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

February 12, 2009

Mike Blevins, Executive Vice President
and Chief Nuclear Officer
Luminant Generation Company, LLC
ATTN: Regulatory Affairs
Comanche Peak Steam Electric Station
P.O. Box 1002
Glen Rose, TX 76043

Subject: COMANCHE PEAK STEAM ELECTRIC STATION - NRC INTEGRATED INSPECTION REPORT 05000445/2008005 AND 05000446/2008005

Dear Mr. Blevins:

On December 31, 2008, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Comanche Peak Steam Electric Station. The enclosed integrated inspection report documents the inspection findings, which were discussed on January 8, 2009, with Mr. R. Flores, Site Vice President, and other members of your staff.

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

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

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, and its enclosure, will be available electronically for public inspection in the NRC Public Document

Room or from the Publicly Available Records component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

George Replogle, Chief
Project Branch A
Division of Reactor Projects

Docket: 50-445: 50-446
License: NPF-87; NPF-89

Enclosure:
NRC Inspection Report 05000445/2008005 and 005000446/2008005
w/Attachment: Supplemental Information

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File located: R:_REACTORS_CPSES\CP2008-005RP-JGK.doc ADAMS ML090430331

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Publicly Avail	☑Yes ☐ No	Sensitive	☐ Yes ☑No	Sens. Type Initials	GDR

SRI:DRP/A	RI:DRP/A	C:DRP/A	C:DRS/EB1	C:DRS/OB
JGKramer	BWTindell	GDReplogle	TRFarnholtz	BWStapleton
/RA – telephonic/	***/not available/***	***/RA/***	***/RA/***	***/RA/***
02/12/2009	02/ /2009	02/10/2009	02/12/2009	02/12/2009
C:DRS/PSB1	**C:DRS/EB2**	**C:DRS/PSB2**	**C:DRP/A**	
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02/12/2009	02/10/2009	02/10/2009	02/12/2009	

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-445, 50-446
License:	NPF-87, NPF-89
Report:	05000445/2008005 and 05000446/2008005
Licensee:	Luminant Generation Company LLC
Facility:	Comanche Peak Steam Electric Station, Units 1 and 2
Location:	FM-56, Glen Rose, Texas
Dates:	September 22 through December 31, 2008
Inspectors:	J. Kramer, Senior Resident Inspector B. Tindell, Resident Inspector K. Clayton, Senior Reactor Inspector, Engineering Branch 1 M. Bloodgood, Reactor Inspector, Engineering Branch 1 L. Carson II, Senior Health Physicist, Plant Support Branch 2 L. Ricketson, P.E., Senior Health Physicist, Plant Support Branch 2 J. Dixon, Senior Resident Inspector, South Texas Project B. Tharakan, Resident Inspector, South Texas Project N. Okonkwo, Reactor Inspector, Engineering Branch 2 G. Tutak, Reactor Inspector, Engineering Branch 2 Accompanied by: D. Reinert, Reactor Inspector, Engineering Branch 1 N. Hernandez, Project Engineer, Project Branch A
Approved By:	George Replogle, Chief, Project Branch A Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000445/2008005, 05000446/2008005; 09/22/2008 - 12/31/2008; Comanche Peak Steam Electric Station, Units 1 and 2, Integrated Resident and Regional Report; Equipment Alignments Inservice Inspection Activities, Maintenance Risk Assessments and Emergent Work Control, Operability Evaluations, Postmaintenance Testing, and Surveillance Testing.

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

A. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, for the licensee's failure to follow procedures that required an evaluation and corrective actions in response to the effects of a borated water leak on primary coolant pressure boundary components. Corrective actions described as "Fix Now" were identified as boric acid deposits or anticipated accumulation of boric acid deposits which directly impact a carbon steel pressure boundary components or subcomponents and could result in increased corrosion rates. The inspectors identified that the inadequate evaluation and corrective actions resulted in the increased corrosion rate. The licensee entered the finding into their corrective action program as Smart Form SMF-2008-003194.

  The finding was more than minor using NRC Inspection Manual Chapter 0612, Appendix E, "Examples of Minor Issues," Example 4.a, because the inadequate evaluation led to the reactor vessel nozzle being adversely affected, in that the corrosion degraded the material condition of the carbon steel portions. The finding was determined to have very low safety significance because assuming worst case degradation, the finding would not result in exceeding the Technical Specification limit for reactor coolant system leakage or affect other mitigation systems resulting in a total loss of their safety function. The cause of the finding was related to the Human Performance crosscutting component of Decision Making in that the licensee failed to use conservative assumptions for decision making when evaluating degraded and nonconforming conditions [H1.b] (Section 1R08).

- Green. The inspectors identified three examples of a noncited violation of 10 CFR 50.65(a)(4) (Maintenance Rule) for the failure to adequately assess and manage the risk of maintenance activities during the outage. In two instances the licensee performed maintenance activities that initiated plant transients and increased the time at midloop without managing the risk. First, workers created a breach of the reactor coolant system boundary and loss of nitrogen cover gas pressure in the system. This caused the pressurizer level to rapidly increase approximately two feet. Second, the licensee removed high pressure seals for the flux thimble tubes creating a cold leg vent path during nozzle dam installation. This also caused spikes in level instrumentation and operators were required to stay in a midloop condition for an additional two hours. The third example involved emergency diesel generator synchronization to the 6.9 kV bus that was supporting the only running residual heat removal pump in a midloop condition with time to boil less than 10 minutes. The testing was originally schedule outside the midloop window. The licensee had started the activity but, after the inspectors raised concerns, the shift manager took actions to back out of the testing. After being properly assessed, the risk for this activity was classified as a red condition (the highest risk threshold), but the licensee was only in an orange condition. The licensee entered the finding into their corrective action program as Smart Forms SMF-2008-003143, SMF-2008-003172, SMF-2008-003196, and SMF-2008-003209.

The finding was more than minor because it was similar to non-minor Example 7.e from Manual Chapter 0612, Appendix E, "Examples of Minor Issues," in that, for the first two examples the activities required additional risk management actions and for the third example, the plant changed from a risk level of Orange to Red. Using Inspection Manual Chapter 0609, Appendix K, “Maintenance Risk Assessment and Risk Management Significance Determination Process,” the finding had very low safety significance because the incremental conditional core damage probability deficit was less than 1x10-6. The cause of the finding was related to the Human Performance crosscutting component of work control for the failure of the licensee to appropriately coordinate work activities [H3.b] (Section 1R13).

- Green. The inspectors documented a self-revealing noncited violation of Technical Specification 5.4.1a (Procedures) for an inadequate test procedure that resulted in inadvertently holding open a main steam safety valve at power. During testing, a test engineer separated a quick disconnect fitting in accordance with the procedural instructions. The action sealed in nitrogen pressure in the test rig and caused the valve to remain held open. In response to the event, operators reduced reactor power to compensate for the partially open safety valve until maintenance personnel closed the valve. The licensee entered the finding into their corrective action program as Smart Form SMF-2008-002946.

  The finding was more than minor because it was associated with the procedure quality attribute of the initiating events cornerstone, and directly affected the cornerstone objective to limit the likelihood of those events that upset plant stability during power operations. Using Manual Chapter 0609, Attachment 4, “Phase 1 - Initial Screening and Characterization of Findings,” the finding had very low safety significance because it did not contribute to the likelihood of mitigating equipment being unavailable. This finding did not have a crosscutting aspect because the procedure section was last revised several years earlier (Section 1R22).

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation of Technical Specification 5.4.1.a (Procedures), for the licensee’s failure to erect scaffolding over safety-related equipment with adequate seismic supports. As a result, the scaffolding would likely fail during a seismic event and impact the service water system. Contract personnel assembled the scaffolding and were under perceived time pressure to finish the work, which was their last task before departing the site. A licensee supervisor inspected the scaffolding and failed to identify the deficiency. The licensee entered the finding into their corrective action program as Smart Form SMF-2008-003683.

  The finding was more than minor because it was similar to non-minor Example 4.a from Manual Chapter 0612, Appendix E, "Examples of Minor Issues," in that the scaffolding could adversely affect safety related equipment during a seismic event. Using the NRC Manual Chapter 0609, Attachment 4, “Phase 1 - Initial Screening and Characterization of Findings,” the finding was a qualification deficiency confirmed not to result in loss of operability or functionality. This finding had a Human Performance crosscutting aspect (work practices component) because the licensee failed to ensure adequate supervisory and management oversight of work activities, including contractors, such that nuclear safety was supported [H4.c] (Section 1R04).

- Green. The inspectors identified a noncited violation of Technical Specification 5.4.1a (Procedures) for the failure to have adequate instructions in place for containment walkdowns looking for fibrous material. As a result, the licensee entered a mode where the containment sumps were required to be operable with unidentified fibrous material in the containment. The licensee had not identified the material during several walkdowns in response to NRC Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors,” and failed to identify several additional instances of fibrous material after inspectors initially identified some of the material. The licensee entered the finding into their corrective action program for resolution as Smart Form SMF-2008-003587.

  The finding was more than minor because it was associated with the procedure quality attribute of the mitigating systems cornerstone, and it affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using NRC Manual Chapter 0609, Attachment 4, “Phase 1 – Initial Screening and Characterization of Findings,” the finding had very low safety significance because it did not represent a loss of system safety function or cause inoperability of a system or train. The finding had a Human Performance crosscutting aspect (work control component) in that the work instructions and pre-job brief failed to effectively incorporate job site conditions into the work instructions and consider that both sides of the seals required inspection [H3.a] (Section 1R15).

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, for the failure to follow procedures to enter a malfunction of a reactor trip bypass breaker into the corrective action program. The breaker tripped slower than permitted during response time testing and was inoperable. Because the condition was not entered into the corrective action program, the licensee did not evaluate the condition or assess the extent of condition. The licensee entered the finding into their corrective action program as Smart Forms SMF-2008-003735 and SMF-2008-003767.

  The finding was more than minor because, if left uncorrected, it would have led to a more safety significant concern. Specifically, because the cause of the failure would not have been fully evaluated and appropriate corrective actions may not be initiated. Once entered into the corrective action program, the licensee identified additional corrective measures. Using NRC Inspection Manual Chapter 0609, Attachment 4, “Phase 1 - Initial Characterization and Screening of Findings,” the finding had very low safety significance because the condition did not result the inoperability of the reactor trip breaker when it was required to be operable. The cause of this finding was related to the Problem Identification and Resolution crosscutting component of the corrective action program, in that, the licensee failed to enter the issue into their corrective action program [P1.a] (Section 1R19).

B. Licensee-Identified Violations

None

REPORT DETAILS

Summary of Plant Status

Comanche Peak Steam Electric Station Unit 1 began the reporting period at approximately 96 percent power in a coast-down to Refueling Outage 1RF13. On September 27, 2008, operators performed a unit shutdown to begin the scheduled refueling outage. On October 16, 2008, the outage ended when the main generator breakers were closed. On October 20, 2008, Unit 1 returned to 100 percent power and operated at essentially 100 percent power for the remainder of the reporting period.

Comanche Peak Steam Electric Station Unit 2 began the reporting period at 100 percent power. On November 8, 2008, operators reduced power to approximately 45 percent to repair a packing leak on Valve 2MS-0088, "Steam Generator 2-03 Level Transmitter 0537 Lower Root Valve." Unit 2 returned to 100 percent power the same day and operated at essentially 100 percent power for the remainder of the reporting period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

Readiness to Cope with External Flooding

a. Inspection Scope

The inspectors evaluated the design, material condition, and procedures for coping with heavy rains as related to roof drains on safety related buildings. The evaluation included a review to check for deviations from the descriptions provided in the Final Safety Analysis Report for features intended to mitigate the potential for flooding. Specific documents reviewed during this inspection are listed in the attachment.

This activity constituted completion of one external flooding sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04)

.1 Partial Equipment Walkdowns

a. Inspection Scope

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

- Unit 1 containment spray system inside containment on October 14, 2008
- Units 1 and 2 station service water system during scaffolding activities on October 30, 2008
- Units 1 and 2 125 VDC systems on November 14, 2008

The inspectors selected these systems based on their risk significance relative to the reactor safety cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could affect the function of the system and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, Final Safety Analysis Report, technical specification requirements, outstanding work orders, Smart Forms, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program with the appropriate significance characterization. Specific documents reviewed during this inspection are listed in the attachment.

These activities constituted completion of three partial system walkdown samples.

b. Findings

Introduction. The inspectors identified a noncited violation of Technical Specification 5.4.1.a (Procedures), for the licensee's failure to erect scaffolding over safety-related equipment with adequate seismic supports. As a result, the scaffolding would likely fail during a seismic event and impact the service water system. Contract personnel assembled the scaffolding and were under perceived time pressure to finish the work, which was their last task before departing the site. A licensee supervisor inspected the scaffolding and failed to identify the deficiency.

Description. On October 30, 2008, while touring the service water intake structure, the inspectors observed that a large scaffold structure above safety-related service water equipment was not adequately seismically supported. Upon notification by inspectors, the licensee installed additional restraints to the scaffolding.

The scaffold structure in the service water intake structure was approximately 24 feet in height, 50 feet in length, and 5 feet wide. The scaffolding spanned both trains of service water in both units. The scaffolding supports did not meet the requirements of Procedure STA 690, "Erecting and Control of Scaffolding," Revision 3. Specifically, the scaffolding did not contain adequate lateral supports and some of the supports were attached to undersized structural members.

After the inspectors identified the issue, the licensee performed an evaluation which determined that the scaffolding would likely fail in a seismic event and impacts the 24-inch service water piping with no adverse effects due to the ruggedness of the pipe and components. Therefore, the service water system would remain operable. The inspectors independently reviewed the scaffolding configuration and potential targets, which included conduit for the service water pump discharge valves, small bore piping for the pump bearing coolers, and manual valves for the service water cross-connect. The inspectors determined that, while it was likely that these targets would be impacted by the scaffolding during a seismic event, the scaffolding would most likely not cause severe enough damage to cause the equipment to lose function. Therefore, the inspectors concluded that the scaffolding had a reasonable potential for impacting and affecting safety related equipment in a seismic event.

Contractors had erected the scaffolding as their last job before departing the site, creating a perceived time pressure. In addition, the supervisory personnel who inspected the scaffolding did not identify the deficiencies. These were significant contributors to this finding.

Analysis. The erection of scaffolding over safety-related service water equipment without adequate seismic supports was a performance deficiency. The finding was more than minor because it was similar to non-minor Example 4.a from Manual Chapter 0612, Appendix E, "Examples of Minor Issues," in that the scaffolding could adversely affect safety related equipment during a seismic event. Using the NRC Manual Chapter 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," the finding was a qualification deficiency confirmed not to result in loss of operability or functionality. This finding had a Human Performance crosscutting aspect (work practices component) because the licensee failed to ensure adequate supervisory and management oversight of work activities, including contractors, such that nuclear safety was supported [H4.c].

Enforcement. Technical Specification 5.4.1.a requires, in part, that written procedures to be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A. Regulatory Guide 1.33, Appendix A, Section 9 recommends, in part, that maintenance that can affect the performance of safety-related equipment should be properly pre-planned and performed in accordance with written procedures. Procedure STA-690, step 6.9.2, states, in part, that scaffolding should be provided with lateral restraints to prevent displacement and or tipping in a seismic event. Procedure STA-690 step 6.12, also requires, in part, that installation of scaffolding that does not meet the applicable requirements of this procedure shall be evaluated by engineering. Contrary to the above, on October 28, 2008, the licensee erected scaffolding without providing proper lateral restraints to prevent the scaffolding from falling on to safety-related equipment in a seismic event and failed to perform an engineering evaluation. Since the violation was of very low safety significance and was documented in the licensee's corrective action program as Smart Form SMF-2008-003683, it is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy:
NCV 05000445/2008005-01; 05000446/2008005-01, "Non-Seismic Scaffolding Installed Over Service Water Equipment."

.2 Semi-Annual Complete System Walkdown

a. Inspection Scope

On December 18, 2008, the inspectors performed a complete system alignment inspection of the Unit 2 turbine driven auxiliary feedwater system to verify the functional capability of the system. The inspectors selected this system because it was considered both safety-significant and risk-significant in the licensee's probabilistic risk assessment. The inspectors performed a walkdown of the system to review mechanical and electrical equipment configuration electrical power availability, system pressure and temperature indications, component lubrication, component and equipment cooling, hangers and supports, operability of support systems, and to ensure that ancillary equipment or debris did not interfere with equipment operation. The inspectors reviewed a sample of past and outstanding work orders to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the corrective action program database to ensure that system equipment alignment problems were being identified and appropriately resolved. Specific documents reviewed during this inspection are listed in the attachment.

These activities constituted completion of one completed system walkdown sample.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Fire Inspection Tours

a. Inspection Scope

The inspectors conducted fire protection walkdowns in the following risk-significant plant areas.

- Alternate power diesel generator area outside Unit 1 containment on September 24, 2008
- Unit 1 reactor coolant pump oil collection systems on September 28, 2008
- Intervening area between redundant train shutdown cooling suction valves in the Unit 1 containment during midloop on September 29, 2008
- Unit 1 Diesel Generator 1-01 on October 8, 2008

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

These activities constituted completion of four quarterly fire-protection inspection samples.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

On November 3, 2008, the inspectors performed a walkdown on the service water intake structure for flood protection measures. The inspectors reviewed the Final Safety Analysis Report and the flooding analysis to assess susceptibilities involving internal flooding. The inspectors also reviewed the corrective action program to determine if licensee personnel identified and corrected flooding problems; verified that operator actions for coping with flooding can reasonably achieve the desired outcomes; and walked down the area listed below to verify the adequacy of equipment, floor, and wall penetration seals. Specific documents reviewed during this inspection are listed in the attachment.

These activities constituted completion of one flood protection measures inspection sample.

b. Findings

No findings of significance were identified.

1R08 In-service Inspection Activities (71111.08)

.1 Nondestructive Examination Activities and Welding Activities (71111.08-02.01)

a. Inspection Scope

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

The inspectors directly observed the following nondestructive examinations:

SYSTEM	WELD IDENTIFICATION	EXAM TYPE
Reactor Coolant System	TBX-1-2100-1 (Pressurizer shell side weld)	Ultrasonic Testing
Reactor Coolant System	TBX-1-2100-6 (Pressurizer shell bottom weld)	Ultrasonic Testing
Reactor Coolant System	TBX-1-4201-H4 (Pipe Hanger for Accumulator “B” Discharge line)	PT, VT-3
Reactor Coolant System	TBX-1-4201- 3 (Pipe weld for Accumulator “B” Discharge line)	PT, VT-3

The inspectors reviewed records, including digital video recordings, for the following nondestructive examinations:

SYSTEM	WELD IDENTIFICATION	EXAM TYPE
Reactor Coolant System	Reactor vessel nozzle to safe end/safe end to pipe hot leg dissimilar metal welds: TBX-1-4400-1/2 for loop 4 TBX-1-4300-1/2 for loop 3 TBX-1-4200-1/2 for loop 2 TBX-1-4100-1/2 for loop 1	Ultrasonic Testing (automated, video)
Reactor Coolant System	Bottom Mounted Instrumentation	BMV (video)
Reactor Coolant System	Reactor Vessel Upper Head	BMV (pictures)
Reactor Coolant System	All pressurizer nozzle weld overlay exams, including: TBX-1-4501-1/2, Safety A TBX-1-4501-12/13, Safety B TBX-1-4501-23/24, Safety C TBX-1-4502-1/2, PORV TBX-1-4503-31/30, Spray TBX-1-4500-8/7, Surge Line	Ultrasonic Testing

During the review and observation of each examination, the inspectors verified that activities were performed in accordance with ASME Boiler and Pressure Vessel Code requirements and applicable procedures. Indications were compared with previous examinations and dispositioned in accordance with ASME Code and approved procedures. The qualifications of all nondestructive examination technicians performing the inspections were verified to be current. None of the observed or reviewed nondestructive examinations identified any relevant indications and cognizant licensee personnel stated that no relevant indications were accepted by the licensee for continued service.

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

These activities constituted completion of one sample.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The licensee performed the required visual inspection of pressure-retaining components above the reactor pressure vessel head. The head was replaced in the last refueling outage (1RF12) during the spring of 2007. Although not required by the inservice inspection program during this outage, a visual inspection was performed. The inspectors reviewed pictures taken during this inspection and confirmed that there was no evidence of leaks or boron deposits on the surface of the reactor pressure vessel head or related insulation. The inspectors verified that the personnel performing the visual inspection were certified as Level III VT-2 examiners. Specific documents reviewed during this inspection are listed in the attachment.

These activities constituted completion of one sample.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

The inspectors reviewed the documentation associated with the licensee's boric acid corrosion control walkdown as specified in Procedure STA-737, "Boric Acid Corrosion Detection and Evaluation," Revision 4. Visual records of the components and equipment were also reviewed by the inspectors. The inspectors verified through record review that the boric acid corrosion control inspection efforts were directed towards locations where boric acid leaks can cause degradation of safety-related components. Additionally, the inspectors independently performed examinations of piping and components containing boric acid during a walkdown of the containment building and the auxiliary building. On those components where boric acid was identified, the engineering evaluations gave assurance that the ASME Code wall thickness limits were properly maintained with the exception of the Loop 4 hot leg reactor nozzle documented in the subsequent finding. The evaluations also confirmed that the corrective actions performed for evidence of boric acid leaks were consistent with requirements of the ASME Code with the exception of the Loop 4 hot leg reactor nozzle. Specific documents reviewed during this inspection are listed in the attachment.

These activities constituted completion of one sample.

b. Findings

Introduction. The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion V, for the failure of the licensee to properly evaluate and perform corrective actions in response to the effects of a borated water leak on primary coolant pressure boundary components. Specifically, on March 3, 2007, the licensee failed to perform an adequate evaluation and take corrective actions in accordance with Procedure STA-737, "Boric Acid Corrosion Detection and Evaluation," Revision 4.

Procedure STA-737 states, in part, that leaks that result in the potential for degrading pressure boundary material due to corrosion require engineering evaluation in addition to rework/repair. Corrective actions described as “Fix Now” were identified as boric acid deposits or anticipated accumulation of boric acid deposits which directly impact a carbon steel pressure boundary component or subcomponent and could result in increased corrosion rates. The inadequate evaluation and corrective actions resulted in the increased corrosion rate identified by the inspectors on October 1, 2008.

Description. On February 28, 2007, the licensee identified a leak of the Unit 1 Loop 4 “Sandbox” plug which allowed for borated water, with a boron concentration of approximately 2500 parts per million, to flow into the sandbox area and overflow into the adjacent loop room and reactor vessel sump. During Outage 1RF12 the refueling cavity was flooded up for up to 55 days due to steam generator and head replacement activities. Due to this leak, equipment in several areas, including this hot leg nozzle, was exposed to borated water for an extended amount of time. Licensee management decided to continue outage activities with the refueling cavity flooded up while equipment in several areas were being exposed to large amounts of borated water from the leak. During a mid-cycle outage on March 3, 2008, the leak was evaluated in Smart Form SMF-2007-000670-03 which stated, in part, that:

- Large amounts of boric acid were present in the cavity
- The insulation was not removed from the hot-leg piping to allow for inspection of the nozzle
- A “red residue” was present on most carbon steel components including supports and penetration areas
- Small boric acid deposits were present in the insulation crevices
- Boric acid residue was noted on the walls

During the next outage (1RF13), specifically on September 29, 2008, a bare metal visual inspection was performed on the Loop 4 hot leg nozzle’s dissimilar metal weld. The inspectors observed a slight residue of what appeared to be boric acid was present on Loop 4 hot leg and that the condition was reported to the boric acid program coordinator for evaluation. The boric acid program coordinator performed an evaluation of the boric acid residue and determined that no further action was required. The evaluation was not documented in a smart form in accordance with Procedure STA-737.

On October 1, 2008, the inspectors questioned the indications of corrosion present on the Loop 4 hot leg nozzle carbon steel section during the review of pictures taken by a vendor during measurements for possible mitigation techniques in the future on these nozzles with respect to the dissimilar metal welds. The corrosion was located on the carbon steel portion of the hot-leg nozzle dissimilar metal weld and was observed by the inspectors as flaking corrosion. These pictures were not evaluated by the boric acid corrosion engineer until after the licensee was questioned on the extent of the corrosion. In addition, the inspectors requested pictures from the bare metal visual inspection conducted during the previous Outage 1RF12, on February 25, 2007. The licensee could not produce pictures of the condition of this nozzle during this inspection. The licensee also failed to document the inspection as required by the “Reactor Coolant System Pressure Boundary Dissimilar Metal Weld Visual Examination Plan,” Revision 2. The examination was documented as part of a work order instead of Figure 1, “Visual

Examination Report," and Figure 2, "Boric Acid Deposit Evaluation Report," of the examination plan, with the results of no indications.

On October 1, 2008, the licensee discovered the same type of leak on the same cavity seal cover (sandbox cover) for the loop 4 cavity as the previous outage, which was documented in Smart Form SMF-2008-003194. During this outage, the total flood up time was eight days. Consequently, equipment in several areas was exposed to 2500 parts-per-million borated water during two outages totaling 63 days. These areas included:

- Loop cavity seal areas, including the reactor vessel nozzle for Loop 4 hot leg
- The loop rooms
- Whip restraint areas for each loop's associated pipe
- Reactor vessel bottom
- Reactor vessel walls

Once the licensee was alerted to the corrosion of the Loop 4 hot leg reactor vessel nozzle during the current outage (1RF13), they developed a strategy for inspecting those areas affected by the borated water once the refueling cavity was drained and the areas were accessible. The inspectors determined that the following items should be included in the operability evaluation on equipment exposed to borated water that the licensee would complete prior to startup:

- If a 0 degree ultrasonic test probe is used to get pipe thickness remaining or amount lost, ensure that the focus is on the outside diameter of the pipe

- Include the insulation effects (i.e. trapped water under the insulation) on the corrosion rate and overall material loss discussion in the report

- Evaluate all major items in the evaluation/operability that were exposed to the borated water. These evaluations must include ASME code compliance verifications

- Any parallels drawn between corrosion rates of different materials should be explained in detail (example: if seismic restraints for the pipe are 106-B grade steel and the nozzle is standard grade carbon steel, a direct corrosion rate/material loss would not be possible without some correlation data)

- Reasons for not removing the Marinite insulation to inspect the remainder of the nozzle and portions of the vessel that were exposed to borated water

On October 13, 2008, the licensee completed an evaluation (SMF-2008-003194-05) of the current conditions of the areas affected by the 2007 and 2008 borated water leaks. The inspectors reviewed the evaluation and discussed the results with the licensee. All the related equipment was declared operable in the evaluation with the following conclusions:

- The components in the sandbox and below were exposed to borated water from the leaking sandbox cover for a short period of time

- The borated water was at low temperature (less than 104 degrees Fahrenheit) for the duration of the exposure

- The concentration of boron in the water was low (less than 2600 ppm)

- The components dried quickly after the refueling cavity was drained

- After the leak was stopped and the electrolyte was gone, the corrosion stopped

- Past and current walk downs show no degradation of materials that would affect operability

- When inspected using a boroscope, hidden areas of water intrusion showed no degradation that would affect operability

Analysis. The performance deficiency associated with this finding was the failure of the licensee to properly evaluate and perform corrective actions in response to the effect on components due a leak of borated water. This finding is associated with the initiating events cornerstone attribute of equipment performance and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during at power operations. Using Inspection Manual Chapter 0612, Appendix E, "Examples of Minor Issues," Example 4.a, the finding is greater than minor due to the inadequate evaluation that led to the reactor vessel nozzle being adversely affected, in that, the corrosion degraded the material condition of the carbon steel portions of the nozzle. Using NRC Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to have very low safety significance because assuming worst case degradation, the finding would not result in exceeding the Technical Specification limit for any reactor coolant system leakage or affect other mitigation systems resulting in a total loss of their safety function. This finding has a cross-cutting aspect associated with the decision making component of the human performance area in that the licensee failed to use conservative assumptions for operability decision-making when evaluating degraded and nonconforming conditions [H1.b].

Enforcement. The inspectors determined that 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings and shall be accomplished in accordance with these instructions, procedures, or drawings. Procedure STA-737 "Boric Acid Corrosion Detection and Evaluation," Revision 4, states, in part, that leaks that result in the potential for degrading pressure boundary material due to corrosion require engineering evaluation in addition to rework/repair. Contrary to the above, on March 3, 2007, the licensee failed to accomplish an adequate evaluation and take corrective actions as required by Procedure STA-737, following a leak from the Sandbox cover. Corrective actions described as "Fix Now" were identified as boric acid deposits or anticipated accumulation of boric acid deposits which directly impact a carbon steel pressure boundary component or subcomponent and could result in increased corrosion rates. Because the licensee failed to adequately evaluate the condition resulting from the Sandbox cover leak on February 28, 2007, the licensee failed to inspect the carbon steel portion of the Loop 4 hot leg reactor vessel nozzle and identify any material degradation present on the component. Since the violation was of very low significance and was documented in the licensee's corrective action program as Smart Form SMF-2008-003194-05, it is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000445/2008005-02), Failure to Adequately Evaluate Material Condition Following a Boric Acid Leak.

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

a. Inspection Scope

The inspectors assessed the insitu screening criteria to assure consistency between assumed nondestructive examination flaw sizing accuracy and data from the EPRI examination technique specification sheets. The inspectors determined that no conditions were identified that warranted insitu pressure testing. The steam generators were replaced in the 1RF12 outage during the spring of 2007 with Delta 76 models containing Alloy 690 thermally treated 690 U-tubes. During the outage that immediately followed the replacement of the steam generators, a 100 percent review of all tubes in all steam generators is required by the Technical Specifications. Therefore a 100 percent review of all tubes in all steam generators was performed during this outage.

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

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. Because these steam generators were replaced, no comparison could be made with the previous outage results. The number of identified indications fell within the range of prediction and was quite consistent with predictions from the vendor for the first outage. No new damage mechanisms were identified during this inspection. The total number of plugged tubes prior to this inspection was one tube. Tube R32C90 in Steam Generator 3 was plugged at the factory. No tubes were plugged during this outage.

The inspection procedure specified confirmation that the steam generator tube eddy current test scope and expansion criteria meet Technical Specification requirements, EPRI guidelines, and commitments made to the NRC. The inspectors evaluated the recommended steam generator tube eddy current test scope established by Technical Specification requirements and the licensee’s degradation assessment report. The inspectors compared the recommended test scope to the actual test scope and found that the licensee had accounted for all known flaws and had, as a minimum, established a test scope that met Technical Specification requirements, EPRI guidelines, and commitments made to the NRC.

The base scope inspection plan required 100 percent tube inspection for this outage (1RF13). The inspection scope (all steam generators) for Outage 1RF13 included:

- 100 percent full length bobbin inspection (except straight legs only in Rows 1-3)
- 20 percent plus-point inspection of hot leg top of tube sheet from 3 inches above to 3 inches below
- 100 percent plus-point inspection of the U-bends in Rows 1-3
- 100 percent plus-point inspection of all dents/dings greater than 5 volts

- 100 percent plus-point inspection of the traceable anomaly signals (23 tubes)

- Special interest rotating pancake coil (freespan signals without historical resolution, bobbin I-code indications)

- 100 percent tube plug video inspection (1 tube)

- Top of tube sheet secondary side video inspection including foreign object search and retrieval

- Upper bundle video inspection in steam generator 1

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

These activities constituted completion of one sample.

b. Findings

No findings of significance were identified.

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

a. Inspection scope

The inspection procedure requires review of a sample of problems associated with inservice inspections documented by the licensee in the corrective action program for appropriateness of the corrective actions. The inspectors reviewed 20 Smart Forms, which dealt with inservice inspection activities and found the corrective actions were appropriate. Specific documents reviewed during this inspection are listed in the attachment.

b. Findings and Observations

The inspectors concluded that the licensee has 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 has an effective program for applying industry operating experience. No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

On December, 9, 2008, the inspectors observed a crew of licensed operators in the plant's simulator to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- Licensed operator performance
- Crew's clarity and formality of communications
- Crew's ability to take timely actions in the conservative direction
- Crew's prioritization, interpretation, and verification of annunciator alarms
- Crew's correct use and implementation of abnormal and emergency procedures

- Control board manipulations
- Oversight and direction from supervisors
- Crew’s ability to identify appropriate technical specification actions
- Crew’s ability to implement appropriate emergency plan actions and notifications

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

These activities constituted completion of one quarterly licensed-operator requalification program sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors evaluated following risk significant systems, components, and degraded performance issues:

- Work practices related to safety-related tubing
- Safety injection accumulators exceeded unavailability criteria
- Units 1 and 2 centrifugal charging pumps

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

- Implementing appropriate work practices
- Identifying and addressing common cause failures
- Scoping of systems in accordance with 10 CFR 50.65(b)
- Characterizing system reliability issues for performance
- Charging unavailability for performance
- Trending key parameters for condition monitoring
- Ensuring proper classification in accordance with 10 CFR 50.65(a)(1) or (a)(2)

The inspectors verified appropriate performance criteria for structures, systems, and components classified as having an adequate demonstration of performance through preventive maintenance, as described in 10 CFR 50.65(a)(2), or as requiring the establishment of appropriate and adequate goals and corrective actions for systems classified as not having adequate performance, as described in 10 CFR 50.65(a)(1).

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the corrective action program with the appropriate significance characterization. Specific documents reviewed during this inspection are listed in the attachment.

These activities constituted completion of three quarterly maintenance effectiveness samples.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

- Unit 1 overall scheduled outage risk profile on September 23, 2008
- Scheduled solid state protection system testing with Steam Generator 2-03 level bistable tripped
- Mobile crane movement of main steam isolation valves on September 12, 2008
- Unit 1 hot midloop activities

The inspectors selected these activities based on potential risk significance relative to the reactor safety cornerstones. As applicable for each activity, the inspectors verified that licensee personnel performed risk assessments as required by 10 CFR 50.65(a)(4) and that the assessments were accurate and complete. When licensee personnel performed emergent work, the inspectors verified that the licensee personnel promptly assessed and managed plant risk. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed the technical specification requirements and inspected portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met. Specific documents reviewed during this inspection are listed in the attachment.

These activities constituted completion of four maintenance risk assessments and emergent work control inspection samples.

b. Findings

Introduction. The inspectors identified three examples of a Green noncited violation of 10 CFR 50.65(a)(4) (Maintenance Rule) for the failure to adequately assess and manage the risk of maintenance activities during the outage. In two instances the licensee performed maintenance activities that initiated plant transients and increased the time at midloop without managing the risk. First, workers created a breach of the reactor coolant system boundary and loss of nitrogen cover gas pressure in the system. This caused the pressurizer level to rapidly increase approximately two feet. Second, the licensee removed high pressure seals for the flux thimble tubes creating a cold leg vent path during nozzle dam installation. This also caused spikes in level instrumentation and operators were required to stay in a midloop condition for an additional two hours. The third example involved emergency diesel generator synchronization to the 6.9 kV bus that was supporting the only running residual heat removal pump in a midloop condition with time to boil less than 10 minutes. The testing was originally schedule outside the midloop window. The licensee had started the activity but, after the inspectors raised concerns, the shift manager took actions to back out of the testing. After being properly assessed, the risk for this activity was classified as a red condition (the next risk threshold), but the licensee was only in an orange condition.

Description. On September 29, 2008, the work window manager authorized maintenance personnel to perform preparatory work for removal of a pressurizer safety valve. The communications between the craft and the work window manager were not clear concerning the work scope. As a result, the craft disassembled a flange connection between the safety and the pressurizer relief tank. This caused a breach of the reactor coolant system boundary and loss of nitrogen cover gas pressure in the system. As a result, the pressurizer level rapidly increased approximately two feet. The time-to-boil of the reactor coolant system was less than 15 minutes when this event occurred. The licensee documented this issue in Smart Form SMF-2008-003143.

On September 30, 2008, a misunderstood communication between the work window manager and a contractor representative resulted in work in a different plant condition than originally planned. With the unit in a midloop condition and time-to-boil less than 10 minutes, craftsmen removed high pressure seals for the flux thimble tubes, which created a cold leg vent path during nozzle dam installation. As a result of the removal, operators observed spikes in indicated pressurizer level. In addition, operators stayed in a midloop condition for an additional two hours before completing the nozzle dam installation. The licensee determined that vague procedural guidance concerning the removal of the high pressure seals and thimble tubes contributed to the event. The licensee documented this issue in Smart Form SMF-2008-003172.

On September 30, 2008, the licensee started Diesel Generator 1-01 to perform post maintenance testing of the diesel generator. The inspectors observed the operators were preparing to synchronize the diesel generator to the 6.9 kV bus. This bus was supplying the only running residual heat removal pump, with time to boil less than 10 minutes. The inspectors questioned the shift manager about the necessity for performing the testing at that time. The shift manager determined that the testing was neither prudent nor necessary and stopped the testing prior to placing the diesel on the bus. The licensee documented this issue in Smart Form SMF-2008-003196.

The inspectors reviewed the licensee’s evaluation of the diesel generator issue documented as Evaluation EVAL-2008-003196-02. In the evaluation, the licensee documented that two senior reactor operators performed a risk assessment during the pre-job brief for the evolution. The senior reactor operators had discussed the test, risk, consequence, and contingencies during the brief and were in agreement that no reasonable failure could cause a loss of the residual heat removal pump. However, the inspectors noted that the operators failed to recognize the risk associated with synchronizing the emergency diesel generator to the bus. Specifically, if the generator was not paralleled and synchronized to the bus properly, supply breakers could trip on over-current. Therefore, there was some additional risk associated with the action.

The licensee performed and documented a deterministic risk assessment of the issue in Evaluation EVAL-2008-003196-01. In the evaluation, the licensee used the Outage Risk Assessment and Management software to determine the risk. Using the plant conditions at the time of testing, the revised risk level was RED, the most risk significant of the four levels of risk in the Comanche Peak program. The original risk assessment was Orange (the next lower threshold).

Analysis. The failure to properly manage risk associated with maintenance activities was a performance deficiency. The finding was more than minor because it was similar to non-minor Example 7.e from Manual Chapter 0612, Appendix E, "Examples of Minor Issues," in that, for the first two examples the activities required additional risk management actions. For the third example, the plant changed from a risk level of Orange to Red (the highest risk threshold). Using Inspection Manual Chapter 0609, Appendix K, “Maintenance Risk Assessment and Risk Management Significance Determination Process,” the finding had very low safety significance because the incremental conditional core damage probability deficit was less than 1x10-6. The cause of the finding was related to the Human Performance crosscutting component of work control for the failure of the licensee to appropriately coordinate work activities [H3.b].

Enforcement. 10 CFR 50.65(a)(4) requires, in part, that the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. Contrary to the above, on September 29 and 30, 2008, the licensee failed to assess and manage the risk associated with maintenance activities during the Unit 1 refueling outage. Since the violation was of very low safety significance and was documented in the licensee’s corrective action program as Smart Form SMF-2008-003209-00, it is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000445/2008005-03, "Failure to Assess and Manage Risk Associated with Maintenance Activities."

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- SMF-2008-003013-00, heavy load stored on safeguards building roof exceeded assumed live load
- SMF-2008-003229-00, Fibrous damming material for joint gap seal found in containment
- SMF-2008-003245-00, gas void found in residual heat removal line during ultrasonic testing
- SMF-2008-003278-00, Unit 1 lower core plate foreign objects including fuel assembly protective bottom grids, P-grids
- Procedures MSM-C0-8722 and OPT-206B, Unit 1, Turbine-Driven Auxiliary Feedwater Pump trip and throttle valve trip hook and latch-up lever not having complete engagement
- SMF- 2008-4057-00, annunicator and trip status light fail to illuminate when steam pressure Loop 4, Protection Set IV, Channel 0546 placed in trip

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

These activities constituted completion of six operability evaluation inspection samples.

b. Findings

Introduction. The inspectors identified a Green noncited violation of Technical Specification 5.4.1a (Procedures) for the failure to have adequate instructions in place for containment walkdowns looking for fibrous material. As a result, the licensee entered a mode where the containment sumps were required to be operable with unidentified fibrous material in the containment. The licensee had not identified the material during several walkdowns in response to NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," and failed to identify several additional instances of fibrous material after inspectors initially identified some of the material.

Description. Fibrous material inside containment may be an adverse condition if it is able to transport to the emergency core cooling system sump during an accident that requires emergency core cooling system recirculation from the containment, in that, it may clog the sump screens and reduce the suction head of the emergency core cooling system pumps. The NRC issued Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," which included a request for a response from licensees detailing the head loss postulated from debris accumulation on the sump screen during certain accidents. As part of the response, the licensee performed detailed walkdowns of containment to identify debris that could affect the emergency core cooling system sumps.

The unit containment buildings have multiple concrete joints with environmental seals, which contain a foam material. The seals were installed with a fibrous material in the expansion joint as a dam for the foam. The licensee was aware of this, however, they believed that the fibrous material was removed after the seal was installed, and did not verify this during the debris walkdowns.

On October 2, 2008, during a Unit 1 containment tour, inspectors identified fibrous material in a concrete expansion joint that was part of a doorway to the pressurizer cubicle. The licensee removed the fibrous material and documented a plan in SMF-2008-003229-00 to identify and remove other fibrous damming material in containment. The licensee personnel performed a walkdown using drawings that identified potentially locations of fibrous material. The personnel performing the walkdown failed to view some of the environmental penetration seals from both sides, which meant that in some cases, the fibrous material on the other side of the seal was not seen. In one case, the seal extended on both sides of the penetration, which most likely means that the fibrous material was still inside and not visible. The licensee had noted in SMF-2008-003229-00 on October 10, 2008 that the soft foam seal would most likely blow out under forces generated from a pipe break. Therefore, fibrous material that was concealed from view by the foam seal could still transport to the sump. However, this information was not effectively communicated in the pre-job brief or in the work instructions to the personnel performing the walkdown two days later so that they understood the necessity of viewing both sides of each seal.

The inspectors toured the Unit 1 containment on October 14, 2008, while the unit was in Mode 4, when one train of emergency core cooling system was required to be operable. The inspectors noted that an expansion joint in the wall of the letdown orifice valve room contained fibrous material. Later, the licensee determined that their walkdown had failed to identify at least 3 areas of fibrous material. The fibrous material was partially removed after the unit reached Mode 3 on October 15, 2008, when both trains of the Emergency core cooling system sumps were required to be operable; however, a small portion was left in place. The licensee determined that the fibrous material found, while adverse to the performance of the sump, was not enough to challenge operability of the emergency core cooling system. The licensee's failure to identify the additional areas with fibrous material delayed the removal of the fibrous material until after entry into Mode 3, when both trains of emergency core cooling system were required to be operable.

Analysis. The failure to have adequate instructions to identify fibrous material in the Unit 1 containment that could adversely affect emergency core cooling system sump function was a performance deficiency. The finding was more than minor because it was associated with the procedure quality attribute of the mitigating systems cornerstone, and it affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using NRC Manual Chapter 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," the finding had very low safety significance because it did not result in a loss of operability of a train of the emergency core cooling system. The finding had a Human Performance crosscutting aspect (work control component) in that the work instructions and pre-job brief failed to effectively incorporate job site conditions into the work instructions and consider that both sides of the seals required inspection [H3.a].

Enforcement. Technical Specification 5.4.1.a (Procedures) requires the licensee, in part, to implement the procedures recommended by Appendix A to Regulatory Guide 1.33, Revision 2. Appendix A to Regulatory Guide 1.33, Section 9 recommends procedures for performing maintenance. Contrary to the above, on October 13, 2008, the licensee failed to provide personnel with an adequate procedure to identify fibrous material that could affect emergency core cooling system sump strainer performance. Since the violation was of very low safety significance and was documented in the licensee's corrective action program as Smart Form SMF-2008-003587-00, it is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000445/2008005-04, "Inadequate Instructions Causes Failure to Identify Fibrous Material in Containment."

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

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

- Unit 1 Diesel Generator 1-02 cylinder head assembly replacement and testing in accordance with Procedure OPT-214-A, "Diesel Generator Operability Test," Revision 19, on October 13, 2008
- Unit 1 turbine driven auxiliary feedwater pump run following turbine teardown on October 15, 2008
- Feedwater control valve stroke time testing following packing adjustment on November 12, 2008
- Boric acid tank level indication check in accordance with Procedure INC-4623x, "Channel Calibration Boric Acid Tank 2 Level, Channel 0106," Revision 4, performed on November 24, 2008
- 125 VDC Train B Battery Charger BC1ED4-1 test in accordance with Procedure MSE-SO-5713, "Class 1E Battery Charger Load Test," Revision 5, observed on November 25, 2008
- Unit 2 Train A emergency diesel generator in accordance with Procedure OPT-214B, "Diesel Generator Operability Test," Revision 13, following the repair of #1 Left Header air start piping gasket leak and replacement of the packing for 4L injector pump, performed on December 10, 2008, per procedure MSM-CO-3831, Revision 3.
- Unit 1 positive displacement pump ultrasonic testing and pump run following drain and fill on December 19, 2008
- Unit 1, Train A bypass breaker replacement test in accordance with Procedure ETP-455A, "Unit 1 Train A Bypass Breaker Replacement Test," Revision 0, observed on November 5, 2008

The inspectors selected these activities based upon the structure, system, or component's ability to affect risk. The inspectors evaluated the activities to ensure the testing was adequate for the maintenance performed, the acceptance criteria were clear, and the test ensured equipment operational readiness.

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

These activities constituted completion of eight postmaintenance testing inspection samples.

b. Findings

Introduction. The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion V, for the failure to follow procedures to enter a malfunction of a reactor trip bypass breaker into the corrective action program. The breaker tripped slower than permitted during response time testing and was inoperable. Because the condition was not entered into the corrective action program, the licensee did not evaluate the condition or assess the extent of condition.

Description. At the start of refueling outage 1RF13, the Unit 1 Train A reactor trip bypass breaker was removed from service for maintenance. During the subsequent testing, the as-left undervoltage trip failed to open within the response time test acceptance criteria of 0.067 seconds. The breaker was removed from service and was replaced with a functioning breaker on November 5, 2008. Upon completion of the post-maintenance test of the replacement breaker, the inspectors questioned the licensee about the cause of the Unit 1 Train A reactor trip bypass breaker malfunction. Upon researching the inspectors' questions, the licensee determined that the cause equipment malfunction had not been entered into their corrective action program with the initiation of a Smart Form.

The licensee determined that, at the beginning of the outage, the as-found tests for the breaker were satisfactory. This as-found condition demonstrated that the breaker was operable during the previous operating cycle. However, none of the maintenance performed between the as-found and as-left tests should have affected breaker performance. Therefore, this was not an in-process finding.

The licensee's Procedure, ETP-455A, "Unit 1 Train A Bypass Breaker Replacement Test," Attachment 10.1.2, Section 1.4 required the licensee to compare the current reactor trip and bypass breaker response times to the previous times and evaluate for adverse trends, and initiate appropriate corrective actions if necessary. The licensee's corrective action program Procedure, STA-421, "Initiation of Smart Forms," also required licensee personnel to ensure that equipment malfunctions/deficiencies were documented with the initiation of a Smart Form. These actions were not taken.

Analysis. The licensee's failure to enter the breaker malfunction into the corrective action program was a performance deficiency. The finding was more than minor because, if left uncorrected, it would have led to a more safety significant concern. Specifically, because the cause of the failure would not have been fully evaluated and appropriate corrective actions may not be initiated. Once entered into the corrective action program, the licensee identified additional corrective measures. Using NRC Inspection Manual Chapter 0609, Attachment 4, "Phase 1 - Initial Characterization and Screening of Findings," the finding had a very low safety significance because the condition did not result the inoperability of the reactor trip breaker when it was required to be operable. The cause of this finding was related to the Problem Identification and Resolution crosscutting component of the corrective action program, in that, the licensee failed to enter the issue into their corrective action program [P1.a].

Enforcement. The inspectors determined that 10 CFR Part 50, Appendix B, Criterion V, requires that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Attachment 8.B of Procedure STA-421, required, in part, that equipment malfunctions, damage, or degradation, other than anticipated wear be documented. Step 6.1.2.1 of the procedure required, in part, that personnel shall ensure the issue is documented on a Smart Form. Contrary to the above, the licensee failed to initiate a Smart Form for the malfunction of the Unit 1, Train A reactor trip bypass breaker. Since the violation was of very low safety significance and was documented in the licensee's corrective action program as Smart Forms SMF-2008-003735 and SMF-2008-003767, it is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000445/2008005-05, "Failure to Initiate Corrective Actions for the Malfunction of a Reactor Trip Bypass Breaker."

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the outage safety plan and contingency plans for the Unit 1 refueling outage, conducted September 27 thru October 16, 2008, to confirm that licensee personnel had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the refueling outage, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below:

- Configuration management, including maintenance of defense-in-depth, is commensurate with the outage safety plan for key safety functions and compliance with the applicable technical specifications when taking equipment out of service

- Clearance activities, including confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing

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

- Status and configuration of electrical systems to ensure that technical specifications and outage safety-plan requirements were met, and controls over switchyard activities

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

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

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

- Controls over activities that could affect reactivity

- Maintenance of secondary containment as required by the technical specifications

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

- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the containment to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing

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

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

These activities constituted completion of one refueling outage sample.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the Final Safety Analysis Report, procedure requirements, and technical specifications to ensure that the surveillance activities listed below demonstrated that the systems, structures, and/or components tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed test data to verify that the significant surveillance test attributes were adequate to address the following:

- Preconditioning
- Evaluation of testing impact on the plant
- Acceptance criteria
- Test equipment
- Procedures
- Jumper/lifted lead controls
- Test data
- Testing frequency and method demonstrated technical specification operability
- Test equipment removal
- Restoration of plant systems
- Fulfillment of ASME Code requirements
- Updating of performance indicator data
- Reference setting data
- Annunciators and alarms setpoints

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

- Unit 1 main steam safety valve testing in accordance with Procedure MSM-S0-8702, “Main Steam Safety Valve Testing,” Revision 3, on September 17, 2008

- Unit 1 safety injection with a loss of offsite power integrated testing in accordance with Procedure OPT-430A, “Train A Integrated Test Sequence,” Revision 5 on October 1, 2008

- Unit 1 Diesel Generator 1-01 in accordance with Procedure OPT-214-A, “Diesel Generator Operability Test,” Revision 19, on October 2, 2008

- Secondary chemistry sample performed, in accordance with Procedure STA-610, “Secondary Water Chemistry Control Program,” Rev. 10, on October 15, 2008

- Unit 1 Train A containment spray eductor flow testing in accordance with Procedure OPT-226A, “Containment Spray Additive System Test,” Revision 3, on December 9, 2008

- Unit 1 auxiliary feedwater remote shutdown operability test performed in accordance with Procedure OPT-216A, “Remote Shutdown Operability Test,” Revision 3, on December 12, 2008

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

These activities constituted completion of six surveillance testing inspection samples (one in-service test sample and five routine surveillance testing samples).

b. Findings

Introduction. The inspectors documented a Green self-revealing noncited violation of Technical Specification 5.4.1a (Procedures) for an inadequate test procedure that resulted in inadvertently holding open a main steam safety valve at power. During testing, a test engineer separated a quick disconnect fitting in accordance with the procedural instructions. The action sealed in nitrogen pressure in the test rig and caused the valve to remain held open. In response to the event, operators reduced reactor power to compensate for the partially open safety valve until maintenance personnel closed the valve.

Description. On September 17, 2008, the licensee performed a test of the Unit 1 main steam safety valves using Procedure MSM-S0-8702, “Main Steam Safety Valve Testing.” During the testing of one of the valves, as allowed by procedure, a quick disconnect was used to relieve the nitrogen pressure of test rig. When the quick disconnect was opened it sealed in the nitrogen pressure and held the valve partially open (an unintended consequence). Control room operators reduced power approximately 5 megawatts electric to compensate for the excess steam demand from the open valve. After noticing the valve had not properly seated closed, maintenance personnel entered the room containing the valve and disconnected another connection from the test rig motor. This action resulted in the valve reseating after being partially open for approximately 45 seconds. The inspectors considered the surveillance procedure inadequate because it resulted in unexpectedly holding open a main steam safety valve while at power.

Analysis. The failure to have appropriate instructions for the operation of the main steam safety test rig was a performance deficiency. The finding was more than minor because it was associated with the procedure quality attribute of the initiating events cornerstone, and directly affected the cornerstone objective to limit the likelihood of those events that upset plant stability during power operations. Using Manual

Chapter 0609, Attachment 4, “Phase 1 – Initial Screening and Characterization of Findings,” the finding was determined to have very low safety significance because it did not contribute to the likelihood of mitigating equipment being unavailable. This finding did not have a crosscutting aspect because the procedure section was last revised several years earlier.

Enforcement. Technical Specification 5.4.1.a (Procedures) requires the licensee, in part, to implement the procedures recommended by Appendix A to Regulatory Guide 1.33, Revision 2. Appendix A to Regulatory Guide 1.33, Section 8.b recommends procedures for surveillance testing. Contrary to the above, on September 17, 2008, Procedure MSM-S0-8702, “Main Steam Safety Valve Testing,” Revision 3, was inadequate, in that it did not provide appropriate guidance for the operation of the test rig. Since the violation was of very low safety significance and was documented in the licensee’s corrective action program as Smart Form SMF-2008-002946, it is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000445/2008005-06, "Failure to Have an Adequate Procedure to Test Main Steam Safety Valves."

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

The inspectors assessed the licensee’s performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, high radiation areas, and worker adherence to these controls. The inspectors used the requirements in 10 CFR Part 20, the Technical Specifications, and the licensee’s procedures required by Technical Specifications as criteria for determining compliance. During the inspection, the inspectors interviewed the radiation protection manager, radiation protection supervisors, and radiation workers. The inspectors 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
- Adequacy of the licensee’s internal dose assessment for any actual internal exposure greater than 50 millirem committed effective dose equivalent
- Radiation work permit briefings and worker instructions

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

These activities constituted completion of 6 of the required 21 samples as defined in Inspection Procedure 71121.01-05.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspectors assessed licensee performance with respect to maintaining individual and collective radiation exposures as low as is reasonably achievable. The inspectors used the requirements in 10 CFR Part 20 and the licensee's procedures required by Technical Specifications as criteria for determining compliance. The inspectors interviewed licensee personnel and reviewed the following:

- Current 3-year rolling average collective exposure
- Six outage work activities from previous work history data which resulted in the highest personnel collective exposures
- Exposures of individuals from selected work groups
- Records detailing the historical trends and current status of tracked plant source terms and contingency plans for expected changes in the source term due to changes in plant fuel performance issues or changes in plant primary chemistry
- Source-term control strategy or justifications for not pursuing such exposure reduction initiatives
- Specific sources identified by the licensee for exposure reduction actions, priorities established for these actions, and results achieved since the last refueling cycle
- Declared pregnant workers during the current assessment period, monitoring controls, and the exposure results

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

These activities constituted completion of 4 of the required 15 samples and 3 of the optional samples as defined in Inspection Procedure 71121.02-05.

b. Findings

No findings of significance were identified.

4OA1 Performance Indicator Verification (71151)

.1 Data Submission Issue

a. Inspection Scope

The inspectors performed a review of the data submitted by the licensee for the third Quarter 2008 performance indicators for any obvious inconsistencies prior to its public release in accordance with NRC Inspection Manual Chapter 0608, "Performance Indicator Program."

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

b. Findings

No findings of significance were identified.

.2 Safety System Functional Failures

a. Inspection Scope

The inspectors sampled licensee submittals for the Safety System Functional Failures performance indicator for Units 1 and 2 for the period from July 2007 through September 2008. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, and NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73," definitions and guidance were used. The inspectors reviewed the licensee's operator narrative logs, operability assessments, maintenance rule records, maintenance work orders, issue reports, event reports and NRC Integrated Inspection reports to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator.

These activities constituted completion of two safety system functional failure samples.

b. Findings

No findings of significance were identified.

.3 Mitigating Systems Performance Index

a. Inspection Scope

The inspectors sampled licensee submittals for the mitigating systems performance index for Units 1 and 2 performance indicators for the period from July 2007 through September 2008. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, was used. The inspectors reviewed the licensee's operator narrative logs, mitigating systems performance index derivation reports, issue reports, event reports and NRC integrated inspection reports to validate the accuracy of the submittals. The inspectors reviewed the mitigating systems performance index component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator. The following performance indicators were reviewed:

- Emergency AC Power System
- Heat Removal System
- High Pressure Injection Systems
- Residual Heat Removal System
- Cooling Water Systems

These activities constituted completion of ten mitigating systems performance index samples.

b. Findings

No findings of significance were identified.

.4 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors sampled licensee submittals for the Occupational Radiological Occurrences performance indicator for the period from the first quarter 2008 through the third quarter 2008. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, was used. The inspectors reviewed the licensee's assessment of the performance indicator for occupational radiation safety to determine if indicator related data was adequately assessed and reported. To assess the adequacy of the licensee's performance indicator data collection and analyses, the inspectors discussed with radiation protection staff, the scope and breadth of its data review, and the results of those reviews. The inspectors independently reviewed electronic dosimetry dose rate and accumulated dose alarm and dose reports and the dose assignments for any intakes that occurred during the time period reviewed to determine if there were potentially unrecognized occurrences. The inspectors also conducted walkdowns of numerous locked high and very high radiation area entrances to determine the adequacy of the controls in place of these areas.

These activities constituted completion of the occupational radiological occurrences sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

.5 Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual Radiological Effluent Occurrences

a. Inspection Scope

The inspectors sampled licensee submittals for the Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual Radiological Effluent Occurrences performance indicator for the period from the first quarter 2008 through the third quarter 2008. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, was used. The inspectors reviewed the licensee's issue report database and selected individual reports since this indicator was last reviewed to identify any potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose. Additionally, the inspectors reviewed the licensee's historical 10 CFR 50.75(g) file and selectively reviewed the licensee's analysis for discharge pathways resulting from a spill, leak, or unexpected liquid discharge focusing on those incidents which occurred over the last few years.

These activities constituted completion of the radiological effluent technical specifications/offsite dose calculation manual radiological effluent occurrences sample.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

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

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

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

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

b. Findings

No findings of significance were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

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

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

b. Findings

No findings of significance were identified.

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's corrective action program and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors focused their review on repetitive equipment issues, but also considered the results of daily corrective action item screening discussed in Section 4OA2.2 licensee trending efforts, and licensee human performance results. The inspectors nominally considered the 6-month period of June 23, 2008 through December 31, 2008, although some examples expanded beyond those dates where the scope of the trend warranted.

The inspectors also included issues documented outside the normal corrective action program in major equipment problem lists, repetitive and/or rework maintenance lists, departmental problem/challenges lists, system health reports, quality assurance audit/surveillance reports, self-assessment reports, and Maintenance Rule assessments. The inspectors compared and contrasted their results with the results contained in the licensee's corrective action program trending reports. Corrective actions associated with a sample of the issues identified in the licensee's trending reports were reviewed for adequacy.

These activities constituted one semi-annual trend inspection sample.

b. Findings

No findings of significance were identified. The inspectors noted that they had recently identified five examples of damaged safety related tubing which spanned several different systems and areas. However, the licensee did not appear to be identifying similar issues on other safety related tubing.

.4 Selected Issue Follow-up Inspection

a. Inspection Scope

On December 10, 2008, the inspectors reviewed the cumulative effects of the operator workarounds and burdens to determine: the reliability, availability, and potential for misoperation of a system; if multiple mitigating systems could be affected, the ability of operators to respond in a correct and timely manner to plant transients and accidents; and if the licensee has identified and implemented appropriate corrective actions associated with operator workarounds.

These activities constituted completion of one in-depth problem identification and resolution operator workaround sample.

Findings

No findings of significance were identified.

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

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

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

b. Findings

No findings of significance were identified.

.2 Temporary Instruction 2515/172, "Reactor Coolant System Dissimilar Metal Butt Welds"

a. Inspection Scope

In October 2008, the inspectors performed portions of Temporary Instruction 2515/172, "Reactor Coolant System Dissimilar Metal Butt Welds," during Outage 1RF13. The reactor coolant system for this unit contains the following dissimilar welds:

- One 14-inch pressurizer surge line nozzle weld was mitigated during Outage 1RF12, spring 2007, using a Full Strength Weld Overlay. This weld is identified as Category F in accordance with MRP-139, "Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guidelines," Section 6, "Examination Schedules." The Visual Category is no longer applicable due to the weld mitigation.

- One 4-inch pressurizer spray nozzle weld was mitigated during Outage 1RF12, spring 2007, using a Full Strength Weld Overlay. This weld is identified as Category F in accordance with MRP-139, "Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guidelines," Section 6, "Examination Schedules." The Visual Category is no longer applicable due to the weld mitigation.

- Three 6-inch pressurizer safety nozzle welds were mitigated during Outage 1RF12, spring 2007, using a Full Strength Weld Overlay. These welds are identified as Category F in accordance with MRP-139, "Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guidelines," Section 6, "Examination Schedules." The Visual Category is no longer applicable due to the welds mitigation.

- Four 29-inch reactor coolant system hot leg nozzles were inspected using ultrasonic testing during Outage 1RF13, fall of 2008. These welds are identified as Volumetric Category D and Visual Category J in accordance with MRP-139, "Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guidelines," Section 6, "Examination Schedules."

* Four 27.5-inch reactor coolant cold leg nozzles are scheduled to be inspected during Outage 1RF14, spring of 2010. These welds are identified as Volumetric Category E and Visual Category K in accordance with MRP-139, "Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guidelines," Section 6, "Examination Schedules."

Licensee's Implementation of the MRP-139 Baseline Inspections (03.01)

The licensee did not perform baseline volumetric inspection activities on the following Dissimilar Metal Butt Welds (DMBWs):

Weld Identification	Description
TBX-1-4501-1/2	Pressurizer Safety A
TBX-1-4501-12/13	Pressurizer Safety B
TBX-1-4501-23/24	Pressurizer Safety C
TBX-1-4502-1/2	Pressurizer PORV
TBX-1-4503-31/30	Pressurizer Spray
TBX-1-4500-8/7	Pressurizer Surge Line

The licensee performed full structural weld overlay (FSWOL) for the mitigation and conducted post mitigation non-destructive examinations on the pressurizer DMBWs during Unit 1 refueling Outage 12 for the welds listed in the above table. The welds are currently classified as Category F in accordance with MRP-139, "Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guidelines," Section 6, "Examination Schedules." The weld identifications were updated to reflect the performance of the FSWOL changing the weld identifications respectively to TBX-1-4500-7/8 OL, TBX-4503-30/31 OL, TBX-1-4501-1/2 OL, TBX-1-4501-12/13 OL, TBX-1-4501-23/24 OL, and TBX-1-4502-1/2 OL. These welds are classified using the Visual Category due to the weld mitigation.

The licensee performed the following ultrasonic testing for the unmitigated nondestructive examinations of Unit 1 hot leg DMBWs during refueling Outage 1RF13:

Weld Identification	Description
TBX-1-4100-1/2	Loop 1 Reactor Coolant System Hot Leg
TBX-1-4200-1/2	Loop 2 Reactor Coolant System Hot Leg
TBX-1-4300-1/2	Loop 3 Reactor Coolant System Hot Leg
TBX-1-4400-1/2	Loop 4 Reactor Coolant System Hot Leg

These welds are currently classified as Volumetric Category D and Visual Category J in accordance with MRP-139, "Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guidelines," Section 6, "Examination Schedules." The licensee is currently evaluating the method of mitigation that will be performed for these welds.

The licensee is currently scheduled to perform the following unmitigated nondestructive examinations on the cold leg DMBWs during refueling Outage 1RF14 (Spring 2010):

Weld Identification	Description
TBX-1-4100-13/14	Loop 1 Reactor Coolant System Cold Leg
TBX-1-4200-13/14	Loop 2 Reactor Coolant System Cold Leg
TBX-1-4300-13/14	Loop 3 Reactor Coolant System Cold Leg
TBX-1-4400-13/14	Loop 4 Reactor Coolant System Cold Leg

These welds are currently classified as Volumetric Category E and Visual Category K in accordance with MRP-139, "Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guidelines," Section 6, "Examination Schedules." The licensee is currently evaluating the method of mitigation that will be performed for these welds.

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

Volumetric Examinations (03.02)

The inspectors reviewed the ultrasonic examination and eddy current examination records of the unmitigated hot leg DMBWs performed on October 4 and 5, 2008. These examinations were conducted in accordance with ASME Code, Section XI, Supplement VIII Performance Demonstrated Initiative requirements regarding personnel, procedures, and equipment qualifications. The licensee is planning on performing nondestructive testing on the unmitigated cold leg DMBWs during refueling Outage 1RF14. No relevant conditions were identified during the examinations of the hot leg DMBWs.

Inspectors reviewed records of NDE performed on pressurizer weld overlays. This effort is documented in Section 1R08 of this inspection report. Inspection coverage met requirements of MRP-139 and no indications requiring repair were identified.

The certification records of ultrasonic examination personnel used in the examination of the unmitigated hot leg DMBWs, and the mitigated pressurizer DMBWs were reviewed. All personnel records showed that they were qualified under the EPRI Performance Demonstration Initiative.

Three laminar indications were identified on welds TBX-1-4501-12/13 OL which were evaluated as within required acceptance conditions in accordance with ASME Code Section XI. No other deficiencies were identified during the NDE.

Weld Overlays (03.03)

The inspectors reviewed records pertaining to the pressurizer nozzles weld overlay and determined that welding was performed in accordance with ASME Code Section IX requirements.

The licensee submitted and received NRC verbal approval on February 20, 2007, to install weld overlays followed by letter dated September 12, 2007, "Comanche Peak Steam Electric Station Relief From ASME Code, Section XI for Implementation of the

EPRI-PDI Supplement 11 Program Requirements, and Weld Overlays, Relief Request B-6."

The qualification records of welders were reviewed and all qualifications were current. No relevant conditions were identified.

<u>Mechanical Stress Improvement (03.04)</u>

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

<u>Inservice Inspection Program (03.05)</u>

The licensee MRP-139 inservice inspection program has been controlled through the use of designated procedures and the corrective action program using Smart Forms to assure that requirements identified in the MRP-139 guidelines are not inadvertently missed. The MRP-139 inservice inspection program is in-process and will receive further inspection at a later date. The inspectors' review determined that the DMBWs nozzles are appropriately categorized in accordance with MRP-139 requirements.

b. <u>Findings</u>

No findings of significance were identified.

.3 <u>Temporary Instruction 2515/173, "Review of the Implementation of the Industry Ground Water Protection Voluntary Initiative"</u>

a. <u>Inspection Scope</u>

An NRC assessment was performed of the licensee's implementation of the Nuclear Energy Institute Ground Water Protection Initiative, dated August 2007 (ML072610036). Inspectors interviewed personnel, performed walk-downs of selected areas, and reviewed the following items:

- Site characterization of the geology and hydrology to verify that it provides an understanding of the predominant ground water gradients based upon current site conditions
- Evaluation of work practices that could lead to leaks and spills
- Evaluation of systems, structures, and components that contain licensed radioactive material to determine potential leak or spill mechanisms
- Implementation of an onsite ground water monitoring program to monitor for potential licensed radioactive leakage into groundwater
- Verify that ground water monitoring results are being reported in the annual effluent and/or environmental monitoring report (see http://www.nrc.gov/reactors/operating/ops-experience/tritium/plant-info.html)
- Procedures for the decision making process for potential remediation of leaks and spills, including consideration of the long term decommissioning impacts

- Verify that records of leaks and spills are being recorded in the licensee's decommissioning files in accordance with 10 CFR 50.75(g)

- Determine if the licensee has identified the appropriate local and state officials and has conducted briefings on the licensee's ground water protection initiative

- Protocols for notification to the local and state officials, and to the NRC regarding detection of leaks and spills

- Licensee and industry assessments of implementation of the ground water protection initiative

The licensee is scheduled to complete the Nuclear Energy Institute assessment on January 9, 2009.

b. Findings

No findings of significance were identified.

.4 Temporary Instruction 2515/176, "Emergency Diesel Generator Technical Specification Surveillance Requirements Regarding Endurance and Margin Testing"

a. Inspection Scope

The objective of Temporary Instruction 2515/176 was to gather information to assess the adequacy of nuclear power plant emergency diesel generator endurance and margin testing as prescribed in plant-specific Technical Specifications. The inspectors reviewed the licensee's Technical Specifications, procedures, and calculations and interviewed licensee personnel to complete the Temporary Instruction. The information gathered while completing this Temporary Instruction was forwarded to the Office of Nuclear Reactor Regulation for further review and evaluation on December, 11, 2008.

b. Findings

No findings of significance were identified.

.5 (Closed) Unresolved Item 05000445; 446/2008007-01: Failure to Correctly Test the Primary Plant Ventilation System

a. Inspection Scope

The inspectors reviewed an in-place testing procedure for the comparison of two sampling devices and the test results. Additionally, the inspector interviewed the system engineer who conducted the comparison test.

b. Description

During Inspection 05000445/2008007; 05000446/2008007, the inspectors identified a identified a potential concern with the implementation of the licensee's testing program of high efficiency particulate air filters. The licensee did not use rigid sampling nozzles, pitot tubes, or similar devices to ensure particulates were collected from the air stream. Instead, the licensee used flexible plastic, small-diameter tubing (0.19-inch inside diameter) inserted into larger hoses or ducts. Engineering representatives stated this method had been established by vendor personnel before commercial operation of the plant and had been used by licensee personnel since that time, but they produced no documentation validating the effectiveness of the testing procedure.

In response to inspector questions, the licensee performed a comparison test using Primary Plant Filter X-04 to validate their method. This was a non-engineered safety feature air cleaning system, but it was identical to Primary Plant Filter X-16, the system which was subject of the concern. The licensee fabricated a rigid sampling nozzle to use in a comparison with its original flexible plastic tube. The licensee simulated a leak in the air cleaning system by allowing a small amount of the test aerosol to bypass the high efficiency particulate air filter. Then, the licensee tested both sampling devices and recorded the results. The inspector reviewed the licensee's testing procedure and the tests results and concluded both sampling devices worked equally well to identify the simulated leak. Therefore, this unresolved item is closed.

c. Findings

No findings of significance were identified.

4OA6 Meetings

Exit Meeting Summary

On October 9, 2008, the inspectors debriefed the inservice inspection activities inspection results to Mr. R. Flores, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. On November 10, 2008, the inspectors telephonically exited with Mr. T. Hope, Manager, Nuclear Licensing. The inspectors acknowledged review of proprietary material during the inspection which had been or will be returned to the licensee.

On November 20, 2008, the inspectors presented the occupational radiation safety inspection results to Mr. F. Madden, Director, Nuclear Oversight and Regulatory Affairs and other members of the licensee staff. The licensee acknowledged the issues presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On January 8, 2008, the inspectors presented the resident inspection results to Mr. R. Flores, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors acknowledged review of proprietary material during the inspection. No proprietary information has been included in the report.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

M. Blevins, Executive Vice President and Chief Nuclear Officer
M. Bozeman, Manager, Nuclear Emergency Planning
R. Flores, Site Vice President
D. Goodwin, Director, Operations
B. Hamilton, Director, Engineering Support
T. Hope, Manager, Nuclear Licensing
D. Kross, Plant Manager
M. Lucas, Vice President, Nuclear Engineering and Support
F. Madden, Director, Nuclear Oversight and Regulatory Affairs
B. Mays, Director, Site Engineering
E. Meaders, Manager, Work Control/Outage
B. Patrick, Manager, Radiation and Industrial Safety
M. Pearson, Director, Performance Improvement
S. Smith, Director, Maintenance
K. Tate, Manager, Security
D. Walling, Manager, Training
D. Wilder, Manager, Plant Support

NRC Personnel

J. Kramer, Senior Resident Inspector
B. Tindell, Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None.

Opened and Closed

05000445/2008005-01 05000446/2008005-01	NCV	Non-Seismic Scaffolding Installed Over Service Water Equipment (Section 1R04)
05000445/2008005-02	NCV	Failure to Adequately Evaluate Material Condition Following a Boric Acid Leak (Section 1R08)
05000445/2008005-03	NCV	Failure to Assess and Manage Risk Associated with Maintenance Activities (Section 1R13)
05000445/2008005-04	NCV	Inadequate Instructions Causes to Failure to Identify Fibrous Material in Containment (Section 1R15)

05000445/2008005-05	NCV	Failure to Initiate Corrective Actions for the Malfunction of a Reactor Trip Bypass Breaker (Section 1R19)
05000446/2008005-06	NCV	Failure to Have an Adequate Procedure to Test Main Steam Safety Valves (Section 1R22)

Closed

05000445/2008007-01 05000446/2008007-01	URI	Failure to Correctly Test the Primary Plant Ventilation System (Section 4OA5.5)

Discussed

None.

LIST OF DOCUMENTS REVIEWED

Section 1RO4: Equipment Alignment

Drawings

NUMBER	TITLE	REVISION
M-206	Flow Diagram, Auxiliary Feedwater System	CP-14
M-206, Sh-01	Flow Diagram, Auxiliary Feedwater System, Pump Trains	CP-10
M-206, Sh 02	Flow Diagram, Auxiliary Feedwater System, yard layout	CP-9
DBD-ME-206	Auxiliary Feedwater System Design Basis Document	24
E1-0018, Sh 02A	125VDC Switchboard One Line Diagram	CP-15
E1-0001, Sh. A	One line Diagram Unit 1 and Common Distribution panels	CP-18
E1-0018, Sh 02	125VDC Switchboard One Line Diagram	CP-18
E1-0018, Sh 02B	125VDC Switchboard One Line Diagram	CP-4
E1-0020	125VDC One Line Diagram	CP-20
E1-0020, Sh A	125VDC One Line Diagram	CP-14

Smart Forms

SMF-2007-000955-00	SMF-2007-001684-00	SMF-2007-001899-00
SMF-2007-002010-00	SMF-2007-002066-00	SMF-2007-002254-00
SMF-2007-002909-00	SMF-2008-002067-00	SMF-2008-003321-00
SMF-2008-003408-00	SMF-2008-003683-00	

Section 1RO5: Fire Protection

Documents

NUMBER	TITLE	REVISION
SOP-614A	Alternate Power Generator Operation	8
	Fire Protection Report	25

Smart Forms

SMF-2008-003315-00 SMF-2008-003484-00

Section 1RO6: Flood Protection Measures

Documents

NUMBER	TITLE	REVISION
DM 99-023	SWIS Insect Abatement	1
SI-CA-0000-693	Flooding Analysis	

Section 1RO8: In-service Inspection Activities

Procedures

NUMBER	TITLE	REVISION
TX-ISI-210	Ultrasonic Testing Exam Procedure for Welds in Ferritic Steels	6
WDP-9.2		0
	Qualification and Certification of Personnel in Nondestructive Examination	
TX-OPS-101		9
	Preservice and Inservice Examination Documentation for Comanche Peak Steam Electric Station	
TX-ISI-11		11
	Liquid Penetrant Examination for Comanche Peak Steam	
TX-ISI-302	Electric Station	3
TX-ISI-8	Ultrasonic Testing Examination of Austenetic Piping Welds	6
MSM-C0-8807	VT-1 and VT-3 Examination Procedure for Comanche Peak Steam Electric Station	2
TX-ISI-214		4
	Comanche Peak Steam Electric Station Maintenance Section-Mechanical Manual	
STA-703		13
	Ultrasonic Examination Procedure for Welds in Piping	
EPG-703	Systems and Vessels	1

Procedures

NUMBER	TITLE	REVISION
EPG-731	Inservice Inspection Program	1
STA-731	Inservice Inspection Program	6
STA-737	ASME Section XI Repair/Replacement Activities	4
STA-760	ASME Section XI Repair & Replacement Activities	1
EPG-9.02	Boric Acid Corrosion Detection and Evaluation	0
WCI-607	RCS Material Management Program	11
	Comanche Peak Steam Electric Station Alloy 600 management Program	
	Fluid Leak Management Process	

Weld Documents

NUMBER	TITLE	REVISION
TBX-2100-1/6	Pressurizer Weld 1 and 6 Ultrasonic Testing Results	10/01/08
TBX-1-4201-H4	package	10/07/08
TBX-1-4201-3	Accumulator B Discharge H4 PT/VT-3 results package	10/07/08
WO-3529285	Accumulator B Discharge weld Ultrasonic Testing results	10/04/08
WR-080356	package	10/04/08
WPS-CP-301-1	U1 Emergency Borate Line Check Valve	11
MIR-080356-01	Weld Record for 1CS-8842	10/04/08
PQR-01,-02,-60	Welding Procedure Specification Technical Sheet	10/04/08
	Material Issue Record (Return)	
	Various Personnel Qualification Records	

Drawings

NUMBER	TITLE	REVISION
TBX-1-4201	Accumulator B Discharge Pipe Profile	10/07/08
TBX-1-2100	Pressurizer Shell Weld Profile for Welds 1 and 6	10/06/08
DR-4278C-3	Outlet and Inlet Nozzle Sections	5/11/79
DR-4278C-2	Outlet and inlet nozzles horizontal Section	12/15/78
DR-4278C-3A	Inlet and Outlet Nozzle Non-Crush Insulation	3/25/80

Section 1R12: Maintenance Effectiveness

Procedures

NUMBER	TITLE	REVISION
STA-744	Maintenance Effectiveness Monitoring Program	3
CPES-I-1018	Installation of Piping/Tubing and Instrumentation	20
STA-109	Conduct of Maintenance	4

Work Orders

WO 403448 WO 398002 WO 3507894

Smart Forms

SMF-2007-002232-00 SMF-2008-002034-00 SMF-2008-002906-00

Section 1R15: Operability Evaluations

Documents

NUMBER	TITLE	REVISION
MDA-304	Control of Heavy Loads and Critical Lifts	6
2323-SI-0617	Safeguards Building 896'-4" Outline	3
STA-661	Non-Plant Equipment Storage and Use Inside Seismic Category I Structures	4
JWI-1629	Inspector for Kaowool Inside Containment	
NEI 02-01	Condition Assessment Guidelines: Debris Sources Inside Pressurized Water Reactor Containments	1
2323-MS-38F	Specification for Fire Rated, Radiation Shielding, and Pressure Penetration Seals	4

Smart Forms

SMF 2001-002201-00 SMF 2008-003013-00 SMF 2008-003229-00
SMF 2008-003245-00 SMF 2008-003480-00 SMF 2008-003587-00

Section 1R19: Postmaintenance Testing

Procedures

NUMBER	TITLE	REVISION
OPT-206B	Auxiliary Feedwater System	20
MSM-C0-8722	Auxiliary Feedwater Turbine Trip Throttle Valve	1
OPT-214B	Maintenance	13
MSM-CO-3831	Diesel Generator Operability Test	3
INC-4623X	Emergency Diesel Engine Cylinder Head Maintenance	4
MSE-SO-5713	Channel Calibration Boric Acid Tank 2 Level, Channel 0106	5
TSP-509	Class 1E Battery Charger Load Test	5
STA-124	Predictive Maintenance Thermographic Analysis Program	1
SOP-609A	Electrical Safe Work Practices	17
MSM-C0-3831	Diesel Generator System	3
	Emergency Diesel Engine Cylinder Head Maintenance	

Work Orders

WO 3621177 WO 3621206 WO 3637723 WO 408923 WO 408198

Smart Forms

SMF-2008-003784-01 SMF-2008-002425-00 SMF-2008-003697-00 SMF-2008-003735-00
SMF-2008-003767-00

Section 2OS1: Access Controls to Radiologically Significant Areas

Documents

NUMBER	TITLE	REVISION
RPI-602	Radiological Surveillance and Posting	35
RPI-606	Radiation Work and General Access Permit	17
STA-653	Contamination Control Program	11

Section 2OS2: ALARA Planning and Controls

Procedures

NUMBER	TITLE	REVISION
STA-651	ALARA Program	0
STA-656	Radiation Work Control	0
STA-657	ALARA Job Planning/Debriefing	4

Smart Forms

SMF-2006-001571-00	SMF-2008-002971-00	SMF-2008-002983-00
SMF-2008-003107-00	SMF-2008-003110-00	SMF-2008-003145-00
SMF-2008-003151-00	SMF-2008-003182-00	SMF-2008-003184-00
SMF-2008-003277-00	SMF-2008-003283-00	SMF-2008-003277-00
SMF-2008-003285-00	SMF-2008-003305-00	SMF-2008-003335-00
SMF-2008-003714-00	SMF-2008-003781-00	SMF-2008-003853-00

Radiation Work Permits

2008-0102 Unit-2 Leak Investigation
2008-0107 Unit -2 Containment Entry to Work 2MS-0088
2008-1208 1RF13 Room 1-155ª Letdown Orifice Valve Room
2008-1215 Scaffolding Activities
2008-1217 Unit-1 905" Pressurizer Spray Room Insulation Work
2008-1600 Refueling Activities

Miscellaneous
ALARA Committee Meeting 2008-009, August 21, 2008
ALARA Committee Meeting 2008-012, November 13, 2008
ALARA Plan U3C15
One Declared Pregnant Worker records and dose evaluations

Section 4OA1: Performance Indicator Verification

Documents

NUMBER	TITLE	REVISION
ENV-323	Groundwater Sampling Program	2
STA-654	Groundwater Protection Program	5
SA-2007-44	Groundwater Monitoring	11/15/07
EVAL-2008-011	Radiation Protection	10/09/08

Section 4OA2: Identification and Resolution of Problems

Miscellaneous
Operations Guideline 36, "Operator Burdens and Work – Arounds," dated June 5, 2008
Operations Work Around List as of December 10, 2008
Operations Burden List as of December 10, 2008

Section 4OA5: Other Activities (Section .2)

Miscellaneous Documents

NUMBER	TITLE	REVISION
TX-ISI-210	Ultrasonic Testing Exam Procedure for Welds in Ferritic Steels	6
TX-ISI-8		6
TX-ISI-302	VT-1 and VT-3 Examination Procedure for Comanche Peak Steam Electric Station	3
RVCHVEP	Ultrasonic Testing Examination of Austenetic Piping Welds	3
RPBDMWVEP	Reactor Vessel Closure Head Visual Examination Plan	2
WO-3611130	Reactor Coolant System Pressure Boundary Dissimilar Metal Weld Visual Examination Plan	10/11/08
WO-3611884	Determine Origin of Borated Leakage, Decontaminate, Remove Boric Deposits	10/10/08
WO-406166193	Determine Origin of Borated Leakage, Decontaminate, Remove Boric Deposits	2/25/07
	Perform Alloy 600 Inspection for Reactor BMI in 1RF12”	3/17/2007
	Comanche Peak Unit 1 Safety Nozzle 'A' SWOL Examination Coverage Summary	3/23/2007
	Comanche Peak Unit 1 Safety Nozzle 'B' SWOL Examination Coverage Summary	3/19/2007
	Comanche Peak Unit 1 Safety Nozzle 'C' SWOL Examination Coverage Summary	3/19/2007
	Comanche Peak Unit 1 PORV Nozzle SWOL Examination Coverage Summary	3/23/2007
	Comanche Peak Unit 1 Spray Nozzle SWOL Examination Coverage Summary	3/19/2007
SR(A)-003	Comanche Peak Unit 1 Surge Nozzle SWOL Examination Coverage Summary	3/15/2007
SR(B)-003	Calibration Sheet, Pressurizer Safety 'A' SWOL	3/23/2007
SR(C)-003	Calibration Sheet, Pressurizer Safety 'B' SWOL	3/15/2007

Miscellaneous Documents

NUMBER	TITLE	REVISION
SR(P)-003	Calibration Sheet, Pressurizer Safety 'C' SWOL	3/15/2007
SP 03	Calibration Sheet, Pressurizer PORV SWOL	3/23/2007
SU 003	Calibration Sheet, Pressurizer Spray SWOL	3/23/2007
	Calibration Sheet, Pressurizer Surge SWOL	

Smart Forms

SMF-2007-000220-00
SMF-2007-000515-00
SMF-2007-000670-00
SMF-2007-000707-00
SMF-2007-000799-00
SMF-2007-001059-00
SMF-2007-001617-00
SMF-2007-001680-00
SMF-2007-001739-00
SMF-2007-002163-00
SMF-2007-003073-00
SMF-2007-003123-00
SMF-2008-000044-00
SMF-2008-000344-00
SMF-2008-000991-00
SMF-2008-002229-00
SMF-2008-002829-00
SMF-2008-002831-00
SMF-2008-002948-00
SMF-2008-002963-00
SMF-2008-003131-00
SMF-2008-003194-00
SMF-2008-003356-00

Section 4OA5: Other Activities (Section .5)

EVAL-2008-000638-01-00, High Efficiency Particulate Air Filter Testing, dated March 3, 2008