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

February 3, 2004

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 05000482/2003006

Dear Mr. Muench:

On December 31, 2003, the NRC completed an inspection at your Wolf Creek Generating Station. The enclosed integrated report documents the inspection findings which were discussed on January 9, 2004, with you and members of your staff.

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

This report documents one NRC-identified finding concerning a missing fire barrier in the main steam enclosure. This finding has potential safety significance greater than very low significance. The finding did present an immediate safety concern, which was immediately corrected. In addition, the NRC identified two issues that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has also determined that violations are associated with these two issues. These violations are being treated as a noncited violations, consistent with Section VI.A of the Enforcement Policy. The noncited violations are described in the subject inspection report. Additionally, the licensee identified violations which were determined to be of very low safety significance and are listed in Section 4OA7 of this report. If you contest these noncited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; 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 facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) 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/

David N. Graves, Chief
Project Branch B
Division of Reactor Projects

Docket: 50-482
License: NPF-42

Enclosure:
NRC Inspection Report 05000482/2003006
w/attachment: Supplement Information

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RIV:SRI/DRP/B	RI/DRP/B	PE:DRP/B	SPE:DRP/B	DRS/OB
FLBrush:sa	TBRhoades	RBCohen	RAKopriva	JFDrake
E - DNGraves	***E - Graves***	***/RA/***	***/RA/***	***/RA/***
1/30/04	1/30/04	1/30/04	1/30/04	2/2/04
C:DRS/PSB	C:DRS/EMB	C:DRP/B		
TWPruett	CSMarschall	DNGraves		
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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-482
License:	NPF-42
Report:	05000482/2003006
Licensee:	Wolf Creek Nuclear Operating Corporation
	Wolf Creek Generating Station
Location:	1550 Oxen Lane NE Burlington, Kansas
Dates:	October 5 through December 31, 2003
Inspectors:	F. L. Brush, Senior Resident Inspector T. B. Rhoades, Resident Inspector R. A. Kopriva, Senior Project Engineer M. O. Miller, Resident Inspector B. W. Henderson, Reactor Engineer R. Cohen, Project Engineer D. R. Carter, Health Physicist M. P. Shannon, Senior Health Physicist J. F. Drake, Operations Engineer L. E. Ellershaw, Senior Reactor Inspector G. B. Miller, Reactor Inspector
Approved By:	D. N. Graves, Chief, Project Branch B

ATTACHMENT: Supplemental Information

Enclosure

CONTENTS

SUMMARY OF FINDINGS

IR 05000482/2003006; 10/4/03 - 12/31/03; Wolf Creek Generating Station. Fire Protection, Access Control to Radiologically Significant Areas, ALARA Planning and Controls

The report covered the period of resident inspection and announced inspections by nine Region IV inspectors. Two Green noncited violations and one unresolved item with potential safety significance greater than Green were identified. The significance of issues is indicated by their color (Green, White, Yellow, or Red) and was determined by the Significance Determination Process in Inspection Manual Chapter 0609. Findings for which the significance determination process does not apply are indicated by the severity level of the applicable violation. 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

- TBD. The inspectors identified a violation of License Condition 2.C(5)(a) of the Wolf Creek Generating Station Facility Operating License having a potential safety significance greater than very low significance because approximately 20 inches of fire barrier between the main steam enclosure and auxiliary feedwater system flow control valve rooms was missing.

  The finding is unresolved pending completion of a significance determination. The finding is greater than minor because it is associated with a degraded fire protection fire barrier and affected the Reactor Safety Mitigating System Cornerstone. The finding was determined to have potential safety significance greater than very low significance because all the main steam atmospheric relief and auxiliary feedwater system flow control valves could be affected by a fire in either area (Section 1RO5).

Cornerstone: Occupational Radiation Safety (OS)

- Green. The inspector identified a noncited violation of 10 CFR 20.1602 because the licensee failed to institute measures to ensure that an individual was not able to gain unauthorized access to a very high radiation area. Specifically, on October 28, 2003, the inspector observed that the area surrounding a locked ladder leading down to the reactor under-vessel area, a very high radiation area, was not provided with a physical barrier that completely enclosed the area. Radiation levels at the bottom of the ladder, one meter away from the withdrawn in-core instrument thimbles, were approximately 640 RADs per hour. An individual could have climbed over the handrail and climbed down the outside of the ladder using the fall protection cage. The finding is in the licensee's corrective action program as Performance Improvement Request 2003-3220.

The finding was greater than minor because it affected the Occupational Radiation Safety cornerstone objective to ensure adequate protection of worker health and safety from exposure to radiation and the finding was associated with the cornerstone attribute (program and process). The finding involved an individual's potential for unplanned or unintended dose. When processed through the Occupational Radiation Safety Significance Determination Process, the finding was determined to be of very low safety significance because the finding was not associated with as low as is reasonably achievable planning or work controls, there was no overexposure nor a substantial potential for overexposure, and the ability to assess dose was not compromised (Section 2SO1).

- Green. The inspectors identified four examples of a noncited violation of 10 CFR 20.1501(a), because the licensee failed to perform required radiological surveys to ensure compliance with 10 CFR 20.1204(a) and 10 CFR 20.1902(b). On October 19, 2003, the licensee did not perform adequate surveys to assess changes in radiological conditions during chemical cleaning of the reactor coolant system. On October 22, 2003, the licensee did not perform an adequate survey of the workers' breathing zone while decontaminating the reactor cavity seal ring. These findings are in the licensee's corrective action program as Performance Improvement Requests 2003-3069 and -3136, respectively.

  The finding is greater than minor because it affected the Occupational Radiation Safety cornerstone objective to ensure adequate protection of worker health and safety from exposure to radiation and the finding is associated with the cornerstone attribute (program and process). The finding involved an individual's potential for unplanned or unintended dose. When processed through the Occupational Radiation Safety Significance Determination Process, the finding was determined to be of very low safety significance because the finding was not associated with as low as is reasonably achievable planning or work controls, there was no overexposure or a substantial potential for overexposure, and the ability to assess dose was not compromised (Section 2SO2).

B. Licensee-Identified Violations

Violations of very low 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. The violations and corrective actions are listed in Sections 4OA7 and 4OA3 of this report.

REPORT DETAILS

Summary of Plant Status

The plant operated at essentially 100 percent power for the report period with the following exceptions. On October 18, 2003, the licensee opened the main generator output breaker to start the Refueling (RF) 13 outage. The outage ended on December 2 and the plant reached full power on December 4, 2003. The plant operated at essentially 100 percent power the remainder of the report period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather (71111.01)

a. Inspection Scope

On November 20, 2003, the inspectors completed a walkdown of plant systems and various buildings using licensee Procedure STN GP-001, “Plant Winterization,” Revision 31, to verify that the onset of cold weather would not affect mitigating systems. The inspectors also discussed aspects of cold weather preparations with licensee personnel. The inspectors also reviewed the following:

- Work Orders 01-225731-000 and 00-215832-000
- Technical Specifications
- Updated Safety Analysis Report

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

Partial Walkdowns

The inspectors performed one walkdown to verify equipment alignment and identify discrepancies that could impact redundant system operability. The inspectors used the system drawings and lineup checklists to perform the walkdowns. The inspectors also discussed the walkdown with various licensee personnel. The inspectors performed the following partial walkdown:

- Centrifugal charging Pump B during a centrifugal charging Pump A outage, October 9, 2003

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

Quarterly Fire Area Walkdowns

a. Inspection Scope

The inspectors toured the following six areas to assess the licensee's control of transient combustible materials, the material condition and lineup of fire detection and suppression systems, and the material condition of manual fire equipment and passive fire barriers. The licensee's fire preplans and fire hazards analysis report were used to identify important plant equipment, fire loading, detection and suppression equipment locations, and planned actions to respond to a fire in each of the plant areas selected. Compensatory measures for degraded equipment were evaluated for effectiveness.

- Component cooling water pump and heat exchanger rooms, December 4, 2003
- Control building 2016 foot level, November 14, 2003
- Emergency diesel Generator B room, October 30, 2003
- Main steam enclosure, December 12, 2003
- Spent fuel pool cooling Pump A and B rooms, December 18, 2003
- Turbine building 2033 foot level, October 17, 2003

b. Findings

Introduction

The inspectors identified a finding which is a violation of License Condition 2.C(5)(a) of the Wolf Creek Generating Station Facility Operating License in that the 3-hour fire barrier between the main steam enclosure and the auxiliary feedwater system flow control valve rooms was degraded. This finding has a potential of being of greater than very low safety significance. This is an unresolved item (URI) pending completion of the Significance Determination Process.

Description

The inspectors identified that approximately 20 inches of fire barrier between the main steam enclosure and the auxiliary feedwater system flow control valve rooms was missing. The fire barrier material was missing from the approximately 4-inch wide seismic gap between the reactor and auxiliary buildings. The fire barrier area also contained a significant amount of debris. The licensee determined that the fire barrier had been degraded since initial plant construction. The licensee immediately cleaned out the debris, placed fire barrier material in the gap, and wrote Performance Improvement Request (PIR) 2003-3704 to document the condition.

Analysis

The finding adversely impacted the fire barrier between the main steam enclosure and auxiliary feedwater system flow control valve rooms. The finding is greater than minor because it affected the reactor safety mitigating system cornerstone objective. The finding has the potential to be of greater than very low safety significance because all steam generator atmospheric relief valves and auxiliary feedwater flow control valves could be affected by a fire in either of the rooms.

Enforcement

License Condition 2.C(5)(a) of the Wolf Creek Generating Station Facility Operating License requires, in part, that the licensee implement and maintain in effect all provisions of the approved fire program. The fire protection program required a 3-hour fire barrier between the main steam enclosure and auxiliary feedwater area. Contrary to the above, approximately 20 inches of the fire barrier between the main steam enclosure and auxiliary feedwater system flow control valve rooms was missing. This is a violation of License Condition 2.C(5)(a) of the Wolf Creek Generating Station Facility Operating License. Pending determination of the finding's safety significance, this finding is identified as URI 50000482/2003-006-01, fire barrier in the main steam enclosure missing.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

On November 4, 2003, the inspector completed the observation and review of the as-found condition and eddy current testing results for component cooling water heat Exchanger B in accordance with Attachment 71111.07, heat sink performance. The inspector also reviewed a sample of past performance test data for the heat exchanger, the tube plugging criteria, and the number of tubes plugged. The inspector verified that there were no heat exchanger deficiencies which could mask degraded performance.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope

.1 Performance of Nondestructive Examination Activities Other than Steam Generator Tube Inspections

Inspection Procedure 71111.08 specified a review of two or three types of nondestructive examination activities be conducted: Volumetric (radiographic or ultrasonic), surface (magnetic particle or liquid penetrant), and visual (VT-1 to determine the surface condition of a part or component, VT-2 to locate evidence of leakage, and VT-3 to determine the general mechanical and structural condition of parts or components). The inspectors reviewed multiple examples of all three types, as noted in column three of the table below.

The inspection procedure also specified a review be conducted of one or two examinations from the previous outage with recordable indications that were accepted for continued service be conducted. The inspectors reviewed one such examination (weld repair on essential service water Valve EFV-0058).

The inspection procedure further specified that, if the licensee completed welding on the pressure boundary for ASME Code Class 1 or 2 systems since the beginning of the previous outage, then verification should be performed that acceptance and preservice examinations were done in accordance with the ASME Code for one to three welds. The inspectors verified one weld (chilled water horizontal heat Exchanger SGK-04B slip-on-flange fillet weld).

The inspection procedure also specified verification that one or two ASME Code Section XI repairs or replacements met ASME Code requirements. The inspectors verified one Section XI repair (replacement of seal weld on chemical and volume control Valve BGV-0496).

System	Component/Weld Identification	Examination Method
Reactor coolant	Pressurizer seismic support lug Welds TBB03-LUG-D-W and TBB03-LUG-B-W	Liquid penetrant
Chilled water	Horizontal heat Exchanger SGK-04B spool piece slip-on-flange connection fillet weld	Liquid penetrant and visual (VT-2)
Essential service water	Valve EFV-0058, repair welds to valve body, ASME Code Section XI repair	Visual (VT-1), radiography (3 shots), and liquid penetrant

System	Component/Weld Identification	Examination Method
Reactor coolant	RTD bypass lines on crossover leg of Loop A: Welds BB-06-F001, BB-06-PW6000, and HB-24-W5001	Ultrasonic
Chemical and volume control	Valve BGV-0496 seal weld	Liquid penetrant

The specific nondestructive examination reports associated with the above listed examinations are identified in the section labeled “Nondestructive Examination Test Reports” in the attachment to this report.

Finally, the inspection procedure specified verification that activities are performed in accordance with ASME Code requirements and that indications and defects, if present, were dispositioned in accordance with the ASME Code. The inspectors verified, through direct observation or record review, that ultrasonic, liquid penetrant, radiographic, and visual examinations of the above systems/ components were performed in accordance with the ASME Code. The inspectors determined that the correct nondestructive examination procedures were used, that examinations and conditions were as specified in the procedure, and that test instrumentation or equipment was properly calibrated within the allowable calibration period. Defects were not identified by the licensee during the inspector-observed examinations. Indications, however, were revealed by the examinations, compared against the ASME Code specified acceptance standards, and properly dispositioned.

The inspectors verified the nondestructive examination certifications of those personnel observed performing examinations or identified during review of completed examination packages.

.2 Steam Generator Tube Inspection Activities

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

The inspectors selected and reviewed the following acquisition technique sheets and their qualifying EPRI examination technique specification sheets to verify that the essential variables regarding flaw sizing accuracy had been identified and qualified through demonstration.

Acquisition technique sheet	EPRI's examination technique specification sheets
SAP-01-03	96001.1, 96004.3, 96005.2, 96008.1, and 96010.1
SAP-02-03	96001.1, 96004.3, 96005.2, 96008.1, and 96010.1
SAP-04-0	21409.1, 21410.1, 20510.1, 20511.1, and 22401.1
SAP-05-03	96910.1 and 21998.1
SAP-06-03	96511.1 and 96511.2
SAP-08-03	21409.1, 21410.1, 20510.1, and 20511.1

At the time of this inspection, the licensee had not identified conditions which warranted the need for conduct of in-situ pressure testing.

The inspection procedure specified comparing the estimated size and number of tube flaws detected during the current outage against the previous outage operational assessment predictions to assess the licensee's prediction capability. The inspectors reviewed Report SG-SGDA-03-19, "Steam Generator Degradation Assessment for Wolf Creek RF13 Refueling Outage - October 2003," Revision 2. The purposes of the report were to provide: (1) a comprehensive review and overall plan for detection and assessment of degradation to be addressed during RF13 and (2) predictions as to the type and extent of degradation expected to be found. At the time of the inspectors' review, the licensee had completed approximately 85 percent of the scheduled eddy current examinations (ET), the results of which appeared to be on track with the predictions identified in the report.

In addition, the inspectors reviewed Report SG-SGDA-03-18, "Wolf Creek RF 11 Condition Monitoring Assessment and Final Operational Assessment - March 2001," Revision 2, which evaluated and summarized the results of the 2001 RF steam generator inspection and testing activities for Steam Generators A and D. The purpose was to demonstrate that the structural and leakage integrity criteria were expected to be maintained throughout Cycles 12 and 13 for those two steam generators (which ended with the initiation of the current RF13).

The inspection procedure specified confirmation be made that the steam generator tube ET scope and expansion criteria meet Technical Specification requirements, EPRI guidelines, and commitments made to the NRC. The inspectors review determined that the steam generator tube ET scope and expansion criteria were being met.

The inspection procedure also specified that, if the licensee identified new degradation mechanisms, then verify that the licensee had fully enveloped the problem in an analysis and had taken appropriate corrective actions before plant startup. At the time of this inspection, no new degradation mechanisms had been identified.

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

The inspection procedure further required that repair processes being used were approved in the Technical Specifications for use at the site. At the time of this inspection, the licensee had not performed or used the designated Technical Specification-approved repair processes, thus there was no opportunity to observe implementation of any potential repairs (e.g., plugging operations) or in-situ pressure testing.

The inspection procedure also required confirmation that the Technical Specification plugging limit was being adhered to and determination whether depth sizing repair criteria were being applied for indications other than wear or axial primary water stress corrosion cracking in dented tube support plate intersections. At the time of this inspection, the licensee had not initiated any plugging or repair activities, thus the inspectors were unable to make this confirmation. The inspectors did determine, however, that the licensee, in response to Information Notice 2002-21, did account for crack-like indications in dented tube support plate intersections by including these parameters in their ET computer programming and the acquisition and analysis technique sheets. Further, the ET data analysts had been presented with specialized training associated with this type of indication.

The inspection procedure stated that, if steam generator leakage greater than 3 gallons per day was identified during operations or during postshutdown visual inspections of the tubesheet face, then assess whether the licensee had identified a reasonable cause and corrective actions for the leakage based on inspection results. The inspectors did not conduct any assessment because this condition did not exist.

The inspection procedure required confirmation that the ET probes and equipment were qualified for the expected types of tube degradation and assessment of the site-specific qualification of one or more techniques. The inspectors observed portions of ET performed on the following locations in Steam Generators A and D: full length, U-bends, special interest locations, hot-leg side between 3 inches above the top of tubesheet to 3 inches below the top of tubesheet, and cold-leg side dent locations. During these examinations, the inspectors verified that: (1) the probes appropriate for identifying the expected types of indications were being used, (2) probe position location verification was performed, (3) calibration requirements were adhered to, and (4) probe travel speed was in accordance with procedural requirements. The assessment of site-specific qualifications of the techniques being used, including a listing of the specific techniques and qualifications reviewed, is addressed and identified in the table above.

The inspection procedure specified that, if loose parts or foreign material on the secondary side of the steam generators were identified, assess the licensee's corrective actions. At the time of this inspection, no foreign material or loose parts had been identified on the secondary side.

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

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

Routine Maintenance Effectiveness Inspection

a. Inspection Scope

The inspectors reviewed the licensee's maintenance rule implementation for the following two structures, systems, or components to assess the effectiveness of maintenance efforts in accordance with 10 CFR 50.65.

- Auxiliary feedwater system, December 19, 2003
- Main steam code safety valves, November 20, 2003

The inspectors reviewed work practices, scoping in accordance with 10 CFR 50.65(b), performance, 10 CFR 50.65(a)(1) or (a)(2) classification and reclassification goals, and identification of common cause failures. The inspectors reviewed various documentation and discussed maintenance rule items with licensee personnel.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed three of the licensee's risk assessments for equipment outages as a result of planned and emergent maintenance in accordance with the requirements of 10 CFR 50.65(a)(4) and licensee Procedure AP 22C-003, "Operational Risk Assessment Program," Revision 9. The inspectors also discussed the planned and emergent work activities with planning and maintenance personnel. The inspectors reviewed the following:

- Operational risk assessments for planned maintenance for the weeks of October 13, November 24, and December 8, 2003

- Actual, planned, and emergent work schedules for the same weeks

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

On November 14 and 16, 2003, the inspector observed the control room operators' performance while draining the reactor coolant system to midloop. The first evolution was to allow removal of the steam generator nozzle dams. The second was to prepare for the reactor coolant system vacuum fill and vent. The inspector observed plant parameters to ensure that system operation complied with the procedure limitations. The inspectors also verified that the required reactor vessel level instrumentation was in service.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors selected one operability evaluation conducted by the licensee during the report period involving risk-significant systems or components to review. The inspectors evaluated the technical adequacy of the licensee's operability determination, verified that appropriate compensatory measures were implemented, and verified that the licensee considered all other pre-existing conditions, as applicable. Additionally, the inspectors evaluated the adequacy of the licensee's problem identification and resolution program as it applied to operability evaluations. The specific operability evaluation reviewed was OE EM-03-09, "Structural Integrity of the Intermediate Head SI Piping Protected by Relief Valves EM 8853A, -B and 8851," Revision 2, October 16, 2003.

The inspectors also reviewed applicable portions of the Updated Safety Analysis Report, Technical Specifications, and system drawings and discussed the operability evaluations with licensee personnel.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

On December 22, 2003, the inspector reviewed an emergent operator workaround for the auxiliary feedwater system to determine the following:

- Effect of the workaround on the system functional capability to respond to an initiating event

- Whether the workaround could affect human reliability in responding to an event

- Effect of the workaround on the operator's ability to implement abnormal or emergency operating procedures

The inspectors identified an operator workaround during the auxiliary feedwater system maintenance effectiveness inspection. A note in Procedure SYS AL-120, "Feeding Steam Generators with a Motor-Driven or Turbine-Driven AFW Pump," Revision 27, stated that the motor-driven auxiliary feedwater pump discharge throttle valves may not close from a near-closed position with the pump running, and in order to close the valves it would have to be opened first or the associated auxiliary feedwater pump stopped.

The licensee stated that the note met the definition of an operator workaround in licensee Procedure AI 22A-001, "Operator Work Arounds/Burdens," Revision 2. The note had been in the procedure since 1988 but was not identified as an operator workaround. The licensee could not determine why the note was in the procedure. Additionally, the licensee stated that the valves had been operated from a variety of positions without any known problems since 1996. The licensee considered the valves operable. The licensee wrote PIRs 2003-3747 and -3755 to document the issue.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed or observed four postmaintenance tests on the following equipment to verify that procedures and test activities are adequate to verify system operability:

- Emergency diesel Generator A, November 12, 2003
- Emergency diesel Generator A and essential service water System A, December 15, 2003
- Emergency diesel Generator B, November 10, 2003
- Residual heat removal Pump A, October 9, 2003

In each case, the associated work orders and test procedures were reviewed to determine the scope of the maintenance activity and determine if the test adequately tested components affected by the maintenance. The Updated Final Safety Analysis Report, design basis documents, and selected calculations were also reviewed to determine the adequacy of the acceptance criteria listed in the test procedures.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors observed and reviewed activities for the fall 2003 refueling and maintenance outage. The inspection-completion dates were from the outage start date of October 18, 2003, through the outage end date of December 2, 2003. Specific inspection activities required by Attachment 71111.20 are documented in the following paragraphs.

Monitoring of Shutdown Activities

The inspectors observed portions of the plant cooldown. The inspectors monitored the reactor coolant system cooldown rate and reviewed the Technical Specification cooldown restrictions. The inspectors verified that the cooldown rate did not exceed requirements.

Clearance Activities

The inspectors verified that various clearance order tags were properly hung and that associated equipment was appropriately configured. The inspectors also verified that appropriate foreign material controls were established. The inspectors specifically reviewed the emergency diesel generator clearance tagouts.

Reactor Coolant System Instrumentation

The inspector verified that the reactor coolant system pressure, level, and temperature instruments were installed and configured to provide accurate indication. The inspector walked down the tygon tube used for reactor coolant system water level during midloop draindown and operation. The licensee also used installed wide- and narrow-range level instrumentation to monitor the reactor coolant system water level. The inspectors verified that the various level indications were within the tolerances specified in the licensee's procedures.

Electrical Power

The inspectors verified the operability and availability of electrical power sources required for outage activities based on walkdowns and discussions with various licensee personnel. This included during refueling, hot midloop, cold midloop, and shutdown operations. The licensee removed the emergency diesel Generator B from service for a major inspection and overhaul. The inspectors verified that emergency diesel Generator A was operable.

Decay Heat Removal System Monitoring

The inspectors verified that the decay heat removal system functioned properly. The inspectors monitored system parameters, reviewed system lineups, and observed system operation.

Spent Fuel Pool Cooling System Operation

The inspector verified that outage work did not impact the ability of the operators to monitor and operate the spent fuel pool cooling system.

Inventory Control

The inspectors reviewed and observed various system lineups and operation to ensure that the risk for a loss of reactor coolant system inventory was minimized. The licensee monitored reactor coolant system water level in the control room throughout the outage with installed instrument and computer point displays. The licensee designated a control room operator to monitor water level during reduced inventory operations.

Reactivity Control

The inspector verified that the licensee controlled reactivity in accordance with Technical Specifications. The outage risk plan, as well as the twice daily outage updates, identified risk significant evolutions. The licensee held thorough prejob briefings for all major evolutions.

Containment Closure

The inspectors verified that the containment was in the proper configuration during various outage activities. These included midloop and refueling operations. The inspectors discussed and reviewed with various licensee personnel their ability to close the containment personnel and equipment hatches within the 30-minute time frame, if warranted. The licensee had procedures and personnel in place to accomplish containment closure in the required time.

Reduced Inventory and Midloop Conditions

The inspectors reviewed the licensee’s commitments from Generic Letter 88-17 and confirmed that they are still in place. The inspectors also verified, during reduced inventory and midloop operations, that the plant systems were in the required configuration. The inspectors observed the draindown and midloop operations and confirmed that there were no negative impacts to control room operations as a result of distractions.

Refueling Activities

The inspectors observed refueling operations. The refueling activities were performed in accordance with Technical Specifications and licensee procedures. The inspector spot checked over half of the incore as-left fuel assembly locations against the fuel loading map and did not identify any discrepancies.

Monitoring of Heatup and Startup Activities

The inspectors observed various plant heatup and startup activities. The licensee conducted these activities in accordance with the plant Technical Specifications and procedures. The inspectors verified that the appropriate equipment was available for mode changes. The reactor coolant system leak rates were within the required limits.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed or observed all or part of four surveillance activities in accordance with inspection Attachment 71111.22 to verify that risk significant structures, systems, and components are capable of performing their intended safety functions and assessing their operational readiness:

- STS EG-100A, “Component Cooling Water Pumps A/C Inservice Pump Test,” Revision 19, December 4, 2003

- STS KJ-001A, “Integrated D/G and Safeguards Actuation Test - Train A,” Revision 27, November 13, 2003

- STS KJ-011B, “DG NE02 24 Hour Run,” Revision 12, October 22, 2003

- STS EM-100A, “Safety Injection Pump A Inservice Pump Test,” Revision 22, October 14, 2003

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

To review and assess the licensee's performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, and high radiation areas, the inspector interviewed supervisors, radiation workers, and radiation protection personnel that had the potential to be involved in high dose rate and high exposure jobs during routine and RF13 operations. The inspector discussed changes to the access control program with the radiation protection manager. The inspector also conducted plant walkdowns within the radiologically controlled area and conducted independent radiation surveys of selected work areas. The following items were reviewed and compared with regulatory requirements:

- Area postings, radiation work permits, radiological surveys, and other controls for airborne radioactivity areas, radiation areas, and high radiation areas
- High radiation area key control
- Internal dose assessment for exposures exceeding 50 millirem committed effective dose equivalent (none observed during this inspection period)
- Setting, use, and response of electronic personal dosimeter alarms
- Conduct of work by radiation protection technicians and radiation workers in areas with the potential for high radiation dose work associated with RF13 activities
- Dosimetry placement when work involved a significant dose gradient (installation and removal of steam generator nozzle dams/covers, personnel access for reactor head modification, and reactor under-vessel inspection)
- Controls involved with the storage of highly radioactive items in the spent fuel and refuel pools
- Audits, licensee event reports (LERs), special reports, and self-assessments involving high radiation area controls and staff performance (no LERs or special reports were recorded during this inspection period)
- Summary of corrective action documents written since the last inspection and selected documents related to high radiation area incidents, radiation protection technician and radiation worker errors, and repetitive and significant individual deficiencies

Performance indicator reviews associated with occupational exposure control effectiveness are documented in Section 4OA1 of this report. The inspector completed all 21 of the required inspection samples.

b. Findings

Introduction. The inspectors identified a Green, noncited violation because the licensee failed to institute measures to ensure that an individual was not able to gain unauthorized access to a very high radiation area under the reactor vessel.

Description. On October 28, 2003, during a tour of the containment building, the inspectors observed that a ladder leading down to the reactor under-vessel area was surrounded with a safety cage with a locked cover and was posted, “Grave

Danger- Very High Radiation Area." The area around the enclosed safety caged ladder was an open area approximately 10 by 15 feet. This area was not provided with a physical barrier that completely enclosed the area and would not prevent an individual from gaining unauthorized or inadvertent access to the very high radiation area under the reactor vessel. Additionally, the ladder safety cage had horizontal supports that were approximately 3.5 feet apart down the length of the ladder. This arrangement would allow an individual to climb over the handrail and down the outside of the ladder using the safety cage.

The licensee performed a radiation survey that determined the radiation levels at the bottom of the ladder, one meter away from the withdrawn in-core instrument thimbles, were approximately 640 RADs per hour, and were approximately 1350 RADs per hour at 1 foot from the thimbles.

Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants," states, in part, that very high radiation areas require much stricter monitoring and controls since failure to adequately implement effective radiological controls can result in radiation doses that result in significant health risk. Additionally, physical barriers should, to the extent practical, completely enclose very high radiation areas sufficient to thwart undetected circumvention of the barrier (i.e., fencing around very high radiation areas should extend to the overhead and preclude anyone from climbing over the fencing).

Analysis. The inspectors determined that the licensee's failure to completely enclose a very high radiation area to ensure individuals were not able to gain unauthorized or inadvertent access was a performance deficiency. The finding was greater than minor because it affected the Occupational Radiation Safety cornerstone objective to ensure adequate protection of worker health and safety from exposure to radiation and the finding is associated with the cornerstone attribute (program & process). The finding involved an individual's potential for unplanned or unintended dose. When processed through the Occupational Radiation Safety Significance Determination Process, the finding was determined to be of very low safety significance because the finding was not associated with ALARA planning or work controls, there was no overexposure nor a substantial potential for overexposure, and the ability to assess dose was not compromised.

Enforcement. 10 CFR 20.1602 requires that, in addition to the requirements of 10 CFR 20.1601 (controls for high radiation areas), the licensee shall institute additional measures to ensure that an individual is not able to gain unauthorized or inadvertent access to areas in which radiation levels could be encountered at 500 RADs per hour at one meter from the source. However, the licensee failed to a provide a physical barrier that completely enclosed the area and that would ensure that an individual was not able to gain unauthorized or inadvertent access to the very high radiation area. Because the failure to adequately control access to a very high radiation area was determined to be of very low safety significance and has been entered into the station's corrective action program as

PIR 2003-3220, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000482/2003006-02, Failure to adequately control access to a very high radiation area.

2OS2 As Low as is Reasonably Achievable (ALARA) Planning and Controls (71121.02)

a. Inspection Scope

The inspector interviewed radiation protection personnel and radiation workers involved in high dose rate, high exposure, and potential airborne area work activities. The inspector assessed the licensee's performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, high radiation areas, and radiation worker practices. The inspector observed high dose work involving reactor head and steam generator primary side activities to determine if personnel ALARA practices complied with regulatory and procedural requirements.

The inspector interviewed radiation protection staff, and other radiation workers to determine the level of planning, communication, ALARA practices, and supervisory oversight that was integrated into work planning and work activities. Additionally, the inspector attended ALARA job briefing for steam generator primary side activities. The inspector reviewed initial and emergent work scopes and estimated person-hours provided to the radiation protection group for accuracy. The following items were reviewed and compared with procedural and regulatory requirements to assess the licensee's program to maintain occupational exposures ALARA:

- Plant collective exposure history for the past 3 years, current exposure trends, source-term measurements, and 3-year rolling average dose information

- ALARA program procedures

- The use and result of administrative and engineering controls to achieve dose reductions

- Permanent and temporary shielding program and implementation

- Plant source-term evaluation and control strategy/program

- Hot spot tracking and reduction program

- ALARA committee meeting minutes and presentations

- Summary of corrective action documents written since the last inspection and selected documents relating to exposure tracking, higher than planned exposure levels, radiation worker practices, and repetitive and significant individual deficiencies

The inspector completed 10 of the required samples.

b. Findings

Introduction. The inspectors identified four examples of a Green, noncited violation of 10 CFR 20.1501(a), because the licensee failed to perform required radiological surveys to ensure compliance with 10 CFR 20.1204(a) and 10 CFR 20.1902(b).

Description. October 19, 2003, the inspector noted that dose rates had increased in the normal charging pump room and the volume control tank valve galley of the auxiliary building. The inspector notified the health physics staff who responded and identified that both rooms had general radiation levels greater than 100 millirem per hour, requiring the areas to be posted as high radiation areas. During the investigation of the event, the licensee also identified that the seal water heat exchanger room had general area radiation levels as high as 250 millirem per hour, which required the area to be posted as a high radiation area. From discussion with the licensee, the inspector concluded that the cause for the elevated dose rates were from the chemical flush of the reactor coolant system. Once identified, the licensee took appropriate timely actions to properly control the areas.

During the observation of work associated with decontamination of the reactor cavity seal ring on October 22, 2003, the inspector identified that the job coverage air sample was approximately 10 feet behind the workers and not representative of the workers breathing zone or in the path of negative ventilation. From a review of the survey information, the inspector determined that contamination levels were as high as 350,000 disintegrations per minute per 100 centimeters squared.

Analysis. The inspector determined that the licensee's failure to perform surveys required by 10 CFR 20.1501(a) are four examples of a performance deficiency. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function and was not the result of any willful violation of NRC requirements or licensee's procedures. The finding is greater than minor because it is associated with the occupational radiation safety cornerstone attribute (program and process) and affected the cornerstone objective to provide adequate protection to a worker's health and safety from exposure to radiation. When these issues were processed through the Occupational Radiation Safety Significance Determination Process, it was determined to be a Green finding because it was not an ALARA planning and control issue, there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised.

Enforcement. 10 CFR 20.1501(a) requires, in part, that a licensee make or cause to be made, surveys that are necessary to comply with regulations in this part and are reasonable under the circumstances to evaluate the radiation levels, the concentrations or quantities of radioactive material, and the potential radiological hazards. 10 CFR 20.1902(b) requires high radiation areas to be conspicuously posted. 10 CFR 20.1204(a) requires, in part, that a licensee take suitable and timely measurements of the concentrations of radioactive materials in the air in the work area.

10 CFR 20.1003 defines a high radiation area as an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 0.1 rem (100 millirem) in 1 hour at 30 centimeters from the radiation source or 30 centimeters from any surface that the radiation penetrates.

The failure to survey the above areas is being identified as four examples of a 10 CFR 20.1501(a) violation. Because the four examples of the finding were of very low safety significance and were entered into the corrective action program as PIRs 2003-3069 and -3136, these examples of a violation were treated as an noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000482/2003006-003, Four examples of the failure to perform radiological surveys.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification (71151)

a. Inspection Scope

Reactor Safety Cornerstone

The resident inspectors performed a review of two PI data. The inspectors reviewed the licensee's data submittal using Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 2. The inspectors reviewed various licensee indicator input information to determine the accuracy and completeness of the PI:

- Safety system unavailability - emergency ac power system, June 2002 through September 2003, completed October 21, 2003

- Safety system unavailability - high pressure injection system, June 2002 through September 2003, completed December 4, 2003

The inspectors discussed system status with various licensee personnel. The inspectors also reviewed licensee information, including control room logs, and the applicable Technical Specifications.

Occupational Radiation Safety Cornerstone

The inspectors sampled licensee submittals for the PI listed below for the period from October 2002 through September 2003. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 2, were used to verify the basis in reporting for each data element.

- Occupational exposure control effectiveness

Licensee records reviewed included corrective action documentation that identified occurrences of locked high radiation areas (as defined in Technical Specification 5.7.2), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in NEI 99-02). Additional records reviewed included ALARA records and whole body counts of selected individual exposures. The inspectors interviewed licensee personnel that were accountable for collecting and evaluating the PI data. In addition, the inspectors toured plant areas to verify that high radiation, locked high radiation, and very high radiation areas were properly controlled. The inspector completed one of the required inspection samples.

Public Radiation Safety Cornerstone

- Radiological effluent Technical Specification/offsite dose calculation manual radiological effluent occurrences

Licensee records reviewed included radiological effluent release corrective action records and annual effluent release reports during the past 4 quarters (no licensee event or special reports were submitted) to determine if any doses resulting from liquid or gaseous effluent releases exceeded PI thresholds. The inspectors interviewed licensee personnel that were accountable for collecting and evaluating the PI data. The inspector completed one of the required inspection samples.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution

The inspectors evaluated the effectiveness of the licensee's problem identification and resolution processes related to high radiation area incidents and radiation protection technician and radiation worker errors during the Access Control to Radiologically Significant Areas, Section 2OS1, inspection.

The inspectors evaluated the effectiveness of the licensee's problem identification and resolution processes relating to ALARA planning and control programs during the ALARA Planning and Controls, Section 2OS2, inspection.

The inspectors reviewed inservice inspection-related condition reports issued during the current and past RF and verified that the licensee identified, evaluated, corrected, and trended problems. In this effort, the inspectors evaluated the effectiveness of the licensee's corrective action process, including the adequacy of the technical resolutions.

4OA3 Event Followup (71153)

.1 (Closed) LER 50-482/2003-002-00: Reactor Vessel Level Indication System Inoperable for a Period Longer Than Allowed by Technical Specifications

On March 25, 2003, the licensee identified that the reactor vessel water level indicating system Trains A and B were inoperable. The level instruments were part of the postaccident monitoring instrumentation and were included in Technical Specification Section 3.3.3. The licensee determined that the level instruments had been inoperable for a number of years due to the failure to provide adequate surveillance activities. The alternate methods available to determine vessel level indication were core exit thermocouples, pressurizer level instrumentation, and the reactor coolant system subcooling monitor indications. The licensee restored the level indicators to operable status during the fall 2003 RF. The inspector did not identify any new findings during the LER review. This finding constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. The licensee documented the problem in PIR 2003-0805. This LER is closed.

.2 (Closed) LER 50-482/2003-003-00: Reactor Protection System Actuation and Reactor Trip Due To Feedwater Isolation Valve Closure

On August 18, 2003, the reactor tripped on low/low steam Generator B water level when its associated main feedwater isolation valve failed closed. NRC Inspection Report 50-482/2003-005, Sections 1R14 and 4OA3, discussed the operator and plant equipment response to the trip. The inspectors reviewed this LER and did not identify any findings of significance. The licensee documented the trip in PIR 2003-2449. This LER is closed.

.3 (Closed) LER 50-482/1998-004-00,01: Volume Control Tank Isolation Valve Does Not Have Redundant Fusing

The licensee identified in PIR 98-3012, initiated on October 8, 1998, that the switch for Valve BG-LCV-112C, volume control tank isolation valve, did not have redundant fusing. A fire affecting this valve switch could have prevented the electrical closure of the valve. This condition would not have been apparent to the operators and could have allowed hydrogen gas intrusion from the volume control tank into the centrifugal charging pump suction lines. Licensee Procedure OFN RP-017, "Control Room Evacuation," was revised to include appropriate guidance for the operators to ensure that the isolation valve was shut.

This finding is more than minor because it had a credible impact on safety in that, if the hydrogen gas had entered the charging pumps suction lines, it could have resulted in gas binding of the pumps. The finding affects the mitigating systems cornerstone and was considered to have very low safety significance (Green) using Appendix F of the significant determination process because of the low ignition frequencies in the areas, the low combustible loading in the areas, the automatic fire detection capabilities, and the ability of operator actions to extinguish the postulated fire and restore equipment necessary for postfire safe shutdown. This licensee-identified finding involved a violation of License Condition 2.C(5)(a) of Facility Operating License NPF-42. The enforcement aspects of the violation are discussed in Section 4OA7. This LER is closed.

4OA5 Other Activities

.1 Temporary Instruction 2515/152, "Reactor Pressure Vessel Lower Head Penetrations (NRC Bulletin 2003-02)

a. Inspection Scope

The inspector completed the review of the licensee's reactor pressure vessel lower head bare metal visual examination on October 23, 2003. On October 21, 2003, licensee personnel were able to access the lower head area and did not need remote-controlled camera equipment to perform the examination. A certified Level III nondestructive examination licensee employee performed the examination. The other licensee personnel supporting the examination attended a training class taught by the Level III examiner.

The examination was conducted in accordance with Procedures STN PE-040D, "RCS Pressure Boundary Integrity Walkdown," Revision 2, and STN PE-040F, "RPV BMI Inspection," Revision 0. Licensee personnel were able to identify, disposition, and resolve any deficiencies. They were also able to identify whether there was any pressure boundary leakage or reactor pressure vessel lower head corrosion as described in the bulletin. The examiner and other support personnel made an extensive video of the bottom head and bottom head area.

There were no obstructions to the visual inspection of the head. The head did have boric acid stains which the licensee attributed to cavity seal ring leakage during past refueling outages. The licensee performed nondestructive testing of the cavity seal ring welds and identified discontinuities that will be repaired in the spring 2005 RF. The licensee also cleaned the bottom head after draining the refueling cavity at the end of the fall 2003 RF.

The licensee's examination consisted of a 360-degree coverage of all the nozzles. The licensee could identify small boric acid leaks as described in Bulletin 2003-02. The nondestructive examination personnel noted a small amount of boric acid material around a few penetrations. The examiners determined that this boric acid was not due to a penetration leak but was from a cavity seal ring leak. The examiners were able to check the annulus between the bottom head and penetration piping. There was no boric acid in the interface between the vessel and penetrations.

The licensee did not take any chemical samples of the deposits. The licensee stated that there was not enough material available for an analysis. The licensee also examined the reactor vessel for pressure boundary leaks and none were identified.

The licensee concluded that the deposits were from cavity seal ring leakage due to the discontinuities in the welds and flow paths of the boric acid residue. There were no material deficiencies that required repair. There were no impediments to an effective examination.

On Thursday, October 23, 2003, NRC and licensee personnel participated in a conference call to discuss the initial results of the examination. All personnel were provided printed pictures as well as pictures on a compact disk of the bottom head examination. There were no issues identified concerning reactor pressure vessel bottom head integrity during the call.

b. Findings

No findings of significance were identified.

.2 Temporary Instruction 2515/153, "Reactor Containment Sump Blockage"

a. Inspection Scope

On November 12, 2003, the inspector completed the review of the licensee's implementation of compensatory measures for the containment recirculation sumps. The compensatory measures were delineated in the Wolf Creek Nuclear Operating Corporation's response to NRC Bulletin 2003-001, Letter WO 03-0049, dated August 8, 2003.

The compensatory measures that had been implemented before the response was sent were:

- Ensuring that alternative water sources were available to refill the refueling water storage tank or to otherwise provide inventory to inject into the reactor core and spray into the containment

- More aggressive containment cleaning and increased foreign material controls

The inspector reviewed the following licensee procedures that addressed these two items:

- Emergency Procedure EMG C-11, “Loss of Emergent Coolant Recirculation,” Revision 14

- Alarm response Procedure ALR 00-047E, “RWST Level HiLo,” Revision 11

- Surveillance Procedure STN EJ-002, “Containment Inspection,” Revision 7

- Administrative Procedure AP 12-004, “Containment Entry and Material Control,” Revision 3

Additionally, the licensee also stated that the above surveillance and administrative procedures were enhanced for the fall 2003 RF. The inspector verified that the licensee had made the changes to these procedures prior to the RF.

The licensee stated in their response letter that the following interim compensatory measures were implemented prior to or planned for the fall 2003 RF:

- Operator training on indications of and response to sump clogging
- More aggressive containment cleaning and increased foreign material controls
- Ensuring containment drainage paths are unblocked
- Ensuring sump screens are free of adverse gaps and breaches
- Additional plant-specific measures

The inspector verified that the licensee implemented the prerefueling outage interim measures as stated in their response letter. The inspectors reviewed the following:

- AP 12-004, “Containment Entry and Material Control,” Revision 3
- Change Package 011154, Revision 0
- ES131010, “Engineering Support Program Continuing Training,” Revision 16
- LR5004001, “Plant Shutdown to Mode 5,” Revisions 7 and 8
- STN EJ-002, “Containment Inspection,” Revision 7
- STS EJ-002, “Containment Sump Inspection,” Revision 11
- Work Order 03-253912-003

Two additional measures, operator classroom and simulator training on potential recirculation sump screen blockage and emergency response organization staff training on sump blockage with possible compensatory actions, were scheduled for completion in early 2004.

The licensee performed a containment walkdown to quantify potential debris sources and check for gaps in the sumps’ screened flowpath. There were no major obstructions in the containment upstream of the sumps. The licensee removed the four access control gates at the entrances to the bioshield. This was the only sump-related modification the licensee planned to do and no preparations for future potential modifications resulting from the sump evaluations were noted by the inspector.

The inspector identified that the as-installed screen hole size of some of the inner and middle screens were larger than the 1/8- and ½-inch criteria stated in the response letter. The licensee determined that the screens were manufactured with some holes slightly larger than the criteria in the letter. Also, the screens were damaged during installation so that some holes were greater than the established criteria. The licensee reviewed their design basis and performed a containment recirculation sump evaluation, which was documented in Change Package 011213, Revision 0.

The licensee determined that the maximum size an opening in the inner screen could be without affecting plant equipment was 1/4 inch. The licensee determined that, even though some holes were greater than 1/4 inch, the total area of the enlarged holes was less than one percent of the total screen wetted area. The licensee stated that the operability of the sumps was not compromised. The licensee repaired the damaged screens and submitted a supplement to their response changing the allowed screens' hole sizes to 3/16 inch for the inner and 5/8 inch for the middle screens. The supplement was Wolf Creek Nuclear Operating Corporation's Letter WO 03-0064, dated November 21, 2003.

b. Findings

No findings of significance were identified.

4OA6 Meetings

.1 Exit Meeting Summaries

The inspectors presented the resident inspection results to Mr. R. Muench, Chief Executive Officer, and other members of licensee management after the conclusion of the inspection on January 9, 2004.

On October 24, 2003, the inspector presented the ALARA planning and control inspection results to Mr. R. Muench and other members of his staff who acknowledged the findings.

The inspectors presented the results of the inservice inspection effort to Mr. R. Muench and other members of licensee management on October 31, 2003.

On October 31, 2003, the inspector presented inspection results of the access controls to radiologically significant areas to Mr. R. Muench and other members of his staff who acknowledged the findings.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. The licensee furnished proprietary information to the NRC during the inspection period. The information was returned to the licensee prior to the end of the report period.

4OA7 Licensee-Identified Violations

The following violations of very low safety significance (Green) were identified by the licensee and are a violation of NRC requirements, which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as a noncited violation.

1. Volume Control Tank Isolation Valve Does Not Have Redundant Fusing

License Condition 2.C(5)(a) of Facility Operating License NPF-42 requires, in part, that the licensee shall maintain all provisions of the approved fire protection program as described in the SNUPPS Final Safety Analysis Report for the facility through Revision 17 and the Wolf Creek site addendum through Revision 15. In 1987, Revision 00 of the Wolf Creek Updated Safety Analysis Report was issued and combined the SNUPPS Final Safety Analysis Report, Revision 17, and the Wolf Creek site addendum, Revision 15, into the Updated Safety Analysis Report. Table 9.5E, Section III.G, of the Updated Safety Analysis Report details the licensee's methods of ensuring that one of the redundant trains of postfire safe shutdown equipment is free of fire damage.

Contrary to this, on October 8, 1998, the licensee identified that Switch BG-LCV-112C, volume control tank isolation valve, did not have redundant fusing. A fire affecting this valve switch could have prevented the electrical closure of the valve. This condition would not have been apparent to the operators and could have allowed hydrogen gas intrusion from the volume control tank into both of the centrifugal charging pump suction lines. This condition could have resulted in the gas binding of the pumps. The conditions have been entered into the licensee's corrective action program as PIR 98-3012. The finding affects the mitigating systems cornerstone and was considered to have very low safety significance (Green) using Appendix F of the significant determination process because of the low ignition frequencies in the areas, the low combustible loading in the areas, the automatic fire detection capabilities, and the ability of operator actions to extinguish the postulated fire and restore equipment necessary for postfire safe shutdown and is being treated as a noncited violation.

2. High Radiation Area Improper Entry

Technical Specification 5.7.1 allows entry into high radiation areas only after individuals have been made knowledgeable of the dose rates in the area. However, on May 20, 2002, an individual entered residual heat removal pump Room A, which was posted as a high radiation area, without having been made knowledgeable of the dose rates in the area. The finding was processed through the Occupational Radiation Safety Significant Determination Process and was determined to be of very low safety significance (Green) because the finding was not associated with ALARA planning or work controls, there was no overexposure nor a substantial potential for overexposure, and the ability to assess dose was not compromised. This finding is documented in PIR 2002-1272.

ATTACHMENTS: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

R. Muench, President and Chief Executive Officer
K. A. Harris, Director, Performance Improvement and Learning
B. T. McKinney, Site Vice President
D. Jacobs, Plant Manager
K. L. Scherich, Director, Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-482/2003006-01	URI	Fire protection (Section 1R05)
50-482/2003006-02	NCV	Access control to radiologically significant areas (Section 2OS1)
50-482/2003006-03	NCV	ALARA planning and controls (Section 2OS2)

Closed

50-482/2003-002-00	LER	Reactor vessel level indication system inoperable for period longer than allowed by Technical Specifications (Section 4AO3)
50-482/2003-003-00	LER	Reactor protection system actuation and reactor trip due to feedwater isolation valve closure (Section 4AO3)
50-482/1998-004-00,01	LER	Volume control tank isolation valve does not have redundant fusing (Section 4AO3)
50-482/2003006-02	NCV	Access control to radiologically significant areas (Section 2OS1)
50-482/2003006-03	NCV	ALARA planning and controls (Section 2OS2)

LIST OF DOCUMENTS REVIEWED

Equipment Alignment

- CKL BG-120, “Chemical and Volume Control System Normal Valve Lineup,” Revision 33

- CKL BG-130, “Chemical and Volume Control System Switch and Breaker Lineup,” Revision 24

Fire Protection

- AP 10-100, “Fire Protection Program,” Revision 7
- AP 10-106, “Fire Preplans,” Revision 2
- Generic Letter 86-10
- Updated Safety Analysis Report
- 10 CFR Part 50, Appendix R, paragraph 3.g.2

Heat Sink Performance

- STN PE-033, “CCW Heat Exchanger Performance Test,” Revision 7

- EG-06-W, “Engineering Calculations, CCW System Calculations,” Revision W-2

- USNRC Generic Letter 89-3, “Service Water System Problems Affecting Safety-Related Equipment,” July 18, 1989

- EPRI NP-7552, Heat Exchanger Performance Monitoring Guide, December 1991

- Drawing M-1HX001, “Heat Exchanger Tube Sheet Map Component Cooling Water Heat Exchanger B (Inlet/Outlet) EEG01B,” Revision 10

- Updated Safety Analysis Report

Maintenance Rule Documents

- Final scope evaluations for AL-0, auxiliary feedwater system

- Functional failure evaluations for AB-04, main steam code safety valves

- Functional failure evaluations for AL-0, auxiliary feedwater system

- Maintenance rule basis Information for AB-04, main steam code safety valves

- Maintenance rule expert panel meeting minutes for AB-04, main steam code safety valves

- Maintenance rule expert panel meeting minutes for AL-0, auxiliary feedwater system

- Maintenance rule performance evaluation for AB-04, main steam code safety valves

- Maintenance rule performance evaluation for AL-0, auxiliary feedwater system

- PIRs 2002-1495 and 2003-0854, -1134, -1159, -1275, -1881, -1885, -3090, and -3752

- SYS AL-120, “Feeding Steam Generators with a Motor Driven or Turbine AFW Pump,” Revision 27

- Technical Specifications

- Updated Safety Analysis Report

- Work Orders 02-233555-000, 02-235259-000, and 02-235313-000

- Work Request 2029823

Operability Evaluations

- Reportability Evaluation Request 2003-011

Personnel Performance During Nonroutine Plant Evolutions

- GEN 00-008 “Reduced Inventory Operations,” Revision 14

Performance Indicator Verification

- Licensee performance indicator worksheets
- Performance indicator summary reports
- Selected NRC inspection reports
- Selected control room operator logs

Postmaintenance Testing

- STS EF-210A, “ESW System Inservice Check Valve Test,” Revision 9

- SYS EJ-100A, “RHR System Inservice Pump A Test,” Revision 27

- SYS KJ-123, “Postmaintenance Run of Emergency Diesel Generator A,” Revision 22

- SYS KJ-124, “Postmaintenance Run of Emergency Diesel Generator B,” Revision 19

- TMP 03-001, “Emergency Diesel Governor Retest,” Revision 2

- Work Orders 01-229446-001, 03-247744-001, 03-254470-000, 03-257196-001, and 03-252728-001

Refueling Outage

Monitoring of Shutdown Activities

- GEN 00-004, “Power Operation,” Revision 46
- GEN 00-005, “Minimum Load to Hot Standby,” Revision 49
- GEN 00-006, “Hot Standby to Cold Shutdown,” Revision 53

Reactor Coolant System Instrumentation

- INC S-286, “Mid Loop Transmitters Fill and Drain,” Revision 6
- STN IC-286, “RCS Mid Loop Level Instrumentation Calibration,” Revision 5
- STN IC-490, “Pressurizer Wide Range Level Calibration,” Revision 5

Reduced Inventory and Mid-Loop Conditions

- Generic Letter 88-17, “Loss of Decay Heat Removal”
- GEN 00-008, “Reduced Inventory Operations,” Revision 14
- SYS BB-112, “Vacuum Fill of the RCS,” Revision 20
- Wolf Creek Letter ET 92-0001, January 2, 1992
- Wolf Creek Letter ET 88-0193, December 23, 1988
- Wolf Creek Letter WM 89-0041, February 2, 1989

Refueling Activities

- FHP 02-011, “Fuel Shuffle and Position Verification,” Revision 28
- STS KE-001, “Refueling Machine Operability Test,” Revision 19

Access Control to Radiologically Significant Areas

Radiation Work Permits

033220, Perform Eddy Current testing and associated support work
033230, Install and remove steam generator nozzle dams/covers
034051, Personnel access for reactor head modification
037001, Access to incore instrument tunnel, very high radiation area, under vessel work activities

Corrective Action Documents

2002-1929 and 2003-0401, -1932, -1335, and -3116

Procedures

AP 25A-001, “Radiation Protection Manual,” Revision 8
AP 25A-200, “Access to Locked High or Very High Radiation Areas,” Revision 12
AP 25B-100, “Radiation Worker Guidelines,” Revision 19
AP 26A-007, “NRC Performance Indicators,” Revision 2
RPP 02-215, “Posting of Radiological Controlled Areas,” Revision 19
RPP 03-106, “Use of Special Dosimetry,” Revision 13

RPP 08-105, “Underwater Dive Operations,” Revision 6

Quality Assurance Audits and Surveillances

WCNOC QE Audit K-569, Radiation protection
WCNOC QE Audit K-581, Radiation protection
Self-assessment, SEL 02-014, Effectiveness of radiological controls at Wolf Creek
Self-assessment, SEL 03-004, Health physics operations

ALARA Planning and Controls

PIRs 2003-2299 and -2484.

Site ALARA committee minutes for September 30 and October 13, 2003.

Radiation Work Permits

Reactor coolant Pump D seal work (RWP 034208)
Reactor head modification work (RWP 034051)
Split pin work on the upper internals (RWP 036061)

Procedures

AP 25A-401, “ALARA Program,” Revision 9
AP 25A-410, “ALARA Committee,” Revision 7
AP 25A-700, “Use of Temporary Lead Shielding,” Revision 7

Inservice Inspection Activities

Procedures

WCRE-10, “Inservice Inspection Program Plan, Interval 2,” Revision 5

WCRE-12, “Risk-Informed Inservice Inspection Basis Document,” Revision 0

UT-95, “Ultrasonic Examination of Austenitic Piping Welds,” Revision 0

UT-98, “Reactor Vessel Closure Head Welds and Adjacent Base Metal,” Revision 0

PDI-UT-6, “Performance Demonstration Initiative Generic Procedure for the UT Exam of Reactor Pressure Vessel Welds,” Revision E

PDI-UT-7, “Performance Demonstration Initiative Generic Procedure for the Manual Ultrasonic Through Wall and Length Sizing of Ultrasonic Indications in Reactor Pressure Vessel Welds,” Revision F

QCP20-520, “Visual Examination (VT-2),” Revision 4

QCP20-508, “Radiographic Examination Procedure,” Revision 1

QCP20-501, “Liquid Penetrant Procedure,” Revision 4

AP 29A-003, “Steam Generator Monitoring,” Revision 7

I-ENG-023, “Steam Generator Data Analysis Guidelines,” Revision 4

LMT-QA-6, “Qualification and Certification of NDE and Visual Examination Personnel,” Revision 34

LMT-QA-37, “Qualification of Nondestructive Examination Personnel for Ultrasonic Examination,” Revision 7

MRS-GEN-1127, “Guidelines for Steam Generator Eddy Current Data Quality Requirements,” Revision 0

QCP-20-501, “Liquid Penetrant Examination,” Revision 4

Gas Tungsten Arc Welding Procedure WPS1-0808TO1, Revision 4

GWS-ASME, “ASME General Welding Standard,” Revision 6

Shielded Metal Arc Welding Procedure WPS1-0808S01, Revision 4

Shielded Metal Arc Welding Procedure WPS1-0101S01, Revision 7

Work Orders

03-256481-001, 03-256482-001, 01-231340-006, 01-231340-008, 02-216675-006, 00-216675-005, and 02-246224-002

PIRs

2002-1757, 2003-3232, and 2003-3236

Miscellaneous Documents

Letter (ET 01-0009), Richard A. Muench to U.S. Nuclear Regulatory Commission, “Relief Request for Application of an Alternative to the ASME Boiler and Pressure Vessel Code Section XI Examination Requirements for Class 1 and 2 Piping Welds, Wolf Creek Generating Station,” February 15, 2001

Letter (ET 01-0028), Richard A. Muench to U.S. Nuclear Regulatory Commission, “Response to NRC Request for Additional Information Regarding Relief Request for Application of an Alternative to the ASME Boiler and Pressure Vessel Code Section XI Examination Requirements for Class 1 and 2 Piping Welds, Wolf Creek Generating Station,” September 27, 2001

NRC Letter to O. L. Maynard, WCNOC, from S. Dembek, USNRC, "Approval of Relief Request for Application of Risk-Informed Inservice Inspection Program for ASME Boiler and Pressure Vessel Code Class 1 and 2 Piping for Wolf Creek Generating Station," December 13, 2001

SG-SGDA-03-19, "Steam Generator Degradation Assessment for Wolf Creek RF 13 Refueling Outage - October 2003," Revision 2

SG-00-10-018, "Wolf Creek RF 11 Condition Monitoring Assessment and Final Operational Assessment - March 2001," Revision 2

Nondestructive Examination Test Reports

Ultrasonic Examination Reports

RF13-11
RF13-12
RF13-13
DMH-002 (calibration report)
TMC-004 (calibration report)

Liquid Penetrant Examination Reports

2826, 2794, 2793, 2859, and 2861

Radiographic Reports

RT-3370 (3 shots: A-A, B-B, and C-C)

Visual Examination Reports

VT-2 Leakage examination report for WO 01-231340-008
VT-1 Visual examination report for WO 02-216675-006

Eddy Current Acquisition Technique Sheets and the Electric Power Research Institute's Examination Technique Specification Sheet (ETSS) Used For Qualification

SAP-01-03; ETSS 96001.1, 96004.3, 96005.2, 96008.1, 96010.1
SAP-02-03; ETSS 96001.1, 96004.3, 96005.2, 96008.1, 96010.1
SAP-04-03; ETSS 21409.1, 21410.1, 20510.1, 20511.1, 22401.1
SAP-05-03; ETSS 96910.1, 21998.1
SAP-06-03; ETSS 96511.1, 96511.2
SAP-08-03; ETSS 21409.1, 21410.1, 20510.1, 20511.1

Welding Procedure Qualification Records

PQR 234, -235, -236, -208, -209, and -238

LIST OF ACRONYMS

ADAMS	agency-wide document access management system
ALARA	as low as is reasonably achievable
CFR	*Code of Federal Regulations*
EPRI	Electric Power Research Institute
ET	eddy current examinations
LER	licensee event report
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PARS	publicly available records system
PI	performance indicator
PIR	performance improvement request
RAD	radiation absorbed dose
RF	refueling
URI	unresolved item