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

August 5, 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 05000445/2007003 AND 05000446/2007003

Dear Mr. Blevins:

On June 22, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Comanche Peak Steam Electric Station, Units 1 and 2 facility. The enclosed integrated inspection report documents the inspection findings which were discussed on July 9, 2007, with you 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 licenses. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents one NRC-identified finding and two self-revealing findings of very low safety significance (Green). Two of the findings were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they were entered into your corrective action program, the NRC is treating these findings as noncited violations (NCV) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, TX 76011-4005 ; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington DC 20555-0001; and the NRC Resident Inspector at Comanche Peak Steam Electric Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be 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, 50-446
License Nos.: NPF-87, NPF-89

Enclosure: NRC Inspection Report 05000445/2007003 and 05000446/2007003
w/Attachment: Supplemental Information

cc w/Enclosure:
Fred W. Madden, Director
Regulatory Affairs
TXU Power
P.O. Box 1002
Glen Rose, TX 76043

George L. Edgar, Esq.
Morgan Lewis
1111 Pennsylvania Avenue, NW
Washington, DC 20004

Terry Parks, Chief Inspector
Texas Department of Licensing
and Regulation
Boiler Program
P.O. Box 12157
Austin, TX 78711

The Honorable Walter Maynard
Somervell County Judge
P.O. Box 851
Glen Rose, TX 76043

Richard A. Ratliff, Chief
Bureau of Radiation Control
Texas Department of Health
1100 West 49th Street
Austin, TX 78756-3189

Environmental and Natural
Resources Policy Director
Office of the Governor
P.O. Box 12428
Austin, TX 78711-3189

Brian Almon
Public Utility Commission
William B. Travis Building
P.O. Box 13326
Austin, TX 78711-3326

Susan M. Jablonski
Office of Permitting, Remediation
and Registration
Texas Commission on
Environmental Quality
MC-122
P.O. Box 13087
Austin, TX 78711-3087

Lisa R. Hammond, Chief
Technological Hazards Branch
National Preparedness Division
FEMA Region VI
800 N. Loop 288
Denton, TX 76209

Electronic distribution by RIV:
Regional Administrator **(BSM1)**
DRP Director **(ATH)**
DRS Director **(DDC)**
DRS Deputy Director **(RJC1)**
DRS Deputy Director **(WBJ)**
Senior Resident Inspector **(DBA)**
Branch Chief, DRP/A **(CEJ1)**
Senior Project Engineer, DRP/A **(TRF)**
Team Leader, DRP/TSS **(CJP)**
RITS Coordinator **(MSH3)**

Only inspection reports to the following:
DRS STA **(DAP)**
M. Kunowski, OEDO RIV Coordinator **(MAK3)**
D. Pelton, OEDO RIV Coordinator **(DLP)**
ROPreports
CP Site Secretary **(ESS)**

SUNSI Review Completed: __CEJ_ ADAMS: √ Yes ☐ No Initials: ___CEJ___
√ Publicly Available ☐ Non-Publicly Available ☐ Sensitive √ Non-Sensitive

R:_REACTORS_CPSES\2007\CP2007-03RP-DBA.wpd

RIV:RI:DRP/A	SRI:DRP/A	C:DRS/EB1	C:DRS/OB	C:DRS/EB2
AASanchez	DBAllen	DAPowers	ATGody	LJSmith
E-ZKDunham	***E-ZKDunham***	***/RA/***	***/RA/***	***/RA/***
8/2/07	8/2/07	**7/23/07**	7/23/07	**7/19/07**
C:DRS/PSB	C:DRP/A			
MPShannon	CEJohnson			
/RA/	***/RA/***			
7/19/07	8/5/07			

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

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Dockets:	50-445, 50-446
Licenses:	NPF-87, NPF-89
Report:	05000445/2007003 and 05000446/2007003
Licensee:	TXU Generation Company LP
Facility:	Comanche Peak Steam Electric Station, Units 1 and 2
Location:	FM-56, Glen Rose, Texas
Dates:	March 24, 2007 through June 22, 2007
Inspectors:	D. Allen, Senior Resident Inspector A. Sanchez, Resident Inspector P. Goldberg, Reactor Inspector S. Alferink, Reactor Inspector G. Guerra, CHP, Health Physicist D. Stearns, Health Physicist K. Clayton, Operations Engineer A. Fairbanks, Reactor Inspector E. Uribe, Reactor Inspector M. Young, Reactor Inspector
Approved by:	Claude Johnson, Chief, Project Branch A Division of Reactor Projects
Attachment:	Supplemental Information

TABLE OF CONTENTS

SUMMARY OF FINDINGS . . . -3-

REPORT DETAILS . . . -5-

REACTOR SAFETY . . . -5-
1R01 Adverse Weather Protection (71111.01) . . . -5-
1R04 Equipment Alignment (71111.04) . . . -5-
1R05 Fire Protection (71111.05Q) . . . -6-
1R06 Flood Protection Measures (71111.06) . . . -7-
1R07 Biennial Heat Sink Performance . . . -8-
1R11 Licensed Operator Requalification Program (71111.11) . . . -11-
1R12 Maintenance Effectiveness (71111.12) . . . -13-
1R13 Maintenance Risk Assessments and Emergent Work Evaluation (71111.13) . . . -13-
1R15 Operability Evaluations (71111.15) . . . -14-
1R17 Permanent Plant Modifications (71111.17A) . . . -15-
1R19 Postmaintenance Testing (71111.19) . . . -16-
1R20 Refueling and Outage Activities (71111.20) . . . -16-
1R22 Surveillance Testing (71111.22) . . . -19-
1R23 Temporary Modifications (71111.23) . . . -20-

RADIATION SAFETY . . . -20-
2OS1 Access Control To Radiologically Significant Areas (71121.01) . . . -20-
2OS2 ALARA Planning and Controls (71121.02) . . . -23-

OTHER ACTIVITIES . . . -25-
4OA1 Performance Indicator Verification (71151) . . . -25-
4OA2 Problem Identification and Resolution (71152) . . . -27-
4OA6 Meetings, Including Exit . . . -30-
4OA7 Licensee-Identified Violations . . . -31-

SUPPLEMENTAL INFORMATION . . . A-1

KEY POINTS OF CONTACT . . . A-1

ITEMS OPENED, CLOSED, AND DISCUSSED . . . A-1

LIST OF DOCUMENTS REVIEWED . . . A-2

LIST OF ACRONYMS . . . A-11

SUMMARY OF FINDINGS

IR 05000445/2007003, 05000446/2007003; 03/24/2007-06/22/2007; Comanche Peak Steam Electric Station, Units 1 and 2. Access Control to Radiologically Significant Areas; ALARA Planning and Controls; Problem Identification and Resolution.

This report covered a 3-month period of inspection by two resident inspectors, five reactor inspectors, two health physicists and one operations engineer. Three Green findings, two of which were NCVs, 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

Cornerstone: Initiating Event

- Green. The inspectors reviewed a self-revealing finding for the inadequate restoration from valve maintenance which resulted in a manual turbine runback. While Unit 1 was at 100 percent power, the 2A Feedwater Heater Normal Level Control Valve 1-LV-2509 failed closed. Operators initially ran the turbine back to 1100 MWe, but eventually reduced load to 700 MWe due to main feedwater pump suction oscillations. The root cause of the event was determined to be inadequate maintenance work practices upon restoration from maintenance on the level control valve.

  The finding is more than minor because it is related to the human performance attribute and affected the initiating event cornerstone objective to limit the likelihood of those events that upset plant stability during power operations. The finding was determined to have a very low risk significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. (Section 4OA2)

Cornerstone: Occupational Radiation Safety (OS)

- Green. The inspectors reviewed a self-revealing noncited violation of 10CFR20.1501(a) for the failure to adequately evaluate radiological conditions in a work area. While performing maintenance on proximity switch cable sleeves on an assembly from the spent fuel pool up-ender, one worker was exposed to concentrations of airborne radioactivity higher than anticipated, resulting in the internal contamination and unplanned dose to the individual. A committed effective dose equivalent of 27 millirem was assigned to the individual. Additionally, after the initial alarm of the airborne activity monitor, a contamination survey of the work area was not performed to evaluate conditions prior to resuming work.

The finding is more than minor because it is associated with the occupational radiation safety attribute of program and process and affected the cornerstone objective because it involves unplanned and unintended dose to a worker. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined that the finding was of very low safety significance because: (1) it was not an ALARA finding, (2) there was no overexposure, (3) there was no substantial potential for an overexposure, and (4) the ability to assess dose was not compromised. In addition, this finding has a cross-cutting aspect in the area of human performance associated with work control because the licensee failed to appropriately coordinate work activities by incorporating actions to keep personnel apprised of conditions at the job site which impacted radiological safety (H3.b). (Section 2OS1)

- Green. The inspectors identified a noncited violation of Technical Specification 5.4.1.a for the failure to develop an adequately detailed work plan for the maintenance of proximity switch sleeves which resulted in the internal contamination of one individual. Specifically, the licensee did not provide adequately detailed work instructions in the work order to allow the ALARA planners to develop an adequate Radiation Work Permit and radiological controls for the maintenance evolution.

  The finding is more than minor because it is associated with the occupational radiation safety attribute of program and process and affected the cornerstone objective because it involves unplanned and unintended dose to a worker. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined that the finding was of very low safety significance because: (1) it was an ALARA work planning finding, (2) the 3-year rolling average collective dose is less than 135 person-rem/unit. In addition, this finding has a cross-cutting aspect in the area of human performance associated with work control because the licensee failed to appropriately plan work activities by incorporating job site conditions which may impact radiological safety (H3.a). (Section 2OS2)

B. Licensee-Identified Violations

A violation of very low safety significance which was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and its corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Comanche Peak Steam Electric Station (CPSES) Unit 1 began the reporting period in refueling outage 1RF12, with the reactor fuel in the spent fuel pool. At the beginning of the reporting period, the new steam generators were in containment with associated pipe welding in progress, and preparation for repair of the containment alternate access was also in progress. Reactor core reload began on April 3, and the refueling outage ended on April 20, at 3:19 p.m. when the main generator output breakers were closed. Unit 1 achieved 100 percent power on April 24, 2007 at 8:47 a.m. On April 27 reactor power was reduced to 80 percent for final testing. Unit 1 returned to 100 percent power on April 28 and remained at essentially full power for the remainder of the reporting period.

CPSES Unit 2 operated at essentially 100 percent power for the entire reporting period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors reviewed Abnormal Conditions Procedure Manual ABN-907, “Acts of Nature,” Revision 11, in the Unit 1 control room in anticipation of severe weather conditions (thunderstorms, tornados, and high winds) predicted for April 24, 2007. The inspectors interviewed the work week coordinator to determine the scheduled work activities and the potential risk impact due to the weather. The inspectors performed a walkdown of the exterior areas inside the protected area to assess the plant’s readiness for high wind velocities, including the material staged in the laydown areas and the status of missile shields, access hatches and exterior doors. The Smart Form (SMF) data base was reviewed for weather related problems that could impact mitigating systems and their support systems to determine if the problems had been properly addressed for resolution.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors: (1) walked down portions of the below listed risk important systems and reviewed plant procedures and documents to verify that critical portions of the selected

systems were correctly aligned; and (2) compared deficiencies identified during the walkdown to the licensee's corrective action program to ensure problems were being identified and corrected.

- Unit 1 Turbine Driven Auxiliary Feedwater (TDAFW) system in accordance with Operations Testing Manual (OPT) Procedure OPT-206A, “AFW System,” Revision 27, while Emergency Diesel Generator (EDG) 1-02 was inoperable for scheduled surveillance testing on April 25, 2007

- Unit 1 Train A 6.9 KV and 480 VAC electrical systems in accordance with System Operating Procedure (SOP) SOP-603A, “6900 V Switchgear,” Revision 14, and SOP-604A, “480 VAC Switchgear and MCCs,” Revision 10, following restoration from the Unit 1 refueling outage, reviewed on April 30 and May 1, 2007

- Unit 2 TDAFWP system in accordance with SOP-304B, “Auxiliary Feedwater System,” Revision 11, while EDG 2-02 was inoperable for scheduled maintenance on May 16, 2007

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q)

Fire Area Tours

a. Inspection Scope

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

- Fire Zone 1SB005, 1SB006, 1SC007 - Unit 1 TDAFW and Motor Driven Auxiliary Feedwater (MDAFW) Pump rooms on the 790 foot elevation on April 3, 2007

- Fire Zone 2SB005, 2SB006, 2SC007 - Unit 2 TDAFW and MDAFW Pump rooms on the 790 foot elevation on April 3, 2007

- Fire Zone 1SK017A, B, C - Unit 1 non-radioactive penetration room on April 3, 2007

- Fire Zone 2SK017A, B, C - Unit 2 non-radioactive penetration room on April 3, 2007

- Fire Zone 1SB008 - Unit 1 safeguards building 810 foot elevation corridor on April 5, 2007

- Fire Zone 2SB008 - Unit 2 safeguards building 810 foot elevation corridor on April 5, 2007

- Fire Zone 1SA142, 1SB143, and 1SB144 - Unit 1 Trains A and B radioactive penetration area rooms, the 831 foot elevation corridor and non-radioactive pipe penetration room on April 5, 2007

- Fire Zone 2SA142, 2SB143, and 2SB144 - Unit 2 Trains A and B radioactive penetration area rooms, the 831 foot elevation corridor and non-radioactive pipe penetration room on April 5, 2007

The inspectors completed eight samples.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

Internal Flood Protection

a. Inspection Scope

The inspectors: (1) reviewed the Updated Safety Analysis Report, the internal flooding analysis, and plant procedures to identify areas that can be affected by internal flooding; (2) reviewed the corrective action program to determine if the licensee identified and corrected flooding problems; (3) verified that operator actions for coping with flooding can reasonably achieve the desired outcomes; and (4) walked down the below listed areas to verify the adequacy of: (a) equipment seals located below the floodline, (b) floor and wall penetration seals, (c) watertight door seals, (d) common drain lines and sumps, (e) sump pumps, level alarms, and control circuits, and (f) temporary or removable flood barriers.

- Units 1 and 2 fuel building 810 foot elevation on May 1-2, 2007

- Unit 2 safeguards building 790 foot elevation on May 8-10, 2007

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

1R07 Biennial Heat Sink Performance

.1 Performance of Testing, Maintenance, and Inspection Activities

a. Inspection Scope

Inspection Module 71111.07, "Heat Sink Performance," requires that two to three safety-related heat exchangers, either directly or indirectly connected to the safety-related service water system, be reviewed to ensure they are either tested or inspected and cleaned. The inspectors selected the following five heat exchangers that were ranked high in the plant specific risk assessment and were directly or indirectly connected to the safety-related service water system:

- Unit 2 engineered safety feature lube oil coolers for the safety injection pumps, centrifugal charging pumps, and containment spray pumps

- Unit 1 residual heat removal system (RHR) heat exchanger

- Unit 2 EDG system jacket water heat exchanger

For the heat exchangers directly connected to the safety-related service water system, the inspectors verified whether testing, inspection, maintenance, and the biotic fouling monitoring program provided sufficient controls to ensure proper heat transfer. The inspectors reviewed chemical controls used to avoid fouling and heat exchanger test, inspection, and cleaning results.

For the chosen heat exchangers, the inspectors verified proper extrapolation of test conditions to design conditions, appropriate use of test instrumentation, and appropriate accounting for instrument inaccuracies. The inspectors reviewed the methods and results of heat exchanger inspection and cleaning, verified that the methods used to inspect and clean were consistent with industry standards, and ensured that the as-found results were appropriately dispositioned such that the final conditions were acceptable. Additionally, the inspectors verified that the licensee appropriately trended these inspection and cleaning results, assessed the causes of the trends, and took necessary actions for any step changes in these trends.

The inspectors observed the inspection and cleaning of the EDG jacket water heat exchanger. The inspectors evaluated the extent of fouling and blockage prior to cleaning, inspected the condition of the surfaces after cleaning, and verified that the number of plugged tubes removed was within the limit of operability of the heat exchanger.

The inspectors completed five inspection samples.

b. Findings

No findings of significance were identified.

.2 Verification of Conditions and Operations Consistent with Design Bases

a. Inspection Scope

For the selected heat exchangers, the inspectors verified that the licensee established heat sink and heat exchanger conditions and operation and test criteria that were consistent with the design assumptions. Specifically, the inspectors reviewed the applicable calculations to ensure that the thermal performance test acceptance criteria for the heat exchangers were being applied consistently throughout the calculations. In addition, the inspectors reviewed test data for the heat exchangers and design and vendor-supplied information to ensure that the heat exchangers were within their design bases. The inspectors reviewed eddy current testing results for the EDG jacket water heat exchanger tubes to verify the structural integrity of the heat exchanger.

The inspectors verified the performance of the ultimate heat sink. The inspectors walked down the heat sink to verify that the heat sink was free from clogging because of macro-fouling, such as silt and debris. The inspectors evaluated chemistry controls that were in place to verify if the controls for biotic fouling were adequate. The inspectors reviewed the performance testing for the pumps and valves in the service water system to verify that the pumps and valves were capable of performing their design function.

b. Findings

Introduction: An unresolved item was identified regarding inadequate design control measures for verifying the adequacy of the safety-related RHR system heat exchangers. The licensee stated that the RHR heat exchangers were not inspected and cleaned, due to ALARA dose consideration. The licensee also stated that the heat exchangers were not tested. Calculation RXE-LA-CPX/0-020, "RHR Cooldown Calculations," and Calculation Number ME-CA-0229-2188, "Component Cooling Water Heat Exchanger Fouling Factor Analysis," were used by CPSES to determine if the RHR heat exchanger would meet its design basis. The calculations only established an overall component cooling water (CCW) heat exchanger fouling factor and the allowable fouling for continued operations. This issue is unresolved for both significance and enforcement, since additional technical review by NRC was needed to assess this issue.

Description: The inspectors reviewed Calculation RXE-LA-CPX/0-020, "RHR Cooldown Calculations," Revision 9, which was prepared to demonstrate the RHR cooldown requirements could be met under various conditions. The licensee assumed an overall fouling factor for the CCW heat exchanger. In addition, the licensee used their computer code "Cooldown" program to determine the performance of the RHR heat exchanger under various operating conditions. The team reviewed the calculation and noted that many assumptions were used. The assumptions included CCW heat exchanger flow rate through the heat exchanger tubes, and an assumed CCW fouling

factor. The CCW flow rate was 7604 gpm/train, which was about the same as the design basis flow rate of 7600 gpm/train. The inspectors noted that if the assumptions made were changed, the results of the calculation could vary.

The inspectors reviewed Calculation ME-CA-0229-2188, “Component Cooling Water Heat Exchanger Fouling Factor Analysis,” Revision 6, which was used by CPSES personnel to determine if the RHR heat exchanger would meet its design basis by determining algorithms necessary to calculate an overall CCW heat exchanger fouling factor and the allowable fouling or margin for continued operation. The fouling factor for the CCW heat exchanger was determined by using recorded temperatures for the inlet and outlet of the CCW heat exchanger and the inlet and outlet temperatures of the safety-related station service water heat exchanger. In addition, the station service water system flow was measured. However, the inspectors noted the calculation stated that instrument uncertainties were not considered in the calculation. The inspectors noted that the instrument uncertainties could cause a large change in the calculation results which could make the results of the calculation using no instrument uncertainties meaningless.

In order to complete the review of the RHR heat exchangers, the inspectors request the following information:

- The margin for the heat transfer rate is needed. The margin consists of the vendor determined heat transfer rate (BTU/hr) at the licensee’s design basis conditions for the supplied heat exchangers, and the required design basis heat transfer rate for the plant.

- Instrument uncertainties are required for Calculation ME-CA-0229-2188 in order to determine the worst case fouling factor for the CCW heat exchanger.

- Information should be supplied to the inspectors concerning the licensee’s computer code “Cooldown” and if it has been verified and validated.

Analysis: At the time of writing, CPSES had not demonstrated that the RHR heat exchangers would meet their safety function. This issue is potentially more than minor because it could affect the Mitigating Systems Cornerstone objective by causing the safety-related RHR system to not transfer sufficient heat to the CCW system to support the safety-related systems. The licensee issued Smart Form SMF-2007-001669-00, dated May 17, 2007, to determine what monitoring or testing should be performed on the RHR heat exchangers. The licensee stated that the RHR heat exchangers were operable due to clean water in both the RHR and CCW closed system loops.

Enforcement: Part 50 of Title 10 of the Code of Federal Regulations, Appendix B, Criterion III, "Design Control," requires, in part, that design control measures shall provide for verifying or checking the adequacy of the design, such as by the performance of design reviews, by the use of alternative or simplified calculation methods, or by performance of a suitable testing program. Additional review by NRC is needed to determine if the RHR heat exchangers would meet their design safety

function. Therefore, this item will be treated as an unresolved item pending additional review of material to determine if the RHR heat exchangers will meet their safety function: URI 05000445;446/2007003-01, Residual heat removal heat exchangers meet design safety function.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

The inspector observed a licensed operator requalification training scenario in the control room simulator on June 21, 2007. The scenario began with a short event to practice immediate actions for a main feedwater pump trip, which included a turbine run back to 700 MWe. The main scenario began with Unit 1 reactor at 100 percent power. The following events then took place: (1) the controlling pressurizer level channel failed low; (2) failure of the controlling Steam Generator 4 level channel; (3) inadvertent start of the TDAFW pump; (4) a trip of Heater Drain Pump 1; (5) two faulted steam generators; and (6) two stuck control rods. The scenario required entry into emergency operating procedures and a Notification of Unusual Event declaration. The scenario was stopped after the crew began safety injection termination.

Simulator observations included formality and clarity of communications, group dynamics, the conduct of operations, procedure usage, command and control, and activities associated with the emergency plan. The inspectors also verified that evaluators and operators were identifying crew performance problems as applicable.

Also on June 21, 2007, the inspectors attended a classroom lecture concerning the chemical volume and control system.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.2 Biennial Licensed Operator Requalification (71111.11B)

a. Inspection Scope

The following inspection activities were performed using Inspection Procedure 71111.11, "Licensed Operator Requalification Program," and 10 CFR 55.46, "Simulation Facilities," as acceptance criteria. The purpose of this review was to determine if the simulator was capable of supporting initial examinations, supporting requalification training required for all licensed operators on shift, and supporting reactivity and control manipulations for initial license applications.

The inspector reviewed the simulator annual performance test book for 2006, in which most of the annual tests were conducted in November 2006, using ANS/ANSI 3.5 -1985, "Nuclear Power Plant Simulators for Use in Operator Training and Examination," as committed to by the licensee in their simulator testing procedure "Simulator Configuration Management," SOMI-009, Revision 8. Because the licensee communicated to the inspector that the simulator would not be used for reactivity manipulation credits on the next exam (scheduled for April 16, 2007), a small sample of the core performance test documents were reviewed in order to assess the adequacy of the simulator in supporting reactivity and control manipulations for future exams as documented on NRC Form 398 "Personal Qualification Statement." While simulator use for reactivity and control manipulation is permitted by 10 CFR 55.46, the simulator must meet the appropriate standards of fidelity, as required by 10 CFR 55.46(c)(2). Documents reviewed during the inspection are listed in the back of this report. The inspector reviewed the criteria in 10 CFR 55.46(c)(2) against the core performance test document samples with Cycle 11 test data from the Start Up and Operations Report for Cycle 11. The simulator was using the Cycle 11 core load for the current training cycle.

One transient test, one malfunction test, and a work package closeout test were ran on the simulator with data capture enabled in order to verify data collected from previous tests was an accurate representation of the test data ran during the testing in December 2006 and also a verification of reasonable model performance based on the current design of the plant. These tests were: (1) Main Steam Line Break-Transient Test Nine; (2) Reactor Coolant Pump Seal Malfunction Test CV-27; and (3) work package closeout for repair of containment model discrepancy in which containment pressure continued to decrease during a Loss of Coolant Accident without any containment spray.

As part of this review, the inspector interviewed one instructor, two reactor operators, two senior reactor operators, all three simulator engineers, the simulator supervisor, and the nuclear training manager. The interviews were performed in order to collect feedback regarding the fidelity of the simulator, the simulator discrepancy reporting system effectiveness, and training on differences between the simulator and the plant. The inspector reviewed several program documents that describe the overall simulator program. One item specifically related to this review was how management groups such as the simulator review board coordinate discrepancy priorities and subsequent repair decisions. These decisions include cost versus training impact for major model upgrades that would improve training on the emergency operating procedures and integrated plant operations. These items were reviewed in order to satisfy the requirements of 10 CFR 55.46(d) for continued assurance of simulator fidelity through problem identification and resolution, proper reporting, root cause evaluations, and a planned schedule for implementing timely corrective actions with proper content. The licensee communicated to the inspector that the chemical and volume control, safety injection, RHR, and CCW system models were replaced earlier this year, with work scheduled to be completed and rolled out to training by the end of the calendar year. The licensee also communicated to the inspector that all of the major hardware components were being replaced on a four-year plan that would be completed by the end of this year.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors independently verified that CPSES personnel properly implemented 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," for the following equipment performance items:

- Unit 1 Train B solid state sequencer placed into a(1) status due to exceeding the unavailability performance criteria for three failures within the past 24 months. These failures were entered into the licensee's corrective action program as SMF-2005-003369-00, 2005-003469-00, and 2007-000090-00.

- Recent failures of the Unit 1 Train B RHR system that caused plant risk to enter a Red condition unexpectedly. These failures resulted in placing the system into a(1), and have been entered into the licensee's corrective action program as SMF-2007-001225-00 and 2007-001250-00.

- Control room radiation monitoring system was returned to a(2), after an extended stay in a(1), following an expert maintenance review panel meeting on June 14, 2007. These issues were placed into the licensee's corrective action program as SMF-2002-004321-00, and 2005-003866-00.

The inspectors reviewed whether the structures, systems, or components (SSCs) that experienced problems were properly characterized in the scope of the Maintenance Rule Program and whether the SSC failure or performance problem was properly characterized. The inspectors assessed the appropriateness of the performance criteria established for the SSCs where applicable. The inspectors also independently verified that the corrective actions and responses were appropriate and adequate.

The inspectors completed three samples.

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:

- Rescheduling of Unit 2 TDAFW pump surveillance test, due to inclement weather, March 29-30, 2007

- Unexpected entry into Risk level Yellow due to severe storm warnings with scheduled maintenance and surveillance activities on April 3, 2007

- Unexpected entry into Outage Risk Assessment Monitor level Red when Unit 1 Train B RHR Pump 1-02 failed to start while attempting to place it in shutdown cooling mode on April 5, 2007

- Unexpected entry into Outage Risk Assessment Monitor level Red when Unit 1 Train B RHR Heat Exchanger Flow Control Valve 1-HCV-607 failed to open while attempting to terminate reactor coolant system (RCS) heatup on April 8, 2007

- Rescheduling of switchyard activities when the Unit 1 schedule for entry into reduced inventory was moved to April 9, 2007

The inspectors completed five samples.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

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

- Quick Turnaround Evaluation (QTE) 2007-001225-01, determine operability of the Unit 1 RHR Pump 1-02 following the failure of the pump to start during a manual start and breaker replacement, on April 7, 2007

- Evaluation (EVAL) 2007-001279-01, determine the operability of the Unit 2 Containment CCW Drain Tank Isolation Valve 2-HV-4725, following the valve stroke time exceeding the Alert limit, on April 11, 2007

- EVAL- 2007-000778-01, determine the operability of the Unit 1 Safety Chiller 1-06 following the identification of a hairline fracture on a three-way solenoid valve that feeds the purge unit, review completed June 22, 2007

- EVAL-2007-000968-01, determine the operability of the Unit 1 CCW Heat Exchanger 1-01 following the plugging of two tubes of the heat exchanger, cleaning and eddy current testing, reviewed June 22, 2007

- QTE-2007-001890-02, determine the operability of Containment Spray Pumps 1-01, 1-03, 1-04, and 2-04 following the discovery that the mechanical seals that were in service were classified as Non-Safety Related, review completed June 22, 2007

The inspectors completed five samples.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17A)

a. Inspection Scope

For the following permanent plant modification described below, the inspectors reviewed the Final Design Authorization (FDA) FDA-2003-002426-03-01, 02; 10 CFR 50.59 screenings, implementing work orders, installation and post-installation testing procedures, and observed installation and testing of portions of the modification to verify that design bases, license bases, and performance capability had not been degraded through this modification.

- The installation of a solenoid valve and key lock open/close hand switch on control room Control Board 1-CB-08 that is powered by the opposite train power source for each atmospheric relief valve on Unit 1. This modification was necessary due to the reduced secondary side water inventory of the new Westinghouse D-76 Steam Generators. Two atmospheric relief valves are now required for a rapid cooldown of the RCS to stop a primary to secondary tube leak. This modification only affected Unit 1, required an update to the Final Safety Analysis Report, design basis documents, and procedures, but did not require a license amendment.

b. Findings

No findings of significance were identified

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors witnessed or reviewed the results of the postmaintenance test for the following maintenance activity:

- Unit 1 Containment Integrated Leakage Rate Test, in accordance with Procedure PPT-S1-7014, Revision 1, "Containment Integrated Leakage Rate Test" following the repair of the containment alternate access made to accommodate the steam generator and reactor vessel head replacements, performed on April 14-15, 2007

Nine other postmaintenance samples are documented as a part of Inspection Report 05000445/2007006, Section 1R19, for the steam generator and reactor vessel head replacement inspection.

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 completed ten samples total.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

.1 Refueling Outage 1RF12

a. Inspection Scope

Refueling outage 1RF12 began on February 24 and included replacement of the steam generators and reactor pressure vessel head. Other significant work completed during this outage included installing a reactor head assembly upgrade with a new control rod drive mechanism air handling unit, applying weld overlay on the pressurizer nozzles, installing new containment recirculation sump screens, and creating and then repairing a hole in the containment, the containment alternate access, to accommodate the removal and installation of the new steam generators. 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 Specifications requirements were implemented to maintain the appropriate defense-in-depth. Specific outage inspections performed and outage activities reviewed and/or observed by the inspectors included:

- Continuing discussions and review of the outage schedule concerning risk with the Outage Manager

- Containment walkdowns to identify safety and quality issues related to the work activities, evaluate material condition of equipment not normally available for inspection, inspect fire protection equipment and fire hazards, observe radiation protection (RP) postings and barriers, and evaluate coatings and debris for potential impact on the recirculation containment sumps

- Reduced inventory activities to perform vacuum fill of the RCS

- RCS instrumentation including Mansell level instrumentation

- Defense in depth and mitigation strategy implementation

- Containment closure capability

- Verification of decay heat removal system capability

- Spent fuel pool cooling capability

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

- Controls over activities that could affect reactivity

- Refueling activities that included fuel transfer and core reloading

- Implementation of procedures for foreign material exclusion

- Electrical power source arrangement

- Containment cleanup and inspection

- Containment recirculation sump inspection after modification of sump filters

- Weld overlay activities on the pressurizer nozzles

- Unit heatup and startup

- Startup testing following replacement of steam generators

- Licensee identification and resolution of problems related to refueling activities

Additional inspections were performed in accordance with Inspection Procedure 71007, "Reactor Vessel Head Replacement Inspection," and Inspection Procedure 50001, "Steam Generator Replacement Inspection," and were documented in Inspection Report 05000445/2007006.

b. Findings

No findings of significance were identified.

.2 Review of Operating Experience Smart Sample (OpESS) FY2007-03, Crane and Heavy Lift Inspection, Supplemental Guidance for IP-71111.20

a. Inspection Scope

Heavy load handling at nuclear power plants may involve risk to stored irradiated fuel and to equipment necessary for a safe shutdown of the reactor. Through the issuance of NUREG-0612,"Control of Heavy Loads at Nuclear Power Plants," July 1980, Generic Letters 80-113, 81-007, and 85-011, and NRC Regulatory Issue Summary 2005-25, the NRC has tried to ensure that the probability of accidents involving dropped heavy loads are kept as low as possible. Because of recent events concerning heavy loads at some nuclear sites, inspectors have completed this supplemental inspection to ensure that facilities have implemented and continue to operate in accordance with the guidance listed above.

The inspectors reviewed licensee's procedures and outage plans for crane use inside and outside containment (due to steam generator and reactor vessel head replacement activities), design basis documents, licensee responses to NUREG-0612, crane maintenance and inspection documents, and interviewed the system engineer.

b. Findings and Observations

No findings of significance were identified. The following inspection items were specifically addressed:

- Determine whether the crane used to lift the reactor vessel head is "single-failure-proof."

  The polar crane at CPSES is placed in "single-failure-proof" mode during heavy lifts, including the reactor vessel head.

- Verify that the licensee has a preventive maintenance program in place based on vendor recommendations for their type of crane, and that crane testing and inspection procedures are completed just prior to use for reactor disassembly (head lift).

  The inspectors verified that CPSES does have a preventative maintenance program and procedures for inspection and testing in place at the site. Maintenance and testing of the polar crane is performed just prior to every outage to ensure proper performance capability.

- This question concerns the verification of commitments and review of load drop analysis if the licensee does not have "single-failure-proof" cranes.

This question is not applicable to CPSES because their polar crane is "single-failure-proof."

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors evaluated the adequacy of periodic testing of important nuclear plant equipment, including aspects such as preconditioning, the impact of testing during plant operations, and the adequacy of acceptance criteria. Other aspects evaluated included test frequency and test equipment accuracy, range, and calibration; procedure adherence; record keeping; the restoration of standby equipment; test failure evaluations; system alarm and annunciator functionality; and the effectiveness of the licensee's problem identification and correction program. The following surveillance test activities were observed and/or reviewed by the inspectors:

- Unit 2 Train A MDAFW pump (inservice test) surveillance in accordance with OPT-206B, "AFW System," Revision 19, observed on April 5, 2007
- Unit 2 Train B MDAFW pump (inservice test) surveillance in accordance with OPT-206B, "AFW System," Revision 19, observed on April 12, 2007
- Unit 1 Containment Purge Supply Penetration MV-0001 local leak rate test in accordance with OPT-834A, "Appendix J Leak Rate Test of Penetration MV-0001 (Containment Purge Supply)," Revision 3, observed on April 13, 2007
- Unit 1 Containment Purge Exhaust Penetration MV-0002 local leak rate test in accordance with OPT-844A, "Appendix J Leak Rate Test of Penetration MV-0002 (Containment Purge Exhaust)," Revision 3, observed on April 13, 2007
- Unit 1 Containment Sump No. 1 Leak Detection System (an RCS leak detection surveillance) in accordance with Instrumentation and Control Manual Procedure (INC) INC-7837A, "Channel Calibration Containment Sump No. 1 Leak Detection System Channel 5142," Revision 1, reviewed on April 17 and 20, 2007
- Unit 1 Containment Sump No. 2 Leak Detection System (an RCS leak detection surveillance) in accordance with INC-7838A, "Channel Calibration Containment Sump No. 2 Leak Detection System Channel 5152," Revision 1, reviewed on April 17 and 20, 2007

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R23 Temporary Modifications (71111.23)

Unit 1 Containment Alternate Access temporary modification was reviewed as part of the steam generator and reactor vessel head replacement and is documented in Section 1R23 of Inspection Report 05000445/2007006.

The inspectors completed one sample.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety [OS]

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 RP manager, RP 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 (RWPs), 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 one airborne radioactivity area

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

- 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

- RWP briefings and worker instructions

- Adequacy of radiological controls, such as required surveys, RP 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 RP technician performance with respect to RP work requirements

The inspectors completed 20 of the required 21 samples.

b. Findings

Introduction. The inspectors reviewed a Green, self-revealing noncited violation of 10 CFR 20.1501(a) for the failure to evaluate the actual radiological hazards during maintenance of proximity switch sleeves, resulting in the internal contamination of one individual.

Description. On November 20, 2006, an instrumentation and control maintenance worker received an internal uptake of radioactive material while replacing sleeves on the fuel transfer system proximity tree which had been removed from the spent fuel pool transfer canal. The sleeves secure electrical cables to the proximity switches. Contamination levels on the equipment prior to the maintenance activity were measured and found to be as high as 500,000 disintegrations per minute per 100 cm^2. The work was performed in a tent constructed on the 860 foot elevation of the fuel handling building, at the top of the ladder from the Unit 1 transfer canal. The proximity tree had been removed from the fuel up-ender assembly and placed in the tent during a previous work evolution.

An airborne radiation monitor, AMS-4, was utilized during the evolution to monitor airborne radioactivity levels within the tent area. The alarm setpoint for the monitor was set at 1 x E-8 microcuries per cubic centimeter. A high efficiency particulate air filter was also utilized to reduce airborne contamination. Continuous coverage by the RP technicians was performed. As the plastic proximity switches were removed, they began to disintegrate. The job continued for approximately 75 minutes at which time the airborne radiation monitor alarmed. Work inside the tent was halted and the tent area evacuated. The RP technician obtained a backup air sample using a high volume air sampler. Field analysis of the sample indicate a derived air concentration (DAC) of 0.18. Since the maintenance evolution was not in progress at the time the backup air sample was obtained, the airborne activity level measured by the backup air sample was not

representative of the airborne activity during the work evolution. No area radiation surveys or surface contamination surveys were conducted prior to resuming work in the tent area.

The maintenance worker was allowed to re-enter the tent and continue replacement of the sleeves. Soon after resuming work, a second airborne monitor alarm was received and the worker was again evacuated from the tent. The airborne monitor indicated an airborne radiation level of three DAC. A surface contamination survey of the tent indicated contamination levels up to 40 milliRad per 100 cm^2. The maintenance worker was monitored for contamination and sent for a whole body count. Based on the results of the whole body count, the individual was assigned a Committed Effective Dose Equivalent of 27 milliRem.

Analysis. The failure to conduct adequate surveys and evaluate the radiological hazards during the work activity resulted in the internal contamination of one individual, constituted a performance deficiency. The finding is more than minor because it is associated with the occupational radiation safety attribute of program and process and affected the cornerstone objective because it involves unplanned and unintended dose to a worker. Using the "Occupational Radiation Safety Significance Determination Process," the inspectors determined that the finding was of very low safety significance because: (1) it was not an ALARA finding, (2) there was no overexposure, (3) there was no substantial potential for an overexposure because the licensee cleared personnel of the area when the air monitor alarmed, and (4) the ability to assess dose was not compromised. In addition, this finding has a crosscutting aspect in the area of human performance associated with work control because the licensee failed to appropriately coordinate work activities by incorporating actions to keep personnel apprised of conditions at the job site which impacted radiological safety (H3.b).

Enforcement. Title 10 CFR 20.1501(a) states that each licensee shall make, or cause to be made, surveys that: (1) may be necessary for the licensee to comply with the regulations in this part; and (2) are reasonable under the circumstances to evaluate the magnitude and extent of radiation levels, concentrations or quantities of radioactive material, and the potential radiological hazards. Title 10 CFR 20.1003 defines survey as an evaluation of the radiological conditions and potential hazards incident to the production, use, transfer, release, disposal, or presence of radioactive material or other sources of radiation. Under the circumstances, a survey of the tent area was necessary to verify compliance with 10 CFR 20.1201, "Occupational Dose Limits for Adults," and 10 CFR 20.1902(d), "Posting of Airborne Radioactivity Areas." However, the licensee failed to conduct adequate airborne and contamination surveys inside the tent to evaluate changes in radiological conditions. Because the failure to evaluate changing radiological conditions during maintenance of the proximity tree was of very low safety significance and was documented in the licensee's corrective action program as SMF-2006-003877, this finding is being treated as a NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000445;446/2007003-02; "Failure to evaluate radiological hazards."

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspectors assessed licensee performance with respect to maintaining individual and collective radiation exposures 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. The inspectors interviewed licensee personnel and reviewed:

- Current 3-year rolling average collective exposure
- 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
- Five work activities of highest exposure significance completed during the last outage (2RF09)
- 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 RWP 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 since the last refueling cycle

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

- Declared pregnant workers during the current assessment period, monitoring controls, and the exposure results

- 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 inspectors completed the required 15 samples and 9 of the optional samples.

b. Findings

Introduction. The inspectors identified a Green, NCV of Technical Specification 5.4.1.a for the failure to develop an adequately detailed work plan for the maintenance of proximity switch sleeves that resulted in the internal contamination of one individual.

Description. On November 20, 2006, an instrumentation and control maintenance worker received an internal uptake of radioactive material while replacing sleeves on the fuel transfer system proximity tree which had been removed from the spent fuel pool transfer canal. The sleeves secure electrical cables to the proximity switches. Contamination levels on the equipment prior to the maintenance activity were measured and found to be as high as 500,000 disintegrations per minute per 100 cm^2. The work was performed in a tent constructed on the 860 foot elevation of the fuel handling building, at the top of the ladder from the Unit 1 transfer canal. The proximity tree had been removed from the fuel up-ender assembly and placed in the tent during a previous work evolution.

Radiation Work Permit 2006-603, with task specific radiological controls, was prepared for this maintenance activity. Three tasks were developed for this RWP. Specifically, Task 1 was for RP technician coverage; Task 2 was for support work above the transfer canal and outside of the tented area; and Task 3 was for work in the transfer canal. According to statements made by an ALARA planner familiar with the task, no task was

written for working on assemblies taken out of the transfer canal and worked in another location such as inside the tented area. The maintenance work order instructions associated with the work activity included three statements for the work to be done. None of the three statements indicated that work was to be performed in a temporary tent located at the top of the fuel transfer canal. The instructions also stated that a hydraulic cylinder was to be removed and a rebuilt cylinder installed. Nothing in the work order indicated that work would be performed on the proximity tree and its associated cabling. Therefore, ALARA Planning did not develop adequate radiological controls for this work evolution.

Analysis. The failure to provide adequate instructions to the ALARA planning group resulted in inadequate radiological controls for the work activity and ultimately resulted with the internal contamination of one individual, constituted a performance deficiency. The finding is more than minor because it is associated with the occupational radiation safety attribute of program and process and affected the cornerstone objective because it involves unplanned and unintended dose to a worker. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined that the finding was of very low safety significance because: (1) it was an ALARA finding, and (2) the 3-year rolling average collective dose is less than 135 person-rem/unit. In addition, this finding has a crosscutting aspect in the area of human performance associated with work control because the licensee failed to appropriately plan work activities by incorporating job site conditions which may impact radiological safety (H3.a).

Enforcement. Licensee Technical Specification 5.4.1.a, states that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, Section 9e(2). Section 9e(2) lists procedures for the control of maintenance evolutions including the necessity for minimizing radiation exposure in preparing the detailed work procedures. Procedure WCI-606, "Work Control Process," Section 6.8.8, states that the work organization will prepare work order instructions which identify significant personnel hazards and place appropriate cautions and warnings within the work instructions. Hazards include radiation, contamination, flammable or explosive gases, hazardous chemicals, asbestos, etc. Contrary to the above, the licensee failed to provide work order instructions that were detailed enough to allow the ALARA Planners to develop an RWP which would provide proper guidance on radiological controls to be used during work on the proximity switch sleeves. Because the finding was of very low safety significance and was documented in the licensee's corrective action program as SMF-2006-003877, this finding is being treated as a NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000445;446/2007003-03; "Failure to Provide a Detailed Work Plan."

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Barrier Integrity Cornerstone

a. Inspection Scope

The inspector reviewed a sample of the performance indicator (PI) data submitted by the licensee regarding the barrier integrity cornerstone to verify that the licensee's data was reported in accordance with the requirements contained in Nuclear Energy Institute

(NEI) 99-02, "Regulatory Assessment Indicator Guideline," Revision 4. The sample included data taken from shift operations logs for OPT-303, "Reactor Coolant System Water Inventory," and the dose equivalent Iodine-131 data from Form CHM-120-101-01, "Reactor Coolant System Control, Technical Specification, and Fuel Performance, Mode 1-3," Revisions 4 through 9, for the period April 2006 to March 2007 for both Units 1 and 2. The inspectors interviewed licensee personnel accountable for collecting and evaluating the PI data. The inspector compared this to the information available on the NRC web page for April 2006 to March 2007 for both Units 1 and 2 for the following PIs:

- Units 1 and 2 RCS Activity
- Units 1 and 2 RCS Leakage

The inspectors completed four samples in this cornerstone.

b. Findings

No findings of significance were identified.

.2 Occupational Radiation Safety Cornerstone

a. Inspection Scope

- Occupational Exposure Control Effectiveness

The inspectors reviewed licensee documents from October 2006 through March 2007. The review included corrective action documentation that identified occurrences in locked high radiation areas (as defined in the licensee's technical specifications), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in NEI 99-02). Additional records reviewed included ALARA records and whole body counts of selected individual exposures. The inspectors interviewed licensee personnel that were accountable for collecting and evaluating the PI data. In addition, the inspectors toured plant areas to verify that high radiation, locked high radiation, and very high radiation areas were properly controlled. PI definitions and guidance contained in NEI 99-02, were used to verify the basis in reporting for each data element.

The inspectors completed the required sample (1) in this cornerstone.

b. Findings

No findings of significance were identified.

.3 Public Radiation Safety Cornerstone

- Radiological Effluent Technical Specification/Offsite Dose Calculation Manual Radiological Effluent Occurrences

The inspectors reviewed licensee documents from October 2006, through March 2007. Licensee records reviewed included corrective action documentation that identified occurrences for liquid or gaseous effluent releases that exceeded PI thresholds and those reported to the NRC. The inspectors interviewed licensee personnel that were accountable for collecting and evaluating the PI data. PI definitions and guidance contained in NEI 99-02, were used to verify the basis in reporting for each data element.

The inspectors completed the required sample (1) in this cornerstone.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Review of Items Entered into the Corrective Action Program

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a routine screening of all items entered into the licensee's corrective action program. This review was accomplished by reviewing the licensee's computerized corrective action program database SMFs, reviewing daily SmartForm Detail Reports, reviewing hard copies of selected SMFs and attending related meetings such as Plant Event Review Committee meetings.

b. Findings

No findings of significance were identified.

.2 Semiannual Trend Review

a. Inspection Scope

On June 22, 2007, the inspectors completed a semiannual review of licensee internal documents, reports, and performance indicators to identify trends that might indicate the existence of more safety significant issues. The inspectors reviewed the following types of documents:

- Corrective Action Documents (Smart Forms)
- System Health Reports
- Planned Maintenance Work Week Critiques
- CPSES Nuclear Overview Department Evaluation Reports (Audits)
- Human Performance Program Health Indicators Package
- Corrective Action Program Health report

- Station Reliability Issues

- CPSES Self-Assessment Reports

b. Findings and Observations

No findings of significance were identified. However, during the review the inspectors did note a continuing trend in foreign material exclusion issues. The inspectors determined that the licensee had adequately identified adverse trends and entered them into the corrective action program using an appropriate threshold.

.3 Selected Issue Follow-up - Review of Test Results from the NSSS Upgrade Project Return to Service Test Program (following Replacement of Unit 1 Steam Generators and Reactor Vessel Head) and documentation, evaluation, and resolution of test deficiencies

a. Inspection Scope

This issue was selected because of the importance of the restart test program in demonstrating the capability of the unit to operate safely and within the design limits. The testing also established baseline performance for important control systems and demonstrated the unit's response to anticipated operational transients. The test program was developed and conducted under the supervision and direction of the Project Joint Test Group. Test results were reviewed by the Project Test Review Group, as a subcommittee of the Station Operation Review Committee. Test deficiencies were documented, evaluated, and resolved in accordance with the existing plant corrective action program and work order processes.

The inspectors reviewed the test results and associated SMFs, evaluations and corrective actions. These were assessed for compliance with the licensee's requirements. Other attributes assessed included: complete and accurate identification of the problem in a timely manner; evaluation and appropriate disposition of the issue; and identification of corrective actions which were appropriately focused to correct the problem. The documents reviewed are listed in the attachment to this report.

The inspector completed one sample.

b. Findings

No findings of significance were identified.

.4 Selected Issue Followup - Inadequate Restoration From Valve Maintenance Resulted in Manual Turbine Load Runback

a. Inspection Scope

The inspectors reviewed SMF-2006-003938 for an operation event that occurred on November 30, 2006 in Unit 1. The inspectors selected this issue because of the unexpected and unplanned transient on reactor power, via a manual turbine runback to 700 MWe.

b. Findings

Introduction. The inspectors identified a Green, self-revealing finding for the inadequate restoration of Unit 1 2A Feedwater Heater Normal Level Control Valve 1-LV-2509, which resulted in the valve failing closed. Operations took action for the failure and eventually ran the turbine load back to 700 MWe.

Description. On November 30, 2006, while at 100 percent power and turbine load at approximately 1220 MWe, the 2A Feedwater Heater Normal Drain Valve 1-LV-2509 unexpectedly failed closed causing a secondary perturbation. Operations responded initially by manually running back turbine load to 1100 MWe. Operations eventually lowered load to 700 MWe to gain control of the main feedwater pump suction pressure oscillations.

The licensee's root cause analysis determined that the root cause for the valve failure was inadequate work practices that led to a fitting on the valve positioner becoming disconnected. Specifically, on November 19, 2005, the Valve Team conducted as found performance testing of Valve 1-LV-2509 as part of scheduled preventative maintenance activities. After testing, the test equipment was removed and the valve positioner was reconnected to the pneumatic signal flex hose via the normal brass Swagelok fitting. The actual mechanic was not available for interview, but the licensee assumed that the mechanic used "skill of the craft" to tighten and leak test the connection. On November 30, 2006, the flex hose fitting became disconnected causing the valve to fail closed, as designed.

The licensee concluded that 1-LV-2509 failed due to poor work practices in that the Valve Team failed to properly tighten the signal flex hose to the positioner following testing, and did not properly leak test the connection with the normal operating system pressure applied.

Analysis. The failure of licensee personnel to properly tighten the Swagelok fitting to the positioner on Valve 1-LV-2509 and adequately perform a leak test on that connection in accordance with Procedure INC-210, "Instrumentation Tubing and Supports Installation and Rework," Revision 4, was the performance deficiency. The performance deficiency is considered to be a finding, and not a violation, because the positioner in question is not safety-related. Using Manual Chapter 0612, the finding was determined to be more than minor because it is related to the human performance attribute and affected the initiating event cornerstone objective to limit the likelihood of those events that upset plant stability during power operations. Manual Chapter 0609, Appendix A (dated March 23, 2007), was used to assess the significance of this finding. The finding was determined to have a very low risk significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. In addition, this finding did not have a crosscutting aspect, because the finding was not reflective of current licensee performances, in that the error was committed in 2005 by contractor who is no longer on site and licensee personnel who perform similar work have not demonstrated similar behavior.

Enforcement. Enforcement action does not apply because the performance deficiency did not involve a violation of regulatory requirements. The licensee has entered this issue into the corrective action program as SMF-2006-003938 and SMF-2007-000425. FIN 05000445/2007003-04; "Inadequate Restoration Following Valve Maintenance."

.5 Radiation Safety Inspection

a. Inspection Scope

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

- Access Control to Radiologically Significant Areas (Section 2OS1)
- ALARA Planning and Controls (Section 2OS2)

b. Findings and Observations

No findings of significance were identified. However, the inspectors noted in reviewing corrective action documents that corrective actions were not implemented in a timely manner. Specifically, a trend in workers entering the radiological controlled area on an incorrect RWP causing dose rate and/or dose alarms was identified by the licensee as early as 1RF11 and the third to fourth quarter of 2005. Although corrective actions were implemented such as heightened awareness and training reinforcement, this did not correct the trend which continued in the 2RF9 outage (October 2006) where a significant number of these events occurred. These events continued to occur at the onset of the 1RF12 outage (February 2007) when the licensee determined that an error trap existed in the RWP log-in process. Specifically, the RWP numbering system for general and specific RWPs was confusing to workers causing them to log-in on a general RWP (lower alarm settings) and not on the specific RWP (higher alarm settings). Corrective actions taken by the licensee included changing the numbering system for general and specific RWPs making them similar in structure and making the computer log-in process steps identical for both. Additionally, radiological controlled area access cards were implemented such that workers write down RWP requirements and keep them for reference. This issue is being continually evaluated by the licensee in SMF-2006-3455.

.6 Biennial Heat Sink Performance Inspection

a. Inspection Scope

The inspectors verified that the licensee had entered significant heat exchanger/heat sink performance problems into the Corrective Action Program. The inspectors reviewed 33 condition reports listed in the attachment.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On April 4, 2007, the inspector presented the results of the Biennial Licensed Operator Requalification inspection to Mr. R. Flores, Site Vice-President, and other members of the licensee's staff. The licensee acknowledged the findings presented in the exit meeting. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On April 6, 2007, the inspectors presented the results of the Occupational Radiation Safety inspection to Mr. R. Flores, Site Vice President and other members of his staff. The licensee acknowledged the findings presented in the exit meeting. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On May 18, 2007, the inspectors presented the results of the Biennial Heat Sink Performance inspection to Mr. M. Kanavos, Plant Manager and other members of his staff. The licensee acknowledged the findings presented in the exit meeting. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On June 29, 2006, the inspectors presented the resident inspection results to Mr. M. Blevins, Senior Vice President and Chief Nuclear Officer, and other members of licensee management. The licensee acknowledged the findings presented in the exit meeting. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violations

The following violation of very low significance (Green) was identified by the licensee and is a violation of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

- Licensee Technical Specification Section 5.7.1.a. requires that each entryway to high radiation areas not exceeding 1.0 rem per hour be barricaded and conspicuously posted as a High Radiation Area. Contrary to this requirement, on February 19, 2007, a high radiation area boundary rope was not attached and the sign not conspicuous to workers with access to the area. This issue was entered into the licensee's corrective action program as SMF-2007-000535-00. This finding is of very low safety significance because it did not involve a very high radiation area or personnel overexposure.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

M. Belvins, Senior Vice President and Chief Nuclear Officer
S. Bradley, Radiation Protection Manager, (Acting)
G. Casperson, Operations Training Manager
A. Caves, ALARA Coordinator
W. Crowe, System Engineer
B. Emanuel, Radiation Protection ALARA
R. Flores, Site Vice President
R. Garcia, Supervisor, RMC
D. Goodman, Simulator Support Supervisor
N. Harris, Consulting Licensing Analyst
J. Henderson, Engineering Smart Team Manager
T. Hope, Manager, Regulatory Performance
M. Kanavos, Plant Manager
B. Knowles, Supervisor, S&C
S. Maier, Design Engineering Analysis Manager
G. Merka, Regulatory Affairs
D. O'Connor, Radiation Protection
J. Patton, Supervisor, Quality Assurance
M. Quick, Engineering Smart Team Manager
W. Reppa, JET Manager
J. Rincon, Radiation Protection ALARA
S. Sewell, Training Manager
J. Simmons, Manager, Radiation Protection, Steam Generator Replacement Project
S. Smith, Director, Site Engineering
J. Stansbury, Radiation Protection, Sr. Technician
D. Wilder, Radiation & Industrial Safety Manager

NRC

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

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000445;446/2007003-01	URI	Residual Heat Removal Heat Exchangers Meet Design Safety Function (Section 1R07.2)

Opened and Closed

05000445;446/2007003-02	NCV	Failure to Evaluate Radiological Hazards (Section 2OS1)

05000445;446/2007003-03	NCV	Failure to Provide a Detailed Work Plan (Section 2OS2)
05000445/2007003-04	FIN	Inadequate Restoration Following Valve Maintenance (Section 4OA2.4)

Closed

None

Discussed

None

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment (71111.04)

DBD-EE-039, Revision 15, Onsite Power Supply

DBD-EE-040, Revision 13, 6.9 KV Electrical Power System

DBD-EE-041, Revision 26, 480 V and 120 V AC Electrical Power System

Drawing E1-0001, Plant One Line Diagram Units 1 and 2, Revision CP-25

Drawing E1-0004, 6.9 KV Auxiliaries One Line Diagram Safeguard Buses, Revision CP-35; Sheet A, Revision CP-25; and Sheet B, Revision CP-10

Drawing E-0005, 480V Auxiliaries One Line Diagram Safeguard Buses, Revision CP-26; and Sheet A, Revision CP-22

Section 1R06: Flood Protection

Calculation Notebook for R&R-PN-021, Revision 3, Internal Flooding Analysis

Section 1R07: Biennial Heat Sink Performance Inspection

Calculations

ME-CA-0229-5129, “Component Cooling Water Flow Rate during a P Signal,” Revision 0

RXE-CPX/0-018, “Ultimate Heat Sink and Maximum Sump Temperature,” Revision 6

ME-CA-0229-5154, “Boundary Conditions for the CCW Fouling Monitoring Program for Increased CCW Flows,” Revision 0

MEB-391, “Minimum Allowable Service Water Flow to Diesel Generators,” Revision 4

RXE-LA-CPX/0-015, "Containment Analysis for Postulated LOCAs Inside Containment at CPSES Units 1 & 2," dated December 16, 2005

ME-CA-0229-2188, "Component Cooling Water Heat Exchanger Fouling Factor Analysis," Revision 6

Smart Forms

2007-001669-00	2005-001065-00	2006-001850-00	2006-001208-00
2004-003650-00	2003-003798-00	2005-004193-00	2006-001236-01
2005-003654-00	2007-001669-00	2006-000707-00	2007-001479-00
2006-002772-00	2002-002859-00	2006-001065-00	2007-001491-00
2005-002747-00	2004-002579-00	2004-001704-00	
2004-002582-00	2006-001255-00	2004-002290-00	
2006-002557-00	2007-000204-00	2007-001160-00	
2006-003289-00	2007-000798-00	2007-001344-00	

Work Orders

3-04-328362-01	3-04-329081-01	3-05-343865-01	3-06-343864-01
3-05-328362-01	3-04-329082-01	3-06-328363-01	3-06-343865-01
3-04-328363-01	3-04-329083-01	3-06-343852-01	5-01-505221-AA
3-05-328363-01	3-04-329084-01	3-06-343853-01	5-02-505218-AA
3-03-342679-01	3-04-330072-01	3-06-343854-01	5-02-505326-AA
3-03-342680-01	3-04-343853-01	3-06-343855-01	5-03-505220-AA
3-04-328362-00	3-05-343861-01	3-06-343860-01	5-03-505222-AA
2003-002830-00	3-05-343864-01	3-06-343861-01	5-04-500682-AA

5-05-500682-AA	5-05-505395-AE	5-06-505395-AB
5-05-505219-AA	5-05-505395-AF	5-06-505395-AC
5-05-505223-AA	5-05-505396-AA	5-06-505395-AD
5-05-505327-AA	5-05-505396-AB	5-06-505395-AE
5-05-505395-AA	5-05-505396-AC	5-06-505396-AA
5-05-505395-AB	5-05-505396-AD	5-06-505396-AB
5-05-505395-AC	5-05-505396-AE	5-06-505396-AC
5-05-505395-AD	5-06-505395-AA	5-06-505396-AE

Procedures

04-034, "Service Water System Fouling Monitoring Program," Revision 3

TDM-901A, "System Data for Throttled Valves/Flow Rates," Revision 11

CHM-150, "Closed Cooling Water Systems," Revision 1

COP-501, "Station Service Water System," Revision 6

COP502A, "Component Cooling Water System," Revision 3

COP-502B, "Component Cooling Water System," Revision 3

COP-609A, "Diesel Generator System," Revision 7

COP-609B, “Diesel Generator System,” Revision 3

ENV-318, “Safe Shutdown Impoundment Algae Control,” Revision 1

ISU-223B, “Remote Shutdown Capability Tests,” Revision 0

PPT-P2-6200, “CCW to RHR/CS HX Outlet Valve flow Control Test,” Revision 2

PPT-SX-7517, “Safe Shutdown Impoundment Inspection,” Revision 2

TDM-901A, “Systems Data Throttled Valves/flow rates,” Revision 11

STA-734, “Service Water System Fouling Monitoring Program,” Revision 3

TSP-711, “Motor Operated Valve Performance Monitoring,” Revision 3

TSP-712, “Motor Operated Valve Margin Determination,” Revision 1

Miscellaneous

CPSES System Status Unit 1 for Component Cooling Water

CPSES System Status Unit 2 for Component Cooling Water

CPSES System Status Unit 2 for Residual Heat Removal

Letter from Cooper Energy Services concerning Engine Heat Rejection for Enterprise DSRV-16-4 Diesel Generator

Water Plant Chemistry Trend Review, First Quarter 2007

Section 1R11.2: Licensed Operator Requalification Program (71111.11B)

NTP-603, Revision 13, "Simulator Certification Management"

SOMI-009, Revision 8, "Simulator Configuration Management"

SOMI-0010, Revision 16, "Simulator Testing Program"

SOMI-013, Revision 2, "Simulator Core Model Evaluation"

SOMI-014, Revision 2, "Simulator Differences"

SAR 04SA0292, "Nestle Acceptance Test" for Core Model

Simulator Malfunction test CV-27 for "Reactor Coolant Pump Seal Leak"

Simulator Malfunction test FW-01 for "Feedwater Piping Leak Outside Containment"

Open Simulator discrepancy report (all)

Closed Simulator discrepancy report from March 2005 thru March, 2007

Annual Operability Test packages
- a. Steady state power test (100%, 80%, 28%)
- b. Transients Reviewed (All 10)
- c. Core test packages for cycles 11 and 12

Work package closeout and post-test for simulator discrepancy # 02SA0397, where containment response was incorrect during LOCA without Containment Spray

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

SMF-2007-001225-00
SMF-2007-001250-00

Section 1R17A: Permanent Plant Modifications

FDA-2003-002426-03
EVAL-2003-002426-17
59SC-2003-002426-04
59SC-2003-002426-09
IPO-11A, Attachment 7.2.11, "Atmospheric Relief Valve Function Test"
ABN-804A, "Response to Fire in the Safeguards Building," Revision 4

Section 1R20: Refueling and Outage Activities

FDA-2005-000658-02

EVAL-2005-000658-02

FSAR Section 9.1.4.2.3

Design Basis Document DBD-ME-006

MDA-304, "Control of Heavy Loads and Critical Lifts," Revision 6

MDA-316, "Control of Load Handling," Revision 0

NSSS Upgrade Project Containment Crane Plan, Revision 3

1FR12 Containment Crane Operations

GL-80-113, "Control of Heavy Loads"

GL-81-007, "Control of Heavy Loads"

GL-1985-011, "Completion of Phase II of 'Control of Heavy loads at Nuclear Power Plants' NUREG 0612"

RIS 2005-25, "Clarification of NRC Guidelines for Control of Heavy Loads"

NUREG 0612, "Control of Heavy Loads at Nuclear Power Plants"

Section 1R22: Surveillance Testing (71111.22)

Work Order 5-05-503782-AA, INC-7838A, Containment Sump No. 2 Leak Detection System Test

Work Order 5-05-503781-AA, INC-7837A, Containment Sump No. 1 Leak Detection System Test

Work Order 5-06-504860-AA, OPT-843A, Appendix J LLRT for Penetration MV-0001

Work Order 5-06-504861-AA, POT-844A, Appendix J LLRT for Penetration MV-0002

QTE- 2002-0693-01-01

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

Corrective Action Documents

SMF-2006-001795, SMF-2006-002207, SMF-2006-002376, SMF-2006-002748,
SMF-2006-003103, SMF-2006-003222, SMF-2006-003331, SMF-2006-003677,
SMF-2006-003877, SMF-2007-000386, SMF-2007-000535, SMF-2007-000604

Audits and Self-Assessments

QA Surveillance of Radworker Practices and Decon/Contamination Controls dated 3/3/07
SA-2006-036, Airborne Monitoring
SA-2006-042, Steam Generator Replacement Radiation Protection Preparedness
SA-2006-050, Control of High Radiation Areas
EVAL-2006-012, Radiation Protection

Radiation Work Permits

30001677, Drywell/Undervessel Control Rod Drive Remove and Replace
30001697, Drywell ISI/NDE/EC and Support
30001705, Drywell MSRV Maintenance
30001693, Drywell Health Physics Support

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
STA-682	Control of Station Diving Operations, Revision 3
RPI-500	Bioassay Program, Revision 9
RPI-602	Radiological Surveillance and Posting, Revision 29
RPI-610	Radiography Controls, Revision 6

Section 2OS2: ALARA Planning and Controls (71121.02)

Corrective Action Documents

2006-0466, 2006-1618, 2006-1795, 2006-2006, 2006-2070, 2006-2071, 2006-2284
2006-2376, 2006-2382, 2006-3103, 2006-3213, 2006-3284, 2006-3330, 2006-3341
2006-3365, 2006-3402, 2006-3404, 2006-3416, 2006-3417, 2006-3419, 2006-3432
2006-3443, 2006-3455, 2006-3502, 2006-3512, 2006-3526, 2006-3630, 2006-3677
2006-3681, 2006-4137, 2007-0218, 2007-0623, 2007-0632, 2007-0640, 2007-0709
2007-0714, 2007-0721, 2007-0745, 2007-0785, 2007-0838

Audits and Self-Assessments

Self-Assessment Report SA-2006-036, Airborne Monitoring

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

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

Self-Assessment Report SA-2006-052, Analysis of Personnel Contaminations during 2RF09

Shielding Requests

2007-13
2007-15
2007-16

Radiation Work Permits

2006-2215
2006-2228
2006-2300
2006-2406
2006-2600

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, 1/25/07, 2/1/07, and 2/8/07)
1RF12 Iodine Trends
1RF12 Cobalt Trends
1RF12 Comanche Peak NSSS Upgrade Project Manual, Radiation Protection Activity Plans
2RF09 CPSES RADIATION PROTECTION ALARA REPORT
4th Quarter ALARA Health Report

Section 4OA1: Performance Indicator Verification

SMF-2007-001507-00

Shift Operations Desktop Instruction NO. 012, Revision 7, 11-4-04, "Operations Department - NRC ROP Performance Indicator: RCS Identified Leakage"

Chemistry Guideline No. 8, Revision 2, 6/7/2004, "CPSES RCS Specific Activity (DEI), NRC Performance Indicator Desktop"

Chemistry/Radiochemistry Manual Procedure CHM-506, Revision 14, "Chemistry Control of the Primary System"

Section 4OA2.2: Semiannual Trend Review

Corrective Action Program Health, 1st Quarter 2007
Open Generic Letter 91-18 Smart Forms list, dated 03/29/07
Plant Management Group Meeting, April 11, 2007
Plant Health Committee (PHC) Meeting, April 2, 2007
Plant Health Committee (PHC) Meeting, May 7, 2007
Plant Health Committee (PHC) Meeting, June 18, 2007
Station Equipment Issues
SMF-2007-000894-00

Section 4OA2.3: Selected Issue Follow-up - Review of Test Results from the NSSS Upgrade Project Return to Service Test Program (following Replacement of Unit 1 Steam Generators and Reactor Vessel Head) and documentation, evaluation, and resolution of test deficiencies

STA -310, "NSSS Upgrade Project Return to Service Test Program," Revision 0

CPSES UFSAR Chapter 14.2

IPO-011A, "Plant Restart and Testing Following Steam Generator Replacement," Revision 0

Unit 1, Cycle 13 Startup Report, Letter TXX-07101, dated June 13, 2007 and attached "NSSS Upgrade Project Return to Service Test Program Summary of Results," dated May 7, 2007

SMF-2007-000434-00
EVAL-2007-000434-01-00
EVAL-2007-000434-02-00
EVAL-2007-000434-03-00
SMF-2007-001303-00
EVAL-2007-001303-01-00
EVAL-2007-001303-02-00 and Lessons Learned from SMF 2007-001303
SMF-2007-001413-00
EVAL-2007-001413-01-01

LIST OF ACRONYMS

ALARA	as low as reasonably achievable
CCW	component cooling water
CFR	*Code of Federal Regulations*
CPSES	Comanche Peak Steam Electric Station
DAC	derived air concentration
EDG	emergency diesel generator
EVAL	evaluation
MDAFW	motor driven auxiliary feedwater
INC	instrumentation and control manual
NCV	noncited violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
OPT	operations testing manual
PI	performance indicator
QTE	quick turnaround evaluation
RHR	residual heat removal
RP	radiation protection
RWP	radiation work permit
SMF	smart form
SOP	system operating procedure
SSC	structures, systems, or components
TDAFW	turbine driven auxiliary feed water
TI	temporary instruction