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

February 6, 2007

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

SUBJECT: WOLF CREEK GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 05000482/2006005

Dear Mr. Muench:

On December 31, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Wolf Creek Generating Station. The enclosed integrated report documents the inspection findings which were discussed on January 9, 2007, with Mr. S. E. Hedges and other members of your staff.

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

This report documents two NRC-identified findings of very low safety significance (Green). Both of these findings were determined to involve violations of NRC requirements. Additionally, one licensee-identified violation, which was determined to be of very low safety significance, is listed in Section 4OA7 of this report. The NRC is treating these violations as noncited violations consistent with Section VI.A.1 of the NRC Enforcement Policy because of the very low safety significance and because the findings were entered into your corrective action program. If you contest these noncited violations, you should provide a response within 30 days of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region IV; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Wolf Creek Generating Station.

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

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

Sincerely,

/RA/

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

Docket: 50-482
License: NPF-42

Enclosure:
NRC Inspection Report 05000482/2006005
w/attachment: Supplemental Information

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SUNSI Review Completed: _yes___ ADAMS: ☒ Yes ☐ No Initials: __vgg__
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SRI:DRP/B	RI:DRP/B	C:DRS/EB2	SRI:DRP/C
SDCochrum:sa	JRGroom	LJSmith	GBMiller
VGGaddy for	***VGGaddy for***	***JMateychick for***	***VGGaddy for***
1/10/07	1/10/07	1/25/07	1/31/07
C:DRS/OB	C:DRS/PSB	C:DRS/EB1	C:DRP/B
RLNease	DAPowers	WBJones	VGGaddy
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1/19/07	1/25/07	1/19/07	2/6/07

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-482
License:	NPF-42
Report:	5000482/2006005
Licensee:	Wolf Creek Nuclear Operating Corporation
Facility:	Wolf Creek Generating Station
Location:	1550 Oxen Lane NE Burlington, Kansas
Dates:	October 8 through December 31, 2006
Inspectors:	S. D. Cochrum, Senior Resident Inspector J. R. Groom, Resident Inspector G. A. George, Reactor Inspector J. H. Nadel, Reactor Inspector F. L. Brush, Senior Project Engineer P. J. Elkmann, Emergency Preparedness Inspector B. K. Tharakan, Health Physicist, Plant Support Branch
Approved By:	V. G. Gaddy, Chief, Project Branch B

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SUMMARY OF FINDINGS

IR 05000482/2006005; 10/08/06 - 12/31/06; Wolf Creek Generating Station; Refueling and Outage Activities, Access Control to Radiologically Significant Areas.

This report covered a 3-month period of inspection by resident and regional inspectors. The inspection identified two Green findings, both of which were noncited violations. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, “Significance Determination Process.” Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” involving the licensee's failure to properly implement the reactor vessel closure head installation procedure during Refueling Outage 15. Specifically, on October 30, 2006, the licensee performed Procedure FHP-02-007B, “Reactor Vessel Closure Head Installation," Revision 5. During the performance of Procedure FHP-02-007B, the licensee encountered problems with the polar crane that prevented the crane hoist from being lowered. The problems with the polar crane were encountered while the reactor vessel head was being transported along the North-South axis of the refueling cavity towards the reactor vessel. Consequently, the licensee transported the reactor vessel closure head approximately 3 feet over the reactor vessel flange, while suspended approximately 4 feet above the operating deck. This condition was not allowed by procedure and exceeded the maximum analyzed height in the head drop analysis.

The failure to properly implement the reactor vessel closure head installation procedure was considered a performance deficiency. The finding was greater than minor because it affected the human performance attribute of the mitigating systems cornerstone and the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events. Using Manual Chapter 0609, Appendix G, “Shutdown Operations Significance Determination Process,” Phase 1 worksheets, the finding was found to be of very low safety significance because it did not affect decay heat removal or reactor coolant system inventory. The inspectors determined that the finding has crosscutting aspects in the area of human performance associated with work practices because the licensee failed to use appropriate human error prevention techniques, such as self- and peer-checking and not proceeding in the face of uncertainty. The inspectors also determined that the finding has crosscutting aspects in the area of problem identification and resolution associated with operating experience because the licensee failed to effectively communicate internally generated lessons learned following the procedural noncompliance during the Refueling Outage 13 reactor vessel head installation. (Section 1R20)

Cornerstone: Occupational Radiation Safety

Green. The inspector identified a noncited violation of 10 CFR 19.12(a)(2) because the licensee failed to provide instructions to a worker on how to minimize exposure while working with radioactive material and contaminated equipment. Specifically, on October 18, 2006, a worker on the Steam Generator A platform received an intake of Cobalt-58 while removing contaminated conduit from the primary side of the steam generator and placing it in a radioactive material bag for storage. The worker was wearing a face shield; however, the inspector identified that the licensee failed to provide the worker with instructions on how to minimize exposure to radioactive material while performing this task. The licensee's corrective actions included providing workers with powered face shields that blow air away from the face. This finding was entered into the licensee's corrective action program.

The finding was greater than minor because it was associated with the Occupational Radiation Safety Cornerstone attribute of Exposure Control and affected the cornerstone objective to ensure the adequate protection of a worker's health and safety from exposure to radioactive materials because a worker received an unintended internal dose. The finding was processed through the Occupational Radiation Safety Significance Determination Process and determined to be of very low safety significance because it was not an as low as is reasonably achievable finding, there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised. Additionally, this finding has a crosscutting aspect in the area of human performance related to work practices because the licensee did not ensure supervisory oversight of work activities such that exposure to radioactive material was minimized and properly controlled. (Section 2OS1)

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 corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Wolf Creek was shut down for Refueling Outage 15 at the beginning of the inspection period. The licensee completed the refueling outage and synchronized the generator to the grid on November 10, 2006. The licensee returned to full power operations on November 15, 2006. Wolf Creek remained at full power for the remainder of the inspection period.

1. REACTOR SAFETY

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

1R04 Equipment Alignment (71111.04)

Partial System Walkdowns

a. Inspection Scope

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

- October 19, 2006, Emergency Diesel Generator A during an Emergency Diesel Generator B run

- October 20, 2006, Emergency Diesel Generator B while Emergency Diesel Generator A was inoperable

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed two samples.

b. Findings

No findings of significance were identified

1R05 Fire Protection (71111.05)

Fire Protection Tours

a. Inspection Scope

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

- October 11, 2006, Containment
- October 19, 2006, Emergency Diesel Generator B room
- October 20, 2006, Emergency Diesel Generator A room
- November 6, 2006, Turbine-driven auxiliary feed pump room
- December 11, 2006, Essential service water pump house
- December 19, 2006, Lower cable spreading room and common corridor

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

Annual Internal Flooding

a. Inspection Scope

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

- December 11, 2006, Main steam/main feedwater isolation valve enclosure

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08P)

Procedure 71111.08 requires a minimum sample size of four for each section (Sections 02.01, 02.02, 02.03, and 02.04). The inspectors fulfilled the requirements of Inspection Procedure 71111.08. Documents reviewed by the inspectors are listed in the attachment.

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

a. Inspection Scope

The procedure requires the review of two to three types of nondestructive examination activities (i.e., volumetric, surface, or visual). The inspectors reviewed one of each: (1) visual examination, (2) magnetic particle (surface) examination, and (3) seven ultrasonic (volumetric) examinations. Of these examinations, the inspectors witnessed the performance of one visual, one surface, and one ultrasonic examination. The ultrasonic examinations are listed below.

DATA SHEET	ISI NUMBER	SYSTEM/WELD AREA	EXAM METHOD
TMC-002	EM-05-S005-B	HPCI/pipe to elbow	Ultrasonic
TMC-002	EM-05-S005-C	HPCI/elbow to pipe	Ultrasonic
AWJ-004	BB-04-F001	Pressurizer spray line/PZR safe end to pipe	Ultrasonic
JLD-004	TBB03-2-W	PZR spray nozzle	Ultrasonic
AWJ-005	AB-01-F074	Main steam/pipe to pen	Ultrasonic
AWJ-005	AB-01-F007	Main Steam/Pipe to Pen	Ultrasonic
TMC-004*	TBB03-CIRCUM-5-W	Pressurizer circumferential weld	Ultrasonic

For each of the activities reviewed, the inspectors verified that the activities performed were in accordance with site procedures and applicable American Society of Mechanical Engineers (ASME) Code requirements. In addition, the inspectors verified that the test equipment was properly used during the current calibration period. During the performance of the activities, the inspectors verified that certifications of nondestructive examination personnel were current and in accordance with industry standards.

The inspectors reviewed the site procedures to verify that recordable indications were dispositioned in accordance with the ASME Code or an NRC approved alternative. During the performance of the above nondestructive examination activities, no recordable indications were identified or accepted for continued service.

The inspection procedure requires verification that the welding process and welding examinations were performed on one to three welds in accordance with ASME Code requirements or an NRC approved alternative. The inspectors observed three in-process weld overlay activities being performed on pressurizer nozzle dissimilar metal welds that are susceptible to primary water stress corrosion cracking. The inspectors reviewed weld overlays being performed on the pressurizer relief line, pressurizer Safety Line C, and the pressurizer surge line. The licensee personnel performed activities in accordance with ASME Code and 10 CFR 50.55a, Request I3R-05.

The inspectors completed the one sample required by Section 02.01.

b. Findings

No findings of significance were identified.

.2 Vessel Upper Head Penetration Inspection Activities

This section of the inspection procedure was performed using Temporary Instruction 2515/150, "Reactor Pressure Vessel and Vessel Head Penetration Nozzles." The results of this inspection are reported in Section 4OA5, Other Activities.

.3 Boric Acid Corrosion Control Inspection Activities Pressurized Water Reactor)

a. Scope

The inspectors reviewed two boric acid screens and seven boric acid evaluations found on reactor coolant system piping and components that were identified during the previous operating cycle. In addition, the inspectors reviewed the boric acid corrosion control procedures.

Through record review, the inspectors verified that the visual inspections emphasized locations where boric acid leaks can cause degradation and these boric acid leaks were properly documented. On those components where boric acid was identified, the engineering evaluations gave assurance that the ASME Code wall thickness limits were properly maintained. The evaluations also confirmed that the corrective actions performed for evidence of boric acid leaks were consistent with requirements of the ASME Code.

The inspectors completed the one sample required by Section 02.03.

b. Findings

No findings of significance were identified.

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

The inspection procedure requires that inspectors compare the in-situ screening criteria to Electric Power Research Institute (EPRI) guidelines. In particular, the inspectors must assess whether assumed nondestructive examination flaw sizing accuracy is consistent with EPRI guidelines. In addition, the inspectors are to assess whether the appropriate tubes are to be pressure tested, observe those pressure tests, if possible, and review the results. The inspectors reviewed licensee procedures for in-situ pressure testing and concluded that the in-situ pressure testing program meets EPRI guidelines.

During the outage, conditions, that met EPRI guidelines, had been identified for in-situ pressure testing specific steam generator tubes; however, those tubes were taken out of service by plugging. Therefore, no in-situ pressure testing was performed on any steam generator tubes.

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

The inspectors reviewed tube flaws identified in the previous outage comparing them to current outage results. The inspectors verified that minor flaw growth had occurred. Those flaws that reached the 40 percent plugging criteria were plugged using a process approved by Technical Specifications (TS). Those flaws below the 40 percent criteria were evaluated and plugged, if the evaluation indicated an estimated growth above 40 percent during the next operating cycle. The 40 percent tube repair criterion was applied to all indications in tube support plate and antivibration bar intersections.

The inspection procedure requires the inspectors confirm that the eddy current test probes and equipment are qualified for the expected types of tube degradation. Inspectors must also assess the site-specific qualification of one or more techniques. The inspectors reviewed the qualification records of the bobbin coil and motorized rotating pancake coil probes. The calibration and qualification records of the probes were current for the types of detectable tube degradations.

During the performance of steam generator eddy current activities, the licensee identified foreign material in the secondary side of the steam generator. The inspectors verified that the licensee inspected the secondary side of the steam generator and removed those foreign materials. The licensee then reinspected those steam generator tubes that may have been affected by foreign material. No loose parts or foreign material were inaccessible; therefore, all foreign material was removed.

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

The inspectors completed the one sample required by Section 02.04.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed four problem identification and resolutions, which dealt with inservice inspection activities, and found that the corrective actions were appropriate. From this review the inspectors concluded that the licensee had an appropriate threshold for entering issues into the corrective action program and has procedures that direct a root cause evaluation when necessary. The licensee also had an effective program for applying industry operating experience.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

Resident Inspector Quarterly Review

a. Inspection Scope

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

- October 26, 2006, Emergency diesel generator Robert Shaw temperature control valves

- December 6, 2006, Remote shutdown panel and postaccident monitoring instrumentation

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

Risk Assessment and Management of Risk

The inspectors reviewed the assessment activities listed below to verify: (1) performance of risk assessments when required by 10 CFR 50.65 (a)(4) and licensee procedures prior to changes in plant configuration for maintenance activities and plant operations; (2) the accuracy, adequacy, and completeness of the information considered in the risk assessment; (3) that the licensee recognizes, and/or enters as applicable, the appropriate licensee-established risk category according to the risk assessment results and licensee procedures; and (4) that the licensee identified and corrected problems related to maintenance risk assessments.

- October 8, 2006, Overall outage risk
- October 20, 2006, Daily risk assessment following discovery of reduced inventory within spent fuel pool

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

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

- October 21, 2006, Residual Heat Removal Trains A and B seismic loading
- November 7, 2006, Emergency Diesel Generator A fuel line fretting

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

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

- October 26, 2006, Essential Service Water Pump B replacement
- October 27, 2006, Emergency Diesel Generator B break-in run
- November 28, 2006, Steam generator blowdown isolation valve

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the following risk significant refueling items or outage activities to verify defense in-depth commensurate with the outage risk control plan, compliance with TS, and adherence to commitments in response to Generic Letter 88-17, “Loss of Decay Heat Removal”: (1) the risk control plan; (2) tagging/clearance activities; (3) reactor coolant system instrumentation; (4) electrical power; (5) decay heat removal; (6) spent fuel pool cooling; (7) inventory control; (8) reactivity control; (9) containment closure; (10) reduced inventory or midloop conditions; (11) refueling activities; (12) heatup and cooldown activities; (13) restart activities; and (14) licensee identification and implementation of appropriate corrective actions associated with refueling and outage activities. The inspectors' containment inspections included observations of the containment sump for damage and debris, and supports, braces, and snubbers for evidence of excessive stress, water hammer, or aging.

- October 7, 2006, Reactor coolant system cooldown
- October 8, 2006, Initial containment entry, including normally inaccessible areas behind the bio-shield wall
- October 10-11, 2006, Reactor vessel head de-tensioning and removal
- October 12, 2006, Removal of reactor upper internals
- October 13, 2006, Core offload (observed from containment and control room)
- October 17, 2006, Outage risk assessment techniques, including assessment of decay heat removal/spent fuel pool cooling, availability of offsite power, reactivity control, and control of containment.
- October 25, 2006, Core reload (observed from control room and fuel handling platform)
- October 30, 2006, Reactor vessel head set (observed from outage control center)
- November 1, 2006, Reactor coolant system fill and vent, including reduced inventory or midloop conditions, which included adherence to commitments in response to Generic Letter 88-17
- November 6, 2006, Reactor coolant system heatup and establishing bubble in pressurizer
- November 7, 2006, Containment closure walkdown, including inspection of containment recirculation sumps

- November 8, 2006, Under reactor vessel inspection

- November 9-11, 2006, Reactor startup, physics testing, and review of core operating limits report

- December 6, 2006, Verification of core reload (by video)

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

Introduction: The inspectors identified a Green noncited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," involving the licensee's failure to properly implement the reactor vessel closure head installation procedure during Refueling Outage 15.

Description: On October 30, 2006, the inspectors observed the licensee perform Procedure FHP-02-007B, "Reactor Vessel Closure Head Installation," Revision 5. Step 8.1.9 of Procedure FHP-02-007B directs the reactor vessel closure head be raised from the head stand until clear of all obstacles, approximately 4 feet above the operating deck. Following movement of the head to the refueling cavity, step 8.1.11 of Procedure FHP-02-007B directs the crane operator to reposition the polar crane onto the north-south axis of the reactor building and lower the head to approximately 12 inches but no more than 2 feet above the operating deck, which translates to a maximum elevation of 2049'-6" in containment.

During the performance of Procedure FHP-02-007B, the licensee encountered problems with the polar crane that prevented the crane hoist from being lowered. The problems with the polar crane were encountered while the reactor vessel head was being transported along the north-south axis of the refueling cavity towards the reactor vessel. Upon determination that the polar crane was unable to be lowered, the licensee stopped the transport of the head towards the reactor vessel. Following direction from the outage control center manager, the reactor vessel head was moved to the storage stand for troubleshooting. The reactor vessel head was successfully installed on a subsequent attempt with the crane operator utilizing the controls from the local cab.

Following the installation of the reactor vessel closure head, the inspectors reviewed Calculation BB-FW-011, "WCNOC, Reactor Vessel Head Drop Analysis," Revision 0. Calculation BB-FW-011 postulates the maximum stress transmitted to the four support nozzles in the event of a failure of the polar crane. The calculation concluded that a reactor vessel head drop from an analyzed height of 28 feet above the reactor vessel flange would not challenge core cooling capability or the integrity of the fuel cladding. The 28 feet translates to a maximum elevation of 2049'-6" in containment; the limit imposed by step 8.1.11 of Procedure FHP-02-007B.

The inspectors also noted that step 8.1.11 contains commitment step 3.2.6 that references a Performance Improvement Request (PIR) 2003-3589, "Procedure Noncompliance During Reactor Head Move." Commitment step 3.2.6 was implemented in response to a procedural error during the Refueling Outage 13 head installation, where the reactor vessel head was lifted greater than the 28 feet analyzed in Calculation BB-FW-011. No additional precautions, details, or procedural requirements regarding commitment step 3.2.6 were included in the procedure.

Because Precaution/Limitation 5.3 allows operators to deviate from step-by-step procedural compliance and step 8.1.11 of Procedure FHP-02-007B did not contain a specific requirement to lower reactor vessel closure head to the required height prior to moving toward the vessel flange, the inspectors questioned whether any portion of the reactor vessel closure head was suspended greater than the allowed height above the operating deck while over the reactor vessel. Since step 8.1.9 of Procedure FHP-02-007B directs the reactor vessel closure head be raised approximately 4 feet above the operating deck and the controls for the polar crane did not allow for the crane hoist to be lowered, the inspectors concluded that the reactor vessel closure head was approximately 2 feet above the maximum allowed height above the vessel flange while being transported towards the reactor vessel. This conclusion was verified by examining video footage of the Refueling Outage 15 reactor vessel closure head installation.

To determine the proximity of the reactor vessel closure head relative to the reactor vessel flange, the inspectors again reviewed video footage of the Refueling Outage 15 reactor vessel closure head installation. Line of sight bearings were established to closely approximate the position of the reactor vessel closure head within the refueling cavity. Additionally, distinguishing marks along the polar crane were examined and compared to video footage of the head installation that occurred on October 30, 2006. Through review of both the video footage and the actual physical inspection of the polar crane, the inspectors determined that the reactor vessel closure head was approximately 3 feet over the reactor vessel flange while suspended approximately 4 feet above the operating deck. This condition exceeded the maximum allowed height analyzed in the head drop analysis.

Analysis: The failure to properly implement the reactor vessel closure head installation procedure was considered a performance deficiency. This finding was more than minor because it affected the human performance attribute of the mitigating systems cornerstone and the cornerstone objective to ensure the availability, reliability, and capability of system that respond to initiating events. Using Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," Phase 1 worksheets, the finding was found to be of very low safety significance because it did not affect decay heat removal or reactor coolant system inventory. The inspectors determined that the finding has crosscutting aspects in the area of human performance associated with work practices because the licensee failed to use appropriate human error prevention techniques, such as self- and peer-checking and not proceeding in the face of uncertainty. The inspectors also determined that the finding has crosscutting aspects in the area of problem identification and resolution associated with operating experience because the licensee failed to effectively communicate internally generated lessons learned following the procedural noncompliance during the Refueling Outage 13 reactor vessel head installation.

Enforcement: 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these procedures. Contrary to Procedure FHP-02-007B, "Reactor Vessel Closure Head Installation," Revision 5, the licensee failed to lower the reactor vessel closure head to the maximum allowed analyzed height prior to positioning over the vessel flange. Because this violation was of very low safety significance and was entered into the corrective action program as Condition Report 2006-003342, this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000482/2006005-01, "Failure to implement the reactor vessel closure head installation procedure."

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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

- October 19, 2006, STS KJ-005B, "Manual/Auto Start, Synchronization & Loading of Emergency D/G NE02," Revision 44

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed plant drawings, work orders, and Temporary Modification Package 06000BG to ensure that the temporary modification was properly implemented. The inspectors: (1) verified that the modification did not have an affect on system operability/availability, (2) verified that the installation was consistent with the modification documents, (3) ensured that the postinstallation test results were satisfactory and that the impact of the temporary modification on permanently installed SSCs were supported by the test, (4) verified that the modifications were identified on control room drawings and that appropriate identification tags were placed on the affected drawings, and (5) verified that appropriate safety evaluations were completed. The inspectors verified that the licensee identified and implemented any needed corrective actions associated with temporary modifications:

- Normal charging pump (PBG04) balance line, December 13, 2006

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1EP2 Alert Notification System Testing (71114.02)

a. Inspection Scope

The inspector discussed with licensee staff the status of offsite siren and tone alert radio systems to determine the adequacy of licensee methods for testing the alert and notification system in accordance with 10 CFR Part 50, Appendix E. The licensee's alert and notification system testing program was compared with criteria in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, Federal Emergency Management Agency (FEMA) Report REP-10, "Guide for the Evaluation of Alert and Notification Systems for Nuclear Power Plants," and the licensee's current FEMA-approved alert and notification system design report, dated June 12, 1987. The inspector also reviewed the following procedures:

- Procedure EP 06-019, "Alert and Notification System Sirens," Revision 3

- Procedure EP 06-022, "Tone Alert Radio Maintenance/Compensatory Measures," Revisions 0-4

Documents reviewed by the inspectors are listed in the attachment.

The inspector completed one sample.

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization Augmentation Testing (71114.03)

a. Inspection Scope

The inspector discussed with licensee staff the status of primary and backup systems for augmenting the on-shift emergency response staff to determine the adequacy of licensee methods for staffing emergency response facilities. The inspector reviewed the following documents related to the emergency response organization augmentation system to evaluate the licensee’s ability to staff their emergency response facilities in accordance with the emergency plan and the requirements of 10 CFR Part 50, Appendix E:

- Procedure EP 06-015, "Emergency Response Organization Callout," Revisions 7- 9
- Tool Kit-011, "Callout Test Guidance," Revision 4
- Emergency response organization guidance, Revision 1
- Callout system call detail report, dated February 16, 2006

Documents reviewed by the inspectors are listed in the attachment.

The inspector completed one sample.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspector performed an in-office review of Revision 8 to Emergency Plan Implementing Procedure Form APF 06-002-01 "Emergency Action Levels," submitted November 3, 2006. This revision changed the amount of steam generator leakage necessary to enter Emergency Action Level 2, "Steam Generator Tube Failure," consistent with NRC-approved changes to station TS for steam generator leakage.

The revision was compared to the previous revision; to the criteria of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1; to the criteria of Nuclear Energy Institute 99-01, "Methodology for Development of Emergency Action Levels," Revision 2; and to the standards in 10 CFR 50.47(b) to determine if the revision was adequately conducted according to the requirements of 10 CFR 50.54(q). This review was not documented in a safety evaluation report and did not constitute approval of the licensee’s changes; therefore, this revision is subject to future inspection.

Documents reviewed by the inspectors are listed in the attachment.

The inspector completed one sample.

b. Findings

No findings of significance were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies (71114.05)

a. Inspection Scope

The inspector reviewed the licensee’s corrective action program requirements in Procedures AP 28A-100, "Condition Reports," Revision 2; AP 28A-001, "Performance Improvement Request," Revision 26; and AP 28B-001, "Root Cause Analysis," Revision 2. The inspector reviewed summaries of 186 PIRs and condition reports assigned to the emergency preparedness department between May 2004 and November 2006, and selected 19 for detailed review against the program requirements. The inspector evaluated the response to the corrective action requests to determine the licensee’s ability to identify, evaluate, and correct problems in accordance with the licensee program requirements and 10 CFR 50.47(b)(14) and 10 CFR Part 50, Appendix E. The inspector also reviewed other documents listed in the attachment to this report.

Documents reviewed by the inspectors are listed in the attachment.

The inspector completed one sample during the inspection.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

For the below listed drill and simulator-based training evolution contributing to drill/exercise performance and emergency response organization performance indicators, the inspectors: (1) observed the training evolution to identify any weaknesses and deficiencies in classification, notification, and protective action requirements development activities; (2) compared the identified weaknesses and deficiencies against licensee identified findings to determine whether the licensee is properly identifying failures; and (3) determined whether licensee performance is in accordance with the guidance of the Nuclear Energy Institute 99-02 document's acceptance criteria.

- April 20, 2006, fire in vital Buss XNB02 and loss of all alternating current power to the vital busses

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

This area was inspected to assess licensee performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, high radiation areas, and worker adherence to these controls. The inspector used the requirements in 10 CFR Part 20 and the licensee's procedures required by TSs as criteria for determining compliance. During the inspection, the inspector interviewed the radiation protection manager, radiation protection supervisors, and radiation workers. The inspector performed independent radiation dose rate measurements and reviewed the following items:

- Performance indicator events and associated documentation packages reported by the licensee in the Occupational Radiation Safety Cornerstone

- Controls (surveys, posting, and barricades) of three radiation, high radiation, or airborne radioactivity areas

- Radiation work permits, procedures, engineering controls, and air sampler locations

- Conformity of electronic personal dosimeter alarm setpoints with survey indications and plant policy; workers' knowledge of required actions when their electronic personnel dosimeter noticeably malfunctions or alarms

- Barrier integrity and performance of engineering controls in airborne radioactivity areas

- Adequacy of the licensee's internal dose assessment for any actual internal exposure greater than 50 millirem committed effective dose equivalent

- Physical and programmatic controls for highly activated or contaminated materials (nonfuel) stored within spent fuel and other storage pools

- Self-assessments and audits related to the access control program since the last inspection; there were no licensee event reports (LERs) and special reports

- Corrective action documents related to access controls

- Licensee actions in cases of repetitive deficiencies or significant individual deficiencies

- Radiation work permit briefings and worker instructions

- Adequacy of radiological controls such as, required surveys, radiation protection job coverage, and contamination controls during job performance

- Dosimetry placement in high radiation work areas with significant dose rate gradients

- Changes in licensee procedural controls of high dose rate - high radiation areas and very high radiation areas

- Controls for special areas that have the potential to become very high radiation areas during certain plant operations

- Posting and locking of entrances to all accessible high dose rate - high radiation areas and very high radiation areas

- Radiation worker and radiation protection technician performance with respect to radiation protection work requirements

Documents reviewed by the inspectors are listed in the attachment.

The inspector completed 21 of the required 21 samples.

b. Findings

Introduction: The inspector identified a noncited violation of 10 CFR 19.12(a)(2) because the licensee failed to provide workers with instructions on how to correctly handle and store radioactive material and contaminated equipment. The violation had very low safety significance.

Description: On October 18, 2006, a steam generator worker on the Steam Generator A platform received an intake of Cobalt-58 while removing conduit from the primary side of the steam generator and placing it into a radioactive material bag for storage. The worker was observed pushing down on the bag in order to compact its size for easier storage. The release of air from the bag during this compaction was the most likely source of Cobalt-58 that resulted in the intake. The worker was wearing a face shield. However, the inspector interviewed the workers involved and identified that the licensee failed to provide the workers with instructions on how to minimize exposure to radioactive material while performing this task. The licensee's corrective actions included placing workers in powered face shields that blow air away from the face and ensuring that there was supervisory oversight during these types of tasks. The worker received a committed effective dose equivalent of one millirem and a committed dose equivalent of four millirem to the lung.

Analysis: The failure to provide instructions to workers was a performance deficiency. The finding was greater than minor because it was associated with the Occupational Radiation Safety Cornerstone attribute of exposure control, and affected the cornerstone objective to ensure the adequate protection of a worker's health and safety from exposure to radioactive materials because a worker received an unintended internal dose. Because the finding involved unplanned, unintended dose resulting from conditions that were contrary to NRC regulations, the finding was evaluated using the Occupational Radiation Safety Significance Determination Process. The finding was determined to be of very low safety significance because: (1) it was not an ALARA (as low as is reasonably achievable) finding, (2) there was no personnel overexposure, (3) there was no substantial potential for personnel overexposure, and (4) the finding did not compromise licensee's ability to assess dose. Additionally, this finding also had a crosscutting aspect in the area of human performance related to work practices because the licensee did not ensure supervisory oversight of work activities such that exposure to radioactive material was minimized and properly controlled.

Enforcement: Section 19.12(a)(2) of Title 10 of the Code of Federal Regulations states, in part, that all individuals who in the course of employment are likely to receive in a year an occupational dose in excess of 100 millirem shall be instructed in precautions and procedures to minimize exposure to radiation and radioactive materials. Contrary to this requirement, on October 18, 2006, the licensee failed to provide instructions to workers about working with radioactive material and contaminated equipment. This violation was entered into licensee's corrective action program as Condition Report 2006-2668. Because this finding is of very low safety significance and was entered into licensee's corrective action program, it is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000482/2006005-02, Failure to provide instructions to workers.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspector assessed licensee performance with respect to maintaining individual and collective radiation exposures ALARA. The inspector used the requirements in 10 CFR

Part 20 and the licensee's procedures required by TSs as criteria for determining compliance. The inspector interviewed licensee personnel and reviewed:

- Site-specific ALARA procedures
- ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements
- Interfaces between operations, radiation protection, maintenance, maintenance planning, and scheduling and engineering groups
- Integration of ALARA requirements into work procedure and radiation work permit (or radiation exposure permit) documents
- Workers' use of the low dose waiting areas
- Radiation worker and radiation protection technician performance during work activities in radiation areas, airborne radioactivity areas, or high radiation areas

Documents reviewed by the inspectors are listed in the attachment.

The inspector completed 3 of the required 15 samples and 3 of the optional samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

Cornerstone: Emergency Preparedness Cornerstone

The inspector reviewed licensee evaluations for the three emergency preparedness cornerstone performance indicators of: (1) drill and exercise performance, (2) emergency response organization participation, and (3) alert and notification system reliability, for the period October 1, 2005, through September 30, 2006. The definitions and guidance of Nuclear Energy Institute 99-02, "Regulatory Assessment Indicator Guideline," Revisions 2-4, and licensee performance indicator Procedure AI 26A-004, "Emergency Planning Performance Indicators," Revision 2, were used to verify the accuracy of the licensee's evaluations for each performance indicator reported during the assessment period.

The inspector reviewed a sample of drill and exercise scenarios and licensed operator simulator training sessions, notification forms, and attendance and critique records associated with training sessions, drills, and exercises conducted during the verification period. The inspector reviewed 15 selected emergency responder qualification, training, and drill participation records. The inspector reviewed alert and notification system testing procedures, maintenance records, and all records of siren tests performed during the inspection period. The inspector also reviewed other documents listed in the attachment to this report.

The inspector completed three samples.

Occupational Radiation Safety Cornerstone

- Occupational exposure control effectiveness

The inspector reviewed licensee documents from April 1 through September 30, 2006. The review included corrective action documentation that identified occurrences in locked high radiation areas (as defined in the licensee's TSs), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in Nuclear Energy Institute 99-02). Additional records reviewed included ALARA records and whole- body counts of selected individual exposures. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data. In addition, the inspector toured plant areas to verify that high radiation, locked high radiation, and very high radiation areas were properly controlled. Performance indicator definitions and guidance contained in Nuclear Energy Institute 99-02, "Regulatory Assessment Indicator Guideline," Revision 4, were used to verify the basis in reporting for each data element.

The inspector completed the required one sample in this cornerstone.

Public Radiation Safety Cornerstone

- Radiological effluent TS/offsite dose calculation manual radiological effluent occurrences

The inspector reviewed licensee documents from April 1 through September 30, 2006. Licensee records reviewed included corrective action documentation that identified occurrences for liquid or gaseous effluent releases that exceeded performance indicator thresholds and those reported to the NRC. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data. Performance indicator definitions and guidance contained in Nuclear Energy Institute 99-02, "Regulatory Assessment Indicator Guideline," Revision 4, were used to verify the basis in reporting for each data element.

Documents reviewed by the inspectors are listed in the attachment.

The inspector completed the required one sample in this cornerstone.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Identification and Resolutions of Problems

a. Inspection Scope

The inspectors performed a daily screening of items entered into the licensee's corrective action program. This assessment was accomplished by reviewing work requests, work orders, and PIRs, and attending corrective action review and work control meetings. The inspectors: (1) verified that equipment, human performance, and program issues were being identified by the licensee at an appropriate threshold and that the issues were entered into the corrective action program; (2) verified that corrective actions were commensurate with the significance of the issue; and (3) identified conditions that might warrant additional followup through other baseline inspection procedures.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.2 Semiannual Trend Review

a. Inspection Scope

The inspectors completed a semiannual trend review of repetitive or closely-related issues that were documented in trend reports, problem lists, performance indicators, health reports, quality assurance audits, corrective action documents, corrective maintenance documents, and departmental self-assessments and interviewed selected licensee staff to determine if any adverse trends existed. Additionally, the inspectors reviewed the licensee’s trending efforts to identify trends that might indicate the existence of more safety-significant issues. The inspectors’ review consisted of the 6-month period from June to December 2006. When warranted, some of the samples expanded beyond those dates to fully assess the issue. The inspectors also reviewed corrective action program items associated with the below listed issues. The inspectors compared and contrasted their results with the results contained in the licensee's quarterly trend reports. Corrective actions associated with a sample of the issues identified in the licensee's trend report were reviewed for adequacy. These areas were chosen based on information gathered by the inspectors during daily plant status reviews over the previous 6 months.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings and Observations

No findings of significance were identified. However, during the review the inspectors noted the following trends:

- The inspectors noticed a continuing negative trend in the area of clearance orders and plant status control issues. Although the licensee had identified trends previously, they were not considered continuing negative trends. The licensee has completed several root causes in these areas and has ongoing corrective actions in place addressing the previously identified negative trends. Corrective actions developed from the last apparent cause determination for clearance orders contained in PIR 2006-001663 and corrective actions developed from a common cause analysis for plant status control issues are contained in Condition Report 2006-00813; however, the current corrective actions have been slow to adequately correct the problems. Condition Report 2006-000610 had been written to identify a continuing trend in status control.

- The inspectors noticed an increasing negative trend in the area of procedure use and adherence issues. A trend was identified by the licensee in 2005; however, corrective actions have been ineffective to date. Although the licensee had identified a trend previously, they did not consider the trend as an increasing negative trend.

.3 Health Physics Review of Identification and Resolutions of Problems

a. Inspection Scope

The inspector evaluated the effectiveness of the licensee's problem identification and resolution process with respect to the following inspection areas:

- Access control to radiologically significant areas
- ALARA planning and controls

b. Findings and Observations

Section 2OS1 describes a finding associated with the failure to provide instructions to workers to minimize exposure to radioactive materials. The licensee's corrective actions did not identify that an apparent cause of the intake of radioactive material was due to the worker not being provided adequate instructions and supervision with respect to controlling contamination and minimizing exposure to radioactive material.

.4 Annual Sample Review (Emergency Preparedness)

a. Inspection Scope

The inspector selected 19 PIRs and condition reports generated between June 1, 2004, and November 1, 2006, for detailed review. The reports were reviewed to ensure that the full extent of the issues were identified, an appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized. The inspector evaluated the PIRs and condition reports against the requirements of licensee Procedures AP 28A-100, "Condition Reports," Revision 2, and AP 28A-001, "Performance Improvement Request," Revision 26.

b. Findings

No findings of significance were identified.

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

.1 Personnel Performance During Nonroutine Evolutions, Events and Transients

a. Inspection Scope

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

- On November 10, 2006, the inspectors observed the licensee’s response after identifying that a symmetrical circular region of fuel assemblies in the center of the core were operating less than predicted power. This condition was found during the review of the initial power ascension flux maps. The licensee verified all nuclear parameters were within specifications and adequate design margin existed. The licensee’s cause analysis determined that the flux suppression was not caused by manufacturing or modeling errors but was most likely an artifact of design and not indicative of a problem with the core. The inspectors reviewed the flux maps and core map video to verify the fuel locations in the core.

b. Findings

No findings of significance were identified.

.2 (Closed) LER 2006-004-00, Failure to Maintain Closure of Containment Penetrations During Fuel Movement

On October 24, 2006, while in Mode 6, refueling, the licensee discovered an air-to-air pathway existed from the containment recirculation sumps though the containment spray system to the auxiliary building. The containment sump suction valves for Containment Spray Pump B were stroked open while downstream valves in the auxiliary building were uncapped and open under a clearance order. This resulted in an air-to-air pathway during fuel movement that was not administratively controlled. This is contrary to TS 3.9.4, which states, in part, a penetration flow path providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls. Upon discovery, the penetration was immediately placed under administrative controls. The root cause for this event was determined to be human performance resulting from inadequate instructions that provided vague guidance. This licensee-identified violation is documented in Section 4OA7. This LER is closed.

4OA5 Other Activities

.1 Temporary Instruction 2515/150, "Reactor Pressure Vessel Head And Vessel Head Penetration Nozzles (NRC Order EA-03-009)," Revision 3

Background

The reactor pressure vessel (RPV) heads of pressurized-water reactors (PWRs) have penetrations for control rod drive mechanisms and instrumentation systems. Nickel-based alloys (e.g., Alloy 600) are used in the penetration nozzles and weld materials. Primary coolant water and the operating conditions of PWR plants can cause cracking of these nickel-based alloys through a process called PWSCC. The susceptibility of RPV head penetrations to pressurized water stress corrosion cracking (PWSCC) appears to be linked to the operating time and temperature of the RPV head. In early 2001, circumferential cracking of the vessel head penetration nozzles at the Oconee Nuclear Station was identified. Circumferential cracking is a safety concern because of the possibility of a nozzle ejection if the condition is not detected and repaired. In early 2002, axial cracking of a vessel head penetration nozzle was discovered at the Davis Besse Nuclear Power Station. The axial crack resulted in primary water leakage, which in turn created a cavity in the RPV head.

The operating experience with respect to PWSCC of RPV head penetration nozzles has reinforced the need for more effective inspections. Section XI of the ASME Boiler and Pressure Vessel Code, which is incorporated into NRC regulations by 10 CFR 50.55a, "Codes and Standards," currently specifies that inspections of the RPV head need only include a visual check for leakage on the insulated surface or surrounding area. These inspections may not detect small amounts of leakage from an RPV head penetration with circumferential or axial cracks. NRC Order EA-03-009 was issued in February 11, 2003, to establish required inspections of RPV heads and associated penetration nozzles at PWRs. These requirements are necessary to provide reasonable assurance that plant operations did not pose an undue risk to the public health and safety. The requirements of that Order were expected to remain in effect pending long-term resolution of RPV head penetration inspection requirements, which is expected to involve changes to the NRC regulations, specifically 10 CFR 50.55a.

Bare metal visual (BMV) examinations of the Wolf Creek Generating Station RPV head had been performed in Refueling Outage RF-13. No indication of leakage was detected in those examinations. Based on no indications of leakage, BMV inspections were not performed in Refueling Outage 14.

In response to NRC Order EA-03-009, Wolf Creek Generating Station performed a BMV and nonvisual nondestructive examination (NDE) of the RPV head and penetration nozzles during Refueling Outage RF-15. The following paragraphs describe the inspection scope and findings related to the examination.

a. Inspection Scope

From October 12-18, 2006, the inspectors conducted an evaluation and assessment of the RPV head and vessel head penetration examinations performed by Wolf Creek Generating Station. During the inspection, the inspectors performed the following actions.

- A review of the susceptibility ranking, including the effective degradation years calculation.
- Independently observe (via videotape if available and if direct observation of the head is not possible) the RPV head BMV and NDE examinations.
- Independent review and report on the condition of the RPV.
- A report on anomalies, deficiencies, and discrepancies identified with the associated structures or the examination process when such problems are judged to be significant enough to potentially impede the examination process.
- A review of the scope of Wolf Creek's plan to examine the pressure-retaining components above the RPV head to ensure that all possible sources of boric acid leakage are included, that examination would be effective in identifying boric acid leakage in this area, and that appropriate actions are implemented should boron deposits be identified on the RPV head or related insulation.
- A review of the results of the examination to ensure that appropriate actions have been taken in response to identified boron deposits on the RPV head or related insulation.

The inspectors observed approximately 20 percent of the RPV head and vessel head penetration nozzles using direct observation and videotapes of the examination (16 of 78 nozzles). Additionally, the inspectors visited the area of the RPV head to observe its general condition and the movement of insulation to accommodate the examination.

b. Findings

No findings of significance were identified. This completes the inspection requirements for the first review of Temporary Instruction 2515/150.

General

Wolf Creek performed BMV and NDE examinations of the reactor vessel head on October 12-18, 2006, and they did not identify any evidence of primary water leakage from cracks in the RPV head and vessel head penetration nozzles. The inspectors reviewed 20 percent of the RPV head and vessel head penetration nozzles and verified the absence of boric acid deposits that could indicate PWSCC on the RPV head.

Susceptibility Ranking

The inspectors reviewed the bases for the type of examinations performed by Wolf Creek. Wolf Creek determined that the effective degradation period for the reactor vessel head was 4 years, as described in Procedure STS PE-04E, “RPV Head Bare Metal Inspection,” Revision 1. The inspectors reviewed the calculation, verified that the appropriate plant-specific information was used as input to the calculation, and confirmed that the effective degradation period was correct. As a result of the calculated effective degradation years, and absence of previous PWSCC, Wolf Creek was a low susceptible reactor. NRC Order EA-03-009 requires that low susceptible plants perform at least a BMV examination every three refueling outages. At least once over the course of every four refueling outages, low susceptible plants are to perform a BMV examination and a nonvisual exam, such as ultrasonic, eddy current, or dye penetrant tests. The inspectors determined that the BMV and NDE examination of the reactor vessel head met the inspection requirements of NRC Order EA-03-009.

Volumetric Examinations

Specific details of the NDE data gathering and analysis methods used were considered proprietary. Therefore, details of the techniques, data, and analyses are not discussed in this report.

The inspectors concluded that Wolf Creek performed a 100 percent NDE examination of the vessel head penetration nozzles. The inspectors observed that personnel training, examination equipment, and procedures were adequate. The inspectors reviewed the examiner qualifications of two Level II and one Level III eddy current examiners, and three Level II and two Level III ultrasonic examiners and verified the examiners were qualified to ASME certification standards. It was also noted that the examiners performing the examination had performed BMV examinations for other Westinghouse PWR plants.

The inspectors reviewed the applicable surface and volumetric examination requirements and concluded that the techniques were able to identify indications in the vessel upper head penetrations. The inspectors reviewed the acceptance criteria and determined that the acceptance criteria met the flaw evaluation guidelines. Plant specific procedures stated that any relevant indication exceeding 0.170 inch shall be reported. There were no indications that met the criteria.

The inspectors concluded that examiners encountered no impediments, such as debris, boric acid deposits, insulation, or other obstacles, that impacted the effective examination of the RPV head.

Bare Metal Visual Examination

The inspectors concluded that Wolf Creek performed a 100 percent BMV examination of the reactor vessel head and vessel head penetration nozzles. The inspectors observed that personnel training, examination equipment, and procedures were adequate to detect small boron deposits that could arise from circumferential or axial PWSSC of the vessel head penetration nozzles. Wolf Creek used a robotic crawler with a front- and rear-mounted camera to perform the examination. The cameras had the capability to adjust lighting and focus and articulate the view angle up and down to view the control rod drive and reactor vessel head vent nozzles. The crawler and video probe cameras were able to show, with clarity, letters that were as small as 0.02 inches in height (i.e., Jaeger J-1 images). The reactor vessel head insulation package was not in direct contact with the head, which allowed access to the entire head. Wolf Creek was able to inspect completely around all of the nozzles and detect any small amounts of boron. The BMV examination was videotaped, and the inspectors used the videotape, as well as direct observation, to verify the absence of boric acid deposits that would be indicative of PWSCC. There were no portions of the head obscured due to debris, preexisting leaks, or other obstructions. The inspectors also determined that the inspection was performed in accordance with licensee Procedure STS PE-040E, "RPV Head Bare Metal Inspection," Revision 1. The inspectors did not identify any active boric acid leaks on or other concerns with the condition of the vessel head.

The inspectors reviewed the training and qualification of the personnel performing the BMV examination. Persons performing the inspection held a VT-2 Level 2 or Level 3 qualification. Persons performing the BMV examination had performed the previous BMV examinations for other Westinghouse PWR plants. The inspectors observed the examiner's identification and disposition of visual data and found their disposition to be appropriate.

The inspectors verified the adequacy of Procedure STS PE-040E, "RPV Head Bare Metal Inspection," Revision 1. The inspectors verified that the procedure provided adequate guidance and examination criteria. Specifically, the procedure provided guidance for complete reactor vessel head examination, including a documentation and tracking mechanism to ensure all vessel head penetrations are examined. The procedure also provided clear standards and acceptance criteria for which the examiners had training and experience. The inspectors observed that the procedure contained guidance for dispositioning boric acid deposits and reporting of anomalies. During the examination, the inspectors noted that personnel were following the procedural guidance and tracking progress according to the procedure.

As stated above, the inspectors noted the absence of boric acid leakage from the vessel head penetrations. Additionally, the inspectors also noted that there were no boric acid deposits on the head or insulation as a result of leakage from above the reactor vessel head. The inspectors did not identify any significant differences between the condition of the head in this inspection and the bare metal examination during the 2005 refueling outage. In the 2005 outage, the licensee removed portions of the vessel head insulation during which the inspectors conducted a visual inspection of the vessel head and did not identify any issues.

The inspectors concluded that examiners encountered no impediments, such as debris, boric acid deposits, insulation, or other obstacles that impacted the effective examination of the RPV head.

.2 (Closed) Temporary Instruction 2515/166, Pressurizer Water Reactor Containment Sump Blockage

a. Inspection Scope

The inspectors reviewed Wolf Creek's implementation of plant modifications and procedure changes committed to in their response to Generic Letter 2004-02.

The inspectors observed installation of the containment recirculation sump strainers, debris barriers and interceptors, and addition of caps to the top of vertically mounted pipe supports. In addition, the inspectors verified that Wolf Creek has implemented specific procedure changes to control tags, labels, tape, and other objects inside the containment building. At the time of the exit meeting, Wolf Creek was in the final stages of implementing a containment coatings assessment program, a latent debris assessment program, and a containment strainer inspection program.

At the time of the inspection, industry testing for chemical effects on containment recirculation sumps was not complete. Since the testing was not complete, Wolf Creek evaluated the new recirculation sump modifications to the original design basis, Regulatory Guide 1.82, Revision 0. The inspectors verified that the design meets the original design basis. The Office of Nuclear Reactor Regulation will determine the adequacy of the modification in regard to Generic Safety Issue-191, " Assessment of Debris Accumulation on PWR Sump Performance."

b. Findings

No findings of significance were identified. This completes the inspection requirements for this Temporary Instruction.

4OA6 Meetings, Including Exit

On October 20, 2006, the inspector presented the occupational radiation safety inspection results to Mr. R. Muench, President and Chief Executive Officer, and other members of the licensee's staff who acknowledged the findings. The inspector confirmed that proprietary information was not provided or examined during the inspection.

On October 26, the inspectors presented the results of the inservice inspection and completion of Temporary Instruction 2515/166 inspections to Mr. S. E. Hedges, Vice President of Operations & Plant Manager, and other members of licensee management. Licensee management acknowledged the inspection findings. The inspectors identified that they had reviewed proprietary information but had returned it to licensee personnel.

On December 1, 2006, the inspector presented the emergency preparedness inspection results to Mr. R. A. Muench, President and Chief Executive Officer, and other members of his staff who acknowledged the findings. The inspector confirmed that proprietary information was not provided or examined during the inspection.

On January 9, 2007, the resident inspectors presented the inspection results to Mr. S. E. Hedges and other members of licensee management. Licensee management acknowledged the inspection findings. The inspectors identified that they had reviewed proprietary information but had returned it to licensee personnel.

4OA7 Licensee-Identified Violations

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

- TS 3.9.4, containment penetrations, requires, in part, that during core alterations each penetration providing direct access from the containment atmosphere to the outside atmosphere be closed by a manual or automatic isolation valve, blind flange, or equivalent but may be unisolated under administrative controls. Contrary to this requirement, during core alterations on October 24, 2006, the licensee failed to adequately control Valve ENHV-0007, containment recirculation sump suction for containment spray Pump B (containment penetration 13), resulting in the valve being opened intermittently for testing. Opening Valve ENHV-0007 provided a direct path from the containment atmosphere to the outside atmosphere. The licensee entered this item in its corrective action program as Condition Report 2006-002804. This finding was determined to be of very low safety significance because it occurred in a shutdown condition when the refueling cavity water level is at or above the minimum level required for movement of irradiated fuel assemblies within containment as defined by TS.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

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

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000482/2006005-01	NCV	Failure to implement the reactor vessel closure head installation procedure (Section 1R20)
05000482/2006005-02	NCV	Failure to provide instructions to workers (Section 2OS1)
05000482/2006-004-00	LER	Failure to Maintain Closure of Containment Penetrations During Fuel Movement (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

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

1R04 Equipment Alignment (71111.04)
CKL KJ-121, “Diesel Generator NE01 and NE02 Valve Checklist,” Revision 27

Checklist CKL EF-120, Essential Service Water Valve, Breaker and Switch Lineup, Revision 40

Drawing SK-M-13EF06, Piping Isometric Essential Service Water SYS. Aux. BLD. A & B Train Supply and Return, Revision E

Drawing M-13EF02, Piping Isometric Essential Service Water SYS Aux. Bldg. A Train Supply, Revision 9

Wolf Creek Generating Station USAR, Revision 19

1R05 Fire Protection (71111.05)
AP 10-102, “Control of Combustible Materials,” Revision 10
AP 10-106, “Fire Preplans,” Revision 3
AP 21B-003, “Control of Temporary Equipment,” Revision 5

Procedure E-1F9905, Fire Hazard Analysis, Revision 0
Condition Report 2006-002920

1R06 Flood Protection Measures (71111.06)
AE-02, “Main Feedwater Line Break,” Revision 1
LF-FH-002, “Auxiliary Building, Area 5 Drainage,” Revision 1
Wolf Creek Generating Station USAR, Revision 19

1R08 Inservice Inspection Activities (71111.08)
Nondestructive Examination Certifications
UT: 3 Level III
3 Level II

ET: 4 Level III
4 Level II

PT: 1 Level III
3 Level II

MT: 1 Level III
3 Level II

Welding Procedure Specifications
WPS1-0101S01, SMAW for P1 materials, Revision 7
WPS1-0505S01, SMAW of P5A to P5A, Revision 7
WPS1-0505T01, GTAW of P5A to P5A, Revision 7

Procedures
WDI-ET-002, Intraspect Eddy Current Inspection of Vessel Head Penetration J-Welds and Tube OD Surfaces, Revision 8

WDI-ET-003, Eddy Current Imaging Procedure for Inspection of Reactor Vessel Head Penetrations, Revision 10

WDI-ET-004, Intraspect Eddy Current Analysis Guidelines, Revision 10

WDI-ET-008, Intraspect Eddy Current Imaging Procedure for Inspection of Reactor Vessel Head Penetrations with Gap Scanner or UT/ET Neptune, Revision 8

WDI-SSP-1059-SAP, Reactor Vessel Head Penetration NDE Tool Operation for Wolf Creek, Revision 0

PDI-UT-10, PDI Generic Procedure for the Ultrasonic Examination of Dissimilar Metal Welds, Revision C

PDI-UT-11, PDI Generic Procedure for the Ultrasonic Detection and Sizing of Reactor Pressure Vessel Nozzle to Shell Welds and Nozzle Inner Radius, Revision B

MT-7, Magnetic Particle Examination, Revision 3

UT-2, Ultrasonic Examination of Vessel Welds and Adjacent Base Metal, Revision 27

WDI-UT-010, Intraspect Ultrasonic Procedure for Inspection of Reactor Vessel Head Penetrations, Time of Flight Ultrasonic, Longitudinal Wave and Shear Wave, Revision 13

Steam Generator Documents
AP 29A-003, Steam Generator Management, Revision 9

MRS 2.4.2 GEN-35, Eddy Current Inspection of Preservice and Inservice Heat Exchanger Tubing, Revision 12

I-ENG-023, Steam Generator Data Analysis Guidelines, Revision 6

SG-SGDA-06-25, SG Degradation Assessment for Wolf Creek RF15 Refueling Outage October 2006

SG-SGDA-05-21, Wolf Creek RF14 Condition Monitoring Assessment and Operational Assessment 2

Boric Acid Corrosion Control
AP16-F001, Boric Acid Corrosion Control Program, Revision 4

AI16-F001, Evaluation of Boric Acid Leakage, Revision 4

AI16-F002, Boric Acid Leakage Management, Revision 3

Boric Acid Corrosion Control Program 2006 3rd Quarter Inspection/Monitoring Report 3rd Quarter 2006

AIF 16F-001-01, CCP A Disch Iso, Revision 1

AIF 16F-001-01, Boric Acid Xfer Pmp Disch, Revision 2

AIF 16F-001-01, PZR/RCS Sample Line Inside Ctmt Test Conn Iso, Revision 1

OE SJ-06-002, Incore Seal Table, Revision 0

AIF 16F-001-01, Fuel Pool Clean-up Pump 2A Disch Check Vlv, Revision 1

AIF 16F-001-01, Recycle Evap Feed Pump B Suction Pp Iso, Revision 1

AIF 16F-001-01, Letdown Reheat HX From CVCS Inlet Iso, Revision 0

APF 28-001-01, PZR/RCS Sample Line Test Conn/Vent Vlv, Revision 7

Miscellaneous

MRP-139, Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guideline, July 14, 2005

ET 06-0021, Wolf Creek Nuclear Operating Corporation 10 CFR 50.55a Request Number I3R-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 10 CFR 50.55a Request Number I3R-05, August 4, 2006

420-A09, Owner's Specification for Full Structural Weld Overlays for Pressurizer Nozzle to Safe-End Dissimilar Metal Welds, Revision 3

BB-S-024, Calculation to Capture Vendor Report WCAP-16625-P, Wolf Creek Pressurizer Spray/Safety/Relief, and Surge Nozzles Structural Weld Overlay Qualification, Revision 0

M-713A-00002, Welding Procedure Specification WPS 3-8/52-TB MC-GTAW-N638, Revision, September 20, 2006

Work Order 05-277921-000, Wolf Creek Owner Hold Points Associated with the Scope of Work to be Performed by Westinghouse, PCI, and Wesdyne to Complete the Pressurizer Nozzle Weld Overlays

Change Package 11977 Pressurizer (TBB03) DM Weld Overlay, Revision 0

1R12 Maintenance Effectiveness (71111.12)

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

M-018-01033, "Temperature Regulator I-1285-S Series," Revision 0

MPM KJ-004, "Robert Shaw Model 1285 Temperature Control Valve," Revision 1

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

PIR 2004-2159

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

AP 22C-003, Operational Risk Assessment Program, Revision 11
2006 RE-15 Outage risk profile

WO 02-241622-000	WO 04-265608-000	WO 05-272252-000
WO 02-241960-000	WO 04-046371-000	WO 06-001463-000
WO 02 241961-000	WO 05-050922-002	WO 06-001752-000
WO 02-241962-000	WO 05-050922-003	WO 06-054501-000
WO 02-241973-000	WO 05-050922-004	WO 06-057244-000
WO 02-241974-000	WO 05-051659-000	WO 06-289298-000

Work Request 05255-89
Work Request 06099-92

PIR 2004-2159	PIR 2004-2286	PIR 2004-3195
PIR 2004-2160	PIR 2004-3135	PIR 2006-0383

1R15 Operability Evaluations (71111.15)

AP 26C-004, Technical Specification Operability, Revision 14
AP 28-001, "Operability Evaluations," Revision 14
AP 28-011, "Resolving Deficiencies Impacting SSCs," Revision 1
Operability Evaluation OE XX-06-002, Revision 1
Operability Evaluation OE XX-06-003, Revision 1
Reportability Evaluation Request 2006-014
Reportability Evaluation Request 2006-018
PIR 2006-002465
Condition Report 2006-002977

1R19 Postmaintenance Testing (71111.19)

Flowserve Pump Division letter to Wolf Creek Nuclear Operating Company dated October 17, 2006, Subject 37 KXH VCT (RLCA02142) ESW Pump Seismic Restraints

AP 16E-002, Post Maintenance Testing Development, Revision 5

SYS EF-200, Operation of the ESW System, Revision 26

SYS KJ-123, Post Maintenance Run of Emergency Diesel Generator A, Revision 35

STS KJ-011B, DG NE02 24 Hour Run, Revision 16

STS KJ-015A, Manual/Auto Fast Start, Sync and Loading of EDG NE01, Revision 21

STS KJ-001A, Integrated D/G and Safeguards Actuation Test - Train A, Revision 31

Work Order 06-284846-001

1R20 Outage (71111.20)

ALARA Review Package RWP 06-1102, February 16, 2006

ALARA Review Package RWP 06-6031, February 16, 2006

Calculation BB-FW-011, “WCNOC, Reactor Vessel Head Drop Analysis,” Revision 1

CR 2003-003589 and CRs 2006-002468, -003029, -003045, -003055, -003187, -003342, -003141, and -002804

Drawing EID-0004, “Pool Parameters,” Revision 0

Drawing M-13BB01, “Reactor Coolant System Primary Loop,” Revision 7

Drawing M-189-50BB-02-02, “Reactor Coolant Pressurizer Safety Valve Lines,” Revision 0

Drawing M-724-00793, “Nozzle Type Relief Valve,” Revision 0

Drawing SFS-WC/CW-GA-00, “Sure-Flow Strainer General Notes and Information,” Revision 3

FHP 02-007B, “Reactor Vessel Closure Head Installation,” Revision 5

GEN 00-002, “Cold Shutdown to Hot Standby,” Revision 64

Mode Change Checklist - Mode 3 to Mode 2, Performed November 9, 2006

OTSC 06-0119, “On the Spot Change to STS KJ-001B, “Integrated D/G and Safeguards Actuation Test - Train B,” dated October 27, 2006

Reportability Evaluation Request 2003-028

RXE 03-001, “Incore Data Reduction and Analysis,” Revision 15

STS AL-102, “MDAFW Pump B Inservice Pump Test,” Revision 34

STS BB-204, “RCS Inservice Valve Test,” Revision 11

STS KJ-001B, “Integrated D/G and Safeguards Actuation Test - Train B,” Revision 31

USAR Change Request 90-110 and 97-181

WCAP-9198, “Westinghouse Reactor Vessel Head Drop Analysis”

WOs 03-257698-000, WO 03-257698-001, and WO 06-272812-000

1R23 Temporary Plant Modifications (71111.23)

Control room temporary modifications opened and closed with in the last 6 weeks
Work Order 05-275664-001 and 002
Work Package 05-275708-001

1EP2 Alert Notification System Testing (71114.02)

AP 17C-024, "Emergency Planning Responsibilities," Revision 6

1EP3 Emergency Response Organization Augmentation Testing (71114.03)

Evaluation Reports for Callout Surveillances Conducted

June 14, 2004
September 16, 2004
March 3, 2005
June 2, 2005
September 13, 2005
December 6, 2005
February 20, 2006
June 7, 2006,
August 29, 2006

Drive-In Drill Evaluation Report, Drill Conducted October 29, 2004

Emergency Response Organization Rosters for November 2005, April 2006, and September 2006

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

NRC Bulletin 2005-002, "Emergency Preparedness and Response Actions for Security-Based Events"

NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1

NEI 99-01, "Methodology for Development of Emergency Action Levels," Revision 2
AP 06-002, "Radiological Emergency Response Plan," Revision 8

APF 06-02-001, "Emergency Action Levels," Revision 8

EPP 06-06, "Protective Action Recommendations," Revision 4

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies (71114.05)
AP 20A-003, "QA Audit Requirements, Frequencies, and Scheduling," Revision 15
EPP 06-002, "Technical Support Center Operations," Revision 18
EPP 06-009, "Drill and Exercise Requirements," Revision 4
EPF 06-018-11, "Technical Support Center Inventory Checklist," Revision 7
Audit K-616, Emergency preparedness, September 19, 2004
Audit K-636, Emergency preparedness, November 14, 2005

Evaluation Reports for Drills Conducted

March 22, 2004
April 1, 2004
August 26, 2004
October 6, 2004
October 18, 2004
January 19, 2005
June 29, 2005
July 12, 2005
July 28, 2005
October 13, 2005
October 26, 2005
November 9, 2005
April 20, 2006
May 4, 2006
June 12, 2006
July 20, 2006, and,
July 26, 2006

Performance Improvement Requests
2004-1852, 2004-1922, 2004-2110, 2004-2234, 2004-2763, and 2004-3488

2005-0113, 2005-0119, 2005-0265, 2005-0617, 2005-0925, 2005-1783, 2005-2104, 2005-2255, and 2005-2363

Condition Reports
2006-0181, 2006-0232, 2006-0931, 2006-1236, 2006-2919, 2006-3026, and 2006-3426

1EP6 Drill Evaluation (71114.06)

Emergency Planning Drill 06-SA-02

2OS1: Access Controls to Radiologically Significant Areas (71121.01)

Condition Reports

2006-001128, 2006-001268, 2006-02467, 2006-002566, 2006-002569, 2006-002579, and 2006-002682

Radiation Work Permits

061102, Incore tunnel/cavity inspections, Revision 0
063051, Under reactor vessel head ISI full body entry, Revision 1
063220, Primary steam generator Eddy Current, Revision 0
063230, Primary steam generator FME nozzle covers, Revision 0
064007, Excore dosimetry replacement, Revision 0
064065, Plant modifications at the containment recirculation sumps, Revision 0
064200, Secondary side steam generator work, Revision 1
064208, Reactor coolant pump seal replacement, Revision 1

Procedures

AP 25A-001	Radiation protection manual, Revision 11
AP 25A-200	Access to locked high or very high radiation areas, Revision 14
RPP 02-205	Radiological survey frequency requirements, Revision 10
RPP 02-215	Posting of radiologically controlled areas, Revision 22A
RPP 02-405	RCA access control, Revision 12
RPP 02-210	Radiation survey methods, Revision 26
RPP 03-205	DAC-hour tracking, Revision 12
RPP 03-210	Internal exposure calculations and evaluations, Revision 12

2OS2: ALARA Planning and Controls (71121.02)

Audits and Assessments

Assessment 44, Radiation Dose Control-Dosimetry Program, October 5, 2006

Condition Reports

2006-002566, 2006-002664, 2006-002277, 2006-002279, 2006-002525, 2006-002535, 2006-002543, and 2006-002567

Procedures

RPP 02-105	RWP, Revision 24
AP 25B-100	Radiation worker guidelines, Revision 26
RPP 03-505	Selection of protective clothing, Revision 9

Radiation Work Permits

062201, Incore tunnel maintenance
064075, RHR Pump A room cooler replacement
066060, Refueling team activities
06-0029, Resin transfer to primary spent resin storage tank
06-0036, Containment access during power Modes 1, 2, and 3 for maintenance functions

Performance Improvement Requests

2005-2815, 2005-2965, 2005-3113, 2005-3066, 2006-0077, 2006-0212, 2006-0580, 2005-1870, 2005-2044, 2005-2530, 2005-2795, 2006-0154, and 2006-0614

Procedures

AP-25B-100, Radiation worker guidelines, Revision 26

AP 25B-300, Radiation work permit program, Revision 14

RPP 02-105, Radiation work [permit, Revision 24

RPP 02-305, Personnel Surveys/decontamination, Revision 15

RPP 03-122, Skin Dose Calculations, Revision 9

RPP 03-405, Exposure history files, Revision 15

AP 25A-001, Radiation protection manual, Revision 11

AP-25B-100, Radiation worker guidelines, Revision 26

RPP 02-210, Radiation surveys, Revision 26

RPP 02-405, RCA access control, Revision 12

SYS HC-201, Sluice CVCS mixed bed Demineralizer A using primary spent resin sluice pump, Revision 18

Miscellaneous

EAD dose and peak dose rate reports
Radiation Work Permit 06-0029 prejob briefing form
Self-Assessment Report SEL 05-012

ALARA Review Packages and Radiation Work Permits

05-0020, <500 Rem/hr filter media processing activities for radioactive waste preparation

05-0036, Containment access during power Modes 1, 2, & 3 for maintenance functions

05-0100, Technical Specification equipment outage

06-0020, <500 Rem/hr filter media processing activities for radioactive waste preparation

06-0029, Resin transfer to primary spent resin storage tank

06-0036, Containment access during power Modes 1, 2, & 3 for maintenance functions

06-0050, Contaminated system breach of valves, pumps, piping, and strainers for maintenance activities

06-0070, Troubleshoot/repair/plant modifications on the incore detector drive system in the seal table/flux map area during power operation Modes 1 & 2

06-0100, TS equipment outage

Audits and Self-Assessments
Quality Assurance Audit Report K-639
Self-Assessment Report SEL 05-012
Self-Assessment Report SEL 05-022

Miscellaneous
2000 - 2005 Wolf Creek Exposure History - Summary
ALARA Report Refuel 14
Current collective exposure report
EAD dose and peak dose rate reports
Declared pregnant worker records
Radiation Work Permit 06-0029 prejob briefing form
Steam generator channel head surveys

4OA1 Performance Indicator Verification (71151)

Condition Report

2006-000393

Procedures

AP 26A-007, NRC Performance Indicators, Revision 5
NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 4

4OA2 Identification and Resolution of Problems (71152)

CR 2006-1763
CR 2006-2282
CR 2006-2370
CR 2006-2371
CR 2006-2388
CR 2006-2390
CR 2006-2392
CR 2006-2394
CR 2006-2399
CR 2006-2401
CR 2006-2402
CR 2006-2405
CR 2006-2446
CR 2006-2454
CR 2006-2468
CR 2006-2470
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CR 2006-2884
CR 2006-2893
CR 2006-2945
CR 2006-2951
CR 2006-2952
CR 2006-2953
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CR 2006-3055
CR 2006-3067
CR 2006-3088
CR 2006-3105
CR 2006-3118
CR 2006-3123
CR 2006-3126

CR 2006-3129
CR 2006-3141
CR 2006-3154
CR 2006-3166
CR 2006-3167
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CR 2006-3187
CR 2006-3195
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CR 2006-3201
CR 2006-3215
CR 2006-3232
CR 2006-3242
CR 2006-3280
CR 2006-3281
CR 2006-3331
CR 2006-3524
CR 2006-3527
CR 2006-3528
CR 2006-3535
CR 2006-3638
CR 2006-3686
CR 2006-3691
CR 2006-3703
CR 2006-3704
CR 2006-3716
CR 2006-3712
CR 2006-3721
CR 2006-3772
CR 2006-3774
CR 2006-3775
CR 2006-3800

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

Procedure AP 15C-002, “Procedure Use and Adherence,” Revision 22
Procedure AP 21-001, “Conduct of Operations,” Revision 36
LER 2006-004-00, Failure to maintain closure of containment penetrations during fuel movement
Condition Report 2006-002804
Condition Report 2006-003199
Flux Maps at 30, 49, 90, 100 percent reactor power
WCNOC Hour No. 1, No. 2 Refuel 15 core map gap verification dated October 27, 2006

4OA5 Other

Calculations
32-9024157, Wolf Creek Recirculation Sump Level Uncertainty, Revision 0

32-9024157, Wolf Creek Recirculation Sump Level Uncertainty, Revision 1

J-EJ10-000-CN-001, Containment BLDG Recirc Sumps and Normal Sump Level Sensors, Revision 0

EJ-30, RHR Pumps A & B NPSH, Revision 2

EJ-30-002-CN002, RHR Pumps A & B NPSH, Revision 2

EJ-30-002-CN001, RHR Pumps A & B NPSH, Revision 2

Design Change Packages

12012, Tube Steel Caps Inside Containment, Revision 0
12037, Debris Barriers in Containment Loop Access Doors A and D, Revision 3
12038, Containment Sump Level Indication, Revision 0
12069, Containment Recirculation Sump Strainers, Revision 4

Calculation

32-9024157, Wolf Creek Recirculation Sump Level Uncertainty, Revision 0

32-9024157, Wolf Creek Recirculation Sump Level Uncertainty, Revision 1

J-EJ10-000-CN-001, Containment BLDG Recirc Sumps and Normal Sump Level Sensors , Revision 0

EJ-30 RHR Pumps A & B NPSH, Revision 2

LIST OF ACRONYMS

ALARA	as low as is reasonably achievable
ASME	American Society of Mechanical Engineers
BMV	bare metal visual
EPRI	Electric Power Research Institute
FEMA	Federal Emergency Management Agency
LER	licensee event report
NCV	noncited violation
NDE	nondestructive examination
NRC	Nuclear Regulatory Commission
PIR	performance improvement request
PWR	pressurized-water reactor
PWSCC	pressurized water stress corrosion cracking
RPV	reactor pressure vessel
SSC	structure, system, and component
TS	Technical Specification
USAR	Updated Safety Analysis Report