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

July 6, 2007

Mike Blevins, Senior Vice President
and Chief Nuclear Officer
TXU Power
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 05000445/2007006

Dear Mr. Blevins:

On May 21, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Comanche Peak Steam Electric Station, Unit 1, facility. No inspection of Unit 2 was performed under this report number. This inspection was conducted due to the Unit 1 steam generator and reactor vessel head replacement activities. The enclosed inspection report documents the inspection findings, which were discussed on May 21, 2007, with Mr. R. Flores and other members of your staff.

This inspection examined activities conducted under your licenses as they related 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 inspection covers steam generator and reactor vessel head replacement activities.

Based on the results of this inspection, no findings of significance were identified.

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

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

Sincerely,

/RA/
Claude Johnson, Chief
Project Branch A
Division of Reactor Projects

Docket Nos.: 50-445
License Nos.: NPF-87

Enclosure: NRC Inspection Report 05000445/2007006
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Dockets:	50-445
Licenses:	NPF-87
Report:	05000445/2007006
Licensee:	TXU Generation Company LP
Facility:	Comanche Peak Steam Electric Station, Unit 1
Location:	FM-56, Glen Rose, Texas
Dates:	January 1, 2007 through May 21, 2007
Inspectors:	D. Allen, Senior Resident Inspector A. Sanchez, Resident Inspector E. Owen, Reactor Inspector, Engineering Branch 1 R. Kopriva, Senior Reactor Inspector, Engineering Branch 1 W. Sifre, Senior Reactor Inspector, Engineering Branch 1 R. Azua, Reactor Inspector, Engineering Branch 1 S. Rutenkroger, Reactor Inspector, Engineering Branch 1 Gilbert L. Guerra, CHP, Health Physicist Donald L. Stearns, Health Physicist
Approved by:	Claude Johnson, Chief, Project Branch A Division of Reactor Projects
Attachment:	Supplemental Information

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SUMMARY OF FINDINGS

IR 05000445/2007006; 01/01/2007-05/21/2007; Comanche Peak Steam Electric Station, Unit 1. Integrated Resident and Regional Report of Steam Generator and Reactor Vessel Closure Head Replacement Activities.

This report covered a 5-month period of inspection by two resident inspectors, five regional reactor inspectors and two health physicists. No findings were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using the Inspection Manual Chapter 0609, “Significance Determination Process.” Findings for which the Significance Determination Process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process,” Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

No findings of significance were identified.

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

Comanche Peak Steam Electric Station (CPSES) Unit 1 began the period operating at essentially 100 percent power. On February 16, 2007 Unit 1 began a reactor power coastdown. On February 24, at 12:00 noon, Unit 1 entered Mode 3 to begin the steam generator and reactor vessel head replacement outage, 1RF12. On April 20, the Unit 1 replacement outage ended when the main generator beakers were closed. Unit 1 achieved 100 percent power on April 24. On April 27, reactor power was reduced to approximate 80 percent power for final testing. Unit 1 returned to 100 percent power on April 28 and remained at essentially 100 percent power for the rest of the reporting period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R02 Evaluations of Changes, Tests, or Experiments (71111.02)

a. Inspection Scope

The inspectors reviewed the effectiveness of the licensee's implementation of changes to the facility structures, systems, and components; risk-significant normal and emergency operating procedures; test programs; and the updated final safety analysis report in accordance with 10 CFR 50.59, "Changes, Tests, and Experiments." The inspectors utilized Inspection Procedure 71111.02, "Evaluation of Changes, Tests, or Experiments," for this inspection.

The inspectors reviewed one safety evaluation performed by the licensee since the last NRC inspection of this area at CPSES, Unit 1. The evaluation was reviewed to verify that licensee personnel had appropriately considered the conditions under which the licensee may make changes to the facility or procedures or conduct tests or experiments without prior NRC approval. The inspectors reviewed two licensee-performed applicability determinations in which licensee personnel determined that evaluations were not required, to ensure that the exclusion of a full evaluation was consistent with the requirements of 10 CFR 50.59. Evaluations and applicability determinations reviewed are listed in the attachment to this report.

The inspectors reviewed and evaluated a sample of recent licensee condition reports to determine whether the licensee had identified problems related to 50.59 evaluations, entered them into the corrective action program, and resolved technical concerns and regulatory requirements. The reviewed condition reports are identified in the Attachment.

The inspection procedure specifies a required minimum sample of six licensee safety evaluations and 12 applicability determinations and screenings (combined). The inspectors completed review of one licensee safety evaluation and 2 applicability determinations for this effort. The remaining required samples are documented in NRC Inspection Report 05000445;446/2007002, Section 1R02.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

.1 Performance of Nondestructive Examination Activities Other Than Steam Generator Tube Inspections, Pressurized Water Reactor Vessel Upper Head Penetration Inspections, Boric Acid Corrosion Control

a. Inspection Scope

The inspectors used Inspection Procedure 71111.08, “Inservice Inspection Activities,” for this inspection. The inspection procedure requires the review of Nondestructive Examination (NDE) activities consisting of two or three different types (i.e., volumetric, surface, or visual). The inspectors observed and reviewed the performance of radiographic and ultrasonic examinations (volumetric) of welds on the Unit 1 new steam generator (3 and 4) to the reactor coolant loops (hot and cold legs), and auxiliary feedwater piping (FW-TUX-42, 36, 38 and 7). Additionally, the inspectors observed dye penetrant and magnetic particle examinations of welds (surface) on new steam generator No. 4 to reactor coolant loop welds (hot and cold legs) and auxiliary feedwater piping (FW-TUX-7) respectively. In addition, the inspectors observed four visual (VT-1 and VT-3) examinations performed on component supports. The table below identifies the above examinations which were conducted using five methods and three examination types.

System/ Component	**Identity**	**Examination Type**	**Examination Method**
Reactor Coolant System	New Steam Generator (#3) to Hot Leg Weld	Volumetric	Ultrasonic
Reactor Coolant System	New Steam Generator (#3) to Cold Leg Weld	Volumetric	Ultrasonic
Auxiliary Feedwater System	Cap Weld FW-TUX-42	Volumetric	Ultrasonic Radiography
Auxiliary Feedwater System	Cap Weld FW-TUX-36	Volumetric	Radiography
Auxiliary Feedwater System	Cap Weld FW-TUX-38	Volumetric	Radiography
Auxiliary Feedwater System	Cap Weld FW-TUX-7	Volumetric	Ultrasonic

System/ Component	**Identity**	**Examination Type**	**Examination Method**
Reactor Coolant System	New Steam Generator (#4) to Hot Leg Weld	Volumetric	Radiography
Reactor Coolant System	New Steam Generator (#4) to Cold Leg Weld	Volumetric	Radiography
Reactor Coolant System	New Steam Generator (#4) to Hot Leg Weld	Surface	Penetrant
Reactor Coolant System	New Steam Generator (#4) to Cold Leg Weld	Surface	Penetrant
Auxiliary Feedwater System	Cap Weld FW-TUX-7	Surface	Magnetic Particle
Component Cooling Water System	Vertical Spring Can H1: CC-1-RB-049	Visual	Visual (VT-3)
Component Cooling Water System	Welded Attachment H1WA: CC-1-RB-049	Visual	Visual (VT-1)
Component Cooling Water System	Vertical Spring Can H1: CC-1-249-701-C53A	Visual	Visual (VT-3)

For each of the observed nondestructive examination activities, the inspectors verified that the examinations were performed in accordance with the specific site procedures and the applicable American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements.

During review of each examination, the inspectors verified that appropriate nondestructive examination procedures were used, examinations and conditions were as specified in the procedure, and test instrumentation or equipment was properly calibrated and within the allowable calibration period. The inspectors also verified the nondestructive examination certifications of the personnel who performed the above volumetric, surface, and visual examinations. Finally, the inspectors observed that indications identified during the ultrasonic, radiographic, and visual examinations were dispositioned in accordance with the ASME-qualified nondestructive examination procedures used to perform the examinations.

The inspection procedure requires review of one or two examinations with recordable indications that were accepted for continued service to ensure that the disposition was made in accordance with the ASME Code. The inspectors verified that two laminar flaws discovered on the original dissimilar metal welds of the Pressurizer Safety Valve B line (TBX-1-4501-12OL and TBX-1-4501-13OL) were acceptable in accordance with the standards of the ASME Code.

The inspection procedure further requires verification of one to three welds on Class 1 or 2 pressure boundary piping to ensure that the welding process and welding examinations were performed in accordance with the ASME Code. The inspectors verified through observation and record review that the auxiliary feedwater pipe cap welds (FW-TUX-42, 38 and 7) and the welding that was performed on the Unit 1 nuclear steam supply system to join the new steam generators (3 and 4) to their associated reactor coolant loops, in the field, were performed in accordance with Sections IX and XI of the, 1998 Edition of the ASME Code. This included review of welding material issue slips to establish that the appropriate welding materials had been used and verification that the welding procedure specification had been properly qualified.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.2 Reactor Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

The inspection requirements for this section parallel the inspection requirement steps in Section 02.01. The inspectors reviewed records of completed nondestructive examinations, including the eddy current and ultrasonic examination data analyses process used on the reactor vessel upper head penetrations during their preservice inspections.

Additionally, the nondestructive examination procedures used to perform the above examinations were reviewed to assure that they were consistent with ASME Code requirements, and the equipment and calibration requirements were appropriately identified and demonstrated.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control Inspection Activities (Pressurized Water Reactors)

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 deleteriously affected by boric acid corrosion.

The inspection procedure requires review of a sample of boric acid corrosion control walkdown visual examination activities through either direct observation or record review. The inspectors reviewed the documentation associated with the licensee’s boric acid corrosion control walkdown, as specified in Station Administrative Manual (STA) Procedure STA-737, "Boric Acid Corrosion Detection and Evaluation," Revision 4. Samples of documented visual inspection records of inspection walkdowns performed on components and equipment during the previous Refueling Outage 1RF11, and this refueling outage, were reviewed by the inspectors.

Additionally, the inspectors performed independent observations of piping containing boric acid during walkdowns of the containment building and the auxiliary building.

The inspection procedure requires verification that visual inspections emphasize locations where boric acid leaks can cause degradation of safety significant components. The inspectors verified through direct observation and program/record review that the licensee's boric acid corrosion control inspection efforts are directed towards locations where boric acid leaks can cause degradation of safety-related components.

The inspection procedure requires both a review of one to three engineering evaluations performed for boric acid leaks found on reactor coolant system piping and components, and one to three corrective actions performed for identified boric acid leaks. There were no applicable corrective action documents generated since the last inspection period that required formal engineering evaluation (e.g., that resulted in a separate design or structural engineering analysis to determine continued operability). The inspectors reviewed Smart Forms (SMF) documenting minor valve packing leaks on valves in the safety injection system. The planned corrective actions were adequate in each case.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

The inspectors verified through records review that licensee personnel and contractors used properly qualified eddy current probes and equipment for the expected types of tube degradation to assure proper identification and evaluation of indications for the new baseline data. The inspectors verified that the licensee analysts reviewed the areas of potential degradation, based on site-specific and industry experience, to assure proper use of this information. The inspectors reviewed the repair criteria used to assure compliance with technical requirements. The inspectors also verified the licensee's eddy current examination scope and expansion criteria met the Technical Specifications, industry guidelines, and commitments to the NRC.

Regarding plugging and in-situ pressure testing, because the steam generators were new replacement components, the licensee had no need for plugging and in-situ pressure testing onsite. The vendor had plugged one tube in Steam Generator No. 3 prior to its delivery onsite due to a tube bulge in the tubesheet region during fabrication.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed selected activities regarding risk evaluations and overall plant configuration control. The inspectors discussed emergent work issues with work control personnel and reviewed the potential risk impact of these activities to verify that the work was adequately planned, controlled, and executed. The activities reviewed were associated with:

- Probability Risk Analysis Report related to the multiple crane operations inside the Unit 1 containment building during 1RF12, on February 23, 2007

- Defense in depth contingency plan, 1RF-22, for maintaining Unit 1 containment pressure while the containment liner is removed and fuel is being unloaded with 24 or less fuel assemblies remaining in the core, on February 26, 2007

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17B)

a. Inspection Scope

The inspectors reviewed eleven permanent plant modification packages and associated documentation, such as implementation reviews, safety evaluation applicability determinations, and screenings, to verify that they were performed in accordance with regulatory requirements and plant procedures. The inspectors also reviewed the procedures governing plant modifications to evaluate the effectiveness of the program for implementing modifications to risk-significant systems, structures, and components, such that these changes did not adversely affect the design and licensing basis of the facility. Procedures and permanent plant modifications reviewed are listed in the Attachment to this report. Further, the inspectors interviewed the cognizant design and system engineers for the identified modifications as to their understanding of the modification packages and process.

The inspectors evaluated the effectiveness of the licensee's corrective action process to identify and correct problems concerning the performance of permanent plant modifications by reviewing a sample of related condition reports. The reviewed condition reports are identified in the Attachment.

The inspection procedure specifies inspector-review of a required minimum sample of six permanent plant modifications. The inspectors completed review of eleven permanent plant modifications.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

.1 Steam Generator and Reactor Vessel Head Replacement

a. Inspection Scope

The inspectors witnessed or reviewed the results of the postmaintenance tests for the following replacement outage activities:

- Control rod drive mechanism (CRDM) ventilation testing following the complete redesign and replacement of the old Unit 1 CRDM ventilation fan system, in accordance with Integrated Plant Operating Procedures Manual (IPO) IPO-011A, "Plant Restart and Testing Following Steam Generator Replacement," Revision 0, reviewed on April 19, 2007

- CRDM testing following reactor vessel head replacement, in accordance with procedure IPO-011A, "Plant Restart and Testing Following Steam Generator Replacement," Revision 0, reviewed on April 19, 2007

- Steam generator blowdown system flow and vibration testing following the replacement of steam generators, in accordance with procedure IPO-011A, "Plant Restart and Testing Following Steam Generator Replacement," Revision 0, observed and reviewed on April 20, 2007

- Transfer of feedwater bypass control to main feedwater control testing following maintenance and tuning activities, in accordance with procedure IPO-011A, "Plant Restart and Testing Following Steam Generator Replacement," Revision 0, observed and reviewed on April 20, 2007

- Electrical load swing testing to ensure reactor control system interaction and tuning following the replacement of the Unit 1 steam generators, in accordance with procedure IPO-011A, "Plant Restart and Testing Following Steam Generator Replacement," Revision 0, observed and reviewed on April 20, 2007

- Steam generator steam flow calibration following replacement of steam generators, in accordance with procedure IPO-011A, “Plant Restart and Testing Following Steam Generator Replacement,” Revision 0, observed and reviewed on April 24, 2007

- Steam Generator Water Level Control System response testing following adjustments and tuning activities, in accordance with procedure IPO-011A, “Plant Restart and Testing Following Steam Generator Replacement,” Revision 0, observed and reviewed on April 30, 2007

- Large load (275 MWe) reduction test following replacement outage activities, in accordance with procedure IPO-011A, “Plant Restart and Testing Following Steam Generator Replacement,” Revision 0, observed and reviewed on April 30, 2007

- Reactor coolant system flow measurement test following the replacement of the Unit 1 steam generators, in accordance with procedure number INC-7018A, “Reactor Coolant System Flow Measurement,” Revision 3, reviewed on May 2, 2007

In each case, the associated work orders and test procedures were reviewed in accordance with the inspection procedure to determine the scope of the maintenance activity and to determine if the testing was adequate to verify equipment operability. The inspectors also reviewed Chapter 14, “Initial Test Program” of Updated Final Safety Analysis Report to help determine the adequacy of the testing.

The inspectors completed nine samples.

b. Findings

No findings of significance were identified.

.2 Containment Alternate Access

Containment Integrated Leak Rate Test Procedure Review (70307)

a. Inspection Scope

The inspectors reviewed the licensee’s containment integrated leak rate test procedure to verify that the test complies with regulatory requirements, guidance, and licensee commitments to evaluate the technical adequacy to determine containment leak tight integrity. The inspectors ensured that the procedure contained sufficiently detailed guidance for: (1) the alignment and operation of all systems and equipment inside and penetrating containment, (2) inspections of the accessible portions of containment, (3) verification of equipment calibration, and (4) appropriate success criteria.

b. Findings

No findings of significance were identified.

Containment Integrated Leak Rate Surveillance (70313)

a. Inspection Scope

The inspectors verified through observation, records review, and independent calculations whether the containment integrated leak rate test was being properly conducted. In addition, the inspectors independently verified the acceptability of the test results through real time observations and analysis and further in-depth independent analysis. The inspectors: (1) ensured that the alignment and operation of all systems and equipment inside and penetrating containment was appropriate, (2) conducted inspections of the accessible portions of containment, (3) verified equipment calibration, and (4) ensured appropriate success criteria were being followed per the approved procedure.

b. Findings

No findings of significance were identified.

Containment Leak Rate Test Results Evaluation (70323)

a. Inspection Scope

The inspectors verified through direct observation and records review that the licensee had adequately performed, reviewed, and evaluated the as-found and as-left containment integrated leak rate test. This review was to ensure that the containment building function was not impacted by the temporary opening which allowed for the replacement of the steam generators and the reactor vessel head.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors evaluated licensee's 1RF12 activities to ensure that risk was considered when developing and when deviating from the outage schedule, the plant configuration was controlled in consideration of facility risk, mitigation strategies were properly implemented, and Technical Specification requirements were implemented to maintain the appropriate defense-in-depth. The inspectors reviewed and/or observed the following items, listed below, as they pertained to the steam generator and reactor vessel head replacement. Coverage of the full scope of Inspection Procedure 71111.20 is documented in Inspection Reports 05000445/446-2007002 and 05000445/446-2007003.

- Unit shutdown and cooldown

- Reduced reactor coolant inventory activities

- Defense in depth and mitigation strategy implementation
- Containment closure capability
- Refueling activities that included fuel offloading, fuel transfer, and core reloading
- Implementation of procedures for foreign material exclusion
- Electrical power source arrangement
- Containment cleanup and inspection
- Unit heatup and startup
- Licensee identification and resolution of problems related to refueling activities

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR), plant drawings, procedure requirements, Technical Specification and Technical Requirements Manual to ensure that the below listed temporary modification was properly implemented. The inspectors: (1) verified that the modification did not have an affect on system operability/availability; (2) verified that the installation was consistent with the modification documents; (3) ensured that the post-installation test results were satisfactory and that the impact of the temporary modification on permanently installed SSCs were supported by the test; (4) verified that the modification was identified on control room drawings and that appropriate identification tags were placed on the affected equipment; and (5) verified that appropriate safety evaluations were completed. The inspectors verified that licensee identified and implemented any needed corrective actions associated with temporary modification.

- Unit 1 Containment Alternate Access, for steam generator and reactor vessel head replacement, in accordance with Final Design Authorization (FDA) FDA-2005-000658-01-02, observed, and reviewed February 24, 2007 through April 6, 2007

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control To Radiologically Significant Areas (71121.01)

a. Inspection Scope

This area was inspected to assess 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 protection workers.

Additionally, using Inspection Procedure 71121.01, "Access Control to Radiologically Significant Areas," the inspectors reviewed activities associated with the steam generator and reactor vessel head replacement to fulfill the inspection requirements of Inspection Procedure 50001, "Steam Generator Replacement Inspection," and Inspection Procedure 71007, "Reactor Vessel Head Replacement Inspection." Specifically, the inspectors reviewed the controls in place at the old steam generator and reactor head storage facility. The inspectors inspected the facility, took independent dose rate measurements, and reviewed the licensee's survey plan. See NRC Inspection Report 05000445;446/2007003, Section 2OS1, for additional information.

The inspectors reviewed the following items:

- Controls (surveys, posting, and barricades) of radiation, high radiation, or airborne radioactivity areas
- Radiation work permits, procedures, engineering controls, and air sampler locations
- Conformity of electronic personal dosimeter alarm set points with survey indications and plant policy; workers' knowledge of required actions when their electronic personnel dosimeter noticeably malfunctions or alarms
- Barrier integrity and performance of engineering controls in airborne radioactivity areas
- Self-assessments, audits, licensee event reports, and special reports related to the access control program since the last inspection
- Corrective action documents related to access controls
- Licensee actions in cases of repetitive deficiencies or significant individual deficiencies
- Radiation work permit briefings and worker instructions

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

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

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

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

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

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

The samples completed for Inspection Procedure 71121.01 will be tracked in Section 2OS1 of NRC Inspection Report 05000445;446/2007003.

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 (ALARA). The inspectors used the requirements in 10 CFR Part 20 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.

Additionally, using Inspection Procedure 71121.02, "ALARA Planning and Controls," the inspectors reviewed activities associated with the steam generator and reactor vessel head replacement to fulfill the inspection requirements of Inspection Procedure 50001, "Steam Generator Replacement Inspection," and Inspection Procedure 71007, "Reactor Vessel Head Replacement Inspection." Specifically, the inspectors reviewed the controls in place at the old steam generator and reactor head storage facility. The inspectors inspected the facility, took independent dose rate measurements, and reviewed the licensee's survey plan. See NRC Inspection Report 05000445;446/2007003, Section 2OS2, for additional information.

The inspectors reviewed the following items:

- Outage (1RF12) work activities and associated work activity exposure estimates, which were likely to result in the highest personnel collective exposures

- Site specific trends in collective exposures, plant historical data, and source-term measurements

- Site specific ALARA procedures

- ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements

- Intended versus actual work activity doses and the reasons for any inconsistencies

- Integration of ALARA requirements into work procedure and radiation work permit documents

- Shielding requests and dose/benefit analyses

- Post-job work activity reviews

- Assumptions and basis for the current annual collective exposure estimate, the methodology for estimating work activity exposures, the intended dose outcome, and the accuracy of dose rate and man-hour estimates

- Method for adjusting exposure estimates, or re-planning work, when unexpected changes in scope or emergent work were encountered

- Exposure tracking system

- Use of engineering controls to achieve dose reductions and dose reduction benefits afforded by shielding

- Workers use of the low dose waiting areas

- 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 and priorities established for these actions, and results achieved against since the last refueling cycle

- Radiation worker and radiation protection technician performance during work activities in radiation areas, airborne radioactivity areas, or high radiation areas

- Self-assessments, audits, and special reports related to the ALARA program since October 2006

- Resolution through the corrective action process of problems identified through post-job reviews and post-outage ALARA report critiques

- Corrective action documents related to the ALARA program and follow-up activities such as initial problem identification, characterization, and tracking

- Effectiveness of self-assessment activities with respect to identifying and addressing repetitive deficiencies or significant individual deficiencies

The samples completed for Inspection Procedure 71121.02 will be tracked in Section 2OS2 of NRC Inspection Report 05000445;446/2007003.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Problem Identification and Resolution (71152)

.1 Inservice Inspection (71111.08)

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 eight corrective action documents (Smart Forms) which dealt with inservice inspection activities and found that the corrective actions were appropriate. From this review the inspectors concluded that the licensee had an appropriate threshold for entering issues into the corrective action program and has procedures that direct a root cause evaluation when necessary. The licensee also had an effective program for applying industry operating experience.

b. Findings

No findings of significance were identified.

.2 Steam Generator and Reactor Vessel Head Replacement Inspection (50001, 71007)

a. Inspection Scope

The inspectors reviewed a sample of the problems identified and documented in the licensee's corrective action program for appropriateness of the corrective actions. The inspector reviewed over eighty corrective action documents which were related to the steam generator and reactor vessel head replacement project and found that the corrective actions were appropriate. The review concluded that the licensee had an appropriate threshold for entering issues into the corrective action program and has procedures that deal with resolution of the issues, even directing a root cause evaluation if necessary. The inspectors also attended numerous contractor overview meetings, that discussed issues identified in the contractor's corrective action program, to ensure that items were entered into the licensee's corrective action program as necessary. The inspectors also determined that the licensee effectively sought and implemented industry operating experience.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

.1 Steam Generator Replacement Inspection (50001)

Design and Planning Inspections (Section 02.02)

a. Inspection Scope

The inspectors used the guidance in Inspection Procedure 50001, "Steam Generator Replacement Inspection," Section 02.02, and inspection procedures referenced therein, to perform the steam generator removal and replacement activities listed below.

Engineering and Technical Support

The inspection activities specified by Section 02.02.a, "Steam Generator Removal and Replacement Inspections," were accomplished in accordance with Inspection Procedures 71111.02, "Evaluation of Changes, Tests, or Experiments", and 71111.17, "Permanent Plant Modifications." These inspections are documented in Sections 1R02 and 1R17 of this inspection report.

Lifting and Rigging

In accordance with Section 02.02.b, the inspectors reviewed the applicable engineering design, modification, and analysis associated with steam generator lifting and rigging including: (1) crane and rigging equipment, (2) steam generator component drop analysis, (3) safe load paths, and (4) load lay-down areas. The inspection focused on the impact of load handling activities on reactor core or spent fuel and its cooling, and plant support systems for Unit 1 and common systems for the operation of Unit 2.

Radiation Protection

In accordance with Section 02.02.c, the inspectors reviewed radiation protection program controls, planning, and preparation in: (1) as low as reasonably achievable planning, (2) dose estimates and tracking, (3) exposure and contamination controls, (4) radioactive material management, (5) radiological work plans and controls, (6) emergency contingencies, and (7) project staffing and training plans. The results are documented in Sections 2OS1 and 2OS2 above, as well as in NRC Inspection Report 05000445;446/2007003, Sections 2OS1 and 2OS2.

Security Considerations and Adverse Impact to the Other Unit

In accordance with Section 02.02.d, the inspectors interviewed security specialists and officers specifically assigned to the steam generator and reactor vessel head replacement project. The inspectors also made frequent observations of security practices during all stages of the project to verify vital and protected barriers were not affected or compromised. The inspectors also reviewed impacts to Unit 2 (operating unit) stemming from the replacement project as activities and schedules changed.

b. Findings

No findings of significance were identified.

Steam Generator Removal and Replacement Inspections (Section 02.03)

a. Inspection Scope

The inspectors used the guidance in Inspection Procedure 50001, Section 02.03, and inspection procedures referenced therein, to perform the steam generator removal and replacement activities listed below.

Welding and Nondestructive Examination Activities

In accordance with Section 02.03.a, inspections were conducted to review welding and NDE activities including: (1) special procedures, (2) training and qualifications, (3) radiography results and work packages, (4) completion of preservice NDE requirements for welds, and (5) completion of baseline eddy current examination of new steam generator tubes. This inspection was performed as part of Inspection Procedure 71111.08, in Section 1R08 of this report.

Lifting and Rigging Activities

In accordance with Section 02.03.b, and Inspection Procedure 71111.23, "Temporary Plant Modifications," the inspectors observed and reviewed several activities associated with lifting and rigging. The inspectors observed and reviewed preparations, crane and rigging inspections, testing, and equipment lay-down areas associated with the following activities:

- Construction of the outside lift system
- Inspection and testing of the outside lift system
- Temporary lift device (inside containment) construction and removal
- Reactor cavity and containment decking (for storage) construction and removal
- Old steam generator removal
- New steam generator installation

Major Structural Modifications and Containment Access and Integrity

In accordance with Section 02.03.c and .d and Inspection Procedures 71111.17 and 71111.23, the inspectors observed the implementation, restoration, where applicable, and removal of the installation of the following structural modifications to support the two steam generator replacement activities listed below.

- Complete removal of the upper steam generator snubber supports
- Temporary alternate containment access

The testing activities for the repair and recovery of the containment building can be found in Sections 1R19, while more information concerning the temporary modification can be found in Section 1R23 of this report.

Unit 1 Outage Operating Conditions

The inspectors used Section 02.03.e and Inspection Procedure 71111.20, “Refueling and Outage Activities,” to complete this inspection. Section 1R20 contains a more detailed explanation of what was observed and reviewed.

Radiation Protection Controls

This inspection was performed during the outage by regional inspectors and the results are documented in Sections 2OS1 and 2OS2 of this report.

Foreign Material Controls

The inspectors followed the guidance contained in Section 02.03.e. The inspectors reviewed and observed procedural controls, field observations, and the licensee’s Plant Event Review Committee (PERC) meetings and various other meetings discussing the foreign material control issues. The inspectors paid particular attention to the reactor coolant and secondary side openings.

Temporary Services

In accordance with Sections 02.03.e, the inspectors reviewed work orders, procedures and observed activities, and performed walkdowns of temporary systems in the containment building. The inspectors also reviewed the fire protection, and industrial safety aspects for alternate construction power, and welding activities.

Radiological Safety Plans for the Old Steam Generator and Reactor Vessel Head Storage Facility

In accordance with Section 02.03.f, the inspectors reviewed the licensee’s radiological safety plans for the storage facility. The inspectors also performed a complete walkdown of the storage facility. This inspection area was also reviewed by regional health physicists inspectors and is documented in Sections 2OS1 and 2OS2 of this report.

b. Findings

No findings of significance were identified.

Post-installation Verification and Testing Inspection (Section 02.04)

a. Scope

The inspectors used the guidance in Inspection Procedure 50001, Section 02.04, and inspection procedures referenced therein, to perform the steam generator removal and replacement activities. Selective inspections were performed in the following areas: (1) containment testing, (2) post-installation inspections and verifications program and its implementation, (3) conduct or RCS leakage testing, (4) conduct of the SG secondary side leakage testing, (5) calibration and testing of instrumentation affected by the SG replacement, (6) procedures required to confirm design and to establish baseline measurements and conduct of testing, and (7) pre-service inspection of new welds. Specific items reviewed are documented in Section 1R19.

b. Findings

No findings of significance were identified.

.2 Reactor Vessel Head Replacement Inspection (71007)

Design and Planning Inspections (Section 02.02)

a. Inspection Scope

The inspectors used the guidance in Inspection Procedure 71007, “Reactor Vessel Head Replacement Inspection,” Section 02.02, and inspection procedures referenced therein, to perform the reactor vessel head removal and replacement activities listed below.

Engineering and Technical Support

The inspection activities specified by Section 02.02.a of Inspection Procedure 71007, were accomplished in accordance with Inspection Procedures 71111.02, “Evaluation of Changes, Tests, or Experiments”, and 71111.17, “Permanent Plant Modifications.” These inspections are documented in Sections 1R02 and 1R17 of this report.

Lifting and Rigging

In accordance with Section 02.02.b of Inspection Procedure 71007, the inspectors reviewed the applicable engineering design, modification, and analysis associated with Reactor Vessel Head lifting and rigging including: (1) crane and rigging equipment, (2) Steam Generator component drop analysis, (3) safe load paths, and (4) load lay-down areas. The inspection focused on the impact of load handling activities on reactor core or spent fuel and its cooling, and plant support systems for the reactor unit and common systems for the other operation unit at the site.

Radiation Protection

The review of radiation protection program controls, planning, and preparation in: (1) ALARA planning, (2) dose estimates and tracking, (3) exposure and contamination controls, (4) radioactive material management, (5) radiological work plans and controls,

(6) emergency contingencies, and (7) project staffing and training plans are documented in Section 2OS1 and Section 2OS2 above, as well as in NRC Inspection Report 05000445;446/2007003, Section 2OS1 and Section 2OS2.

Security Considerations and Adverse Impact to the Other Unit

In accordance with Section 02.02.d, the inspectors interviewed security specialists and officers specifically assigned to the steam generator and reactor vessel head replacement project. The inspectors also made frequent observations of security practices during all stages of the project to verify vital and protected barriers were not affected or compromised. The inspectors also reviewed impacts to Unit 2 (operating unit) stemming from the replacement project as activities and schedules changed.

b. Findings

No findings of significance were identified.

Reactor Vessel Head Fabrication Inspections at Licensee Facility (Section 02.03)

a. Inspection Scope

The inspectors used the guidance in Inspection Procedure 71007, Section 02.03, and inspection procedures referenced therein, to perform the following reactor vessel head fabrication inspection activities.

Heat Treatment

The inspectors verified that the material heat treatment used to enhance the mechanical properties of the Reactor Vessel head material carbon, low alloy, and high alloy chromium steels was conducted per the ASME Code, Section III requirements. Also, the inspections were performed to verify that adequate heat treatment procedures were available to assure that the following requirements were met: (1) furnace atmosphere, (2) furnace temperature distribution and calibration of measuring and recording devices, (3) thermocouple installation, (4) heating and cooling rates, (5) quenching methods, and (6) record and documentation requirements.

Nondestructive Examination

Inspections were conducted to ensure the manufacturing control plan included provisions for monitoring NDE, and to ascertain that the NDE was performed in accordance with applicable code, material specification, and contract requirements.

Welding

The inspectors reviewed the documentation for the weld overlay welding operations that established a layer of stainless steel cladding on the inside of the reactor vessel head to determine if it was accomplished per design. The inspectors also selected a sample of dome-to-flange and control rod drive mechanism (CRDM) flange-to-nozzle welds and reviewed the following items: (1) certified mill test reports of the dome, flange, weld material rods, and CRDM nozzles; (2) certified mill test reports for the welding material for the reactor vessel head cladding; (3) cladding weld records, weld rod material control requisitions, traceability of weld material rods, weld procedure qualification, welder qualifications, and nonconformance reports; (4) CRDM nozzle cladding welding inspection records, weld rod material control requisitions, traceability of weld material rods, weld procedure qualification, welder qualifications, and nonconformance reports; (5) CRDM to nozzle welding and welds inspection records, weld rod material control requisitions, traceability of weld material rods, weld procedure qualification, welder qualifications, and nonconformance reports; and (6) NDE procedures, NDE records of the welds, NDE personnel qualifications, and certification of NDE solvents.

Procedures

Inspections were completed to ensure that repair procedures had been established and that these procedures were consistent with applicable ASME Code, material specification, and contract requirements by verifying: (1) repair welding was conducted in accordance with procedures qualified to Section IX of the ASME Code, (2) all welders had been qualified in accordance with Section IX of the ASME Code, (3) records of the repair were maintained, and (4) that requirements had been established for the preparation of certified material test reports and that the records of all required examinations and tests were traceable to the procedures to which they were performed.

Code Reconciliation

The inspectors reviewed the required documentation, supplemental examinations, analysis, and ASME Code documentation reconciliation to ensure that the original ASME Code N-Stamp remains valid, and that the replacement head complies with appropriate NRC rules and industry requirements. The inspectors also ensured that the design specification was reconciled and a design report was prepared for the reconciliation of the replacement head, verifying that they were certified by professional engineers competent in ASME Code requirements.

Quality Assurance Program

Inspections were conducted to ensure that machining was carried out under a controlled system of operation, a drawing/document control system was in use in the manufacturing process, and that part identification and traceability was maintained throughout processing and was consistent with the manufacturer’s Quality Assurance program. In addition, the inspectors ensured that only the specified drawing and document revisions were available on the shop floor and were being used for fabrication, machining, and inspection through review of applicable procedures.

Compliance Inspection

The inspectors verified that the original ASME Code, Section III, data packages for the replacement Reactor Vessel head were supplemented by documents included in the ASME Code Section XI, (preservice inspection) data packages; examined selected manufacturing and inspection records of the finished machined Reactor Vessel head; and verified compliance with applicable documentation requirements.

b. Findings

No findings of significance were identified.

Reactor Vessel Head Removal and Replacement (Section 02.04)

Lifting and Rigging Activities

In accordance with Section 02.04.a, and Inspection Procedure 71111.23, "Temporary Plant Modifications," the inspectors observed and reviewed several activities associated with lifting and rigging. The inspectors observed and reviewed preparations, crane and rigging inspections, testing, and equipment lay-down areas associated with following activities:

- Construction of the outside lift system
- Inspection and testing of the outside lift system
- Temporary palfinger crane for servicing and preparing the reactor vessel head
- Reactor cavity and containment decking (for storage) construction and removal
- Old reactor vessel head removal
- New reactor vessel head installation, and vessel set

Major Structural Modifications

This inspection was not applicable due to the lack of major structural modifications. The only modification was the installation of the new control rod drive mechanism vent fan modification. The modification was reviewed as part of the 71111.02, "Evaluations of Changes, Tests, or Experiments," and 71111.17, "Permanent Plant Modification," inspection. This modification item is documented in Section 1R02 and 1R17.

Containment Access and Integrity

The inspection is documented in Sections 1R19, 1R23, and Section 4OA5.1.

Unit 1 Outage Operating Conditions

The inspectors used Section 02.04.d and Inspection Procedure 71111.20, "Refueling and Outage Activities," to complete this inspection. Section 1R20 contains a more detailed explanation of what was observed and reviewed.

Radiation Protection Controls

This inspection was performed during the outage by regional inspectors and the results are documented in Sections 2OS1 and 2OS2 of this report.

Foreign Material Controls

The inspectors followed the guidance contained in Section 02.04.d. The inspectors reviewed and observed procedural controls, field observations, and the licensee's plant event review committee (PERC) meetings and various other meetings discussing the foreign material control issues. The inspectors paid particular attention to the reactor coolant and secondary side openings.

Temporary Services

In accordance with Sections 02.04.d, the inspectors reviewed work orders, procedures and observed activities, and performed walkdowns of temporary systems in the containment building. The inspectors also reviewed the fire protection, and industrial safety aspects for alternate construction power, and welding activities.

Radiological Safety Plans for the Old Steam Generator and Reactor Vessel Head Storage Facility

In accordance with Section 02.04.e, the inspectors reviewed the licensee's radiological safety plans for the storage facility. The inspectors also performed a complete walkdown of the storage facility.

b. Findings

No findings of significance were identified.

Post-installation Verification and Testing Inspection (Section 02.05)

a. Scope

The inspectors used the guidance in Inspection Procedure 71007, Section 02.05, and inspection procedures referenced therein, to perform the steam generator removal and replacement activities. Selective inspections were performed in the following areas: (1) containment testing, (2) post-installation inspections and verifications program and its implementation, (3) conduct or RCS leakage testing, (4) conduct of the SG secondary side leakage testing, (5) calibration and testing of instrumentation affected by the SG replacement, (6) procedures required to confirm design and to establish baseline measurements and conduct of testing, and (7) pre-service inspection of new welds. Specific items reviewed are documented in Section 1R19.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On February 9, 2007, the inspectors presented the safety evaluation and permanent plant modifications inspection results to Mr. Steve L. Smith, Site Engineering Director, and other members of the staff who acknowledged those results. No proprietary information was included in this report.

On March 29, 2007, the inspectors presented the In-Service Inspection, Steam Generator and Reactor Vessel Closure Head Replacement Activities inspection results to Mr. Steve L. Smith, Site Engineering Director, and other members of the staff who acknowledged those results. No proprietary information was included in this report.

On May 21, 2007, the inspectors presented the resident inspection results to Mr. R. Flores, Vice President Nuclear Operation, and other members of licensee management. No proprietary information was included in this report.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

S. Abbott, Engineer, Westinghouse
W. Bamford, Engineer, Westinghouse
D. Bersi, Steam Generator Replacement Project, Component Design/Fabrication Lead
M. Blevins, Senior Vice President and Chief Nuclear Officer
J. Brabec, Steam Generator Replacement Project, Installation Manager/Asst. Project Manager
S. Bradley, Supervisor, Health Physics, Radiation Protection & Safety Services
A. Caves, ALARA Coordinator
T. Clouser, Manager, Shift Operations
W. Crosby, Bechtel, NDE Level III
J. Curtis, Radiation Protection Manager, Radiation and Industrial Safety
T. Dorris, Bechtel, Lead Weld Engineer
B. Emanuel, Radiation Protection ALARA
J. Finneran, Steam Generator Replacment Project, Project Engineering Manager
R. Flores, Vice President, Nuclear Operations
J. Gallman, Senior Nuclear Analyst (Work Week Coordinator)
R. Garcia, Supervisor, Radioactive Material Control
D. Haggerty, Project Engineer, Bechtel
N. Harris, Consulting Licensing Analyst
B. Henley, Engineering Consultant (Seismic Analysis)
G. Hietpas, AREVA, Site Director
D. Holland, Senior Nuclear Analyst (Work Week Coordinator)
N. Hood, Project Engineering Manager
T. Hope, Regulatory Performance Manager
M. Kanavos, Plant Manager
S. Karpyak, Risk & Reliability Engineering Supervisor
R. Kidwell, Sr. Nuclear Technologist, Regulatory Affairs
M. Killgore, Engineering Support Director
D. Kissinger, Design Engineering Analysis Engineer
G. Krishnan, Procurement Engineering & Program Manager, SHAW
D. Kross, Director, Maintenance
J. Lamarca, Engineering Smart Team Manager
S. Lantis, Lead Weld Superintendent/Dayshift
B. Lichtenstein, Engineer, Risk and Reliability, Westinghouse
F. Madden, Director, Regulatory Affairs
F. Maddy, JET Engineer
S. Maier, Design Engineering Analysis Manager, Technical Support
B. Mays, Steam Generator Project Manager
E. Meaders, Outage Manager
J. Mercer, Maintenance Rule Coordinator
G. Merka, Regulatory Affairs
J. Meyer, Technical Support Manager
S. Miller, Senior Engineering Analyst, Results Engineering
G. Morini, Westdyne, Project Manager
W. Morrison, Maintenance Smart Team Manager

D. O’Connor, Supervisor, Radiation Protection, Radiation Protection & Safety Services
W. Olsen, Bechtel, Lead Mechanical Weld QC
P. Passalugo, SHAW, ISI Program Lead
J. Patton, Supervisor, Quality Assurance
C. Peters, Bechtel, CAD Weld QC
K. Pitilli, Design Engineering Analysis Engineer
L. Pope, System Engineer
H. Quach, AREVA, Principal Engineer
W. Reppa, JET Manager
J. Rincon, Radiation Protection ALARA
J. Seawright, Consulting Engineer, Regulatory Affairs
R. Segura, Nuclear Analyst Consultant (Electrical Systems)
J. Simmons, Manager, Radiation Protection, Steam Generator Replacement Project
R. Smith, Director, Operations
S. Smith, Site Engineering Director
D. Snow, Regulatory Affairs
D. Sparks, Senior Nuclear Analyst (Work Week Coordinator)
J. Stansbury, Radiation Protection, Sr. Technician
J. Taylor, Engineering Smart Team Manager
D. Tirsun, Engineer, Risk and Reliability, Westinghouse
C. Tran, Engineering Programs Manager
I. Whitt, Engineer, Boric Acid Corrosion Detection Program
D. Wilder, Radiation and Industrial Safety Manager
H. Winn, System Engineer
T. Wright, Bechtel
G. Yezefski, System Engineer

NRC

D. Allen, Senior Resident Inspector
A. Sanchez, Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Opened and Closed

None

Closed

None

Discussed

None

LIST OF DOCUMENTS REVIEWED

Section 1R02: Evaluations of Changes, Tests, or Experiments (71111.02)

Evaluations

Document Number	Title/Description	Revision
59EV-2005-000224-01-00	Main Feedwater Modification Due to Replacement Steam Generators	0

10 CFR 50.59 Screenings

Document Number	Title/Description	Revision
59SC-2005-000658-02-01	Rigging and Transport of OSG and RSG	1
59SC-2005-000224-01-00	Main Feedwater Piping Modifications Due to Replacement Steam Generators	0

Section 1R08: Inservice Inspection Activities

Reports:

Report No.	Title	Date
12UT-11	Calibration Data Sheet	3/2007
12UT-14	Calibration Data Sheet	3/2007
12UT-17	Calibration Data Sheet	3/2007
12UT-20	Calibration Data Sheet	3/2007

Procedures:

Procedure No.	Title	Revision
5069-000-4MP-T040-W0109	Bechtel Welding Procedure Specification P1-T	2
25069-000-4MP-T040-W0458	Bechtel Welding Procedure Specification P8-T(RA)	1
25069-000-4MP-T040-W0576	Bechtel Welding Procedure Specification P1-AT-Lh(CVH+30ºF)	1

Procedures:

Procedure No.	Title	Revision
25069-000-4MP-T040-W0579	Bechtel Welding Procedure Specification P1-T-0(CVN+30)	0
ENSA-GP-7.1	Distribution and Control of Documents	33
EPG-9.02	CPSES Alloy 600 Management Program	0
EPG-703	Inservice Inspection Program	1
EPG-731	ASME Section XI Repair/Replacement Activities	1
GWS-1	General Welding Standard	2
NDE 7.10	Steam Generator Tube Selection and Examination	11
NDE 3.02	ASME Section XI Magnetic Particle Examination	3
PQR 1041	Bechtel Welding Procedure Qualification Record	10
RT-1	NDE Procedure Radiographic Examination	10
STA-703	Inservice Inspection Program	13
STA-731	ASME Section XI Repair & Replacement Activities	6
STA-733	Steam Generator Reliability Program	10
STA-760	RCS Materials Management Program	1
TX-ISI-08	VT-1 and VT-3 Examination Procedure for CPSES	6
TX-ISI-11	Liquid Penetrant Examination for Comanche Peak Steam Electric Station	11
TX-ISI-302	Ultrasonic Examination of Austenitic Piping Welds	2
VL-04-002930	ENSA Welding Procedure Specification	0
VL-04-002931	ENSA Welding Procedure Specification	0
VL-04-002933	ENSA Welding Procedure Specification	0

Procedures:

Procedure No.	Title	Revision
VL-05-000111	Acceptance Testing for Weld Overlay Cladding (Welding Strip ER 309L + Flux)	0
VL-05-000120	Quality Plan for Closure Head Forging	2
VL-05-000426	Visual Examinations	2
VL-05-000679	ENSA Welding Procedure Specification	2
VL-05-000996	ENSA Welding Procedure Specification	0
VL-05-001245	ENSA Welding Procedure Specification	0
VL-05-001329	Measurement of Cladding Thickness by Ultrasonics	1
VL-05-001484	ENSA Welding Procedure Specification	1
VL-05-001564	Acceptance Testing for Weld Overlay Cladding (Welding Strip ER 308L + Flux)	2
VL-05-002108	ENSA Welding Procedure Specification	0
VL-05-002708	Preheating and Hydrogen Bake Requirements	0
VL-05-002709	Magnetic Particles Examination	1
VL-05-002245	Post Weld Stress Relief Heat Treatment	1
VL-05-003073	ENSA Welding Procedure Specification	3
VL-05-003074	ENSA Welding Procedure Specification	3
VL-05-003331	Ultrasonic Examination of the CRDMH/CETNA/RVLMS Full Penetration Welds	2
VL-06-000078	ENSA Welding Procedure Specification	1
VL-06-000172	Radiographic Examinations	4
VL-06-000410	Liquid Penetrant Examinations	2
VL-06-000413	ENSA Welding Procedure Specification	4

Procedures:

Procedure No.	Title	Revision
VL-06-000599	ENSA Welding Procedure Specification	1
VL-06-000741	Preservice Examinations-Manual Ultrasonic Inspection of the CRD Full Penetration Welds of the Replacement RPV Head of Comanche Peak Unit 1	1
VL-06-000870	Non Destructive Examinations after the Hydrostatic Pressure Testing	2
VL-06-000918	ENSA Welding Procedure Specification	0
VL-06-001356	Procedure for Ultrasonic Examination of the Reactor Vessel Closure Head Penetrations During the Comanche Peak 1 RRVCH Pre-service Inspection	1
VL-06-001357	Procedure for the Remote Visual Examination of the Reactor Pressure Vessel Head During the Comanche Peak 1 RRVCH Pre-service Inspection	1
VL-06-001510	Preservice Examinations-Examination using Dye Penetrants, not Soluble in Water, and Directly Visible by Color Contrast on Replacement RPV Head of Comanche Peak Unit 1	1
VL-06-001514	Procedure for the Eddy Current Pre-service Inspection of the Outer Surface of Penetration Nozzle and the J-Groove Weld (CRD Area) of Comanche Peak 1 RRVCH	1
VL-06-001515	Procedure for the Eddy Current Pre-service Inspection of the J-Groove Weld (Cladding Area) of Comanche Peak 1 RRVCH	1
VL-06-001516	Procedure for the Eddy Current Pre-service Inspection of the Vent Pipe (J-Groove Weld and Inner Surface) of Comanche Peak 1 RRVCH	1
VL-06-001517	Procedure for the Eddy Current Pre-service Inspection of Open Penetration Nozzles of Comanche Peak 1 RRVCH	1

Procedures:

Procedure No.	Title	Revision
VL-06-001804	Guidelines for Analyzing Data from PWR Reactor Vessel Head Penetrations Using MASERA and MASERA-TOFD During the Comanche Peak 1 RRVCH Pre-Service Inspection	2
VL-06-002182	Project M505 - RRVCH Arc Strike Repair Procedure (MRR No. 1571X)	0
VL-05-002245	Post Weld Stress Relief Heat Treatment	0
VL-06-003406	Comanche Peak Unit 1 Replacement RV Closure Head - ASME Design Summary	3
WD-1	Bechtel Welding Standard Documentation of Welds	3
WLD-103	Welder Performance Qualifications	6
WCI-606	Work Control Process	9

Design Documents:

Document No.	Title	Revision
DBD-CS-018	Design Criteria for Pipe Stress and Pipe Supports	7
2EP-5.13	Guidelines for Wall Thinning Evaluation for ASME Code Class 2, 3, and ANSI B31.1 Piping	0
900580-07	Comanche Peak Unit 1: Operational Qualification for Dimetrics Gold Track II Welding System	0

Calculations:

Calculation No.	Title	Revision
CT-2-030	Pipe Stress Calculation for Containment Spray Piping Stress Problem CT-2-030	2
CT-2-031	Pipe Stress Calculation for Containment Spray Piping Stress Problem CT-2-031	2

Miscellaneous Documents:

Document No.	Title	Revision
Letter	Fort Calhoun Station, Unit No. 1 - Relief Request for the Use of Radiography using Phosphor Imaging Plate (TAC No. MC8843)	5/2006
Letter	Comanche Peak Steam Electric Station (CPSES) Unit 1 - Summary of Conference Calls with TXU Energy to Discuss the 2004 Steam Generator Tube Inspections (TAC No. MC2564)	7/2004
Letter	Comanche Peak Steam Electric Station Unit 1 - Summary of the Tenth Refueling Outage (1RF10) Steam Generator Tube Inservice Inspection (TAC No. MC4458)	7/2005
EVAL-2006-000751-01-00	Relief Request for use of Phosphor Imaging Plates	
EVAL-2003-002426-25-00	Evaluation to Allow the use of Digital Radiographic Examination	½007
TXX-04141	Comanche Peak Steam Electric Station (CPSES) Unit 1, Docket No. 50-445 Submittal of Unit 1 Tenth Refueling Outage (1RF10) GL 95-05 Report	7/2004
TXX-04157	Comanche Peak Steam Electric Station (CPSES) Unit 1 Tenth Refueling Outage (1RF10) GL 95Steam Generator Twelve Month Report	8/2004
TXX-04172	Comanche Peak Steam Electric Station (CPSES) Unit 1, Docket No. 50-445 Submittal of Corrected Unit 1 Tenth Refueling Outage (1RF10) GL 95-05 Report	9/2004
TXX-05059	Comanche Peak Steam Electric Station (CPSES) Unit 1, Docket No. 50-445 CPSES Response to Request for Additional Information Concerning the Spring 2004 (1RF10) Steam Generator Inservice Inspection Reports	3/2005
TXX-07013	Comanche Peak Steam Electric Station (CPSES) Docket Nos. 50-445 and 50-446 Inspection and Mitigation of Alloy 82/182 Pressurizer Butt Welds	½007

Smartforms:

SMF-2006-002066-00

Codes:

ASME Code Section III, 1989 Edition
ASME Code Section IX, 1998 Edition
ASME Code Section XI, 1998 Edition

Specifications:

CPSES-P-1079, Rev. 6

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

CPSES Containment Crane Plan

Design Basis Document DBD-ME-006

NSSS Upgrade Project Containment Crane Plan

Procedure MDA-304, Control of Heavy Loads and Critical Lifts

Engineering Report: PRA considerations Related to Multiple Crane Operations Inside Containment During 1RF12

CPSES 1RF12 Outage Scope Presentation

Section 1R17: Permanent Plant Modifications (71111.17A)

Final Design Authorization

Number	Title	Revision/Date
FDA-2005-000224-07-02	Instrumentation and Control Change to the SG Level Instrumentation System for the SGRP	2
FDA-2003-002426-01-00	FSAR update for Steam Generator Replacement	0
FDA-2005-000224-04-02	Modify the Main Steam Piping System to Support Replacement of the Unit1 Steam Generators in 1RF12	2
FDA-2005-000658-03-01	Design and Construct the Systems and Structures Needed to Move the Old and New Steam Generators and Reactor Vessel Heads	1
FDA-2005-000658-02-01	Rigging and Transporting of Steam Generators and Reactor Vessel Head	1
FDA-2005-000658-03-00	Roads and Haul Route Engineering Basis	0

Final Design Authorization

Number	Title	Revision/Date
FDA-2005-000658-01-01	Design and construct the systems and structures required to create and restore the Unit-1 Steam Generators and Reactor Vessel Head Replacement Project Containment Alternate Access.	1
FDA-2004-002711-01-00	Develop design modification for Replacement of the Unit-1 Reactor Vessel Closure Head and Control Rod Drive Mechanisms	0
FDA-2005-000224-02-01	Modify the Auxiliary Feedwater (AFW) Piping System to Support Replacement of the Unit 1 Steam Generator in 1RF12	1
FDA-2005-000224-01-00	Main Feedwater Piping Modifications Due to Replacement Steam Generators	0
FDA-2005-000658-02-01	Rigging and Transport of OSG and RSG	1

Calculations

Number	Tittle	Revision/Date
ME-CA-0000-5208	Main Steam and Feedwater Penetration Area Environmental Analysis	3
NUB-099	Subcompartment Analysis for Main Steam Line Penetration Area	1
NUB-168	Steam Generator Main Steamline Break with Gradual Pipe Separation at Break	2
25069-100-COC-1000-0001	Evaluation of Buried Utilities and At-Grade Structures Along OSG and RSG Route	1

Drawings

Number	Title	Sheet No.
SK-F16-05-000658-03-00	Heavy Haul Route Modifications and Existing Commodities Protection	Sheet 14
SK-F18-05-000658-03-00	Heavy Haul Route Modifications and Existing Commodities Protection	Sheet 15
SK-F01-05-000658-03-01	Heavy Haul Route Location Plan	Sheet 1

Drawings

Number	Title	Sheet No.
SK-F02-05-000658-03-01	Replacement Steam Generators and Replacement Reactor Vessel Head Offload Area	Sheet 1
SK-F03-05-000658-03-01	Heavy Haul Route Modifications and Existing Commodities Protection	Sheet 1
SK-F04-05-000658-03-00	Heavy Haul Route Modifications and Existing Commodities Protection	Sheet 2
SK-F05-05-000658-03-00	Heavy Haul Route Modifications and Existing Commodities Protection	Sheet 3
SK-F06-05-000658-03-01	Heavy Haul Route Modifications and Existing Commodities Protection	Sheet 4
SK-F07-05-000658-03-00	Heavy Haul Route Modifications and Existing Commodities Protection	Sheet 5
SK-F08-05-000658-03-00	Heavy Haul Route Modifications and Existing Commodities Protection	Sheet 6
SK-F09-05-000658-03-00	Heavy Haul Route Modifications and Existing Commodities Protection	Sheet 7
SK-F10-05-000658-03-00	Heavy Haul Route Modifications and Existing Commodities Protection	Sheet 8
SK-F11-05-000658-03-01	Heavy Haul Route Modifications and Existing Commodities Protection	Sheet 9
SK-F12-05-000658-03-00	Heavy Haul Route Modifications and Existing Commodities Protection	Sheet 10
SK-F13-05-000658-03-00	Heavy Haul Route Modifications and Existing Commodities Protection	Sheet 11
SK-F14-05-000658-03-00	Heavy Haul Route Modifications and Existing Commodities Protection	Sheet 12
SK-F15-05-000658-03-00	Heavy Haul Route Modifications and Existing Commodities Protection	Sheet 13
SK-F17-05-000658-03-01	Replacement Steam Generator Storage Facility and Replacement Reactor Vessel Head Storage Facility Enclosure Plan and Sections	Sheet 1
SK-F19-05-000658-03-01	Heavy Haul Route and Protection Details	Sheet 1
25069-100-V14-UA30-00512-001	“Rigging International Drawing of Runway System Decking and Handrail,”	0

Drawings

Number	Title	Sheet No.
25069-100-V14-UA30-00190-001	“Rigging International Drawing - Outside Lift System (OLS) Load Test	0
25069-100-V14-UA30-00189-001	“Rigging International Drawing - Outside Lift System (OLS) Load Test,”	0
25069-100-V14-UA30-00188-001	“Rigging International Drawing - Outside Lift System (OLS) Load Test,”	0
25069-100-V14-UA30-00264-002	“Rigging International Drawing - Handling SG’s Inside Containment - General Arrangement,”	0
25069-100-V14-UA30-00263-002	“Rigging International Drawing - Handling SG’s Inside Containment - General Arrangement,”	0
25069-100-V14-UA30-00262-002	“Rigging International Drawing - Handling SG’s Inside Containment - General Arrangement,”	0
25069-100-V14-UA30-00261-001	“Rigging International Drawing - Handling SG’s Inside Containment - General Arrangement,”	0
25069-100-V14-UA30-00513-001	“Rigging International Drawing of Runway System Decking and Handrail,”	0

Procedures

Number	Title	Revision/Date
ECE-5.01-03	Design Change Notices and Related Process Documentation	10
ECE-5.01-04	Technical Evaluation of Replacement Items	3
ECE-5.01-08	Electronic Design Change Process	10
STA-716	Modification Process	16
STA-707	10CFR50.59 Reviews	16
P-2786-31:	Rigging International Procedure.	31

Miscellaneous Documents

Number	Title	Revision/Date
WCAP-16469P	Comanche Peak Unit 1 Replacement Steam Generator Program NSSS Engineering Report	1
DBD-ME-206	Auxiliary Feedwater System	19

Section 1R19: Postmaintenance Testing (71111.19)

Engineering Position Paper -1RF12 IPO-011A Start-up Testing Review

Smart Forms

2007-1419
2007-1413
2007-1303
2007-0434

Section 1R23: Temporary Plant Modifications (71111.23)

Final Design Authorizations (FDA)

2005-3364-02-04
2005-000658-01-02

CPSES Design Basis Document DBD-CS-073, “Concrete Containment Structure,” Revision 7

CPSES Design Basis Document DBD-CS-074, “Containment Liner and Penetrations,” Revision 7

CPSES Design Basis Document DBD-CS-083, “Containment Concrete Internals,” Revision 5

CPSES Design Basis Document DBD-ME-029, “Seismic Qualification of Equipment,” Revision 10

Magnetic Particle Nondestructive Examination Report # MT-095, MT-096

Leak Testing - Vacuum Box Bubble Test Nondestructive Examination Report # VB-002

Bechtel Specification 25069-100-3PS-DG00-Q0002, “Technical Specification for Installation of Q (Safety Related) Cadweld Splices,” Revision 2

Bechtel General Construction Procedure 25069-200-GPP-GCPC-00002, “Cadweld Rebar Splices/Testing of Cadweld Rebar Splices,” Revision 0

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

Audits and Self-Assessments

SA-2006-042, Steam Generator Replacement Radiation Protection Preparedness

Procedures

STA-650	General Health Physics Plan, Revision 5
STA-653	Contamination Control Program, Revision 10
STA-656	Radiation Work Control, Revision 12
STA-660	Control of High Radiation Areas, Revision 10
RPI-602	Radiological Surveillance and Posting, Revision 29
RPI-610	Radiography Controls, Revision 6

Section 2OS2: ALARA Planning and Controls (71121.02)

Audits and Self-Assessments

Self-Assessment Report SA-2006-042, Steam Generator Replacement Radiation Protection Preparedness

Self-Assessment Report SA-2006-048, Review of CPSES ALARA Program

Shielding Requests

2007-13
2007-15
2007-16

Radiation Work Permits

2007-1302
2007-1305
2006-1306

Procedures

RPI-606	Radiation Work and General Access Permits, Revision 15
STA-651	ALARA Program, Revision 9
STA-657	ALARA Job Planning/Debriefing, Revision 11

Other

CPSES ALARA Committee Meeting Minutes
- 11/30/06, 12/14/06, 1/11/07, ½5/07, 2/1/07, 2/8/07

1RF12 Comanche Peak NSSS Upgrade Project Manual, Radiation Protection Activity Plans

Section 4OA2: Problem Identification and Resolution (71152)

Smart Forms

2007

1786	1324	1227	0948	0780	0662	0611
1671	1312	1155	0916	0760	0657	0609
1516	1303	1153	0864	0759	0649	0436
1451	1301	1152	0864	0753	0647	0434
1438	1285	1150	0862	0745	0646	0356
1406	1247	1146	0850	0743	0640	0256
1396	1230	1119	0843	0738	0639	0098
1393	1221	1058	0836	0731	0635	
1364	1220	0981	0818	0726	0627	
1330	1228	0950	0796	0710	0627	

2006

2492
2291
1322
0751

Comanche Peak Steam Electric Station "Quality Assurance Oversight Plan fo NSSS Upgrade Project," Revision 1

Weekly Quality Performance Meetings Attended

February 28, 2007
March 7, 2007
March 14, 2007
March 21, 2007
March 28, 2007
April 4, 2007
April 11, 2007
April 18, 2007

Section 4OA5: Other Activities (50001, 71007)

Procedures:

PPT-P1-7001, “ILRT Alignment and Equipment Protection,” Revision 0
PPT-P1-7002, “ILRT Instrumentation System,” Revision 0
PPT-S1-7014, “Containment Integrated Leakage Rate Test,” Revision 1

LIST OF ACRONYMS

1RF12	unit 1, twelfth refueling outage
ALARA	as low as reasonably achievable
ASME	American Society of Mechanical Engineers
CFR	*Code of Federal Regulations*
CPSES	Comanche Peak Steam Electric Station
CRDM	control rod drive mechanism
FDA	final design authorization
IPO	integrated plant operating procedures
NCV	noncited violation
NDE	nondestructive examination
NRC	Nuclear Regulatory Commission
OPT	operations testing manual
PERC	plant event review committee
SMF	smart form
SOP	system operating procedure
SSC	structures, systems, or components
STA	station administrative manual
UFSAR	updated final safety analysis report