defense-in-depth strategies involving automatic suppression, fire barriers, and administrative controls were degraded. The senior reactor analyst conducted a Phase 3 review of this finding and concluded that the incremental core damage frequency was 1.6E-8 per year, or very low safety significance (Green). The inspectors found that the cause of the finding had a cross-cutting aspect in the area of problem identification and resolution. Specifically, corrective actions from ineffective fire watches in 2008 did not prevent recurrence of the inadequate fire watch on April 5, 2011 [P.1.d](Section 4OA3.3).

Cornerstone: Mitigating Systems

- Green. The inspectors reviewed a self-revealing noncited violation of Technical Specification 5.4.1a, “Administrative Procedures,” for a loss of component cooling water train B inventory caused by inadequate clearance order verification. Valve HBV110 was stuck in position and was partially open. When the clearance order was implemented, the operators concluded the valve was already closed. Subsequently, the valve created a leakage path which exceeded the surge tank makeup flow capacity and required manual isolation by the control room operators to protect safety-related components. Wolf Creek has taken corrective actions to include communication of expected as-found equipment positions in pre-job briefings and the clearance order template. This issue is captured in the corrective action program as Condition Reports 34505 and 40219.

  Failure to properly establish clearance order boundary isolation was a performance deficiency. The performance deficiency is more than minor because it is associated with the equipment performance and human performance attributes of the Mitigating Systems Cornerstone and impacted the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609.04, the finding was determined to be of very low safety significance because the finding did not result in the loss of operability or functionality of the component cooling water train or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The inspectors found that the finding had a cross-cutting aspect of work practices in the area of human performance associated with the communication of human error prevention techniques, such as holding pre-job briefings, self- and peer-checking, and proper documentation of activities [H.4(a)](Section 1R04).

- Green. The inspectors identified a finding involving the failure to follow the requirements of Procedure AP 16E-002, “Post Maintenance Testing Development,” for the startup feedwater pump. On November 4-6, 2010, Wolf Creek workers disassembled the startup feedwater pump for numerous preventive and corrective activities including removing the rotating element. On November 17, 2010, Wolf Creek conducted surveillance Procedure STN AE-007, “Startup Main Feedwater Pump Operational Test,” following reassembly. The only acceptance criteria listed in this procedure is that the motor-driven feedwater pump starts from the control room with no local operator action. The inspectors found this contrary to Procedure AP 16E-002, which requires acceptance criteria for a pump flow capacity test, vibration, bearing and lubrication temperatures, motor current, external leakage, and lubrication level be found satisfactory. This issue is captured in the corrective action program as Condition Report 39494. Wolf Creek issued a new work package to conduct a single-point pump capacity test and complete the required postmaintenance testing. Wolf Creek found, pending final review, that initial calculations show that the pump design is capable of enough flow to provide a heat sink in emergency operating procedures.

Failure to follow Procedure AP 16E-002 for developing test criteria for plant equipment after the completion of maintenance activities is a performance deficiency. The finding is more than minor because it is associated with the Mitigating Systems Cornerstone attribute of equipment performance and it adversely affects the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609.04, the inspectors determined that the finding had very low safety significance (Green) because it did not result in 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, or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The inspectors determined that the finding had a cross-cutting aspect in the area of problem identification and resolution. Specifically, Wolf Creek created a testing procedure in response to a root cause evaluation, but did not consider acceptance criteria to ensure that the pump performs acceptably [P.1(d)](Section 1R19).

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to translate the design basis into instructions, procedures, and drawings. The inspectors found that the licensee failed to assess whether vortexing occurred in the containment spray additive tank in the event of a design-basis accident. Wolf Creek entered this issue in the corrective action program as Condition Report 38715.

  Failure to implement design control measures to analyze whether containment spray piping remained full of water was a performance deficiency. This finding was more than minor because it affected the design control attribute of the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of the containment spray system to respond to initiating events and prevent undesirable consequences. Specifically, the inspectors had reasonable doubt on the capability of the containment spray system to properly inject because of vortexing in the containment spray additive tank. The inspectors performed the significance determination using Inspection Manual Chapter 0609.04. The finding was determined to be of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in loss of operability or functionality. Although the failure to have this calculation had existed since original construction, the inspectors determined this finding reflected current performance since the licensee was required to evaluate likelihood of tanks allowing gas intrusion into the emergency core cooling systems in response to Generic Letter 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems." Consequently, this finding had problem identification and resolution cross-cutting aspects associated with the corrective action program in that the licensee did not thoroughly evaluate the potential for gas intrusion from all possible tanks [P.1(c)](Section 4OA5).

B. Licensee-Identified Violations

Violations of very low safety significance, which were identified by the licensee, have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and condition report numbers are listed in Section 4OA7.

REPORT DETAILS

Summary of Plant Status

Wolf Creek began the quarter shut down for Refueling Outage 18. Wolf Creek restarted on June 22, 2011. Reactor operators manually tripped the reactor from 82 percent power on June 26 due to a trip of main feedwater pump B. Wolf Creek restarted on June 29 and ended the quarter holding at 55 percent power to complete troubleshooting and repairs on main feedwater pump B.

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

The inspectors performed a review of 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 procedures and 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 reliability. Examples of aspects considered in the inspectors' review included:

- The coordination between the transmission system operator and the control room personnel during off-normal 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

During the inspection, the inspectors focused on plant-specific design features and the procedures used by plant personnel to mitigate or respond to adverse weather conditions. Additionally, the inspectors reviewed the Updated Safety Analysis Report (USAR) and performance requirements for selected systems, and verified that operator actions were appropriate per station procedures. Specific documents reviewed during this inspection are listed in the attachment. The inspectors also reviewed corrective action documents to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their corrective action program. These activities constitute completion of one readiness for summer weather affect on offsite and alternate ac power sample as defined in Inspection Procedure 71111.01-05.

b. Findings

No findings were identified.

.2 Readiness for Impending Adverse Weather Conditions

a. Inspection Scope

When thunderstorms, tornados, and high winds were forecast in the site vicinity on June 1, 9, and 16, 2011, the inspectors reviewed the plant preparations for the expected weather conditions. On June 1, 9, and 16, the inspectors walked down the offsite power system, refueling water storage tank, and reactor makeup water storage tank because their safety functions could be affected by high wind-generated missiles or a loss of offsite power. The inspectors evaluated these preparations against the site procedures and determined that actions by the plant staff were adequate. During the inspection, the inspectors focused on plant-specific design features and the station procedures used to respond to specified adverse weather conditions. The inspectors also toured outdoor areas of the plant to look for any loose debris that could become a wind-driven projectile. The inspectors evaluated operator staffing and accessibility of instrumentation and controls for systems required to operate the plant. Additionally, the inspectors reviewed the USAR and performance requirements for the selected systems and verified that operator actions were appropriate per station procedures. The inspectors also reviewed a sample of corrective action documents to verify that the licensee-identified adverse weather issues at an appropriate threshold and entered them into the corrective action program. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of two readiness for impending adverse weather condition samples as defined in Inspection Procedure 71111.01-05.

b. Findings

Introduction. The inspectors identified a Green noncited violation of Technical Specification 5.4.1.a for having no procedure to address onsite debris impacting plant equipment during severe weather.

Description. On June 1, 2011, a severe thunderstorm watch was announced by the national weather service. The inspectors walked down the transformer yard at 6 p.m., with the storms forecast to arrive later that night. The inspectors found numerous pieces of unsecured plywood and 2”x4” and 2”x8” planks. The inspectors brought this to the licensee’s attention, and Wolf Creek personnel secured the materials. The inspectors reviewed station Procedure AI 14-006, “Severe Weather,” Revision 9A. The procedure directed public address system announcements for national weather service severe weather declarations and instructions on personnel sheltering, but included no steps on equipment protection from onsite debris. The inspectors reviewed Procedure OFN SG-003, "Natural Events," Revision 20A, but it did not direct entry until a tornado is sighted or a tornado warning is issued.

The national weather service issued a tornado warning for the site area on June 9, at 3:20 p.m. The inspectors walked down the transformer yard at 5 p.m. The inspectors again found unsecured debris in the transformer and tank areas. The inspectors reported the debris to the control room and outage control center who sent personnel to secure the material. On June 10, a severe thunderstorm watch was issued at 5 p.m., and the inspectors walked down the transformer and tank yards at 6 p.m. to verify the corrective action from the previous day had been implemented for the pending storms. The inspectors found that some material was removed or secured, but also found numerous unsecured sections of scaffolding, wood, palettes, diamond plate, and debris. The inspectors discussed this with the outage control center. Condition Report 40351 was written with immediate actions to secure the loose materials. The extent of condition description included any area where inclement weather has the potential of creating airborne objects that could challenge plant equipment. On June 16, the inspectors walked down the transformer yard and tank areas during a thunderstorm. The inspectors found numerous unsecured pieces and stacks of wood and other debris that could impact plant equipment if winds were more severe. Wolf Creek responded by securing or removing the debris and writing Condition Report 40573 which implemented a weekly preventive maintenance activity to clean up outside areas and changed Procedure AI 14-006 to perform walkdowns of outside areas prior to severe weather. The inspectors found previously written condition reports on lack of adverse weather preparations for outdoor areas prior to the inspection.

<u>Analysis</u>. Failure to remove potential wind-driven debris from the transformer and tank areas before severe weather is a performance deficiency. This finding was more than minor because it impacted the protection against external factors attribute of the Initiating Events Cornerstone, and it affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors evaluated this finding using Inspection Manual Chapter 0609.04, and determined that it was of very low safety significance (Green) for June 16 because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment would be unavailable since the reactor was shutdown. Inspectors used Manual Chapter 0609, Appendix G, Checklist 4, for the other occurrences because Wolf Creek was in Modes 4 or 5. The finding again screened to Green because it did not increase the likelihood of a loss of inventory, did not cause the loss of reactor coolant system instrumentation, did not degrade the ability of the licensee to terminate a leak path or add inventory when needed, or degrade the ability to recover residual heat removal if it was lost. This finding has a cross-cutting aspect in the area of problem identification and resolution, specifically the corrective action program attribute because licensee short-term corrective actions failed to ensure debris was secured or removed prior to severe weather [P.1(d)].

<u>Enforcement</u>. Technical Specification 5.4.1.a requires, in part, that written procedures shall be established, implemented, and maintained covering the procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, 1978. Regulatory

Guide 1.33, Appendix A, Section 6.w, requires, in part, written procedures for acts of nature (e.g., tornado, flood, dam failure, earthquakes). Contrary to the above, prior to June 16, 2011, Wolf Creek had not established written procedures for acts of nature associated with tornados. Specifically, there were no procedural directions that addressed how the licensee was to protect from wind-driven projectiles, associated with tornados, in the protected area. Because this violation was of very low safety significance and was entered into the licensee's corrective action program as Condition Report 40573, this violation is being treated as a noncited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000485/2011003-01, "No Procedure for Debris in Transformer and Tank Yards Prior to Severe Weather."

1R04 Equipment Alignments (71111.04)

.1 Partial Walkdown

a. Inspection Scope

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

- March 8, 2011, Component cooling water

The inspectors selected this system based on its risk significance relative to the Reactor Safety Cornerstone at the time it was 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, 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 inspected accessible portions of the system 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 one partial system walkdown sample as defined in Inspection Procedure 71111.04-05.

b. Findings

Introduction. The inspectors reviewed a self-revealing Green noncited violation of Technical Specification 5.4.1a, "Administrative Procedures," for an inadequate clearance order verification which caused a loss of component cooling water B inventory.

Description. On March 8, 2011, Wolf Creek was preparing to implement Temporary Modification Order (TMO) 10-017-EG-00 to install temporary equipment to cool the radwaste system heat loads. These preparations included hanging clearance order D-HB-N-029 which required station operators to verify closed manual valves EGV0079 and HBV0110 and open and uncap the associated piping header hose connection valves HBV0088 and HBV0111. Until TMO 10-017 is implemented, component cooling water must be periodically aligned to the radwaste building to cool its associated nonsafety-related heat loads. This nonsafety component cooling water function adds seismic vulnerabilities that render the aligned train inoperable (NCV 05000482/2010007-01). At 9:30 a.m., station operators attempted to move valve HBV0110 in the closed direction and found that the valve would not turn. The operators compared the stem position relative to that of an identical model valve. The operators successfully manipulated travel of valve EGV0079 in the previous step from the fully open to fully closed position. This apparent position verification was made using the naked eye, and was the basis for assuming that the valve was firmly on its seat and signed the clearance order verifications accordingly.

At 2:37 p.m., the control room operators performed a planned routine alignment of component cooling water train B to radwaste. This action immediately resulted in a 200 gpm component cooling water leak through valve HBV0110 and out of the hose connection piping penetrations. The rapidly decreasing component cooling water B surge tank level caused an auto start of the demineralized water makeup to the component cooling water B surge tank and simultaneously sent an alarm to the control room operators. However, the demineralized water makeup capacity is only 60 gpm, resulting in a component cooling water B inventory loss of 140 gpm and a decreasing surge volume. Without prompt manual actions, the 5000 gallon component cooling water train B surge tank volume would have been exhausted in 25 minutes, at which point component cooling water train B would void and fail. For the duration of the leak, component cooling water train B was unavailable because it was unable to meet its accident mission time. Operators identified the cause and isolated the component cooling water supply from the radwaste building. The leak was determined to be approximately 500 gallons total, or 10 percent of the normal component cooling water surge tank inventory.

The leak revealed that valve HBV0110 was not fully closed but was stuck in a throttled position. Station operators were directed by the control room to attempt to move the valve in the open direction, which they did with an approved torque assist device. When the operators subsequently moved the valve in the closed direction, it moved beyond its previous position and was properly seated. Later that evening, when component cooling water was once again aligned to radwaste, no leakage occurred. Wolf Creek entered the event into their corrective action as Condition Report 34505.

The inspectors reviewed the history for valve HBV0110. All four of the subject valves had minimal manipulation since the waste evaporator package they were originally associated with had been abandoned in place in the early 1990s. Also, periodic maintenance to inspect and lubricate the valve internals has not been performed during this time. The last position verification made on valve HBV0110 was conducted April 21, 2006, and indicated that the valve was throttled partially open. The valve was also listed on drawing M-12HB02 as normally throttled. The clearance order paperwork specified to leave the valve 1.4 turns open upon removal of the clearance order.

The inspectors determined that the operators failed to meet the requirements of station Procedure AP 21E-001, step 6.4.2.1, to properly position the equipment/components in the sequence specified on the clearance order tag hang list, as well as step 6.4.3.1, the independent verification of that component or equipment condition. The inspectors' interviews with operators and station management indicated that the cause of the leakage was a lack of information communicated to the operators performing the tagout. Wolf Creek tagout practices did not provide expected, as-found component positions for taggers and verifiers in the clearance order tag "Hang List" nor is this information communicated during pre-job briefings. Wolf Creek initiated Condition Report 40219 which directed oral communication of the expected initial component positions during pre-job briefings and on the clearance order paperwork template.

Analysis. Failure to properly establish clearance order boundary isolation is a performance deficiency. The performance deficiency is more than minor because it impacted the equipment performance and human performance attributes of the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609.04, the finding was determined to be of very low safety significance (Green) because the finding is not a design or qualification deficiency confirmed not to result in loss of operability or functionality; the finding does not represent a loss of system safety function; the finding does not represent actual loss of safety function of a single train for more than its technical specification allowed outage time; the finding does not represent an actual loss of safety function of one or more nontechnical specification trains of equipment designated as risk significant per 10 CFR 50.65 for more than 24 hours; and the finding does not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The inspectors found that the finding had a cross-cutting aspect of work practices in the area of human performance. The licensee communicates human error prevention techniques, such as holding pre-job briefings, self- and peer-checking, and proper documentation of activities. Specifically, Wolf Creek did not communicate the expected as-found condition of valve HBV0110 to the taggers and verifiers of the clearance order [H.4(a)].

Enforcement. Wolf Creek Technical Specification 5.4.1a requires that procedures be established, implemented, and maintained covering the activities described in Regulatory Guide 1.33, Revision 2, Appendix A. Regulatory Guide 1.33, Revision 2, Appendix A, Section 1(c) requires, in part, procedures governing equipment control, including locking and tagging. Licensee Procedure AP 21E-001 "Clearance Orders," steps 6.4.2.1 and 6.4.3.1, specifies that equipment and components be positioned and verified in the sequence specified on the clearance order tag list. Contrary to the above, on March 8, 2011, the licensee failed to ensure the component was positioned and verified in the sequence specified on the clearance order tag list. Specifically, while performing clearance order D-HB-N-029, valve HBV0110 was not properly positioned and verified as specified on the clearance order tag list. These actions directly resulted in a loss of component cooling water train availability. Because this finding is of very low safety significance and was entered into the licensee corrective action program as Condition Reports 34505 and 40219, this violation is being treated as a noncited violation in accordance with Section 2.3.2 of the Enforcement Policy: NCV 05000482/2011003-02, “Failure to Properly Establish Clearance Order Boundary Isolation Resulting in Loss of Component Cooling Water Inventory.”

.2 Complete Walkdown and System Walkdown Associated with Temporary Instruction (TI) 2515/177

a. Inspection Scope

On April 27, 2011, the inspectors performed a complete system alignment inspection of the containment spray 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 inspected the system to review mechanical and electrical equipment lineups, 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.

The inspectors conducted a walkdown of the containment spray system in sufficient detail to reasonably assure the acceptability of the licensee’s walkdowns (TI 2515/177, Section 04.02.d). The inspectors also verified that the information obtained during the licensee’s walkdown was consistent with the items identified during the inspector’s independent walkdown (TI 2515/177, Section 04.02.c.3).

In addition, the inspectors verified that the licensee had isometric drawings that describe the containment spray system configurations and had acceptably confirmed the accuracy of the drawings (TI 2515/177, Section 04.02.a). The inspectors verified the following related to the isometric drawings.

- High point vents were identified

- Other areas where gas can accumulate and potentially impact subject system operability, such as at orifices in horizontal pipes, isolated branch lines, heat exchangers, improperly sloped piping, and under closed valves were acceptably referenced in documentation

- Horizontal pipe centerline elevation deviations and pipe slopes in nominally horizontal lines that exceed specified criteria were identified

- All pipes and fittings were clearly shown

- The drawings were up-to-date with respect to recent hardware changes and that any discrepancies between as-built configurations and the drawings were documented and entered into the corrective action program for resolution

The inspectors verified that piping and instrumentation diagrams accurately described the subject systems; that they were up-to-date with respect to recent hardware changes; and any discrepancies between as-built configurations, the isometric drawings, and the piping and instrumentation diagrams were documented and entered into the corrective action program for resolution (TI 2515/177, Section 04.02.b).

Documents reviewed are listed in the attachment to this report.

These activities constitute completion of one complete system walkdown sample as defined in Inspection Procedure 71111.04-05. Also, this inspection effort counts toward the completion of TI 2515/177. See Section 4OA5 for additional information.

b. Findings

No findings 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:

- March 19, 2011, Safety injection pump room A
- March 19, 2011, Control room ventilation equipment room B
- March 20, 2011, Auxiliary building 1988' pipe chase
- April 6, 2011, Containment building

The inspectors reviewed these 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 in Inspection Procedure 71111.05-05.

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08)

.1 Inspection Activities Other Than Steam Generator Tube Inspection, Pressurized Water Reactor Vessel Upper Head Penetration Inspections, Boric Acid Corrosion Control (71111.08-02.01)

a. Inspection Scope

The inspection procedure required review of two or three types of nondestructive examination activities and, if performed, one to three welds on the reactor coolant system pressure boundary. It also required review of one or two examinations with relevant indications (if any were found) that had been accepted by the licensee for continued service.

The inspectors directly observed the following nondestructive examinations:

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Pressurizer	TBB03-CIRCUM-1-W	Ultrasonic Examination
Pressurizer	TBB03-SEAM-4W	Ultrasonic Examination
Pressurizer	TBB03-10-B-W	Ultrasonic Examination
Pressurizer	TBB03-10-C-W	Ultrasonic Examination
Pressurizer	TBB03-10-B-IR	Ultrasonic Examination
Pressurizer	TBB03-10-C-IR	Ultrasonic Examination
Steam Generator	EBB01A-SEAM-5-W	Ultrasonic Examination

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Steam Generator	EBB01A-SEAM-8-W	Ultrasonic Examination
RV Closure Head Studs and Nuts	CH-STUD-19 through 36	Ultrasonic Examination
Main Steam Piping Support	AB-01-R001	Visual Examination 3
Main Steam Piping Support	AB-01-R003	Visual Examination 3
Feedwater Piping Support	AE05-R028	Visual Examination 3
Feedwater Piping Support	AE-04-R019	Visual Examination 3
Feedwater Piping Support	AE05-C001	Visual Examination 3
Main Steam Integral Attachment	AB-01-R010	Magnetic Examination
Feedwater Integral Attachment	AE-05-R028	Magnetic Examination

During the review and observation of each examination, the inspectors verified that activities were performed in accordance with ASME Boiler and Pressure Vessel Code requirements and applicable procedures. Indications were compared with previous examinations and dispositioned in accordance with ASME code and approved procedures. The qualifications of all nondestructive examination technicians performing the inspections were verified to be current.

Only the visual examination of AE05-R028, "Piping Support," identified any relevant indications. Repairs were made to AE05-R028 and it was reexamined and was satisfactory. Wolf Creek personnel stated that no relevant indications were accepted by the licensee for continued service.

The inspectors directly observed a portion of the following welding activities:

SYSTEM	WELD IDENTIFICATION	WELD TYPE
Reactor Coolant System	13BG22-W-33	Shield Metal Arc Welding
Reactor Coolant System	13BG22-W-34	Shield Metal Arc Welding

The inspectors verified, by review, that the welding procedure specifications and the welders had been properly qualified in accordance with ASME Code, Section IX, requirements. The inspectors also verified through record review that essential variables for the welding process were identified, recorded in the procedure qualification record, and formed the bases for qualification of the welding procedure specifications. Specific documents reviewed during this inspection are listed in the attachment.

b. Findings

.1 Failure to Ensure Fillet Weld Met Size Requirements on Train B Charging Header Vent Line

Introduction. The inspectors documented a self-revealing Green noncited violation of 10 CFR Part 50, Appendix B, Criterion IX, "Control of Special Processes," after the licensee failed to ensure that that the fillet weld between the train B charging header and the half-coupling used to attach two vent valves met 2:1 taper weld requirements. The undersized weld subsequently resulted in a 300 drop-per-minute leak in January 2011.

Description. On January 3, 2011, the licensee identified a 300 drop-per-minute pinhole leak at the weld joint between the train B charging header and/or the half coupling used to attach vent valves BGV0845 and BGV0846. The licensee measured the subject weld and concluded that the weld was undersized and the required 2:1 aspect ratio was not obtained. The weld was performed in the October/November 2009 timeframe during the installation of vent valves in the chemical and volume control system, the residual heat removal system, and the high pressure coolant injection system. Also, quality assurance inspectors inappropriately accepted the undersized weld.

Wolf Creek's extent-of-condition review concluded that 12 additional welds either did not meet the procedurally required 2:1 aspect ratio or did not meet ASME minimum weld size requirements. No other undersized welds developed leaks. After the leak was identified, the train B charging system was declared inoperable and the weld was repaired and built up to the correct 2:1 aspect ratio.

Wolf Creek performed a hardware failure analysis on the failed weld and concluded that although the main characteristics of the weld fracture were consistent with stress corrosion cracking, the crescent shape of the fracture indicated cyclic crack growth. The licensee also concluded that the configuration of the vent line with no lateral support could have created a cantilever effect on the line and in combination with a notch created by the lack of fusion in the weld root served as a stress concentrator. This issue was entered into the licensee's corrective action program as Condition Report 36438.

Analysis. Failure to meet ASME code weld size requirements is a performance deficiency. The finding was more than minor because it was associated with the equipment performance attribute of the Initiating Events Cornerstone. The finding adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. The inspectors performed a Phase 1 screening in accordance with Inspection Manual Chapter 0609.04 and determined that the finding was of very low safety significance (Green) because the issue did not result in exceeding the technical specification limit for identified reactor coolant system leakage or affect other mitigating systems resulting in a total loss of their safety function. This finding had a cross-cutting resources aspect in the area of human performance, because the licensee failed to ensure that welders and quality assurance inspectors were adequately trained in the procedural requirements and methods for measuring weld dimensions to assure nuclear safety [H.2(b)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion IX, "Control of Special Processes," requires in part, that measures be established to ensure that special processes, including welding are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements. Contrary to the above, in October 2009, the licensee failed to ensure that special processes, including welding, were controlled and accomplished using qualified procedures. Specifically, welders failed to ensure that the fillet weld between the train B charging header and the half-coupling used to attach two vent valves met 2:1 taper weld requirements, which subsequently resulted in a 300 drop-per-minute leak in January 2011. This issue was entered into the licensee's corrective action program as Condition Report 36438. Because this finding was determined to be of very low safety significance and was entered into the license's corrective action program, this violation is being treated as a noncited violation consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000482/2011003-03, "Failure to Assure Fillet Weld Met Size Requirements on Train B Charging Header Vent Line."

.2 Failure to Ensure Separation of Stainless Steel and Carbon Steel Grinding and Cutting Tools

Introduction. The inspectors identified a Green noncited violation of 10 CFR 50.55a, "Codes and Standards," after the licensee failed to ensure that stainless steel and carbon steel grinding wheels, flapper wheels, cutting wheels, and files were stored separately and used only for the weld preparation of the designated steel.

Description. During inspection of the tool issue room in the radiologically controlled area, the inspectors identified that tools designated for either stainless steel or carbon steel weld preparation were not stored separately. The inspectors noted that although stainless steel grinding wheels, flapper wheels, and cutting wheels were marked for stainless steel use, they were stored with carbon steel grinding wheels, flapper wheels, and cutting wheels. Additionally, the inspectors identified that although stainless steel files and carbon steel files were stored in separate drawers, there were files in the stainless steel drawer that appeared to have been used on carbon steel, and there was a file marked for use on stainless steel in the carbon steel drawer. There was also no procedure or written guidance pertaining to proper storage and control of the equipment.

The failure to separate tools used for stainless steel weld preparation from tools used for carbon steel preparation could result in the contamination of stainless steel welds by carbon steel and affect the material integrity and corrosion resistance. The licensee immediately removed the tools and replaced them with new tools stored separately for use on specific types of metals. This issue was entered into the licensee's corrective action program as Condition Report 3644.

Analysis. Failure to protect stainless steel welds from deleterious contamination is a performance deficiency. The finding was more than minor because it was associated with the equipment performance attribute of the Initiating Events Cornerstone. The finding adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations and if left uncorrected, the finding would become a more significant safety concern. The inspectors performed a Phase 1 screening in accordance with Inspection Manual Chapter 0609.04 and determined that the finding was of very low safety significance (Green) because the issue did not result in exceeding the technical specification limit for identified reactor coolant system leakage or affect other mitigating systems resulting in a total loss of their safety function. This finding had a resources cross-cutting aspect in the area of human performance, because the licensee did not provide adequate procedures for the preparation of stainless steel and carbon steel welds [H.2(c)].

Enforcement. Title 10 CFR 50.55a, states in part, that "Each operating license for a boiling or pressurized water-cooled nuclear power facility is subject to the conditions in paragraphs (f) and (g)." Title 10 CFR 50.55a(g)(4), requires, in part, that components classified as ASME Code Class 1, Class 2, and Class 3 meet the requirements set forth in Section XI of the ASME Boiler and Pressure Vessel Code and Addenda. Section XI of the ASME Code, Part IWA-4221(b)(2), states that "When adding a new component to an existing system, the Owner shall specify a Construction Code." The licensee specified Section III of the subject code when adding a new component to an existing system. Section III, Part NG4412, states that "The work [weld preparation] shall be protected from deleterious contamination." Contrary to the above, prior to June 2011, the licensee did not ensure that weld preparation was protected from deleterious contamination. Specifically, the licensee failed to ensure weld preparation was protected, in that tools located in the hot tool room drawers containing files, grinding wheels, flapper wheels, and cutting wheels that were used for the purpose of weld preparation, were found to contain a mixture of both stainless steel tools and carbon steel tools. This issue was entered into the licensee's corrective action program as Condition Report 36444. Because this finding was determined to be of very low safety significance and was entered into the licensee's corrective action program, this violation is being treated as a noncited violation consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000482/2011003-04, "Failure to Assure Separation of Stainless Steel and Carbon Steel Grinding and Cutting Equipment."

.2 Vessel Upper Head Penetration Inspection Activities (71111.08-02.02)

a. Inspection Scope

The inspectors witnessed portions of the licensee's performance of the required visual inspection (VT-2) of the reactor head and pressure-retaining components above the reactor pressure vessel head in accordance with requirement of ASME Code Case N-729-1 as mandated by 10 CFR 50.55a. Implementation required ASME Code IWA-2212 VT-2 under the mirror insulation on top of the reactor head through multiple access points. The inspectors reviewed the results of this inspection for evidence of leaks or boron deposits at reactor pressure boundaries and related insulation above the head. Specific documents reviewed during this inspection are listed in the attachment.

These actions constitute completion of the requirements for Section 02.02 of Inspection Procedure PI 71111.08.

b. Findings

No findings were identified.

.3 Boric Acid Corrosion Control Inspection Activities (71111.08-02.03)

a. Inspection Scope

The inspectors evaluated the implementation of the licensee's boric acid corrosion control program for monitoring degradation of those systems that could be adversely affected by boric acid corrosion. The inspectors reviewed the documentation associated with the licensee's boric acid corrosion control walkdown as specified in Procedure STN PE-040D, "RCS Pressure Boundary Integrity Walkdown," Revision 3, and Procedure AP 16F-001, "Boric Acid Corrosion Control Program," Revision 6A. The inspectors also reviewed the visual records of the components and equipment. The inspectors verified that the visual inspections emphasized locations where boric acid leaks could cause degradation of safety-significant components. The inspectors also verified that the engineering evaluations for those components where boric acid was identified gave assurance that the ASME code wall thickness limits were properly maintained. The inspectors confirmed that the corrective actions performed for evidence of boric acid leaks were consistent with requirements of the ASME code. Specific documents reviewed during this inspection are listed in the attachment.

These actions constitute completion of the requirements for Section 71111.08-02.03.

b. Findings

Failure to Assure Configuration Control of Safety-Related Systems

Introduction. The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," involving the failure of the licensee to review the suitability of replacing the design specified stainless steel manifold plugs with test fittings and brass caps on various flow transmitter equalizing block valve drain ports.

Description. During a boric acid walkdown, the inspectors identified that drain ports on the equalizing block of two separate reactor coolant system flow transmitters had brass fittings installed instead of stainless steel fittings. The inspectors brought this condition to Wolf Creek's attention. The licensee determined that a design configuration nonconformance existed in that licensee Drawing J-17D22 specified that stainless steel manifold plugs be installed in the drain ports during plant operation. The licensee failed to review the suitability of installing brass fittings and leaving test fittings on flow transmitter equalizing block valve drain ports instead of the design specified stainless steel manifold plugs. Wolf Creek immediately replaced the brass caps with stainless steel fittings. This issue was entered into the licensee's corrective action program as Condition Report 36439.

Analysis. Plugging instrument lines with test fittings of a different material is a performance deficiency. The finding was more than minor because it was associated with the design control attribute of the Initiating Events Cornerstone. The finding affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. The inspectors screened the finding per Inspection Manual Chapter 0609.04 and determined that the finding was of very low safety significance (Green) because the issue would not result in exceeding the technical specification limit for identified reactor coolant system leakage or affect other mitigating systems resulting in a total loss of their safety function. The inspectors also determined that the finding had a resources cross-cutting aspect in the area of human performance, because the licensee did not provide adequate training of personnel so that the inappropriately installed fittings could be identified during system walkdowns [H.2(b)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures be established for the selection and review of suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems, and components. Contrary to the above, the licensee failed to establish measures for the selection and review for suitability of parts that are essential to the safety-related functions of systems and components. Specifically, the licensee failed to review the suitability of replacing the design specified stainless steel manifold plugs with brass caps and test fittings on various equalizing block valve drain ports for pressure, differential pressure, and flow transmitters. This issue was entered into the licensee's corrective action program as Condition Report 36439. Because this finding was determined to be of very low safety significance and was entered into the licensee's corrective action program, this violation is being treated as a noncited violation consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000482/2011003-05, "Failure to Assure Configuration Control of Safety-Related Systems."

.4 Steam Generator Tube Inspection Activities (71111.08-02.04)

a. Inspection Scope

The inspection procedure specified an assessment of in situ screening criteria to assure consistency between assumed nondestructive examination flaw sizing accuracy and data from the EPRI examination technique specification sheets. The inspection procedure also specified assessment of appropriateness of tubes selected for in situ pressure testing, observation of in situ pressure testing, and review of in situ pressure test results. No conditions were identified that warranted in situ pressure testing. The steam generators are original construction steam generators.

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

Wolf Creek completed steam generator eddy current inspections for Refueling Outage 18 on April 12, 2011. In accordance with the EPRI steam generator examination guidelines, bobbin coil inspections were expanded in steam generator B due to inspection results associated with wear at anti-vibration bar locations that resulted in a C-2 condition. In accordance with the EPRI guidelines, another 20 percent of the tubing in steam generator B was inspected. No other scope expansions were required. In accordance with Technical Specification 5.5.9.c, nine tubes in steam generator B, three tubes in steam generator C, and three tubes in steam generator D were plugged based on inspection results indicating they contained flaws with a depth equal to or exceeding 40 percent of the nominal tube wall thickness. The damage mechanism associated with each of the pluggable indications was wear at anti-vibration bar locations. No tubes in steam generator A required plugging. No new corrosion damage mechanisms were identified.

The following secondary side maintenance and inspections were performed:

- Sludge lancing of all four steam generators. The amount of sludge removed from each steam generator was: steam generator A, 26 lbs; steam generator B, 34 lbs; steam generator C, 30 lbs; and steam generator D, 27.5 lbs.

- Foreign object search and retrieval of all four steam generators to locate, identify, and retrieve foreign objects present on the steam generator tube sheet. Foreign object search and retrieval was also performed to inspect for any possible loose parts identified during the eddy current program. Minor foreign objects were identified and addressed within the corrective action program and plant procedures. Visual examination and eddy current testing verified that no degradation was associated with any tubes surrounding the foreign objects.

- In-bundle inspection of steam generators B and C to inspect the condition of the top of the tube sheet and to augment the foreign object search and retrieval effort. No anomalies were identified during these inspections and the information will be used for trending and to plan future maintenance operations.

- Upper steam drum inspection in steam generators B and C to evaluate the condition of the upper steam drum components with regard to damage of any kind. Ultrasonic testing was also performed on locations susceptible to erosion on the feeding in steam generators B and C. No anomalies were identified and the information will be used for trending and to plan future maintenance.

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

These actions constitute completion of the requirements for Section 71111.08-02.04.

b. Findings

No findings were identified.

.5 Identification and Resolution of Problems (71111.08-02.05)

a. Inspection Scope

The inspectors reviewed 99 condition reports which dealt with inservice inspection activities and found the corrective actions for inservice inspection issues were appropriate. The specific condition reports reviewed are listed in the documents reviewed section. From this review the inspectors concluded that the licensee had an appropriate threshold for entering inservice inspection issues into the corrective action program and had procedures that direct a root cause evaluation when necessary. The licensee also had an effective program for applying industry inservice inspection operating experience. Specific documents reviewed during this inspection are listed in the attachment.

These actions constitute completion of the requirements for Section 71111.08-02.05.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Inspection Scope

On June 14, 2011, 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
- Compliance with assumptions for manual action timing in Chapter 15 of the USAR

The inspectors compared the crew’s performance in these areas to preestablished 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 Inspection Procedure 71111.11.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk significant system:

- Vital switchgear air conditioning units

The inspectors reviewed events such as where ineffective equipment maintenance has resulted in valid or invalid automatic actuations of engineered safeguards systems and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- Implementing appropriate work practices
- Identifying and addressing common cause failures
- Scoping of systems in accordance with 10 CFR 50.65(b)

- Characterizing system reliability issues for performance
- Charging unavailability for performance
- Trending key parameters for condition monitoring
- Ensuring proper classification in accordance with 10 CFR 50.65(a)(1) or -(a)(2)
- Verifying appropriate performance criteria for structures, systems, and components classified as having an adequate demonstration of performance through preventive maintenance, as described in 10 CFR 50.65(a)(2), or as requiring the establishment of appropriate and adequate goals and corrective actions for systems classified as not having adequate performance, as described in 10 CFR 50.65(a)(1)

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered 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 one quarterly maintenance effectiveness samples as defined in Inspection Procedure 71111.12-05.

b. Findings

No findings 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:

- June 3, 2011, Component cooling water train A while train B was inoperable

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 one maintenance risk assessment and emergent work control inspection sample as defined in Inspection Procedure 71111.13 05.

b. Findings

No findings were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- March 12, 2011, Emergency diesel generator A jacket water leak
- May 18, 2011, Source range NI-31 high counts after loss of cavity cooling
- June 16, 2011, Essential service water system flaw evaluations

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 so 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 personnel'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 three operability evaluations inspection samples as defined in Inspection Procedure 71111.15-04.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18)

.1 Temporary Modifications

a. Inspection Scope

To verify that the safety functions of important safety systems were not degraded, the inspectors reviewed the temporary modification identified as TMO 10-017, component cooling water modification to radioactive waste building.

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 one sample for temporary plant modifications as defined in Inspection Procedure 71111.18-05.

b. Findings

No findings were identified.

.2 Permanent Modifications

a. Inspection Scope

The inspectors reviewed key parameters associated with energy needs, materials, timing, heat removal, control signals, licensing basis, and failure modes for the permanent modification identified as the source range gamma metrics equivalency to Westinghouse detectors.

The inspectors verified that modification preparation, staging, and implementation did not impair emergency/abnormal operating procedure actions, key safety functions, or operator response to loss of key safety functions; systems, structures and components' performance characteristics still meet the design basis; the modification design assumptions were appropriate; and licensee personnel identified and implemented appropriate corrective actions associated with permanent plant modifications. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one sample for permanent plant modifications as defined in Inspection Procedure 71111.18-05.

b. Findings

No findings were identified.

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:

- November 17, 2010, Startup feedwater pump testing after rebuild
- March 7, 2011, Feedwater regulating bypass valve controller setting adjustment
- April 1, 2011, Solid state protection system train B after Westinghouse card testing
- May 17, 2011, Offsite power to engineered safety features transformer A after replacement of Raychem splices
- June 12, 2011, Component cooling water to thermal barrier heat exchangers after flow balance Procedure SYS EG-205
- June 24, 2011, Main turbine overspeed testing after replacement

The inspectors selected these activities based upon the structure, system, or component's 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 six postmaintenance testing inspection samples as defined in Inspection Procedure 71111.19-05.

b. Findings

Introduction. The inspectors identified a finding of very low safety significance (Green) involving the failure to follow the requirements of Procedure AP 16E-002, “Post Maintenance Testing Development,” for the startup feedwater pump.

Description. On November 4-6, 2010, Wolf Creek workers performed maintenance on the startup feedwater pump to replace a leaking pump casing gasket. Workers disassembled the pump per the vendor manual instructions and found a split casing gasket and both mechanical seals darkened and cracked from overheating. The pump was reassembled using new parts including bearings, O-rings, mechanical seals, and casing gasket. The service water cooling lines were also replaced. Wolf Creek Procedure AP 16E-002, “Post Maintenance Testing Development,” states that it provides guidelines to develop test criteria for plant equipment after the completion of maintenance activities. The procedure also ensures proper testing is implemented to prove components, systems, and sub-systems perform as designed after the completion of maintenance activities. Furthermore, Procedure AP 16E-002, Revision 9C, step 6.2 and attachments, requires that when a pump is disassembled or replaced, the postmaintenance testing includes a pump-flow capacity test be conducted to determine the capability of the pump to produce the required flow rates within the range of differential pressure limits. Also, it requires that vibration, bearing, and lubrication temperatures, motor current, external leakage, and lubrication level are satisfactory.

The inspectors reviewed root cause corrective action 25817-02-14 which created Procedure STN AE-007 to test the pump with no local actions and ensure a minimum recirculation flow of 60,000 pounds per hour for pump protection. The inspectors did not find a discussion of adequate flow. On November 17, 2010, Wolf Creek conducted surveillance Procedure STN AE-007, “Startup Main Feedwater Pump Operational Test,” following the pump reassembly. The only acceptance criteria listed in this procedure was that the motor-driven feedwater pump started from the control room with no local operator action. The test contained no acceptance criteria to ensure that after the completion of maintenance activities, the pump could produce the required flow rates for either low power or emergency operations.

The purpose of the motor-driven startup feedwater pump is to provide heated feedwater to the steam generators during plant startup and shutdown operations. The startup feedwater pump is a horizontal, multi-stage, centrifugal pump with a rated maximum flow rate of 260,000 pounds per hour. Maximum flow through the startup feedwater pump suction lines is limited to 230,000 pounds per hour to prevent excessive tube vibration in the steam generator blowdown regenerative heat exchanger. According to Wolf Creek training materials, Form APF 30E-004-01, Revision 2, “Main Feedwater System,” the required steam generator flow rate during plant startup is 200,000 pounds per hour. This is based on the maximum steam generator blowdown rate, the heat lost to ambient surroundings from all main steam lines, and the maximum heatup rate of all main steam lines and the turbine stop valve heat. The startup feed pump is also used in Emergency

Management Guideline FR-H1, “Response to Loss of Secondary Heat Sink,” step 17, to feed the steam generator until the steam generator level is restored to greater than the minimum level for ensuring an adequate heat sink. The success criteria in emergency operating procedures for feedwater is based on 270,000 pounds per hour for auxiliary feedwater or greater than 6 percent narrow range steam generator level. The emergency operating procedure setpoint document requires 250,000 pounds per hour from each motor-driven auxiliary feedwater pump.

This issue is captured in Condition Report 39494. Wolf Creek issued a new work package to conduct a single-point pump capacity test and complete the required postmaintenance testing in accordance with Procedure AP 16E-002. Wolf Creek also found that there was not a technical basis for the blowdown heat exchanger vibrations which previously limited the allowable flow through the pump to approximately 200,000 pounds per hour. Wolf Creek initial calculations, pending final review, show that the pump would be capable of enough flow to provide a heat sink.

Analysis. The failure to follow Procedure AP 16E-002 for developing test criteria for plant equipment after the completion of maintenance activities is a performance deficiency. The finding is more than minor because it is associated with the Mitigating Systems Cornerstone attribute of equipment performance and it adversely affects the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609.04, the inspectors determined that the finding had very low safety significance (Green) because it did not result in 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, or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The inspectors determined that the finding had a cross-cutting aspect in the area of problem identification and resolution. Specifically, Wolf Creek created a testing procedure in response to a root cause evaluation, but did not consider acceptance criteria to ensure that the pump performs acceptably [P.1(d)].

Enforcement. Enforcement action does not apply because the performance deficiency did not involve a violation of regulatory requirements. This finding is of very low safety significance and the issue was entered into the licensee's corrective action program as Condition Report 39494: FIN 05000482/2011003-06, “Inadequate Acceptance Criteria for Postmaintenance Testing of the Startup Feedwater Pump.”

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the outage safety plan and contingency plans for the refueling outage, conducted from March 19 through June 22, 2011, and June 26-29, 2011, to confirm that licensee personnel had appropriately considered risk, industry experience, and previous site-specific problems. The inspectors determined that the plan ensured sufficient defense in depth. During the refueling outage, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below.

- Configuration management maintains defense in depth, commensurate with the outage safety plan, and in compliance with technical specifications.

- Clearance activities were properly tagged and equipment configured to safely support work.

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

- 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.

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

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

- Controls over activities that could affect reactivity.

- Maintenance of secondary containment as required by the technical specifications.

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

- Startup and power ascension, tracking of startup prerequisites, walkdown of containment to verify that debris removal, and reactor physics testing.

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

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

These activities constitute completion of one refueling outage sample and one forced outage inspection sample as defined in Inspection Procedure 71111.20-05.

b. <u>Findings</u>

No findings were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the USAR, procedure requirements, and technical specifications to ensure that the surveillance activities listed below demonstrated that the systems, structures, and/or components 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 supported operability or functionality
- Test equipment removal
- Restoration of plant systems
- Fulfillment of ASME code requirements
- Updating of performance indicator data
- Engineering evaluations, root causes, and bases for restoring systems, structures, and components not meeting acceptance criteria were correct
- Reference setting data
- Annunciator and alarm setpoints

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

- September 8, 2010, Main steam valve AB-V85 inservice valve test

- April 24, 2011, Residual heat removal room cooler B test
- April 26, 2011, Filling, venting, and void surveillance of safety injection
- May 13, 2011, Tan-delta testing of offsite power underground cables
- May 14, 2011, STS PE-018, Containment integrated leak rate test
- May 18, 2011, Containment isolation valve EJHV8811B inservice test
- May 24, 2011, Filling, venting, and void surveillance of residual heat removal train B
- June 15, 2011, STS IC-615A, Safety injection signal slave relay test

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

These activities constitute completion of eight surveillance testing inspection samples as defined in Inspection Procedure 71111.22-05.

b. Findings

No findings were identified.

.2 Surveillance Testing Associated with TI 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems"

a. Inspection Scope

When reviewing the April 26, 2011, filling, venting, and void surveillance of safety injection and the May 24, 2011, filling, venting, and void surveillance of residual heat removal train B surveillances listed above in Section 1R22.1, the inspectors verified that the procedures were acceptable for (1) testing with shutdown operation, maintenance, and subject system modifications; (2) void determination and elimination methods; and (3) post-event evaluation.

The inspectors reviewed the procedures used for conducting surveillance tests and the determination of void volumes to ensure that the acceptance criteria were satisfied and would be reasonably assured to remain satisfied until the next scheduled surveillance test (TI 2515/177, Section 04.03.a). Also, the inspectors reviewed procedures used for filling and venting following conditions which may have introduced voids into the subject systems to verify that the procedures acceptably addressed testing for such voids and provided acceptable processes for their reduction or elimination (TI 2515/177, Section 04.03.b). Specifically, the inspectors verified that:

- Gas intrusion prevention, refill, venting, monitoring, trending, evaluation, and void correction activities were acceptably controlled by approved operating procedures (TI 2515/177, Section 04.03.c.1)

- Procedures ensured the system did not contain voids that may jeopardize operability (TI 2515/177, Section 04.03.c.2)

- Procedures established that void criteria were satisfied and will be reasonably ensured to be satisfied until the next scheduled void surveillance (TI 2515/177, Section 04.03.c.3)

- The licensee entered changes into the corrective action program as needed to ensure acceptable response to issues. In addition, the inspectors confirmed that a clear schedule for completion is included for corrective action program entries that have not been completed (TI 2515/177, Section 04.03.c.5)

- Procedures included independent verification that critical steps were completed (TI 2515/177, Section 04.03.c.6)

The inspectors verified the following with respect to surveillance and void detection:

- Specified surveillance frequency was consistent with technical specification requirements (TI 2515/177, Section 04.03.d.1)

- Surveillance frequencies were stated or, when conducted more often than required by technical specifications, the process for their determination was described (TI 2515/177, Section 04.03.d.2)

- Surveillance methods were acceptably established to achieve the needed accuracy (TI 2515/177, Section 04.03.d.3)

- Surveillance procedures included up-to-date acceptance criteria (TI 2515/177, Section 04.03.d.4)

- Procedures included effective follow-up actions when acceptance criteria are exceeded or when trending indicates that criteria may be approached before the next scheduled surveillance (TI 2515/177, Section 04.03.d.5)

- Measured void volume uncertainty was considered when comparing test data to acceptance criteria (TI 2515/177, Section 04.03.d.6)

- Venting procedures and practices utilized criteria such as adequate venting durations and observing a steady stream of water (TI 2515/177, Section 04.03.d.7)

- An effective sequencing of void removal steps was followed to ensure that gas does not move into previously filled system volumes (TI 2515/177, Section 04.03.d.8)

- Qualitative void assessment methods included expectations that the void will be significantly less than allowed by acceptance criteria (TI 2515/177, Section 04.03.d.9)

- Venting results were trended periodically to confirm that the systems are sufficiently full of water and that the venting frequencies are adequate. The inspectors also verified that records on the quantity of gas at each location are maintained and trended as a means of pre-emptively identifying degrading gas accumulations (TI 2515/177, Section 04.03.d.10)

- Surveillances were conducted at any location where a void may form, including high points, dead legs, and locations under closed valves in vertical pipes (TI 2515/177, Section 04.03.d.11)

- The licensee ensured that systems were not preconditioned by other procedures that may cause a system to be filled, such as by testing, prior to the void surveillance (TI 2515/177, Section 04.03.d.12)

- Procedures included gas sampling for unexpected void increases if the source of the void is unknown and sampling is needed to assist in determining the source (TI 2515/177, Section 04.03.d.13)

The inspectors verified the following with respect to filling and venting:

- Revisions to fill and vent procedures to address new vents or different venting sequences were acceptably accomplished (TI 2515/177, Section 04.03.e.1)

- Fill and vent procedures provided instructions to modify restoration guidance to address changes in maintenance work scope or to reflect different boundaries from those assumed in the procedure (TI 2515/177, Section 04.03.e.2)

The inspectors verified the following with respect to void control:

- Void removal methods were acceptably addressed by approved procedures (TI 2515/177, Section 04.03.f.1)

- The licensee had reasonably ensured that the residual heat removal pump is free of damage following a gas-related event in which pump acceptance criteria was exceeded (TI 2515/177, Section 04.03.f.2)

Documents reviewed are listed in the attachment to this report.

This inspection effort counts towards the completion of TI 2515/177. See Section 4OA5 for additional information.

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

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

a. Inspection Scope

The inspector performed an in-office review of the Wolf Creek APF 06-002-01, "Emergency Action Levels," Revision 15A. This revision made two administrative changes to EAL-6, "Loss of Electrical Power/Assessment Capability." The change included replacing the abbreviation "D/Gs" with the capitalized and bolded wording "DIESEL GENERATORS," and capitalizing and bolding the wording "NB TRANSFORMERS."

This revision was compared to its previous revision, to the criteria of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, and to the standards in 10 CFR 50.47(b) to determine if the revision adequately implemented the requirements of 10 CFR 50.54(q). This review was not documented in a safety evaluation report and did not constitute approval of licensee-generated changes; therefore, this revision is subject to future inspection.

These activities constitute completion of one sample as defined in Inspection Procedure 71114.04-05.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Data Submission Issue

a. Inspection Scope

The inspectors performed a review of the performance indicator data submitted by the licensee for the 1st Quarter 2011 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 were identified.

.1 Reactor Coolant System Specific Activity (BI01)

a. Inspection Scope

The inspectors sampled licensee submittals for the reactor coolant system specific activity performance indicator for the period from the 2nd Quarter 2010 through the 1st Quarter 2011. The inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6 to determine the accuracy of the performance indicator data reported during those periods. The inspectors reviewed the licensee's reactor coolant system chemistry samples, technical specification requirements, issue reports, event reports, and NRC integrated inspection reports for the period of April 1, 2010, through March 31, 2011, to validate the accuracy of the submittals. The inspectors also reviewed the licensee's condition 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. In addition to record reviews, the inspectors observed a chemistry technician obtain and analyze a reactor coolant system sample. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of one reactor coolant system specific activity sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

.2 Reactor Coolant System Leakage (BI02)

a. Inspection Scope

The inspectors sampled licensee submittals for the reactor coolant system leakage performance indicator for the period from the 2nd Quarter 2010 through the 1st Quarter 2011. The inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6 to determine the accuracy of the performance indicator data reported during those periods. The inspectors reviewed the licensee's operator logs, reactor coolant system leakage tracking data, issue reports, event reports, and NRC integrated inspection reports for the period of April 1, 2010, through March 31, 2011, to validate the accuracy of the submittals. The inspectors also reviewed the licensee's condition 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 reactor coolant system leakage sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

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 various baseline inspections discussed in previous sections, the inspectors reviewed issues to verify that they were being entered into the Wolf Creek corrective action program at an appropriate threshold. The inspectors verified the program to be addressing issues in a timely manner as well as identifying and correcting adverse trends. The inspectors reviewed attributes that included:

- Complete and accurate identification of the problem
- Timely correction, commensurate with the safety significance
- Evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent of condition reviews, and previous occurrences reviews
- 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 reviews for the identification and resolution of problems did not constitute any additional inspection samples. They were considered a part of the inspections performed during the quarter and documented in Section 1 of this report.

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

The inspectors performed a daily screening of items entered into the licensee's corrective action program through review of the Wolf Creek's daily corrective action documents.

The inspectors performed these daily reviews as part of their plant status monitoring activities and did not constitute any separate inspection samples.

b. Findings

No findings were identified.

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the Wolf Creek corrective action program and associated documents to identify trends that could represent a more significant safety issue. The inspectors focused their review on repetitive equipment issues, but also considered the results of daily corrective action item screenings, licensee trending efforts, and licensee human performance results. The inspectors considered the 6-month period of January through June 2011 although some examples expanded beyond those dates where necessary.

The inspectors also reviewed issues documented outside the normal corrective action program in major equipment problem lists, repetitive and/or rework maintenance lists, departmental problem/challenges lists, system health reports, quality assurance audit/surveillance reports, self-assessment reports, and maintenance rule assessments. The inspectors compared and contrasted their results with the conclusions reached in the Wolf Creek corrective action program trending reports.

The inspectors reviewed corrective actions for problem identification and resolution and human performance cross-cutting themes.

These activities constitute completion of a one semi-annual trend inspection sample as defined in Inspection Procedure 71152-05.

b. Findings

No findings were identified.

.4 Selected Issue Follow-up Inspection

a. Inspection Scope

During a review of Wolf Creek corrective action program items, the inspectors noted a condition report documenting over drilling of stud holes on a feedwater regulating valve body. The inspectors reviewed vendor manuals and station procedures for drilling and installing Heilicoil inserts. The inspectors reviewed vendor calculations for the strength of the joint. The inspectors interviewed engineers regarding the procedure and determined that work performed was consistent with vendor instructions.

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

b. Findings

No findings were identified.

4OA3 Event Follow-up (71153)

.1 March 19, 2011, Safety Injection

a. Inspection Scope

On March 19, 2011, with the reactor in hot standby, the inspectors responded to the control room when Wolf Creek received a safety injection signal for a rapid steamline pressure decrease. The inspectors reviewed control room logs and plant computer data. The inspectors interviewed control room operators about the conditions leading up to the event as well as the plant response. The inspectors reviewed plant operating practices regarding methods of feedwater heating, main steam procedures, and emergency operating procedures. From interviews with several members of the operating crew and plant data before and after the event, the inspectors independently reviewed the sequence of events:

- The crew assumed the watch in Mode 1 and reduced reactor power to Mode 3 for Refueling Outage 18.
- On March 18, 2011, SYS AE-200, "Feedwater Preheating During Plant Startup and Shutdown," Revision 29, was entered for the plant shutdown.
- At midnight, the turbine was tripped in accordance with procedure.
- At 12:07 a.m., feedwater temperature and steam flows begin oscillating. Over the next hour, feedwater temperature and steam flows oscillated. It was later determined that the oscillations were due to manual control of FB-PIC 300 combined with solenoid valve air leakage. This action was not peer-checked and control room supervision was not made aware. PIC-300 controls valve FB-17A which admits steam to the high pressure feedwater heaters. This is a large steam demand. The heaters had several temperature swings. This was not identified until after the safety injection.
- At 12:37 a.m., March 19, the reactor entered Mode 2.
- At 12:54 a.m., March 19, the reactor entered Mode 3.
- At 1:00 a.m., letdown automatically isolated at 17 percent pressurizer level due to a cooldown in progress.

- At 1:01 a.m., reactor coolant temperature could not be maintained, operators shut the main steam isolation valves to stop the cool down. Reactor coolant system temperature subsequently recovered to 560°F. Main feedwater pump A turbine was subsequently tripped from the control room.
- At 2:21 a.m., feedwater preheating was secured using SYS AE-200. Feedwater temperature decreases.
- With main steam isolation valves shut, the feedwater heaters continued to draw steam from the main steam header. Steam line temperature decreases.
- At 3:19 a.m., operators re-opened the main steam isolation valve bypass valves.
- To open the main steam isolation valves, operators entered Procedure SYS AB-120. This procedure is intended for use in Mode 4 with a maximum steam line pressure of 300 psi. Steam line pressure was approximately 1000 psi. Precaution 4.5 and step 6.14.2 require that main steam isolation valve differential pressure be less than 20 psi to open a main steam isolation valve. Senior reactor operators and management oversight mark these steps as not applicable.
- At 4:04 a.m., upon opening main steam isolation valve C, a negative steam line pressure rate on steam line C triggered an automatic safety injection signal.
- Operators entered EMG E-0, “Response to Reactor Trip or Safety Injection.”
- The pressurizer power-operated relief valve began cycling due to the pressure increase from the high head centrifugal charging pumps adding inventory to the reactor coolant system.
- At 4:11 a.m., the safety injection signal was reset and Technical Specification 3.0.3 was entered for both trains of emergency core cooling system inoperable because automatic safety injection signal was blocked.
- At 4:12 a.m., high pressure injection was terminated when the boron injection tank valves were shut.
- At 4:18 a.m., pressurizer power-operated relief valve stops cycling and closes.
- At 4:23 a.m., Wolf Creek transitioned to Procedure EMG ES-03, “Safety Injection Termination.”
- At 4:44 a.m., normal letdown flow from the reactor coolant system was re-established to reduce pressurizer level from a high of 88 percent.
- At 5:20 a.m., Wolf Creek completed Procedure EMG ES-03 and transitioned to Procedure OFN EM-024 “Safety Injection Recovery.”

- At 5:54 a.m., Wolf Creek notified the headquarters operations officer of the event by making event notification 46685 per 10 CFR 50.72(b)(2)(iv)(A) which is a 4-hour report for emergency core cooling system discharge into the reactor coolant system.

- At 6:39 a.m., both reactor trip breakers were closed using SYS SF-120 and Technical Specification 3.0.3 was exited.

- At 11:21a.m., Wolf Creek updated event notification 46685 with additional information regarding safety system actuation and loss of an accident mitigation safety system after the inspectors identified that these 8-hour reporting requirements may also apply to this event. Condition report 34995 was written for the potentially missed reports.

- Wolf Creek subsequently left the main steam isolation valves shut and cooled the plant using the atmospheric relief valves to a temperature where the residual heat removal system could be placed in service.

b. Findings

Introduction. A Green self-revealing cited violation of Technical Specification 5.4.1.a, “Administrative Procedures,” was reviewed involving the failure to correct a previous violation for an inadequate main steam system procedure. Specifically, Procedure SYS AB-120 was not corrected to establish appropriate conditions to open a main steam isolation valve. The inadequate procedure resulted in a safety injection.

Description. The inspectors reviewed a March 5, 2010, event involving excessive steam generator level swell and feedwater isolation following opening of a main feedwater isolation valve described in Condition Report 23938 and noncited violation NCV 05000482/2010004-01. Wolf Creek determined the cause of the March 5, 2010, P-14 feedwater isolation was an inadequate means of determining the pressure difference across the main steam isolation valves using control room pressure indicators. Procedure SYS AB-120, “Main Steam and Steam Dump Startup and Operation,” Revision 24, used an acceptance criterion of less than 25 psi differential pressure to allow opening of a main steam isolation valve. The procedure directed the operators to determine valve differential pressure using control room indicators before opening the main steam isolation valves. The control room instruments have ranges from 0 to 1300 psi or greater with a 25 psi scale and are accurate to within plus or minus 25 to 38 psi. The inspectors concluded the apparent cause evaluation in Condition Report 23938 appropriately determined that instrument uncertainty equal to or greater than the procedure’s acceptance criteria was not reasonable. Subsequently, Wolf Creek revised Procedure SYS AB-120 to direct operators to determine differential pressure using locally installed instruments in lieu of the control room pressure indicators, but this change was only implemented for steam line pressures below 300 psi. Additional changes were made to several procedures which reduced the allowable steam generator level band when opening a main steam isolation valve.
Procedure SYS AB-120 revisions did not address steam pressures above 300 psi nor

were its precautions and limitations updated to reflect main control board instrumentation accuracy.

On March 19, 2011, Wolf Creek was in Mode 3 shutting down for a refueling outage. The steam header pressure was 1000 psi. At 1:01 a.m., operators shut all main steam isolation valves due to an excessive cooldown. Several hours later, the operators began to open the valves using Procedure SYS AB-120. A tighter acceptance criterion of 20 psi differential pressure was specified in the procedure before opening a main steam isolation valve. Wolf Creek operators did not use local instruments as specified by the procedure. Instead they used control room instruments to determine main steam line pressures on both side of the main steam isolation valves without considering that the instrument uncertainty exceeded the range of acceptance criteria. While the control room pressure and temperature instruments indicated that the differential pressure was acceptable, actual differential pressure was about 200 psi. When main steam isolation valve C was opened, a safety injection signal occurred.

The inspectors reviewed corrective actions for Procedure SYS AB-120 and found several missed opportunities to correct the deficiency. On October 18, 2010, Condition Report 29168 was written stating “Guidance for opening MSIVs not good above 35 psi steam press,” as its problem description. Wolf Creek reviewed Condition Report 30453 which responded to noncited violation NCV 05000482/2010004-01 and appropriately concluded that the evaluation was flawed for two reasons. First, Condition Report 30453 failed to incorporate the instrument uncertainty issue previously identified in Condition Report 29168 into the precautions and limitations of Procedure SYS AB-120. Second, Condition Report 30453 failed to address the full range of anticipated plant conditions which may require opening a main steam isolation valve, specifically steam pressures above 300 psi. The inspectors concluded the failure to implement comprehensive corrective actions to address the March 5, 2010, event directly contributed to the March 19, 2011, inadvertent safety injection event and constituted a failure to restore compliance for noncited violation NCV 05000482/2010004-01.

The inspectors reviewed the safety impact of the safety injection transient on Wolf Creek. Actual safety impacts included a waterhammer on the main steam lines. This caused a partial failure of main steam isolation valve actuator to bonnet gaskets. The pressurizer power-operated relief valve 455 cycled seven times. Main feedwater was lost when the feedwater isolation valves received a close signal from the safety injection. Emergency core cooling system injection check valve BB8948C experienced body-to-bonnet gasket leakage. The pressurizer started at 17 percent level and filled to 88 percent level until letdown was reestablished. Inadvertent safety injection has the potential to challenge the pressurizer safety valves and escalate to a loss of coolant accident if not terminated.

Analysis. The failure to correct deficiencies in Procedure SYS AB-120 for steam pressures above 300 psi was a performance deficiency. The inspectors determined that this finding was more than minor because it impacted the equipment performance attribute for the Initiating Events Cornerstone and it affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, this issue relates to the configuration control attribute for shutdown equipment alignment. The inspectors evaluated the significance of this finding using Inspection Manual Chapter 0609.04. Assuming worst case degradation, the finding resulted in exceeding the technical specification limit for reactor coolant system leakage due to the pressurizer power-operated relief valve cycling. Therefore, the inspectors screened the finding to a Phase 2 review by the senior reactor analyst. The senior reactor analyst used the Wolf Creek SPAR Model and concluded that the incremental core damage probability was 3.7E-7, Green. The inspectors found that the cause of the finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program. Specifically, several evaluations failed to include an adequate extent of condition review that identified that the procedures were inadequate for opening a main steam isolation valve at system pressures above 300 psi [P.1(c)].

Enforcement. Technical Specification 5.4.1.a requires that procedures be established, implemented, and maintained covering the activities described in Regulatory Guide 1.33, Revision 2, Appendix A. Regulatory Guide 1.33, Revision 2, Appendix A, Section 3.i, requires procedures for the startup, operation, and shutdown for the main steam system. Wolf Creek Procedure SYS AB-120, “Main Steam and Steam Dump Startup and Operation,” Revision 27, implements these requirements for the main steam system. Contrary to the above, from March 5, 2010, to March 19, 2011, Wolf Creek Procedure SYS AB-120 had not been maintained to cover activities for the startup, operation and shutdown of the main steam system. Specifically, Procedure SYS AB-120, Revision 27, contained inadequate steps necessary to open a main steam isolation valve without causing a safety injection signal.This issue and the corrective actions are being tracked by the licensee in Condition Report 34964. Due to the licensee’s failure to restore compliance from previous NCV 05000482/2010004-01 within a reasonable time after the violation was identified, this violation is being cited as a Notice of Violation consistent with Section 2.3.2 of the Enforcement Policy: VIO 05000482/2011003-07, “Failure to Correct Procedure for Opening Main Steam Isolation Valves” (EA-11-149).

.2 March 21, 2011, Low Temperature Overpressure System Actuation.

a. Inspection Scope

On March 21, 2011, Wolf Creek was shutdown for a refueling outage. While cleaning the reactor coolant system, operators failed to maintain reactor coolant system pressure below 350 psi. When charging was increased for the clean-up, the low temperature overpressure setpoint was exceeded causing pressurizer power-operated relief valve 455 to lift three times. The inspectors interviewed reactor operators, reviewed control room logs, procedures, pressure and temperature limits report, License Amendment 130, and plant computer data.

b. Findings

Introduction. The inspectors reviewed a self-revealing Green noncited violation of Technical Specification 5.4.1.a, "Procedures," for failure to maintain pressure below the low pressure overpressure protection setpoint.

Description. On March 21, 2011, Wolf Creek was adjusting the chemical and volume control system to inject hydrogen peroxide into the reactor coolant system to induce a crud burst to reduce system radioactivity for the refueling outage. Letdown flow was at approximately 63 gpm. The unit was in Mode 5 with the pressurizer solid and maintaining reactor coolant system temperature at 160°F and 350 psig pending reactor coolant system cleanup. The pressurizer is considered 'solid' when it is water filled because water is not compressible when compared with a gas bubble. Charging and letdown were in the process of being increased in order to increase the rate of reactor coolant system cleanup. At 2:52 p.m., power-operated relief valve 455A cycled three times over the following 4 minutes when reactor coolant system pressure increased to its lift setpoint of 415 psig. Reactor coolant system pressure control was subsequently reestablished at 350 psig when letdown flow was increased to approximately 120 gpm.

During interviews, the operators stated that the charging header controller was adjusted before letdown, and that it was sluggish at the low pressure. The procedure only stated to maintain pressure and did not provide specific guidance. At the time, operators had a band of 330-350 psig to maintain, and the operators stated that the normal charging pump controller was sluggish at its reduced operating pressure. The operators stated that the charging pump controller was increased three times and on the third time, a large increase in charging was received.

The inspectors reviewed plant computer data and found that when charging header pressure was initially increased without increasing letdown flow from the residual heat removal system, the reactor coolant system pressure rapidly increased. As the 4 minute event progressed, the normal charging pump controller was adjusted several times while letdown was progressively increased. Charging header pressure and flow drove the increases in reactor coolant system pressure. The three lifts of the power operated relief valve were due to the changes in charging header pressure with a solid pressurizer.

The inspectors reviewed Procedures GEN 00-006 and SYS BG-120 and found that they did not contain any precautions or limitations regarding the reactor coolant system pressure response to a sluggish charging controller with a water-solid plant. There were no instructions that letdown should have been increased first and to adjust charging second, to match. Procedure GEN 00-006, step 6.44.8.3 only stated to maintain a pressure band of 325-350 psig when adjusting charging flow. Although the procedures had several steps to maximize letdown and charging for reactor coolant system clean-up, there were no specific steps on how to perform this, and there were no continuous action steps or precautionary steps to prevent over-pressurizing the reactor coolant system.

The inspectors reviewed the "Just in Time Training" for the refueling outage and identified that it contained guidance on raising letdown to 120 gpm and subsequently taking the plant solid. It did not contain guidance or lessons on manipulating letdown with the plant solid. With the reactor coolant system pressure at the upper end of the band specified in Procedure GEN 00-006, letdown would be appropriate to adjust first to prevent the lifting of relief valves. If the reactor coolant system was solid at the lower end of the pressure band specified in Procedure GEN 00-006, adjusting charging first would be appropriate to avoid a decrease in reactor coolant system pressure that could meet the reactor coolant pump trip criteria.

Analysis. Failure to maintain pressure below the power operated relief valve setpoint was a performance deficiency. The performance deficiency was more than minor because it impacted the Initiating Events Cornerstone objective of configuration control to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The significance of the finding was determined using Inspection Manual Chapter 0609, "Significance Determination Process," Appendix G, Checklist 2, and determined to be of very low safety significance (Green), because it did not cause the loss of mitigating capability of core heat removal, inventory control, power availability, containment control, or reactivity control. Additionally, the finding also did not cause any low temperature overpressure technical specifications to be exceeded. The inspectors found that the cause of the finding had a cross-cutting aspect in the area of human performance. Specifically, operators had to rely on skill of the craft when procedures should have supplied more instruction for manipulating charging and letdown with the pressurizer water solid [H.2.c].

Enforcement. Wolf Creek Technical Specification 5.4.1.a, "Procedures," requires, in part, that written procedures shall be established, implemented and maintained for the activities recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, "General Plant Operating Procedures," Section 2.j, requires procedures for hot standby to cold shutdown. Procedure GEN 00-006, "Hot Standby to Cold Shutdown," Revision 76, implements this procedure. Procedure GEN 00-006, Step 6.44.8.3 required the licensee to maintain a pressure band below 350 psig when manipulating charging flow. Contrary to the above, on March 21, 2011, Wolf Creek did not implement Procedure GEN 00-006, step 6.44.8.3, to maintain a pressure band below 350 psig when manipulating charging flow. Because the finding is of very low safety significance and has been entered into the licensee's corrective action program as Condition Report 35244, this violation is being treated as a noncited violation consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000482/2011003-08, "Failure to Maintain Reactor Coolant System Pressure Below Relief Valve Setpoint."

.3 April 5, 2011, Vital Switchgear Room Fire

a. Inspection Scope

The inspectors responded to a fire in the switchgear rooms and to the control room. The inspectors interviewed fire brigade leaders and the control room shift manager regarding the fire alarms and the fire brigade response and examined the damage inside of nonvital inverter PN009. The inspectors observed postfire actions to ventilate the area to remove the smoke and Halon.

b. Findings

Introduction. The inspectors reviewed a self-revealing Green noncited violation of License Condition 2.C.5 for failure to implement adequate fire impairments which affected both trains of vital ac and dc switchgear.

Description. On March 26, 2011, Wolf Creek implemented a breach authorization requiring a continuous fire watch because the doors between vital ac and dc switchgear rooms were propped open. These doors are 3-hour fire barriers. This was done to allow the train B air conditioning unit and ventilation to provide cooling to the train A switchgear in accordance with Procedure SYS GK-200, "Inoperable Class IE A/C Unit." With the train A air conditioning unit out of service, two sets of double doors were propped open between vital ac switchgear trains A and B on the 2000' elevation of the control building. On April 5, 2010, Wolf Creek completed preventive maintenance on nonvital inverter PN009 which is located in the 2000' elevation train A vital switchgear room. Wolf Creek was preparing to test nonvital inverter PN009 and reenergized it for about 20 minutes. Two electricians were at the inverter cabinet in the train A vital switchgear room when smoke began emanating from the top of the cabinet. The electricians shut off the dc input and opened the ac output breakers on the lower door of the cabinet. The Halon actuation alarm sounded indicating that Halon would discharge into the room in 30 seconds. One electrician told the fire watch that it was necessary to evacuate. The two electricians and the fire watch were egressing through the north missile door when the Halon system discharged. The breached doors between ac switchgear rooms were not shut. The fire brigade responded and removed an extension cord and shut the doors between the vital ac switchgear rooms. The fire brigade found only smoke and Halon in the rooms and no fire at PN009. Subsequent examination by Wolf Creek and the vendor found that vendor errors in labeling the terminals caused an excessive current in an adjacent transformer which caused the fire. Both transformers were replaced. The vendor stated that no other damage occurred. Condition Report 36719 was written on the inadequate fire watch response.

The inspectors interviewed the April 5 fire watch and found that he thought Halon was going to discharge into both the trains A and B vital switchgear rooms. Thus, he would have to egress through the north missile door and not to the train B switchgear room. He understood his duty to shut the doors upon alarm, but indicated that he would not be able to remove the extension cord, shut the doors, and exit within 30 seconds. The fire watch stated that removing the extension cord and shutting the doors would likely take 3 to 4 minutes. The inspectors found that the Halon system was designed to discharge into the switchgear room with the alarming smoke detectors. The fire watch also stated that he left the room without shutting the doors because the electricians instructed him to leave the room prior to the Halon actuation.

The inspectors found that the only written instructions for fire watches was the statement on the fire impairment which said "Per AP 10-104, section 6.1.9 (SYS GK-200), in case of fire or Halon discharge, close doors 33011 & 33023 and exit area and notify control room." Wolf Creek relies on training and reading of the fire impairment to understand the compensatory action. The inspectors reviewed the design of the 1301 Halon system and found that the system was sized to extinguish a fire in one switchgear room only.

The inspectors found that with doors 33011 and 33023 (each a set of double doors) open between vital switchgear rooms, the Halon system would not have been successful at extinguishing a fire.

The inspectors reviewed written statements from the fire brigade, the fire watch, and the electricians. The inspectors reviewed Procedure AP 10-107, “Fire Protection Impairment Control,” and Procedure APF 10-104-01, “Breach Authorization Permit,” and found that the requirements of the breach permit were not met because the fire watch failed to close doors 33011 and 33023 during an actual fire. Procedure AP 10-104, step 5.62, states, in part, that the boundary watch must be able to clear any cord or tool crossing a breached barrier and to notify the control room if any condition in which a breached barrier cannot be closed within the time requirements. The inspectors reviewed form APF 10-104-01, breach authorization, for the 2000’ and 2016’ elevation switchgear rooms and found no quantitative timing requirements for closing the doors. The inspectors concluded that a 30-second acceptance criterion was critical because open doors would prevent the Halon system from reaching the necessary concentration to extinguish a fire. The inspectors found that the breach permit was not met because the fire watch did not close the doors and that the breach permit was inadequate because it did not contain timed acceptance criteria necessary to ensure the success of the Halon system.

On April 12, 2011, the inspectors interviewed a different vital ac switchgear room fire watch and found that the watch was also not clear on their duty to shut the doors regardless of what other workers tell them to do. The fire watch did have correct knowledge of the Halon system, the 30-second delay between the alarm and discharge, and what room to egress to depending on the fire location. The inspectors shared this with the outage control center.

On April 14, 2011, Wolf Creek inhibited the Halon systems for the rod-drive motor generator set room, all vital dc switchgear rooms, and the vital ac switchgear rooms. Wolf Creek judged it more important to ensure that the fire watches shut the fire doors rather than have an ineffective automatic Halon system actuation. On April 13, 2011, Wolf Creek initiated new training for all fire watches to ensure they had copies of the breach permits. The inspectors interviewed fire watches on April 14, 2011, and found that the watches had a complete and correct understanding of their duties.

The inspectors found that the inability to close these fire doors was identified in Refueling Outage 16 in Condition Report 2008-1357. Actions included protective equipment and training to remove cables crossing the open doors, but the inspectors concluded that those corrective actions did not ensure proper fire watch actions on April 5, 2011. Corrective actions included “APF 10-104-06 to include Special Requirements for Boundary Watch.” Although a new section of the breach impairments was created, it was typically not utilized when breaching vital switchgear doors.

As an extent of condition review, the inspectors reviewed previous fire impairments for Procedure SYS GK-200 for open fire doors on the 2000’ and 2016’ control building elevations. The inspectors found that Procedure SYS GK-200 had been implemented

23 times over the prior year of operation representing approximately 36 days of impairments for both trains of ac and dc switchgear and batteries.

Analysis. The failure to implement adequate fire watches that ensured the success of the Halon system was considered a performance deficiency. The performance deficiency was considered more than minor because it impacted the Initiating Events Cornerstone and its objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the fire area of the protection against external factors attribute was impacted. The inspectors used Inspection Manual Chapter 0609.04 to screen the finding to Inspection Manual Chapter 0609, Appendix F, because the fire protection defense-in-depth strategies involving automatic suppression, fire barriers, administrative controls were degraded. Because the subject finding was not clearly covered by the approach used in Appendix F, the senior reactor analyst performed a Phase 3 analysis. The doors were open for 36 days, so a 36-day exposure period (EXP) was used. The analyst used generic values for the Fire Ignition Frequency (λ_{FI}), Severity Factor (P_{SF}) and the probability of manual suppression before damage (P_{MS}). The Fire Mitigation Frequency (λ_{FM}) was calculated as follows:

$$\begin{aligned}\lambda_{FM} &= \lambda_{FI} * P_{SF} * P_{MS} * EXP \\ &= 2.0 \times 10^{-2}/\text{year} * 0.1 * 0.1 * 36 \text{ days} \div 365 \text{ days/year} \\ &= 1.97 \times 10^{-5}\end{aligned}$$

The analyst assumed that if the fire grew to a point that it could spread to the opposite train, it would actuate the opposite train's Halon system and cause an isolation of all ventilation. However, there was no credible source of flammable materials that would cause the growth of the fire into the opposite train's switchgear. Therefore, the analyst quantified the conditional core damage probability (CCDP) for the failure of Switchgear NB01 using the Standardized Plant Analysis Risk Model for Wolf Creek Station, Revision 8.15. The resulting CCDP was 8.3×10^{-4}. The final change in core damage frequency (ΔCDF) was calculated as follows:

$$\begin{aligned}\Delta CDF &= \lambda_{FM} * CCDP \\ &= 1.97 \times 10^{-5} * 8.3 \times 10^{-4} \\ &= 1.6 \times 10^{-8}\end{aligned}$$

Therefore, this finding was determined to be of very low safety significance (Green).

The inspectors found that the cause of the finding had a cross-cutting aspect in the area of problem identification and resolution. Specifically, corrective actions from 2008 ineffective fire watches did not prevent recurrence of the April 5, 2011, inadequate fire watch [P.1.d].

Enforcement. License condition 2.C.(5) states, in part, that the licensee shall maintain in effect all provisions of the approved fire protection program as described in the

Standardized Nuclear Unit Power Plant System USAR for the facility through Revision 17, the Wolf Creek site addendum through Revision 15, and as approved in the safety evaluation report through Supplement 5, Amendments 191 and 193. AP 10-100, fire protection program, states, in part, that AP 10-104, "Breach Authorization," is part of the fire protection program. Procedure AP 10-104, step 5.62, states, in part, that the boundary watch must be able to clear any cord or tool crossing a breached barrier and to notify the control room if any condition in which a breached barrier cannot be closed within the time requirements. Procedure AP 10-104, steps 6.1.8 and 6.1.9, require, in part, that a continuous fire watch shall be established for the vital switchgear rooms because open doors will reduce Halon concentration and expose redundant trains to the same fire. Contrary to the above, prior to April 14, 2011, the licensee failed to implement and maintain in effect all provisions of the approved fire protection program. Specifically, the licensee used an ineffective fire barrier breach permit system that did not ensure that the Halon systems would effectively extinguish fires because the fire watches could not clear any cord or tool crossing a breached barrier and did not notify the control room of a condition in which a breached barrier could not be closed within the time requirements. The licensee entered this issue into their corrective action program as Condition Report 36719. Because this violation was of very low safety significance and it was entered into the corrective action program, this violation is being treated as a noncited violation, consistent with the NRC Enforcement Policy, Section 2.3.2:
NCV 05000482/2011003-09, "Inadequate Fire Watch Defeats Halon Fire Suppression in Vital Switchgear Rooms During Fire."

.4 <u>(Closed) Licensee Event Report (LER) 2006-003-00, Indications Discovered on Pressurizer during Preplanned Inservice Inspections</u>

On October 11, 2006, during Refueling Outage 15, engineering personnel performing preplanned inservice examination of the pressurizer nozzle to safe end dissimilar metal welds identified five circumferential flaw indications. Three indications were located in the surge nozzle dissimilar metal weld, one indication was in the safety nozzle C dissimilar metal weld, and one indication was in the relief nozzle dissimilar metal weld. The locations were all part of the reactor coolant system pressure boundary. There was no evidence of reactor coolant system pressure boundary leakage. The most probable mechanism responsible for the indications was primary water stress corrosion cracking. Wolf Creek Generating Station was in Mode 5, cold shutdown. Weld overlay repairs of the flaw indications were performed prior to the unit's return to power operations. The inspectors reviewed LER 05000482/2006-003-00 to verify that the cause was identified and that corrective actions were appropriate. This LER is closed.

.5 <u>(Closed) Notice of Violation VIO 05000482/2010006-05, Failure to Correct NRC Identified NCV Apparent Cause Evaluation Vice Root Cause Evaluation for Essential Service Water</u>

The violation involved the failure to perform an adequate cause evaluation and to take corrective actions to preclude repetition for a significant condition adverse to quality. Although determined to be of very low safety significance (Green), this violation was cited in Notice of Violation 05000482/2010006-05 because not all of the criteria specified in Section 2.3.2 of the NRC Enforcement Policy were satisfied (EA-10-160). Specifically, the Wolf Creek Generating Station failed to restore compliance within a reasonable time for a previously NRC identified noncited violation as documented in NRC Inspection Report 05000482/2009007-03. The inspectors reviewed the corrective actions completed by the licensee and verified that the cause was identified and that corrective actions were appropriate. This violation is closed.

4OA5 Other Activities

.1 (Closed) NRC TI 2515/177, Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems (NRC Generic Letter 2008-01)

a. Inspection Scope

As documented in Sections 1R04.1 and 1R22 of this report, the inspectors confirmed the acceptability of the described actions for the high pressure safety injection system and the containment spray system. This inspection effort counts towards the completion of TI 2515/177 which is closed in this inspection report.

The inspectors evaluated whether the licensee maintained documents, installed system hardware, and implemented actions with the information provided in their response to NRC Generic Letter 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems." Specifically, the inspectors verified that the licensee has implemented or was in the process of implementing the commitments, modifications, and programmatically controlled actions described in the response to Generic Letter 2008-01. The inspectors conducted their review in accordance with TI 2515/177 and considered the site-specific supplemental information provided by the Office of Nuclear Reactor Regulation (NRR) to the inspectors.

b. Inspection Documentation

The inspectors reviewed the licensing basis, design, testing, and corrective actions as specified in the TI. The specific items reviewed and any resulting observations are documented below.

Licensing Basis. The inspectors reviewed selected portions of licensing basis documents to verify that they were consistent with the NRR assessment report and that the licensee properly processed any required changes. The inspectors reviewed selected portions of technical specifications, technical specification bases, and the USAR. The inspectors also verified that applicable documents that described the plant and plant operation, such as calculations, piping and instrumentation diagrams, procedures, and corrective action program documents, addressed the areas of concern and were changed, if needed, following plant changes. The inspectors confirmed that the licensee performed surveillance tests at the frequency required by the technical specifications. The inspectors verified that the licensee tracked their commitment to evaluate and implement any changes that will be contained in the technical specification task force traveler.

Design. The inspectors reviewed selected design documents, performed system walkdowns, and interviewed plant personnel to verify that the licensee addressed design and operating characteristics. Specifically:

- The inspectors verified that the licensee had identified the applicable gas intrusion mechanisms for their plant.

- The inspectors verified that the licensee had established void acceptance criteria consistent with the void acceptance criteria identified by NRR. If NRR acceptance criteria were not met, then the inspectors verified that the licensee has justified the deviations. The inspectors also confirmed that the range of flow conditions evaluated by the licensee was consistent with the full range of design basis and expected flow rates for various break sizes and locations.

  The inspectors noted that the licensee used the methods developed by Westinghouse to estimate the suction voids emergency core cooling system pumps. Westinghouse documented their review and test results in WCAP-16631-NP, "Testing and Evaluation of Gas Transport to the Suction of ECCS Pumps." Wolf Creek used WCAP-16631-NP to show that GOTHIC can acceptably predict quantitative void transport behavior. However, the inspectors noted that test configuration and conditions differed from actual plant configuration and conditions. These methods relied on industry testing documented by Westinghouse and used the GOTHIC computer code to better estimate the impacts resulting from voiding in the emergency core cooling systems.

  The licensee had received analyses for their facility based upon the simplified equation developed by Westinghouse, which would more accurately estimate the void sizes allowed on the suction of the emergency core cooling pumps without affecting operability. In addition, the license had received a revised estimate of water hammer effects developed by Fauske on the pump discharges for their emergency core cooling systems. These analyses would replace the use of GOTHIC. These analyses allow for a more realistic estimate of void sizes on both the suction and discharge of the emergency core cooling system pumps. The licensee had not accepted these analyses at the time of this inspection.

  The inspectors discussed with NRR that the licensee had used these methods. The ultimate acceptability of these methods required further evaluation by NRR to: (1) better understand the acceptability of the application of the revised test results contained in WCAP-16631-NP to void assessment analysis; (2) better understand and evaluate the use of the simplified equation; and (3) assess potential generic implications. The licensee documented these outstanding issues in Condition Report 39943.

- The inspectors selectively reviewed applicable documents, including calculations, and engineering evaluations with respect to gas accumulation in the emergency core cooling systems. Specifically, the inspectors verified that these documents addressed venting requirements, aspects where pipes were normally void such as some spray piping inside containment, void control during maintenance activities, and the effect of debris on strainers in containment emergency sumps causing accumulation of gas under the upper elevation of strainers and the impact on the required net positive suction head.

- The inspectors conducted a walk down of selected regions of the emergency core cooling systems in sufficient detail to assess the licensee's walk downs. The inspectors completed a full containment spray system alignment as documented in Section 1R04. The inspectors also verified that the information obtained during the licensee's walkdown was consistent with the items identified during the inspectors' independent walk down.

- The inspectors verified that piping and instrumentation diagrams and isometric drawings that describe the residual heat removal and safety injection system configurations. The review of the selected portions of isometric drawings considered the following:

  1. High point vents were identified.

  2. High points without vents were recognizable.

  3. Other areas where gas could accumulate and potentially impact operability, such as at orifices in horizontal pipes, isolated branch lines, heat exchangers, improperly sloped piping, and under closed valves, were described in the drawings or in referenced documentation.

  4. Horizontal pipe centerline elevation deviations and pipe slopes in nominally horizontal lines that exceed specified criteria were identified.

  5. All pipes and fittings were clearly shown.

  6. The drawings were up to date with respect to recent hardware changes, and that any discrepancies between as-built configurations and the drawings were documented and entered into the corrective action program for resolution.

- The inspectors verified that the licensee had completed their walkdowns and selectively verified that the licensee identified discrepant conditions in their corrective action program and appropriately modified affected procedures and training documents. The inspectors determined that the licensee appropriately considered the differing gas intrusion mechanisms with one exception. The inspectors noted that the licensee failed to analyze whether vortexing would occur in their containment spray additive tank. The details of this issue are described in Section 4OA5.1.e of this report.

Testing. The inspectors reviewed selected surveillance, postmodification test, and postmaintenance test procedures and results implemented during power and shutdown operations to verify that the licensee had approved and used procedures that addressed gas accumulation and/or intrusion into the subject systems. This review included the verification of procedures used for conducting surveillances and determination of void volumes to ensure that the licensee satisfied the established void criteria with reasonable assurance until the next scheduled void surveillance. Also, the inspectors reviewed procedures used for filling and venting following conditions that may have introduced voids into the subject systems to verify that the procedures addressed testing for such voids and provided processes for their reduction or elimination. The inspectors observed the performance of the emergency core cooling system void surveillance as documented in Section 1R22.

Corrective Actions: The inspectors reviewed selected actions from the 2011 assessment review and sampled other corrective action program documents to assess how effectively the licensee addressed the issues in their corrective action program associated with Generic Letter 2008-01. In addition, the inspectors verified that the licensee implemented appropriate corrective actions for condition reports identified in the 9-month and supplemental responses. The inspectors determined that the licensee had initiated a large number of corrective actions in response to previous events at their facility. The inspectors determined that the licensee had effectively implemented the actions required by Generic Letter 2008-01.

Based on this review, the inspectors concluded that reasonable assurance exists the licensee will continue to implement the requirements of Generic Letter 2008-01 and will complete all outstanding items. This TI is closed.

1. Findings

Introduction. The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control” for the failure to translate the design basis into instructions, procedures, and drawings. The inspectors found that the licensee failed to assess whether vortexing occurred in the containment spray additive tank during a design-basis accident.

Description. The inspectors evaluated licensee activities related to evaluation of gas intrusion into their emergency core cooling systems. The inspectors questioned whether air entrainment in the containment spray system, as a result of vortexing in the containment spray additive tank, affected the ability of the containment spray system to remain full of water and meet the accident flow requirements. The licensee did not have a calculation to determine whether vortexing would occur in their containment spray additive tank at the required design flow rates. The licensee initiated Condition Report 38715 to document this deficiency; initiated actions to calculate the effects of vortexing in the containment spray additive tank during design basis flows; and established a Mode 3 restraint related to completing the calculation to ensure that containment spray would be operable as required.

The system used an eductor driven by discharge flow from both of the containment spray pumps to draw sodium hydroxide from the single chemical additive tank during a design-basis accident. Vacuum breakers allowed air into the tank as the liquid drained.

Calculation EN M-024, “Critical Submergence in Containment Spray Additive Tank (TEN01) to Avoid Vortex,” Revision 0, concluded that vortexing would not occur.

Analysis. Failure to implement design control measures to analyze whether containment spray piping remained full of water was a performance deficiency. This finding was more than minor because it affected the design control attribute of the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of the containment spray system to respond to initiating events and prevent undesirable consequences. Specifically, the inspectors had reasonable doubt on the capability of the containment spray system to properly inject because of vortexing in the containment spray additive tank. The inspectors performed the significance determination using Inspection Manual Chapter 0609.04. The finding was determined to be of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in loss of operability or functionality. Although the failure to have this calculation had existed since original construction, the inspectors determined this finding reflected current performance since the licensee was required to evaluate likelihood of tanks allowing gas intrusion into the emergency core cooling systems in response to Generic Letter 2008-01, “Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems.” Consequently, this finding had problem identification and resolution cross-cutting aspects associated with the corrective action program in that the licensee did not evaluate thoroughly the potential for gas intrusion from all possible tanks [P.1(c)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, requires, in part, that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program, was identified. Specifically, the design capability of the containment spray system requires that the system be full of water in order to achieve and maintain the design rate of flow. Contrary to the above, as of May 6, 2011, the licensee had not verified the adequacy of the design capability of the containment spray system to remain full of water through design review, calculation, or testing. Specifically, the licensee had not analyzed whether vortexing in the containment spray additive tank would affect system flow. The analysis demonstrated that no air should be entrained as a result of vortexing. The licensee documented this issue in Condition Report 38715. Because this finding was of very low safety significance and has been entered into the corrective action program, it is being treated as a noncited violation consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000482/2011003-10, “Failure to Analyze for Vortexing in Containment Spray Additive Tank.”

.2 (Closed) NRC TI 2515/183, Followup to the Fukushima Daiichi Nuclear Station Fuel Damage Event

a. Inspection Scope

The inspectors assessed the activities and actions taken by the licensee to assess its readiness to respond to an event similar to the Fukushima Daiichi nuclear plant fuel damage event. This included (1) an assessment of the licensee’s capability to mitigate conditions that may result from beyond design basis events, with a particular emphasis on strategies related to the spent fuel pool, as required by NRC Security Order Section B.5.b issued February 25, 2002, as committed to in severe accident management guidelines, and as required by 10 CFR 50.54(hh); (2) an assessment of the licensee's capability to mitigate station blackout conditions, as required by 10 CFR 50.63 and station design bases; (3) an assessment of the licensee's capability to mitigate internal and external flooding events, as required by station design bases; and (4) an assessment of the thoroughness of the walkdowns and inspections of important equipment needed to mitigate fire and flood events, which were performed by the licensee to identify any potential loss of function of this equipment during seismic events possible for the site.

b. Findings and Observations

NRC Inspection Report 05000482/2011008 (ML11133A354) documented detailed results of this inspection activity. Following issuance of the report, the inspectors conducted additional follow-up on the following seven selected issues.

1. Extensive damage mitigation guideline procedures specify that if the control room staff and field operators are compromised, then the shift security commander becomes the incident coordinator until an operator can be found. The inspectors identified that shift security commanders are not trained on reactor technology and mitigating systems, therefore it is not reasonable to assume they would have a sufficient knowledge base or decision making ability to direct technical response to an extensive damage situation. The licensee entered the issue into their corrective action program and is in the process of conducting additional procedural and technical training for security commanders.

   The inspectors reviewed licensee extensive damage mitigation guidelines in greater detail and compared them to the requirements of 10 CFR 50.54(hh)(2) as well as to the expectations outlined in the NRC Staff Guidance for Use in Achieving Satisfactory Compliance with February 25, 2002, Order Section B.5.b dated February 25, 2005, and determined that Wolf Creek's procedures meet agency expectations in that they direct security commanders to seek out persons with the best technical expertise available. This issue of concern is closed with no finding.

2. The licensee identified that extensive damage mitigation guidelines procedures to refill the refueling water storage tank are not viable because the connection point is not readily accessible. The licensee entered this issue into their corrective action program and is evaluating potential design changes to resolve this concern.

   The inspectors reviewed the applicable extensive damage mitigation attachments which direct refilling of the refueling water storage tank in greater detail and compared them to the requirements of 10 CFR 50.54(hh)(2) as well as to the expectations outlined in the NRC Staff Guidance for Use in Achieving Satisfactory Compliance with February 25, 2002, Order Section B.5.b, dated

February 25, 2005, and determined that these procedures do not meet regulatory requirements for compliance with Order 10 CFR 50.54(hh)(2) because the expectation element could not be effectively implemented using existing or readily available resources and because personnel safety concerns associated with the expectation element had not been addressed. There is no guidance as to how the connection is to be accessed, nor is the required equipment needed access and work safely at heights pre-staged in advance. This issue of concern is documented as a licensee identified violation in Section 4OA7.1 of this report.

3. The licensee identified that extensive damage mitigation guideline procedures require additional precautionary guidance to prevent excessive reactor coolant system depressurization which could compromise natural circulation core cooling. The licensee entered this issue into their corrective action program and is evaluating procedural enhancements to remedy this concern.

   The inspectors reviewed the applicable licensee extensive damage mitigation attachments which direct actions which can cool and depressurize the reactor coolant system and determined that this issue of concern was an enhancement only and not a violation of regulatory requirements. Since operators reviewing these procedures identified the same concerns and because the licensee has entered this issue in their corrective action program this issue of concern is closed with no finding.

4. The licensee identified that alternate power sources specified by extensive damage mitigation guidelines procedures are not properly staged in advance. Additional technical guidance on the configuration and use of these sources needs to be added to the extensive damage mitigation guidelines procedures. The licensee entered this issue into their corrective action program and is evaluating alternative equipment and procedural enhancements to resolve this concern.

   The inspectors reviewed the applicable licensee extensive damage mitigation attachments which direct the use of alternate dc sources in greater detail and compared them to the requirements of 10 CFR 50.54(hh)(2) as well as to the expectations outlined in the NRC Staff Guidance for Use in Achieving Satisfactory Compliance with February 25, 2002, Order Section B.5.b dated February 25, 2005, and determined that these procedures do not meet regulatory requirements for compliance with 10 CFR 50.54(hh)(2) because the expectation element could not be effectively implemented using existing or readily available resources. Specifically, the components are not pre-staged in advance. This issue of concern is documented as a licensee identified violation in Section 4OA7.1 of this report.

5. During walkdowns with the inspector, nuclear station operators failed to promptly locate certain station blackout emergency operating procedure components in the plant. The inspectors determined that this was due to inadequate training, lack of specific procedural guidance, and over-reliance on a computer database of equipment locations. The computer database would be unavailable during an

actual station blackout. The licensee agreed with this characterization and entered this issue into their corrective action program.

6. The inspectors determined that this issue of concern was a performance deficiency and a violation of Technical Specification 5.4.1.a which requires, in part, that specified station procedures be established, implemented, and maintained. The inspectors determined that all of the components operators failed to identify for local actions were backed up by components which would fail safe in a loss of ac power event and therefore did not have the potential to further complicate that event. The inspectors evaluated the issue using Inspection Manual Chapter 0612, Appendix B, "Issue Screening," and determined the failure to comply with Technical Specification 5.4.1.a constituted a violation of minor significance that is not subject to enforcement action in accordance with the NRC's Enforcement Policy. The inspectors also found that Wolf Creek had completed appropriate corrective actions in this area. This issue of concern is closed as an NRC identified minor violation.

7. The licensee identified that some fire protection equipment is not stored in seismic or tornado qualified locations. The licensee identified that the water supply pumps and piping used for fire protection and extensive damage mitigation guideline actions is not seismic or tornado qualified. The licensee also identified that equipment used to access underground diesel storage tanks is not seismic or tornado qualified; also the tanker truck used to refill the diesel-driven fire pump and fire truck is not parked in a seismic or tornado qualified building. The licensee entered these issues into their corrective action program.

8. The inspectors reviewed the requirements of 10 CFR 50.54(hh)(2) as well as to the expectations outlined in the NRC Staff Guidance for Use in Achieving Satisfactory Compliance with February 25, 2002, Order Section B.5.b, dated February 25, 2005, and determined that those regulatory requirements apply only to fire and explosion events, not to earthquakes and tornadoes. Because Wolf Creek identified this issue and entered into their corrective action program and because this issue of concern has no associated violation of regulatory requirements, it does not meet the criteria of a finding under the Inspection Manual Chapter 0612. This issue of concern is closed with no finding.

9. The condensate storage tank used in station blackout response and extensive damage mitigation guideline procedures is not seismic or tornado qualified. The licensee entered the issue into their corrective action program. The inspectors found that the safety-related source, from the essential service water system, would not be impacted. The inspectors reviewed applicable sections of Wolf Creek's USAR and determined that this issue is within the boundaries of Wolf Creek's NRC-approved design bases. This issue of concern is closed with no finding.

.3 <u>(Closed) NRC TI 2515/184, Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs)</u>

The inspectors reviewed the licensee's severe accident management guidelines (SAMGs), implemented as a voluntary industry initiative in the 1990's, to determine (1) whether the SAMGs were available and updated, (2) whether the licensee had procedures and processes in place to control and update its SAMGs, (3) the nature and extent of the licensee's training of personnel on the use of SAMGs, and (4) licensee personnel's familiarity with SAMG implementation.

The results of this review were provided to the NRC task force chartered by the Executive Director for Operations to conduct a near-term evaluation of the need for agency actions following the Fukushima Daiichi fuel damage event in Japan. Plant-specific results for Wolf Creek were provided as Enclosure 14 to a memorandum to the Chief, Reactor Inspection Branch, Division of Inspection and Regional Support, dated May 27, 2011 (ML111470264).

.4 <u>(Closed) TI 2515/172, Reactor Coolant System Dissimilar Metal Butt Welds</u>

a. <u>Inspection Scope</u>

Portions of TI 2515/172 were previously performed at Wolf Creek Nuclear Generating Station, during Refueling Outages 15, 16, and 17. The results of those inspections are documented in NRC Inspection Reports 05000482/2006005, 05000482/2008003, 05000482/2009005 and 05000482/2011003, respectively. Specific documents reviewed during this inspection are listed in the attachment. This unit has the following dissimilar metal butt welds:

COMPONENT ID	DESCRIPTION	MRP-139 CATEGORY	BASELINE EXAM
RV-301-121-A	Loop 1 Outlet Nozzle to Safe-End Weld	D	April 2005 RF14
RV-301-121-B	Loop 2 Outlet Nozzle to Safe-End Weld	D	April 2005 RF14
RV-301-121-C	Loop 3 Outlet Nozzle to Safe-End Weld	D	April 2005 RF14
RV-301-121-D	Loop 4 Outlet Nozzle to Safe-End Weld	D	April 2005 RF14

COMPONENT ID	DESCRIPTION	MRP-139 CATEGORY	BASELINE EXAM
RV-302-121-A	Loop 1 Inlet Nozzle to Safe-End Weld	E	April 2005 RF14
RV-302-121-B	Loop 2 Inlet Nozzle to Safe-End Weld	E	April 2005 RF14
RV-302-121-C	Loop 3 Inlet Nozzle to Safe-End Weld	E	April 2005 RF14
RV-302-121-D	Loop 4 Inlet Nozzle to Safe-End Weld	E	April 2005 RF14
TBB03-1-W /MW7090-WOL-DM	Pressurizer Surge Nozzle to Safe-End Weld	F	October 2006 RF15
TBB03-2-W /MW7089-WOL-DM	Pressurizer Spray Nozzle to Safe-End Weld	B	October 2006 RF15
TBB03-3-A-W /MW7086-WOL-DM	Pressurizer Safety Nozzle A to Safe-End Weld	B	October 2006 RF15
TBB03-3-B-W /MW7087-WOL-DM	Pressurizer Safety Nozzle B to Safe-End Weld	B	October 2006 RF15
TBB03-3-C-W /MW7088-WOL-DM	Pressurizer Safety Nozzle C to Safe-End Weld	F	October 2006 RF15
TBB03-4-W /MW7085-WOL-DM	Pressurizer Relief Nozzle to	F	October 2006 RF15

COMPONENT ID	DESCRIPTION	MRP-139 CATEGORY	BASELINE EXAM
	Safe-End Weld		

1. Licensee's Implementation of the Materials Reliability Program (MRP-139) Baseline Inspections (03.01)

   The inspectors reviewed records of structural weld overlays and nondestructive examination activities associated with the licensee's pressurizer and hot leg structural weld overlay mitigation effort. The baseline inspections of the pressurizer dissimilar metal butt welds were completed as noted in the table above. The pressurizer dissimilar metal butt welds had full structural weld overlay applied in Refueling Outage 15. The first Component ID in the preceding table was the designation prior to the overlay; the latter Component ID is the current weld designation (after overlay).

   The licensee requested the deviations from the MRP-139 baseline inspection requirements. These locations are now examined in accordance with the approved alternative of relief request I3R-05. The licensee did not take any other deviations from the baseline inspection requirements of MRP-139, and all other applicable dissimilar metal butt welds were scheduled in accordance with MRP-139 guidelines.

2. Volumetric Examinations (03.02)

   The results of these inspections are documented in NRC Inspection Reports 05000482/2006005, 05000482/2008003, and 05000482/2009005.

3. Weld Overlays (03.03)

   Only the pressurizer nozzles have been mitigated. The mitigation type was full structural weld overlay applied in Refueling Outage 15. A pre-service exam in accordance with relief request I3R-05 was performed. An inservice exam on the MRP-139, category F welds was performed in Refueling Outage 16 in accordance with I3R-05. This examination also falls within the guidelines of MRP-139 for category F welds.

4. Mechanical Stress Improvement (03.04)

   The licensee did not employ a mechanical stress improvement process.

5. Inservice Inspection Program (03.05)

   The licensee has prepared an MRP-139 inservice inspection program. All the welds in the MRP-139 inservice inspection program are appropriately categorized in accordance with MRP-139. The inservice inspection frequencies are consistent with the inservice inspection frequencies called for by MRP-139.

b. Findings

No findings were identified.

4OA6 Meetings

Exit Meeting Summary

On April 1, 2011, the inspectors presented the inservice inspection results to Mr. S. Hedges, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors telephonically re-exited with Mr. Hedges, Site Vice President, and other members of the licensee's staff on June 16, 2011. The inspectors acknowledged review of proprietary material during the inspection which was returned to the licensee.

On April 5, 2011, the Deputy Director of the Division of Reactor Projects conducted a regulatory performance meeting in conjunction with the public annual assessment meeting with Mr. M. Sunseri, President and Chief Executive Officer, and other members of the licensee staff to review the corrective actions related to the previously White performance indicators for unplanned scrams per 7000 critical hours, unplanned scrams with complications, and safety system functional failures.

On May 6, 2011, the inspectors presented the inspection results to Mr. M. Sunseri, President and Chief Executive Officer, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

On June 8, 2011, the inspector communicated the results of the in-office inspection of changes to the licensee's emergency plan to Mr. T. East, Superintendent, Emergency Planning, and to Mr. W. Muilenburg, Licensing Engineer, of the licensee's staff. The licensee acknowledged the issues presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On July 13, 2011, the inspectors presented the inspection results to Mr. S. Hedges, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. Although proprietary information was used during the inspection, it was returned to the licensee or destroyed.

4OA7 Licensee-Identified Violations

The following violations of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section 2.3.2 of the NRC Enforcement Policy for being dispositioned as noncited violations.

.1 Title 10 CFR 50.54(hh)(2)(ii) states: "Each licensee shall develop and implement guidance and strategies intended to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities under the circumstances associated with loss of large areas of the plant due to explosions or fire, to include strategies in the following area of operations to mitigate fuel damage." On April 13, 2011, while performing procedure reviews as part of industry-wide self-assessments in response to the core damage events at Fukushima Daiichi, Wolf Creek engineers identified two instances of mitigating strategy procedures which did not contain sufficient information to accomplish those strategies successfully. The first example was the ability to refill the refueling water storage tank, and the second example involved flashing the diesel generator field using alternate dc sources. These issues were documented in the licensee's corrective action program as Condition Report 37374. The inspectors evaluated these findings under Inspection Manual Chapter 0609, Appendix L, and determined these findings to be of very low safety significance because the findings did not involve unrecoverable unavailability of multiple mitigating strategies such that spent fuel pool cooling, injection to the reactor vessel, or injection to steam generators cannot occur, or unrecoverable unavailability of on-site, self-powered, portable pumping capability, or substantial inability to perform command and control enhancements.

.2 Title 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," requires, in part, that a test program be established to assure that all testing required to demonstrate that structures, systems and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in the applicable design documents. On May 13, 2011, Wolf Creek identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XI, test control for stroking residual heat removal containment sump valve 8811B prior to its as-found diagnostic test. Wolf Creek stroked the valve for a clearance order and as such, preconditioned the valve prior to its test. Plant computer data from this stroke, data from the diagnostic stroke, and valve disassembly showed no deficiencies. Using Inspection Manual Chapter 0609.04, the inspectors determined the finding to be of very low safety significance because it was confirmed not to result in the loss of operability or functionality. This issue is captured in Condition Report 37244.

SUPPLEMENTAL INFORMATION
KEY POINTS OF CONTACT

Licensee Personnel

G. Beckett, Superintendent, Support Engineering
P. Bedgood, Manager, Radiation Protection
R. Evenson, Requalification Program Supervisor
J. Harris, System Engineer
S. Hedges, Site Vice President
S. Henry, Operations Manager
R. Hobby, Licensing Engineer
D. Hooper, Supervisor, Regulatory Affairs
T. Just, Senior Technician, Chemistry
J. Keim, Support Engineering Supervisor
S. Koenig, Manager, Corrective Actions
M. McMullen, Technician, Engineering
C. Medency, Supervisor, Radiation Protection
W. Muilenburg, Licensing Engineer
R. Murray, Simulator Supervisor
B. Norton, Manager, Integrated Plant Scheduling
J. Pankaskie, Engineering Supervisor
G. Pendergrass, Director of Engineering
L. Rockers, Licensing Engineer
G. Sen, Regulatory Affairs Manager
R. Smith, Plant Manager
M. Sunseri, President and Chief Executive Officer
J. Truelove, Supervisor, Chemistry
J. Weeks, System Engineer
M. Westman, Training Manager

NRC Personnel

D. Loveless, Senior Reactor Analyst

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000482/2011003-07	VIO	Failure to Correct Procedure for Opening Main Steam Isolation Valves (EA-11-149) (Section 4OA3.1)

Opened and Closed

05000485/2011003-01	NCV	No Procedure for Debris in Transformed and Tank Yards Prior to Severe Weather (Section 1R01)
05000482/2011003-02	NCV	Failure to Properly Establish Clearance Order Boundary Isolation Resulting in Loss of Component Cooling Water Inventory (Section 1R04)
05000482/2011003-03	NCV	Failure to Assure Fillet Weld Met Size Requirements on Train B Charging Header Vent Line (Section 1R08.1)
05000482/2011003-04	NCV	Failure to Assure Separation of Stainless Steel and Carbon Steel Grinding and Cutting Equipment (Section 1R08.1)
05000482/2011003-05	NCV	Failure to Assure Configuration Control of Safety-Related Systems (Section 1R08.3)
05000482/2011003-06	FIN	Inadequate Acceptance Criteria for Postmaintenance Testing of the Startup Feedwater Pump (Section 1R19)
05000482/2011003-08	NCV	Failure to Maintain Reactor Coolant System Pressure Below Relief Valve Setpoint (Section 4OA3.2)
05000482/2011003-09	NCV	Inadequate Fire Watch Defeats Halon Fire Suppression in Vital Switchgear Rooms During Fire (Section 4OA3.3)
05000482/2011003-10	NCV	Failure to Analyze for Vortexing in Containment Spray Additive Tank (Section 4OA5.1)
05000482-2515/177	TI	Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems (NRC Generic Letter 2008-01) (Section 4OA5.1)

Closed

05000482/2006-003-00	LER	Indications Discovered on Pressurizer during Preplanned In-service Inspections (Section 4OA3.4)
05000482/2010006-05	VIO	Notice Of Violation EA-10-160, Failure to correct NRC identified NCV. Apparent Cause Evaluation vice Root Cause Evaluation for Essential Service Water (Section 4OA3.5)

05000482-2515/183	TI	Followup to the Fukushima Daiichi Nuclear Station Fuel Damage Event (Section 4OA5.2)
05000482-2515/184	TI	NRC Temporary Instruction 2515/184, Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs) (Section 4OA5.3)
05000482/2515/172	TI	Temporary Instruction 2515/172, Reactor Coolant System Dissimilar Metal Butt Welds (Section 4OA5.4)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

PROCEDURES

NUMBER	TITLE	REVISION
OFN AF-025	Unit Limitations	32
Ai 14-008	Severe Weather	9A
AP 21C-001	Wolf Creek Substation	11A
OP1450001	Outage Risk Management	000
APF 22B-001-05	Shutdown Risk Assessment	0
APF 22B-001-10	Shutdown Safety function Status and Assessment Summary	1

CONDITION REPORTS

00040573 00040351

WORK ORDERS

11-344384-000

Section 1R04: Equipment Alignment

PROCEDURES

NUMBER	TITLE	REVISION / DATE
CKL EN-120	Containment Spray System Lineup	15A
AP 21E-001	Clearance Orders	27
SYS EN-400	Containment Spray System Fill and Vent	11
STN EN-003A	Containment Spray Train A Void Monitoring and Venting	3
STN EN-003B	Containment Spray Train B Void Monitoring and Venting	3
CKL HB-122	Liquid Waste Evaporator Normal Lineup	15
D-HB-N-029	Clearance Order Liquid Radwaste System	March 30, 2011
Standing Order 1	Valve Setup and Operation	43
M-12HB01	Piping and Instrumentation Diagram Liquid Radwaste System	19

CONDITION REPORTS

13599	25918	28343	28771	32378
33060	33063	33064	34505	

WORK ORDERS

93-100775-001 94-100830-001 03-257175-003

DRAWINGS

NUMBER	TITLE	REVISION
M-13EN03	Piping Orthographic Containment Spray System Reactor Building “A” & “B” Trains	3
M-13EN05	Piping Orthographic Containment Spray System Reactor Building “A” & “B” Trains	2

M-12EN01	Piping and Instrumentation Diagram Containment Spray System	12
M-13EN01	Piping Isometric Containment Spray System Auxiliary Building “A” Train	7
M-13EN01	Piping Isometric Containment Spray System Auxiliary Building “B” Train	7
M-13EN06	Small Piping Isometric Containment Spray System Auxiliary Building	0

Section 1R05: Fire Protection

PROCEDURES

NUMBER	TITLE	REVISION
E-1F9905	Fire Hazard Analysis	0
AP 10-106	Fire Preplans	7
FPPM-001	Auxiliary Bldg El. 1974’	2

DRAWINGS

NUMBER	TITLE	REVISION
M-663-00017	Penetration Typical Details, Attachment B	W21
AP 10-106	Fire Preplans	7
FPPM-001	Auxiliary Bldg El. 1974’	2

CONDITION REPORTS

15073

Section 1R08: Inservice Inspection Activities

PROCEDURES

NUMBER	TITLE	REVISION
AI 16F-001	Evaluation Of Boric Acid Leakage	5A
AI 16F-002	Boric Acid Leakage Management	7
AI 28A-010	Screening Condition Reports	8A
AP 16F-001	Boric Acid Corrosion Control Program	5A/6A
AP 28-100	Condition Reports	13
AP 29A-003	Steam Generator Management	14
APF 28D-001	Self-Assessment Process	12
I-ENG-023	Steam Generator Data Analysis Guidelines	11
MRS 2.4.2 GEN-35	Eddy Current Inspection of Preservice and Inservice Heat Exchanger Tubing	14
PDI-ISI-254-SE-NB	Remote Inservice Examination of Reactor Vessel Nozzle to Safe End, Nozzle to Pipe, and Safe End to Pipe Welds Using the Nozzle Scanner	1
PDI-UT-2	Generic Procedure for the Ultrasonic Testing of Austenitic Pipe Welds	E
PDI-UT-3	Generic Procedure for Ultrasonic Through Wall Sizing in Pipe Welds	D
PDI-UT-6	Generic Procedure for the Ultrasonic Testing of Reactor Pressure Vessel Welds	F
PDI-UT-8	Generic Procedure for the Ultrasonic Examination of Weld Overlaid Similar and Dissimilar Metal Welds	F
QCP-20-501	PT (Penetrant Testing)	8

QCP-20-502	Magnetic Particle Examination AC/DC Yoke and AC Coil Techniques	8B
QCP-20-503	UT Thickness-Wall Thin	3
QCP-20-504	UT For Flaw Detection	5
QCP-20-508	Radiographic Examination of Welds	4A
QCP-20-510	UT Instrument Linearity	3
QCP-20-511	RT of AWS Groove Welds	1B
QCP-20-514	ET Testing	5B
QCP-20-516	PT/NON-STD Temp	05
QCP-20-517	RT Wall Thinning	2A
QCP-20-520	Pressure Test Examination	8B
QCP-20-521	UT Profile and Plotting	1B
QCP-20-522	UT Ferritic Pipe Welds	1B
QCP-20-523	UT Austenitic Pipe Welds	1B
QCP-20-527	UT- Soldering	1
QCP-20-540	VT-1 Visual Exam	0C
QCP-20-541	VT-3 Visual Exam	2A
QCP-20-543	Fluorescent Dye PT Exam	1
QCP-20-600	Visual Examination Of ASME Welds	9A
SG-CDME-10-8	Wolf Creek Steam Generator Secondary Side Condition Monitoring Assessment and Operational Assessment For Fuel Cycle and Refueling Outage 18, February 2011	0
SG-SGMP-09-23	Wolf Creek, RF18 Condition Monitoring Assessment and Operational Assessment, November 2009	2
SG-SGMP-10-30	Wolf Creek, RF18 Steam Generator Degradation Assessment, March 2011	1

STN PE-370	Foreign Object Search and Retrieval and Secondary Side Inspections	11
STN PE-040D	RCS Pressure Boundary Integrity Walkdown	3
STS PE-022	Steam Generator Tube Inspection	18
STS PE-040E	RPV HEAD VISUAL INSPECTION	2
UT-2	Ultrasonic Examination of Vessel Welds and Adjacent Base Metal	28
UT-95	Ultrasonic Examination of Austenitic Piping Welds	3
WCRE-24	WESDYNE Year 2011 Reactor Vessel Nozzle Safe-end Examinations Program Plan	0
WCAL-002	Pulser/Receiver Linearity Procedure	10
WDI-CAL-102	Calibration Procedure for PCI Eddy Current Card	1
WDI-EQPT-1021	Installation and Removal of the WESDYNE Nozzle Scanner (SQUID)	5
WDI-EQPT-1022	Reactor Vessel Nozzle Scanner Setup and Checkout	4
WDI-STD-146	ET Examination of Reactor Vessel Pipe Welds Inside Surface	11

CONDITION REPORTS

21975	28474	21976	28601	22027	28771
22128	28847	22280	28848	22391	28959
23173	28967	23251	28978	23455	29128
23459	29197	23867	29237	24020	29612
24077	29801	24230	30023	24336	30067
24339	30210	24469	30899	24658	31003
24659	31366	24661	31742	24662	31763
24663	31765	24665	31766	24676	31779
24681	31799	24857	31808	24893	31865
25095	32035	25173	32115	25196	32117
25224	32203	25228	32204	25268	32298
25361	32559	25377	32412	25394	32646
25495	32648	25643	32842	25871	33225
26354	33355	26358	33575	27193	33581

27472	33600	27650	33603	27892	33684
28050	33686	28144	33688	28258	33690
28322	33689	28386	35793	28403	

WORK ORDERS

08-310289	10-326485	10-324683	10-326486	10-325740
10-324621	09-320607	10-325747	10-325738	10-326483
10-325742				

MISCELLANEOUS

NUMBER	TITLE	REVISION / DATE
	2010 3rd Quarter Outside Containment BACCP Monitoring Walk-down	
	Boric Acid Leakage Screening/Evaluation for Normal Train B Charging Pump (PBG04)	May 5, 2010
	Boric Acid Leakage Screening/Evaluation for Reactor Coolant Pump “A” (PBB01A)	March 5, 2010
	Boric Acid Leakage Screening/Evaluation for Accumulator Tank C Discharge Check Valve (EP8956C)	October 19, 2009
	Boric Acid Leakage Screening/Evaluation for RHR HX A/CVCS To SI Pump A Upstream Isolation (EMHV8924)	October 20, 2009
	Boric Acid Leakage Screening/Evaluation for SI Pump B Suction Check Valve (EM8926B)	July 8, 2010
	Boric Acid Leakage Screening/Evaluation for SI Pump A Suction Check Valve (EM8926A)	September 8, 2010
	Boric Acid Leakage Screening/Evaluation for RCS Loop 3 Steam Generator Primary Side Downstream Drain (BBV0476)	November 20, 2009
	Boric Acid Leakage Screening/Evaluation for RCS Loop 1 Steam Generator Primary Side Downstream Drain (BBV0474)	March 5, 2010

	Change Package # 012869, Installation of Vent Valves in the Chemical and Volume Control System (BG), the Residual Heat Removal (EJ), and the High-Pressure Coolant Injection System (EM)	3
	Technical Report No. 11-2039-TR-001, Failure Analysis of Socket Weld on a Vent Valve Assembly from the CVCS	March 2011
	S/G Eddy Current Calibrated Equipment List	October 16, 2009
	Steam Generator data Analysis Desktop Instruction	4
	RF 18 Steam Generator data Analysis Desktop Instruction	0
	SGAMP Self Assessment, Steam Generator Asset Management Program	October 17, 2008
	Wolf Creek RF 17 Fall 2009 Steam Generator Secondary Side Visual Inspection Recommendations	August 17, 2009
	Wolf Creek RF17 Condition Monitoring Assessment and Operational Assessment	November 2009
APF 28D-001-02	Self Assessment Report SEL 04-038, “Steam Generator Program”	4
APF 29A-003-001	Secondary Chemistry Wet Layup Initial Monitoring Frequency	2
ET 09-0016	Revision to Technical Specifications 5.5.9, "Steam Generator (SG) Program," and TS 5.6.10, "Steam Generator Tube Inspection Report”, for a Permanent Alternate Repair Criterion	June 2, 2009
ET-09-0025	Docket No. 50-482: Revision to Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program," and TS 5.6.10, "Steam Generator Tube Inspection Report"	September 15, 2009
ET-10-0030	Revision to Technical Specifications 5.5.9, “Steam Generator (SG) Program,” and TS (Technical Specifications) 5.6.10, “Steam Generator Tube Inspection Report,” for a Temporary Alternate Repair Criterion	November 30, 2010

I3R-01	Wolf Creek Generating Station - Third 10-Year Interval Inservice Inspection Program Relief Request I3R-01 (TAC NO. MD0297)	February 21, 2007
I3R-05	Wolf Creek Generating Station - Authorization Of Relief Request I3R-05, Alternatives To Structural Weld Overlay Requirements (TAC NO. MD1813)	July 19, 2007
13R-06	Wolf Creek Generating Station -Relief Request 13R-06, Alternative To The Examination Requirements Of ASME Code, Section XI For Class 1 Piping Welds Examined From The Inside Of The Reactor Vessel (TAC NO. MD9658)	July 23, 2009
Docket No. 50-482	10 CFR 50.55a Request 13R-06, Alternative to the Examination Requirements of ASME Section XI for Class 1 Piping Welds Examined from the Inside of the Reactor Vessel	September 16, 2008
Docket No. 50-482	Wolf Creek Nuclear Operating Corporation's Response to Request for Additional Information Regarding 10 CFR 50.55a Request 13R-06	April 23 , 2009
ET 05-0014	10 CFR 50.55a Request Number 13R-03 for the Third Ten-Year Interval Inservice Inspection (ISI) Program - Request for Relief to Allow Use of Alternate Requirements for Snubber Inspection and Testing	September 28 2005
ET 06-0042	Wolf Creek Nuclear Operating Corporation's Response to the September 20, 2006 NRC Request for Additional Information Regarding 10 CFR 50.55a Request 13R-05	September 27, 2006
ET 06-0044	Wolf Creek Nuclear Operating Corporation's Revised Commitment Regarding 10 CFR 50.55a Request 13R-05	October 2, 2006
ET 06-001 0	Inservice Inspection Program Plan for the Third Ten-Year Interval and 10 CFR 50.55a Requests 13R-01, 13R-02, and 13R-04	March 2, 2006
ET 06-0021	10 CFR 50.55a Request 13R-05, Installation and Examination of Full Structural Weld Overlays for Repairing/Mitigating Pressurizer Nozzle-to-Safe End Dissimilar Metal Welds and Adjacent Safe End-to-Piping Stainless Steel Welds	May 19, 2006

ET 06-0031	Wolf Creek Nuclear Operating Corporation's Response to Request for Additional Information Regarding I 0 CFR 50.55a Request I3R-05 and Submittal of Revision 1 to 10 CFR 50.55a Request 13R-05	August 4,2006
ET 06-0043	Wolf Creek Nuclear Operating Corporation's Response to NRC Request for Additional Information Regarding 10 CFR 50.55a Request 13R-01	October 5,2006
ET 06-0058	Wolf Creek Nuclear Operating Corporation's Response to the Second NRC Request for Additional Information Regarding 10 CFR 50.55a Request 13R-01	December 20, 2006
MRS-TRC-2087	Use of Appendix H and I Qualified Techniques at Wolf Creek RF18 April 2011 S/G Inspection	0
SAP-+PT-09	Steam Generator Eddy Current Inspection Multi-Frequency Eddy Current Parameters	0
SAP-+PTUB-09	Steam Generator Eddy Current Inspection Multi-Frequency Eddy Current Parameters	0
SAP-01-09	Steam Generator Eddy Current Inspection Multi-Frequency Eddy Current Parameters	0
SAP-02-09	Steam Generator Eddy Current Inspection Multi-Frequency Eddy Current Parameters	0
SAP-03-09	Steam Generator Eddy Current Inspection Multi-Frequency Eddy Current Parameters	0
SAP-04-09	Steam Generator Eddy Current Inspection Multi-Frequency Eddy Current Parameters	0
SAP-05-09	Steam Generator Eddy Current Inspection Multi-Frequency Eddy Current Parameters	0
SAP-06-09	Steam Generator Eddy Current Inspection Multi-Frequency Eddy Current Parameters	0
SAP-07-09	Steam Generator Eddy Current Inspection Multi-Frequency Eddy Current Parameters	0
SAP-08-09	Steam Generator Eddy Current Inspection Multi-Frequency Eddy Current Parameters	0
SAP-09-09	Steam Generator Eddy Current Inspection Multi-Frequency Eddy Current Parameters	0

SAP-10-09	Steam Generator Eddy Current Inspection Multi-Frequency Eddy Current Parameters	0
SAP-11-09	Steam Generator Eddy Current Inspection Multi-Frequency Eddy Current Parameters	0
SAP-12-09	Steam Generator Eddy Current Inspection Multi-Frequency Eddy Current Parameters	0
SAP-BOB-09	Steam Generator Eddy Current Inspection Multi-Frequency Eddy Current Parameters	0
SAP-DELTA-09	Steam Generator Eddy Current Inspection Multi-Frequency Eddy Current Parameters	0
SAP-GHENT-09	Steam Generator Eddy Current Inspection Multi-Frequency Eddy Current Parameters	0
SEL 04-038	Steam Generator Program	4
SG-CDME-08-15	Wolf Creek RF16 Condition Monitoring Assessment and Operational Assessment, April 2008	1
SG-CDME-09-1	Wolf Creek Steam Generator Secondary Side Condition Monitoring and Operational Assessment for Fuel Cycle and Refueling Outage 17	0
SG-SGMP-09-9	Steam Generator Degradation Assessment for Wolf Creek, RF17 Refueling Outage, October 2009	0
SEL 09-151	EPRI-WRTC/Utility Welding Program Best Practices	
	Visual Examination for Leakage of PWR Reactor Head Penetrations	2
WCRE-15	Program Plan For Management Of Alloy 600 Components And Alloy 82/182 Welds	3

Section 1R11: Licensed Operator Requalification Program

NUMBER	TITLE	REVISION
LR5002026	Inadvertent Safety Injection Lab	3

Section 1R12: Maintenance Effectiveness

MISCELLANEOUS

NUMBER	TITLE
GK-01	Final Scope Evaluation, System GK, Control Building HVAC System

CONDITION REPORTS

00035992	00027105	00026250	00028792	00027228
00026251	00034299	00026250		

Section 1R13: Maintenance Risk Assessment and Emergent Work Controls

MISCELLANEOUS

NUMBER	TITLE	REVISION
EDI 23M-050	Engineering Desktop Instruction Monitoring Performance to Criteria and Goals	3
MPE GK-003	Control Room and Class 1E A/C Units Preventive Maintenance Activity	3
EDI 23M-050	Engineering Desktop Instruction Monitoring Performance to Criteria and Goals	3

WORK ORDERS

10-330270-000	10-330269-000	10-330270-000	10-330269-000

Section 1R15: Operability Evaluations

NUMBER	TITLE	REVISION
ALR 501	Standby Diesel Engine System Control Panel KJ-121	13, 14 and 14A

AP 26C-004	Operability Determination and Functionality Assessment	23
OE KJ-10-001	Emergency Diesel Generators KKJ01A and KKJ01B	0

MISCELLANEOUS

NUMBER	TITLE	REVISION / DATE
EDI 23M-050	Engineering Desktop Instruction Monitoring Performance to Criteria and Goals	3
	Final Scope Evaluation, System GK, Control Building HVAC System	
MPE GK-003	Control Room and Class 1E A/C Units Preventive Maintenance Activity	3
EDI 23M-050	Engineering Desktop Instruction Monitoring Performance to Criteria and Goals	3
GK	Final Scope Evaluation - Control Building HVAC System	
	LER 22011-003-00, Diesel Generator Declared Inoperable Due to Inadequate Reinstallation of Pipe connection Resulting in Excessive Governor Oil Coolant Leak	May 12, 2011
	A-EDG Governor Heat Exchanger Water Leak	
9.5-16	USAR	19
2011-1027-0	Training Needs Analysis	
	Operations Requalification Cycle 11-01 Week 0 to Week 6 Schedule	
OP1336001	Plant Changes	0

M-018-00110-W13	Electrical Schematic Engine Guide Panel KJ121	
E-13KJ02	Schematic Diagram Diesel Generator KKJ01A Annunciator and Miscellaneous Circuits	7
M-12EF01	Piping & Instrumentation Diagram Essential Service Water System	57
M-12EF02	Piping & Instrumentation Diagram Essential Service Water System	26
M-K2EF03	Piping & Instrumentation Diagram Essential Service Water System	10
M-13EF07(Q)	Piping Isometric Essential Service WTR.Sys.Control Bldg Cooler(A&B) Train Supply & Return	1
M-13EF08	Piping Isometric Essential Service Wtr,-Diesel Generator Bldg.	01

CONDITION REPORTS

00034661 00038229

REPORTABILITY EVALUATION REPORT

2011-037

Section 1R18: Plant Modifications

NUMBER	TITLE	REVISION
10-017-EG	Temporary Cooling of CCW Radwaste Loads	0

CONDITION REPORTS

00035262

Section 1R19: Postmaintenance Testing

PROCEDURES

NUMBER	TITLE	REVISION
SYS EG-205	CCW Flow Adjustment to Reactor Coolant Pumps, Seal Water Heat Exchanger, and Excess Letdown Heat Exchanger	9
STS AE-209	Main Feed Reg Valve Bypass Valve Inservice Valve Test	2
STN AE-001	Startup Main Feedwater Pump Operational Test	0A
STN AC-007	Turbine Overspeed Trip Set	26
AP 16E-002	Post Maintenance Testing Development	9C

WORK ORDERS

37698	38443	29128	34806	39145
34434	34500	36164		

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION / DATE
	Control Room Turnover Checklist-Day Shift	March 16, 2011
APF 30E-004-01	Main Feedwater System	2
BD-EMG FR-h1	Response to Loss of Secondary Heat Sink	10
	FWIS and Reactor Trip on Low S/G LevelCR 29128 Root Cause Evaluation	

E-0099	Cable Sheath Grounding and Termination Data	7
KD-7496	One Line Diagram	40

CONDITION REPORTS

0037698	00025817	00038443

WORK ORDERS

11-337610-000	11-337610-001	11-337610-002	11-337610-003	11-337610-004
11-337610-005	11-337610-006	11-337610-007	09-322525-000	10-335457-001
11-337610-004	11-337610-005	11-337610-006	11-337610-007	11-337610-000
11-337610-001	11-337610-002	11-337610-003		

Section 1R20: Refueling and Other Outage Activities

PROCEDURES

NUMBER	TITLE	REVISION
FHP 02-007A	Reactor Vessel Closure head Removal	10
SYS BB-215	RCS Drain Down with Fuel in Reactor	28
STS IC-439	Channel Calibration NIS Post Accident Monitoring N60	3A
GEN 00-002	Cold Shutdown to Hot Standby	73
STN EJ-002	Containment Inspection	17

DRAWINGS

NUMBER	TITLE	REVISION
C-OL2901(Q)	Reactor Building Line Plate Floor Details, SHT-1	7

C-OS2919(Q)	Reactor Building Incore Instrumentation Tube Supports and Platforms	8
C-OL2914(Q)	Reactor Building Liner Place Floor Details-Sheet-3	4

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION / DATE
11-2039-L-001	ALTRAN SOLUTIONS	April 13, 2011
Refuel 18, No. 15	The Daily Howl	April 2, 2011
Refuel 18, No. 17	The Daily Howl	April 4, 2011
Information Notice 2008-20	Failure of Motor Operated Value Actuator Motors with Magnesium Allow Rotors	December 8, 2008
MS-02	Piping Class Sheets	53
	Evaluation of Interim Operation	0
	ALARA Planning Survey	
	RF18 High Impact Teams/Major Projects	
	Letter NE 11-0009, dated February 28, 2011, from W. H. Ketchum To R. A. Smith and R. E. Kopecky	
AP 22B-001	Outage Risk Management	13

CONDITION REPORTS

00029149	00029322	00030371	00030371	00032254
00033358	00033698	00033699	00033715	00033716
00034068	00034349	00035261	00035262	00035304
00035314	00035419	00035426	00035516	00035533

00035535	00035537	00035539	00035540	00035540
00035541	00035542	00035544	00035545	00035546
00035547	00035548	00035549	00035550	00035551
00035552	00035553	00035554	00035555	00035556
00035557	00035558	00035559	00035560	00035573
00035614	00035615	00035617	00035619	00035620
00035621	00035622	00035623	00035624	00035625
00035626	00035627	00035628	00035629	00035630
00035632	00035663	00035714	00035963	00035965
00035987	00036031	00036032	00036106	00036186
00036272	00036292	00036300	00036492	00036518
00036798	00036799	00036857	00036876	00036880
00036881	00036957	00036966	00036988	00037110
00037289	00037615	00037909	00038083	00038086
00038113	00038333	00038517	00038680	00039099
00039283	2007-000299	00035429	00039721	

REPORTABILITY EVALUATION REQUESTS

2011-047	2011-102	2011-103	2011-113	00037048
2011-059	2011-057	2011-046	2011-056	

WORK ORDERS

10-324324-000	10-328967-001	10-324322-000	10-339212-003

Section 1R22: Surveillance Testing

PROCEDURES

NUMBER	TITLE	REVISION
STS BG-002A	Train A ECCS System Vent for Mode 4	10
STS BG-007A	ECCS Valve Check and Train A and Common Void Monitoring and Venting	5

Section 1R22: Surveillance Testing

PROCEDURES

NUMBER	TITLE	REVISION
STS BG-007B	ECCS Train B Void Monitoring and Venting	5
SYS EM-410	Fill and Vent of Safety Injection System After Maintenance	17 and 18
SYS EJ-110	RHR System Fill and Vent Including Initial RCS Fill	56, 57, 59 and 60
STS IC-211B	Actuation Logic Test Train B Solid State Protection System	35
STS PE-018	Containment Integrated Leakage Rate Test	9
AP 21-004	Operator Response Time Program	2
STN TCA-001	Manual Time Critical Action Timing	3
SYS GP-519	CILRT-EN System	2
AP 29E-001	Program Plant for Containment Leakage Measurement	13
AI 21-016	Operator Time Critical Actions Validation	2
STS KJ-001B	Integrated Diesel Generator and Safeguards Actuation Testing Train B	42A
STS IC-615B	Slave Relay Test K615 Train B Safety Injection	27

WORK ORDERS

10-326512-001 09-322158-001

CONDITION REPORTS

00037110	00037244	30302	33443	39083

39081

REPORTABILITY EVALUATION REQUESTS

2011-094

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION
AN-99-025	Steam Generator Tube Rupture Overfill Analysis with Revised Operator Action Times	1
AIF 21-016-02	Time Verification Form	0A
EJHV8811B	Analysis Print	
EJHV8811B	Refuel XVIII Preparation Package	

Section 4OA2: Identification and Resolution of Problems

PROCEDURES

NUMBER	TITLE	REVISION
KMS-4	Mechanical Standard	2

VENDOR DOCUMENTS

NUMBER	TITLE	REVISION
TB-68-2	Tensile Strength of Threaded Insert Assembly	2

CONDITION REPORTS

38321

WORK ORDERS

11-339714-000 11-337546-000

Section 4OA3: Event Follow-Up

PROCEDURES

NUMBER	TITLE	REVISION
BD-EMG ES-03	SI Termination	10A
SYS AE-200	Feedwater Preheating During Plant Startup and Shutdown	29 and 30
AP 15C-002	Procedure Use and Adherence33	
EMG E-0	Reactor Trip or Safety Injection	25
EMG ES-03	SI Terminations	18
SYS AB-120	Main Steam and Steam Dump Startup and Operation	27
SYS PN-200	Energizing and Deenergizing Inverters PN09 and PN10	11
ALR KC-888	Fire Protection Panel KC-008 Alarm Response	18A
AP 10-104	Breach Authorization	24A
SYS GK-200	Inoperable Class IE A/C Unit	21A
AP-10-103	Fire Protection Impairment Control	23A
AP 21D-003	Control of Information Tagging	15B
SYS BG-120	Chemical and Volume Control System	42
GEN 00-006	Hot Standby to Cold Shutdown	76
AP 21-001	Conduct of Operation	50
SY1300400	Chemical and Volume Control System – Low Pressure Letdown	13
SY1300400	Chemical and Volume Control System – Plan/Text	25

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION / DATE
CR 34964 White Paper	Streamline Safety Injection When ABHV20 Was Opened	2
Change Package 012410	PRT Sparing Line Bypass	0
Change Package 013674	CRDM Nozzle #6 Scratch Evaluation	0
ES-1.1	Background Information for Westinghouse Owners Group Emergency Response Guideline	April 30, 2005
	Control Room Turnover Checklist	March 11, 19 and 22, 2011
	Corrective Action Review Board Meeting Minutes	March 23, 2011
Chapter 7.3-39	Updated Safety Analysis Report	13
	Site Clock Reset Communication – Condition Report 34964	March 19, 2011
Page 14 of 17	Breach Authorization Permit Log	April 22, 2011
Page 6 of 7	Fire Protection Impairment Control Log	April 22, 2011
2011-118, 119, 121, 122	Fire Protection Impairment Control Permit	
2011-148, 149, 215, 237, 238	Breach Authorization Permit	
	Fire Protection Significance Determination Review 04/05/2011 Halon Discharge in ESF Switchgear Room 1	
	Fire Incident Investigation Report	April 5, 2011
FW1431401	Just-in-Time Training – Alternate Planning and/or Training Record	0

11-339929-001	AMETEK Solidstate Controls	
16577-M-658	Technical specification for Furnishing, Installing, and Testing Halogenated Agent Extinguishing System for the Standardized Nuclear Unit Power Plant System (SNUPPS) Wolf Creek Unit Only	
SU4-KC02	Fire Protection System Halon Preoperational Test	0
	Control Room Turnover Checklist	April14, 15, 21, 2011
	Boundary Watch Duties	
FW1231401	Fire Watch Duties and Responsibilities	10
LR5005012	JIT Plant Shutdown From 100% RTP	3
I-11154	Operation and Maintenance Instructions Solenoid Power Operated Relief Valve	1
59 99-0007	LTOP TS Bases – B 3.4.12	1

DRAWINGS

NUMBER	TITLE	REVISION
M-744-00019	SNUPPS Projects Functional Diagram Reactor Trip Signals	W07
M-744-00024	SNUPPS Projects Functional Diagram Steam Generator Trip Signals	W06
M-744-00025	SNUPPS Projects Functional Diagram Safeguards Actuation Signals	W07

CONDITION REPORTS

00033745	00034963	00034964	00034964	00034967
00034968	00034969	00034970	00034975	00034987
00034995	00035000	00035001	00035012	00035017
00035246	00035249	00035251	00035319	00035333

00035515	00035638	00035648	00035650	00035652
00036164	00036719	00037931	00038232	00038516

REPORTABILITY EVALUATION REQUEST

2011-040

Work Orders

08-310440-001	08-310449-000	08-310449-001	08-310440-000	11-339200-001
11-339027-000				

Section 4OA5: Other Activities

CALCULATIONS

NUMBER	TITLE	REVISION
CN-SEE-III-11-6	Evaluation of Suction Side Gas Void Volumes for Wolf Creek to Address GL-2008-01	0
EN-M-024	Critical Submergence in Containment Spray Additive Tank (TEN01) to Avoid Vortex	0
XX-M-074	Comparison of GOTHIC Gas Transport Calculations with Westinghouse Test Data for Wolf Creek Emergency Core Cooling System	0
XX-M-076	Startup Pressure Pulse Analysis for WCGS ECCS Discharge Piping	0
XX-M-079	ECCS (Emergency Core Cooling System) Horizontal Line Metrology (laser measurements) Data Evaluation,"	1

CONDITION REPORTS

00006250	00008212	00018673	00029160	00033057
00032378	00033060	00033061	00033062	00033063

00033065	00033070	00033071	00038714	00038715
2008-000091				

DRAWINGS

NUMBER	TITLE	REVISION
M-12BG03	Piping and Instrumentation Diagram Chemical & Volume Control System	47
M-12BN01	Piping and Instrumentation Diagram Borated Refueling Water Storage System	14
M-12EJ01	Piping and Instrumentation Diagram Residual Heat Removal System, sheet 1	46
M-12EM01	Piping and Instrumentation Diagram High Pressure Coolant Injection System	37
M-12EM02	Piping and Instrumentation Diagram High Pressure Coolant Injection System	19
M-12EP01	Piping and Instrumentation Diagram Accumulator Safety Injection System	08
M-12EN01	Piping and Instrumentation Containment Spray System	12
M-13EJ01	Piping Isometric Residual Heat Removal System – Auxiliary Building “A” Train	09
M-13EM01	Piping Isometric High Pressure Coolant Injection System – Auxiliary Building	16

INSPECTION REPORTS (05000482/

2008007	2009006	2009007	2010005	2010006
2010007				

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION / DATE
	Brooks Metrology Report	
	Examples of Accumulator Level Alerts	
	List of discharge and suction vent valves for the Generic Letter 2008-01 systems	
	Technical Specifications Surveillance Requirement 3.5.2.3 Bases	
	Updated Final Safety Analysis Report, Section 6.3.2.2	
	Updated Final Safety Analysis Report Change Request 2008-004 to section 6.3.2.2	
2008-01	“Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems	January 11, 2008
2008-0624	Technical Specifications Document Revision Request	
APC 09-20	Generic Letter (GL) 2008-01, “Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems” – Evaluation of Unexpected Voids or Gas Identified in	May 18, 2009

	Plant ECCS and Other Systems	
Form APF 05-002-01	Engineering Screening Form	17
	Gas Voiding Improvement Plan – Project Report	1
NEI 09-10	Guidelines for Effective Prevention and Management of System Gas Accumulation	1
	Reviewed the set of laser metrology isometric drawings	
Report FAI/08-70	Gas Voids Pressure Pulsations Program	1
Report FAI/11-192	Void Acceptance Criteria for Wolf Creek Discharge Piping Based on FAI/08-70 Methodology, Revision 1	March /2011
SEL 2011-196	STARS Gas Team Self-Assessment	January 20, 2011
Specification M-204	Technical Specification for Field Fabrication and Installation of Piping and Pipe Supports to ASME Section III for the Wolf Creek Generating Station	46
Standing Order 33	Accumulator Level Alert E-mail	0
WCAP-16631-NP	Testing and Evaluation of Gas Transport to the Suction of ECCS Pumps – Volume 1	0
WCAP-17276-P	Investigation of Simplified Equation for Gas Transport	January 2011
WCNOC122-PR-01	Study of Vent Requirements for Cooling Water Systems	0

System Walk Down Reports

PROCEDURES

NUMBER	NUMBER	REVISION
AI 23P-001	Gas Intrusion Program	0
AP 21E-001	Clearance Orders	27
QCP-20-526	Ultrasonic Measurement for Liquid Level Measurement	1
STN IC-252A	Calibration of RHR Pump A Mini Flow Valve Control Switch	7A
STN IC-252B	Calibration of RHR Pump B Mini Flow Valve Control Switch	8A
STS BG-002	ECCS Valve Check and System Vent	26
STS BG-002A	Train A ECCS System Vent for Mode 4	5 and 10
STS BG-002B	Train B ECCS System Vent for Mode 4	4 and 10
SYS BG-120	Chemical and Volume Control System Startup	43
SYS EG-400	Component Cooling Water System Fill and Vent	20A
SYS EM-410	Fill and Vent of Safety Injection System After Maintenance	18A
SYS EJ-110	RHR System Fill and Vent Including Initial RCS Fill	60
SYS SJ-002	Void Sampling Using a Sample/Purge Rig	1

SURVEILLANCE TESTS

TITLE	TITLE	DATE
STS BG-007A	ECCS Valve Check and Train A and Common Void Monitoring and Venting	March 3, 2011
STS BG-007B	ECCS Train B Monitoring and Venting	March 18, 2011
STS EG-003A	CCW Train A Monitoring and Venting	March 16. 2011
STS EG-003B	CCW Train A Monitoring and Venting	March 16, 2011
STS EN-003A	Containment Spray Train A and Common Void Monitoring and Venting	March 2, 2011
STS EN-003B	Containment Spray Train B Void Monitoring and Venting	March 15, 2011

Section 4OA5: Other Activities

NUMBER	TITLE	REVISION
EPP 06-021	Training Programs	8
SAM SAG-01	Inject into the Steam Generators	1
SAM SAG-02	Depressurize the RCS	1
SAM SAG-03	Inject into RCS	1
SAM SAG-04	Inject into Containment	1

SAM SAG-05	Reduce Fission Product Releases	1
SAM SAG-06	Control Containment Conditions	1
SAM SAG-07	Reducing Containment Hydrogen	1
SAM SAG-08	Flood Containment	1
SAM SAEG-01	TSC Long Term Monitoring	2
SAM SAEG-02	SAMG Termination	1
SAM SACRG-02	SACRG for Transients after TSC is Functional	2
SAM SACRG-01	Severe Accident Control Room Guideline Initial Response	2
	WOG Severe Accident Management Guidance	1

<u>CONDITION REPORTS</u>

18664	18398