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

May 18, 2006

James J. Sheppard, President and
Chief Executive Officer
STP Nuclear Operating Company
P.O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT ELECTRIC GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 05000498/2006002 AND 05000499/2006002

Dear Mr. Sheppard:

On April 7, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your South Texas Project Electric Generating Station, Units 1 and 2, facility. The enclosed integrated report documents the inspection findings which were discussed on April 13, 2006, with you and members of your staff.

The inspection examined activities conducted under your licenses as they relate 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 two NRC-identified findings and two self-revealing findings of very low safety significance (Green), evaluated under the risk significance determination process (SDP). These findings were determined to involve violations of NRC requirements. Additionally, a licensee-identified violation which was determined to be of very low safety significance is listed in this report. However, because of the 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 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, U.S. Nuclear Regulatory Commission Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011-4005; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington DC 20555-0001; and the NRC Resident Inspector at South Texas Project Electric Generating Station, Units 1 and 2, facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) 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).

Sincerely,

/RA/

Claude E. Johnson, Chief
Project Branch A
Division of Reactor Projects

Dockets: 50-498
50-499
Licenses: NPF-76
NPF-80

Enclosure:
NRC Inspection Report 05000498/2006002
and 05000499/2006002
w/Attachment: Supplemental Information

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RIV:RI:DRP/A	SRI:DRP/A	PE:DRP/A	SPE:DRP/A	C:DRS/PSB	C:DRS/EB1
JLTaylor	FLBrush	MABrown	TRFarnholtz	MPShannon	JAClark
E-CEJohnson	***E-CEJohnson***	***/RA/***	***/RA/***	***/RA/***	***/RA/***
5/17/06	5/17/06	5/12/06	5/18/06	5/12/06	5/12/06

C:DRS/OB	C:DRS/EB2	C:DRP/A			
ATGody	LJSmith	CEJohnson			
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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Dockets:	50-498, 50-499
Licenses:	NPF-76 NPF-80
Report No:	05000498/2006002 05000499/2006002
Licensee:	STP Nuclear Operating Company
Facility:	South Texas Project Electric Generating Station, Units 1 and 2
Location:	FM 521 - 8 miles west of Wadsworth Wadsworth, Texas 77483
Dates:	January 1 through April 7, 2006
Inspectors:	J. Cruz, Senior Resident Inspector F. Brush, Senior Resident Inspector J. Taylor, Resident Inspector T. Brown, Project Engineer P. Elkman, Emergency Prepardness Specialist, Operations Branch B. Tharakan, Health Physicist, Plant Support Branch G. Replogle, Senior Reactor Inspector J. Keeton, Contractor
Approved By:	Claude E. Johnson, Chief Project Branch A Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000498/2006002, 05000499/2006002; 01/01/06 - 04/07/06; South Texas Project Electric Generating Station; Units 1 & 2; Integrated Resident Report, Event Followup, Occupational Radiation Safety, Identification and Resolution of Problems

The report covered a 3-month period of inspection completed by the resident inspectors and project engineers and announced inspections by regional inspectors. Four Green noncited violations were identified. The significance of issues is indicated by their color (Green, White, Yellow, or Red) and was determined by the Significance Determination Process in Inspection Manual Chapter 0609. Findings for which the significance determination process does not apply 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: Occupational Radiation Safety (OS)

- Green. The inspector reviewed a self-revealing noncited violation of Technical Specification 6.8.1a because the licensee failed to correctly install temporary shielding. Specifically, on October 5, 2005, a crew of four workers installed 270 pounds of shielding per Shielding Request 2005-2-001 on the wrong reactor coolant system valve, RC-142, instead of the correct valve, RC-0017A. The error became evident later that morning when the same crew went to install 6 pounds of shielding on Valve RC-142 and discovered it already had 270 pounds of shielding on it. The corrective action was to place the proper amount of shielding on each valve. The failure to correctly install temporary shielding resulted in the work crew receiving an additional radiation dose of 87 millirem, with one individual receiving as high as 27 millirem of additional radiation dose.

  The finding was greater than minor because it was associated with the Occupational Radiation Safety Cornerstone attribute of Program and Process and affected the cornerstone objective to ensure the adequate protection of a worker's health and safety from exposure to radiation because it resulted in additional exposure to radiation due to actions contrary to procedures. When processed through the Occupational Radiation Safety Significance Determination Process, the finding was determined to be of very low safety significance because it was not an ALARA finding, there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised. In addition, the finding had crosscutting aspects associated with human performance because the failure to follow shielding procedures and Shielding Request 2005-2-001 directly contributed to the finding. (Section 2OS2)

Cornerstone: Initiating Events

- Green. A self-revealing noncited violation of Technical Specification 6.8.1.a was identified for failure to adhere to Plant Operating Procedure 0POP02-BR-0001, "Boron Recycle System Operations," Revision 16. The failure to follow procedure resulted in a subsequent evolution that inadvertently transferred borated water to the Unit 2 volume control tank, decreased power by 2.8 percent, and decreased reactor coolant system temperature by 6EF. The licensee entered the performance deficiency into their corrective action program for resolution.

The failure to follow procedure resulting in a subsequent evolution inadvertently transferring borated water to the Unit 2 volume control tank is a performance deficiency. This finding is greater than minor because it had the actual impact of affecting reactor reactivity and is associated with the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. The finding is only of very low safety significance because the reactivity change was negative and the power reduction transient was minor. The cause of the finding is related to crosscutting aspects in the area of human performance related to failure to follow procedure and attention to detail. (Section 4OA3)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix R, Section III.L.3, in that the method used to position motor-operated valves ("hot-sticking") following a fire in the control room was not independent of the fire area. Specifically, a portion of each valve control circuit was located in the control room. A fire affecting those circuits could result in maloperation or overthrusting of the valves.

  The failure to ensure that all circuits relied on for safe shutdown in response to a control room fire were free of the fire area was a performance deficiency. The issue was more than minor because it affected the reliability objective of the Equipment Performance attribute under the Mitigating Systems Cornerstone. Specifically, motor-operated valves that are relied upon to achieve postfire safe shutdown were less reliable because parts of their control circuits could be damaged by the fire. A Senior Reactor Analyst evaluated the safety significance of this finding using Manual Chapter 0609, "Significance Determination Process," Appendix F, and determined that the finding constituted a low level of degradation for postfire safe shutdown equipment. Therefore, the finding was of very low safety significance. (Section 4OA5)

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix R, Section III.L.1, because the thermohydraulic analysis was inconsistent with actions allowed in the South Texas Project licensing basis for a control room evacuation. Specifically, the analysis inappropriately credited certain manual actions from the control room that are required to be performed in the field.

  The failure to have an adequate written evaluation available for a control room fire scenario was a performance deficiency. This issue was more than minor because it affected the Mitigating Systems Cornerstone attributes of protection from external factors (fire). The inadequate analysis overestimated the amount of time available when accomplishing shutdown actions and, during walkdowns, the inspectors could not verify compliance with the requirements. A Senior Reactor Analyst evaluated the safety significance of this finding using Manual Chapter 0609, "Significance Determination Process," Appendix F, and determined that the finding constituted a low level of degradation for postfire safe shutdown analysis. Therefore, the finding was of very low safety significance. (Section 4OA5)

B. Licensee-Identified Violations

Violations of very low safety significance, which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at essentially 100 percent power throughout the inspection period.

Unit 2 operated at essentially 100 percent power throughout the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors completed a review of the licensee's readiness of seasonal susceptibilities involving cold weather. The inspectors: (1) reviewed plant procedures, the Updated Safety Analysis Report, and Technical Specifications to ensure that operator actions defined in adverse weather procedures maintained the readiness of essential systems; (2) walked down portions of the below listed two systems to ensure that adverse weather protection features (heat tracing, space heaters, weatherized enclosures, and temporary chillers) were sufficient to support operability, including the ability to perform safe shutdown functions; (3) evaluated operator staffing levels to ensure the licensee could maintain the readiness of essential systems required by plant procedures; and (4) reviewed the corrective action program to determine if the licensee identified and corrected problems related to adverse weather conditions.

C (Unit 1 and 2) preparation for entry into the extreme cold weather procedure for expected temperature below 34EF, February 9, 2006

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdown

a. Inspection Scope

The inspectors: (1) walked down portions of the four 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) The inspectors verified the alignment and condition of Emergency Diesel Generator System Train A. The inspectors verified that the system equipment and control board were aligned in accordance with Plant Operating Procedure 0POP02-DG-0001, “Emergency Diesel Generator 11(21),” Revision 41, January 23-24, 2006

- (Unit 1) The inspectors verified the alignment and condition of Containment Spray System Train C. The inspectors verified that the system equipment and control board were aligned in accordance with Plant Operating Procedure 0POP02-CS-0001, “Containment Spray Lineup,” Revision 8, January 24, 2006

- (Unit 2) The inspectors verified the alignment and condition of Essential Cooling Water System Train A. The inspectors verified that the system equipment and control board were aligned in accordance with Plant Operating Procedure 0POP02-EW-0001, “Essential Cooling Water Operations,” Revision 37, January 30, 2006

- (Unit 1) The inspectors verified the condition of the Safety Injection System Train B while the Standby Diesel Generator Train A was in an extended outage for its 5-year inspection using Plant Operating Procedure 0POP02-SI-002 "Safety Injection System Initial Lineup," Revision 17, on March 14, 2006

The inspectors completed four samples.

b. Findings

No findings of significance were identified.

.2 Semiannual System Walkdown

a. Inspection Scope

The inspectors reviewed the following: (1) plant procedures, drawings, the Updated Safety Analysis Report, Technical Specifications, and vendor manuals to determine the correct alignment of the system; (2) reviewed outstanding design issues, operator workarounds, and corrective action program documents to determine if open issues affected the functionality of the system; and (3) verified that the licensee was identifying and resolving equipment alignment problems.

- (Unit 2) The inspectors verified the alignment and condition of the accessible portions of the Auxiliary Feedwater System Train B. The inspectors verified that the system equipment and control board were aligned in accordance with Plant Operating Procedure 0POP02-AF-0001, “Auxiliary Feedwater,” Revision 22, January 25, 2006

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Inspection

a. Inspection Scope

The inspectors walked down the six below 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.

- (Unit 2) Essential cooling water intake structure pump room Train B (Fire Zone Z601), January 31, 2006
- (Unit 1) Channel III battery and distribution rooms (Fire Zones Z043), February 1, 2006
- (Unit 2) Channel III battery and distribution rooms (Fire Zones Z043), February 1, 2006
- (Unit 2) Isolation Valves Cubicle, Train C (Fire Zone Z406), February 16, 2006
- (Unit 2) Isolation Valves Cubicle, Train B (Fire Zone Z407), February 16, 2006
- (Unit 1) Diesel Generator Building, Train A, (Fire Zones Z502, 508,511) February 22, 2006

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

.2 Annual Inspection

a. Inspection Scope

The inspectors observed a fire brigade drill on February 9, 2006, to evaluate the readiness of licensee personnel to prevent and fight fires, including the following aspects: (1) use of protective clothing, (2) use of breathing apparatuses, (3) placement and use of fire hoses, (4) entry into the fire area, (5) use of firefighting equipment, (6) brigade leader command and control, (7) communications between the fire brigade and control room, (8) searches for fire victims and fire propagation, (9) smoke removal, (10) use of prefire plans, and (11) adherence to the drill scenario. The licensee simulated a fire in the Load Center 12J7 13.8 kV transformer.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors: (1) reviewed the Updated Safety Analysis Report, the flooding analysis, and plant procedures to assess susceptibilities involving 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 area to verify the adequacy of: (a) equipment seals located below the floodline, (b) floor and wall penetration seals, (c) watertight door seals, (d) common drain lines and sumps, (e) sump pumps, level alarms, and control circuits, and (f) temporary or removable flood barriers.

- (Unit 1) On March 16, 2006, the inspectors conducted a walkdown of the component cooling water heat exchanger area and grade level exterior of the reactor containment building, fuel handling building, mechanical and electrical auxiliary building, and diesel generator building.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07A)

a. Inspection Scope

The inspectors observed the inspections of the tube side of the jacket water cooler, lube oil cooler, and intercoolers of Standby Diesel Generator 11 in accordance with Work Orders 269869 and 269868 on March 14-15, 2006. Review and assessment of the test results were performed against Procedure OPCP01-ZQ-0004, "Cooling Water System Inspection Guidelines," Revision 2. Discussions were held with the system engineer and the results were compared to the results from the last performance.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

The inspectors observed testing and training of senior reactor operators and reactor operators on February 8, 2006, to identify deficiencies and discrepancies in the training, to assess operator performance, and to assess the evaluator's critique. The training scenario involved a failure of Component Cooling Water Pump 1A concurrent with the failure of both Steam Generator Feedwater Pumps 11 and 12 and followed by a large break loss of coolant accident.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Implementation (71111.12)

a. Inspection Scope

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

- (Common) Essential Chill Water System problems as documented in Condition Report (CR) 05-15959 Root Cause Investigation of Common Cause Analysis of Essential Chiller Failures in 2005, March 31

- (Unit 1) Standby Diesel Generator 13 problems as documented in CR 06-4207; during performance of Plant Surveillance Procedure 0PSP03-DG-0018, the diesel generator failed to reach 110 percent load required by the surveillance, April 3

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

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

.1 Risk Assessment and Management of Risk

a. Inspection Scope

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

- (Unit 2) Evaluation of high risk work activity for troubleshooting of Main Generator Voltage Regulator (Evaluation 1453), January 18, 2006

- (Unit 1) Evaluation of medium risk work activity for replacement of a solenoid valve for Steam Generator 1C Feedwater Hydraulic Jockey Pump (Evaluation 1483), January 30, 2006

- (Unit 1) Evaluation of medium risk work activity for calibration of Feedwater Heater 11 discharge header pressure indicator (Evaluation 1468), January 31, 2006

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

.2 Emergent Work Control

a. Inspection Scope

The inspectors: (1) verified that the licensee performed actions to minimize the probability of initiating events and maintained the functional capability of mitigating systems and barrier integrity systems; (2) verified that emergent work-related activities such as troubleshooting, work planning/scheduling, establishing plant conditions, aligning equipment, tagging, temporary modifications, and equipment restoration did not place the plant in an unacceptable configuration; and (3) reviewed the corrective action program to determine if the licensee identified and corrected problems with risk assessment and emergent work control.

- (Unit 2) Evaluation of medium risk work activity for troubleshooting and postmaintenance testing of control rod drive mechanisms for Control Bank D, Group 1 rods (Evaluation 1470), January 9, 2006

- (Unit 1) Evaluation of high risk work activity for troubleshooting of Class 1E Vital 120 Vac Distribution Panel 1202 (Evaluation 1481), January 20, 2006

- (Unit 1) Evaluation of high risk work activity for troubleshooting of Class 1E 10.0 kV Inverter 1202 (Evaluation 1482), January 23, 2006

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions (71111.14)

a. Inspection Scope

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

- (Unit 1) Control room operator response to Control Rod D-4 inserting when an outward rod motion was requested, January 9, 2006

- (Unit 2) Control room and plant operator actions during online repair of Steam Generator 2C Feedwater Control Valve FCV-0553, March 2, 2006

- (Unit 1) Control room operator and fire brigade response to a fire in the ERFDADS/UPS inverter in the 4160 Volt Train B safety-related switchgear room, March 23, 2006

The inspectors completed three 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 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.

- (Unit 2) Evaluation for returning Essential Chiller 22A to operable status following frequent cycling due to low chilled water temperature (CR 06-300), January 11, 2006

- (Unit 1) Evaluation of Essential Cooling Water Train C through-wall de-alloying indications immediately upstream of the Essential Chillers 11C/12C to Essential Chillers 11A/12A essential cooling water crosstie valve (CR 06-360-1), January 17, 2006

- (Unit 1) Evaluation of Class 1E 10.0 kV Inverter 1202 operable status following the identification of degraded capacitors (CR 05-16523-34), January 24, 2006

- (Unit 2) Evaluation of Low Head Safety Injection Pump 2A discharge test/vent line isolation valve leak (CR 06-1911), February 9, 2006

- (Unit 1) Evaluation of Essential Cooling Water Train C aluminum bronze pipe erosion indications immediately downstream of Component Cooling Water Pump 1C Supplementary Cooler 11C Outlet Valve EW-0105 (CR 06-3132-3), March 9, 2006

- (Unit 1) Evaluation of Essential Chilled Water Chillers 12A and 12C failure to stop from control room and failure to control temperature, respectively (CR 06-4478 and 4479), April 5, 2006

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications(71111.17)

a. Inspection Scope

The inspectors reviewed key affected parameters associated with energy needs, materials/replacement components, timing, heat removal, control signals, equipment protection from hazards, operations, flowpaths, pressure boundary, ventilation boundary, structural integrity, process medium properties, licensing basis, and failure modes for the one modification listed below. The inspectors verified that: (1) modification preparation, staging, and implementation does not impair emergency/abnormal operating procedure actions, key safety functions, or operator response to loss of key safety functions; (2) postmodification testing will maintain the plant in a safe configuration during testing by verifying that unintended system interactions will not occur, SSC performance characteristics still meet the design basis, the appropriateness of modification design assumptions and the modification test acceptance criteria has been met; and (3) the licensee has identified and implemented appropriate corrective actions associated with permanent plant modifications.

- (Common) The inspectors reviewed the licensee's modification package associated with changing the spent fuel pool temperature indicating switch setpoints for both units. Additionally, the inspectors reviewed the specific documents associated with the modification. These included Design Change Package 03-11703-5,6, Supplement 0-1; and Work Orders 405920 and 405928, March 28.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

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

- (Unit 2) Plant Surveillance Procedure 0PSP03-RS-0001, “Monthly Control Rod Operability,” Revision 20, review of postmaintenance testing following exercising of control rod drive mechanism gripper latches, January 12, 2006

- (Unit 1) Plant Surveillance Procedure 0PSP03-CH-0002, “Essential Chilled Water Pump 11B(21B) Inservice Test,” Revision 14, review of postmaintenance testing following planned maintenance, January 25, 2006

- (Unit 1) Plant Surveillance Procedure 0PSP03-CC-0008, "Component Cooling Water System Train 1B Valve Operability Test," Revision 10, review of testing performed to demonstrate operability following planned maintenance of CC-MOV-0774 and CC-MOV-0137, January 25, 2006

- (Unit 2) Plant Maintenance Procedure 0PMP04-ZH-0004, “HEPA Filter Removal and Replacement,” Revision 5, and Plant Surveillance Procedure 0PSP11-ZH-0008, “CRE and FHB HVAC In-Place HEPA Filter Leak Test,” Revision 13, review of testing to demonstrate high-efficiency particulate air (HEPA) filter operability of Fuel Handling Building Exhaust Train 21A following a filter change, January 28, 2006

- (Unit 2) Work Order 458570, “Fuel Handling Building Train A Exhaust Filtration Units Outlet Isolation Damper Operator,” and Plant Surveillance Procedure 0PSP03-HF-0001, “Train A Fuel Handling Building Emergency Exhaust System Operability,” Revision 16, review of testing associated with the rebuild and reinstallation of Damper Actuator HV 9507, January 29, 2006

- (Unit 2) Plant Surveillance Procedure 0PSP03-AF-0010, "Auxiliary Feedwater System Valve Operability Test," Revision 20, review of testing performed to demonstrate operability following planned maintenance, March 1, 2006

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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

- (Unit 1) Plant Surveillance Procedure 0PSP03-CS-0002, "Containment Spray Pump 1B(2B) Inservice Test," Revision 11, January 24, 2006
- (Unit 1) Plant Surveillance Procedure 0PSP03-RH-0002, "Residual Heat Removal Pump 1B(2B) Inservice Test," Revision 10, January 25, 2006
- (Unit 1) Plant Surveillance Procedure 0PSP03-DG-0002, "Standby Diesel 12(22) Operability Test," Revision 28, January 25, 2006
- (Unit 1) Plant Surveillance Procedure 0PSP03-RS-000a, "Monthly Control Rod Operability," Revision 20, January 30, 2006
- (Unit 2) Plant Surveillance Procedure 0PSP03-DG-0002, "Standby Diesel 12(22) Operability Test," Revision 28, January 31, 2006
- (Unit 2) Plant Surveillance Procedure 0PSP03-AF-0007, "Auxiliary Feedwater Pump 14(24) Inservice Test," Revision 31, February 16, 2006
- (Unit 1) Plant Surveillance Procedure 0PSP03-DG-0018, "Standby Diesel 13(23) Twenty-Four Hour Load Test," Revision 26, March 25, 2006

The inspectors completed seven samples.

b. Findings

Introduction: Standby Diesel Generator (SBDG) 13 failed to reach 110 percent load as required by Plant Surveillance Procedure 0PSP03-DG-0018. The licensee performs this surveillance once per 18 months.

Description: On March 25, 2006, SBDG 13 did not meet the surveillance requirement of reaching and maintaining 110 percent load for 2 hours. However, the SBDG did reach 100 percent load during the surveillance test. The licensee immediately declared SBDG 13 inoperable and subsequently determined that it had been inoperable since May 28, 2005. The licensee immediately began troubleshooting SBDG 13. In addition, they started and loaded Unit 1 SBDGs 11 and 12 to full load. The TS action statement only required starting the diesels. However, the shift supervisor ensured that the diesels would be able to meet their full load requirement.

In May, 2005, the governor to fuel rack linkage had been incorrectly set during a planned diesel maintenance overhaul outage. The SBDGs have an extended overhaul outage every 5 years. The licensee determined that the maintenance procedure was correct but that the fuel linkage incorrect setup was due to personnel error. The postmaintenance test was inadequate in that it did not require loading the diesel to 110 percent. The diesel was, however, tested to 100 percent load following the maintenance.

The licensee reviewed the maintenance and testing history for the other five Unit 1 and Unit 2 SBDGs. They determined that three of the diesels had been overhauled since the 18-month surveillance had been performed. The licensee then successfully tested these diesels to 110 percent load.

The licensee also determined that, since May 28, 2005, SBDGs 11 and 12 had been inoperable at various times. The licensee was reviewing the Unit 1 diesel outages since May 28, 2005, to determine if more than one diesel had been inoperable for greater than the Technical Specification allowed outage time.

Analysis: Pending further review by the licensee, if the potential existed that more than one diesel may have been inoperable for longer than the TS allowed outage time due to planned and unplanned diesel outages, NRC will conduct a follow-up review of this issue. This issue is considered unresolved.

Enforcement: Pending further review by the licensee and follow-up by NRC, this finding is considered as Unresolved Item (URI) 05000498/2006002-01, DGs Potentially Inoperable for Greater Than the TS Allowed Outage Time (AOT).

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

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

- (Unit 2) T2-06-0300-4, “Essential Chiller 22A Temporary Recorder Installation,” January 13, 2006

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

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

a. Inspection Scope

The inspector performed in-office reviews of Revision 20-4 to the South Texas Project Electric Generating Station Emergency Plan, and Revision 7 to Emergency Plan Implementing Procedure 0ERP01-ZV-IN01, “Emergency Classification,” both submitted January 12, 2006. These revisions changed emergency classification level descriptions and revised emergency action levels as described in NRC Bulletin 2005-002, and added detail regarding Operations Support Center communication systems and the number of satellite telephones deployed onsite.

These revisions were compared to their previous revisions, to the criteria of NUREG-0654, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” Revision 1, to Nuclear Energy Institute (NEI) 99-01, “Methodology for Development of Emergency Action Levels,” Revision 2, to NRC Bulletin 2005–02, and to the requirements of 10 CFR 50.47(b) and 50.54(q) to determine if the licensee adequately implemented 10 CFR 50.54(q).

The inspector completed two samples during this inspection.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

For the below listed drill and simulator-based training evolution contributing to Drill/Exercise Performance and Emergency Response Organization performance indicators, the inspectors: (1) observed the training evolution to identify any weaknesses and deficiencies in classification, notification, and Protective Action Requirements development activities; (2) compared the identified weaknesses and deficiencies against licensee identified findings to determine whether the licensee is properly identifying failures; and (3) determined whether licensee performance is in accordance with the guidance of the NEI 99-02 document's acceptance criteria.

- On February 22, the inspectors observed a drill and simulator-based training evolution in which the scenario consisted of a reactor coolant pump shaft failure with the failure of the reactor protection system to automatically trip the reactor. The manual trip of the reactor was followed by loose parts in the reactor coolant system resulting in fuel cladding damage with a loss of reactor coolant system boundary integrity. The scenario progressed in a manner which required the emergency response organization to declare and respond to a General Emergency.

- On March 29, the inspectors observed the training emergency exercise. Observations in the control room simulator and the emergency operations facility included opportunities for emergency classifications and offsite notifications. The inspectors reviewed the scenario, drill objectives, activity log sheets, evaluations, and final critique notes. The inspectors also observed the licensee's critique in the emergency operations facility and the technical support center and discussed observations with the drill controllers and evaluators from the emergency operations facility, control room simulator, and technical support center. The inspectors verified that the licensee adequately conducted drills and critiqued the drill performance in accordance with the facility guidelines.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

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 radiation protection manager, radiation protection supervisors, and radiation workers. The inspectors performed independent radiation dose rate measurements and reviewed the following items:

- Performance indicator events and associated documentation packages reported by the licensee in the Occupational and Public Radiation Safety Cornerstones
- Controls (surveys, posting, and barricades) of radiation, high radiation, or airborne radioactivity areas
- Corrective action documents related to access controls
- Radiation work permit briefings and worker instructions
- Adequacy of radiological controls such as, required surveys, radiation protection job coverage, and contamination controls during job performance
- Radiation worker and radiation protection technician performance with respect to radiation protection work requirements

The inspectors completed 6 of the required 21 samples.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

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

- Current 3-year rolling average collective exposure
- Site-specific ALARA procedures
- Five work activities of highest exposure significance completed during the last outage
- ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements
- Intended versus actual work activity doses and the reasons for any inconsistencies
- Integration of ALARA requirements into work procedure and radiation work permit (or radiation exposure permit) documents

- Postjob (work activity) reviews

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

- Workers use of the low dose waiting areas

- First-line job supervisors' contribution to ensuring work activities are conducted in a dose efficient manner

- Radiation worker and radiation protection 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 the last inspection

- Resolution through the corrective action process of problems identified through postjob reviews and postoutage ALARA report critiques

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

The inspectors completed 15 of the required 29 samples.

b. Findings

Introduction: The inspectors reviewed a Green self-revealing noncited violation of Technical Specification 6.8.1a because the licensee failed to correctly install temporary shielding on a reactor coolant system valve.

Description: At approximately 1 a.m. on October 5, 2005, the temporary shielding coordinator briefed a crew of three radiation workers about the installation of 270 pounds of temporary shielding on Valve RC-0071A in accordance with Temporary Shielding Request 2005-2-001. The shielding coordinator then led the crew to the work site to complete the job. However, without verifying the correct valve number, the shielding coordinator directed the work crew to install the shielding on what was assumed to be Valve RC-0017A. At approximately 4:45 a.m., the temporary shielding coordinator accompanied the same work crew to install 6 lbs of shielding on Valve RC-142 in accordance with Temporary Shielding Request (TSR) 2005-2-0009. However, upon arriving at Valve RC-142, it became evident to the shielding coordinator that the wrong valve had the 270 pounds of shielding on it. The coordinator informed the radiation protection supervisor and the decision was made to remove the 270 pounds of shielding from Valve RC-142 and place it on Valve RC-0017A. A subsequent engineering analysis concluded that the additional weight on Valve RC-142 did not impact the structural integrity or function of the valve. However, additional radiation dose was incurred by the work crew to correct the error.

Analysis: The failure to correctly install temporary shielding is a performance deficiency. The finding was greater than minor because it was associated with the Occupational Radiation Safety Cornerstone attribute of Program and Process and affected the cornerstone objective to ensure the adequate protection of a worker's health and safety from exposure to radiation because it resulted in additional exposure to radiation due to actions contrary to procedures. When processed through the Occupational Radiation Safety Significance Determination Process, the finding was determined to be of very low safety significance because it was not an ALARA finding, there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised. In addition, the finding had crosscutting aspects associated with human performance because the failure to follow shielding procedures directly contributed to the finding.

Enforcement: Technical Specification 6.8.1a, states, in part, that procedures be implemented covering the activities listed in Appendix A of Regulatory Guide 1.33. Section 7 of Appendix A of Regulatory Guide 1.33 requires procedures for limiting personnel exposure to radiation. Station Procedure OPRP07-ZR-0004, "Shielding," step 7.3.5, requires that shielding be installed in accordance with the instructions described on Form 1 of TSR 2005-2-0009, which requires that only 6 pounds of shielding be placed on Valve RC-142. On October 5, 2005, the licensee failed to meet this requirement when they installed 270 pounds of shielding on Valve RC-142. Because the finding was of very low safety significance and has been entered into the licensee's corrective action program as CR 05-12304, this violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy: Noncited Violation (NCV) 050-499/2006002-02, Failure to Correctly Install Temporary Shielding.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors sampled licensee submittals for the performance indicators listed below for the period from July through December 2005. This review was to verify the accuracy of the performance indicator data reported during that period, performance indicator definitions, and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 2, were used to verify the basis in reporting for each data element.

Occupational Radiation Safety Cornerstone

Occupational Exposure Control Effectiveness Performance Indicators

Licensee records reviewed included corrective action documentation that identified occurrences in high radiation areas with dose rates greater than 1,000 millirem per hour at 30 centimeters (as defined in the 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 inspector interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data. In addition, the inspector toured plant areas to verify that high radiation and very high radiation areas were properly controlled.

Public Radiation Safety Cornerstone

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

Licensee records reviewed included corrective action documentation that identified occurrences for liquid or gaseous effluent releases that exceeded performance indicator thresholds and those reported to the NRC. The inspectors interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data.

The inspectors completed two of two required Performance Indicator samples during this inspection period.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Cumulative Review of the Effects of Operator Workarounds

a. Inspection Scope

The inspectors reviewed the cumulative effects of operator workarounds in Units 1 and 2, on February 9, 2006, to determine: (1) the reliability, availability, and potential for misoperation of a system; (2) if multiple mitigating systems could be affected; (3) the ability of operators to respond in a correct and timely manner to plant transients and accidents; and (4) if the licensee has identified and implemented appropriate corrective actions associated with operator workarounds.

The inspectors completed one sample.

.2 Daily CR Review

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 daily screening of items entered into the licensee's corrective action program. This review was accomplished by reviewing hard copy or electronic summaries of each CR, attending various daily screening meetings, and accessing the licensee's computerized corrective action program database.

b. Findings and Observations

No findings of significance were identified.

.3 Review of Missing Motor-Operated Valve (MOV) T-Drains Issue

The inspectors evaluated the effectiveness of the licensee's problem identification and resolution processes regarding discovery of missing T-drains in the motor for Valve 2-SI-0019C. The issue was documented by the licensee in CR 05-13606 and reported in Licensee Event Report (LER) 0500499/2005-006. On October 18, 2005, with Unit 2 shut down for refueling, the licensee identified that the limitorque motor on Valve 2-SI-MOV-0019C had ordinary pipe plugs installed in the two drilled holes designed to contain T-drains, which are required as part of the motor's environmental qualification. During the subsequent inspection of other Unit 2 MOVs, Valve 2-RA-MOV-0003 (Radiation Monitor RT-8011 inside containment isolation valve) was discovered to have one ordinary pipe plug and one T-drain installed rather than the two required T-drains. It was determined that the condition was reportable because the lack of T-drains would have rendered the associated systems inoperable for a period of time in excess of that allowed in the Limiting Condition for Operation. The absence of T-drains in Valve SI-19C make the valve inoperable per Technical Specifications 3.5.2, which requires, in part, that three independent emergency core cooling system (ECCS) subsystems shall be operable. The licensee determined that the MOV motors had been replaced/refurbished in 1994 and that inadequate corrective actions from LER 93-008, a previous occurrence of missing T-drains, and inadequate maintenance procedure guidance had been the causes. Corrective actions included inspecting all accessible harsh environment MOVs, revising the maintenance procedure, and personnel requalification training. This issue involved problem identification and resolution crosscutting aspects in the area of human performance as several motor inspections had occurred without the issue being discovered. The licensee's extent of condition assessment, operability assessment, and maintenance plan were reviewed and discussed with engineering and operations personnel. The inspectors evaluated the condition records against the requirements in the licensee's corrective action program and 10 CFR Part 50, Appendix B. The enforcement aspects of the violation are discussed in Section 4OA7.

4OA3 Event Follow-up (71153)

.1 Unit 2 Inadvertent Boration.

Introduction: A self-revealing noncited violation of Technical Specification 6.8.1.a was identified for a failure to adhere to Plant Operating Procedure 0POP02-BR-0001, "Boron Recycle System Operations," Revision 16. The failure to follow procedure resulted in a subsequent evolution that inadvertently transferred borated water to the volume control tank, decreased power by 2.8 percent, and decreased reactor coolant system temperature by 6EF.

Description: On October 28, 2005, Operations transferred Recycle Holdup Tank (RHUT) 2B contents per Plant Operating Procedure 0POP02-BR-0001, "Boron Recycle System Operations," Revision 16. When the procedure was completed, step 11.30.0 to close Valve CV-0754 remained unperformed. Subsequent lining up to transfer RHUT 2B contents to the Refueling Water Storage Tank per Plant Operating Procedure 0POP02-FC-0001, "Spent Fuel Pool Cooling and Cleanup System," caused the volume control tank level to increase with more highly borated water, decreasing coolant temperature and reactor power. The licensee entered this performance deficiency into their corrective action program for resolution.

Analysis: This finding is greater than minor because it had the actual impact of affecting reactor reactivity and is associated with the Initiating Events Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. The finding is of very low safety significance because the reactivity change was negative and the power reduction transient was minor. The finding passed the screening criteria of Inspection Manual Chapter 0612, Appendix B. Using the Significance Determination Process of Inspection Manual Chapter 0609, Appendix A, under the Initiating Events Cornerstone, the answer to the Phase 1 screening question of "Does the finding contribute to both the likelihood of a reactor trip AND the likelihood that mitigation equipment or functions will not be available?" was determined to be "No" because there was no degradation of the chemical and volume control system that could have decreased the probability of system operation. Therefore, the finding was determined to be of very low safety significance. The root cause of the performance deficiency involved a failure to follow procedure. This finding involved crosscutting aspects in the area of human performance related to failure to follow procedure and attention to detail.

Enforcement: Technical Specification 6.8.1.a requires that procedures be established, implemented, and maintained covering the applicable procedures in Appendix A of Regulatory Guide 1.33. Appendix A, Item 3.n, requires procedures be maintained for the chemical and volume control system. Plant Operating Procedure OPOP02-BR-0001, "Boron Recycle System Operations," Revision 16, was not properly implemented in that a failure to follow procedure resulted in subsequent inadvertent boration. The inadvertent boration increased the risk of an initiating event of a transient initiatior. Because this finding was entered into the licensee's corrective action program

as CR 05-14545 and is of very low safety significance, this finding is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy:
NCV 05000499/2006002-03, Inadvertent Boration.

.2 <u>(Closed) LER 0500499/2005-006</u>: Unit 2 Missing Motor-Operated Valve T-Drains

On October 18, 2005, with Unit 2 shut down for refueling, the licensee identified that the limitorque motor on Valve 2-SI-MOV-0019C had ordinary pipe plugs installed in the two drilled holes designed to contain T-drains, which are required as part of the motor's environmental qualification. During the subsequent inspection of other Unit 2 MOVs, Valve 2-RA-MOV-0003 (Radiation Monitor RT-8011 inside containment isolation valve) was discovered to have one ordinary pipe plug and one T-drain installed rather than the two required T-drains. It was determined that the condition was reportable because the lack of T-drains would have rendered the associated systems inoperable for a period of time in excess of that allowed in the Limiting Condition for Operation. The absence of T-drains in Valve SI-19C make the valve inoperable per Technical Specification 3.5.2, which requires, in part, that three independent ECCS subsystems shall be operable. The licensee determined that the MOV motors had been replaced/refurbished in 1994 and that inadequate corrective actions from LER 93-008, a previous occurrence of missing T-drains, and inadequate maintenance procedure guidance had been the causes. Corrective actions included inspecting all accessible harse environment MOVs, revising the maintenance procedure, and personnel requalification training. This finding is more than minor because it affected the availability, reliability, and capability objectives of the mitigating systems Reactor Safety Cornerstone and was considered to have very low safety significance (Green) using Appendix A of the significance determination process because of the availability of two other trains and information from the Senior Risk Analyst that the specific system response (low head safety injection pump recirculation mode) was not included in the plant significance determination process workbook. Also, a redundant outside containment isolation valve was available for RT-8011. The licensee entered this performance deficiency into their corrective action program (CR 05-13606) for resolution. This licensee-identified finding involved a violation of Technical Specification 3.5.2, Emergency Core Cooling System. This issue involved problem identification and resolution crosscutting aspects in the area of human performance as several motor inspections had occurred without the issue being discovered. The enforcement aspects of the violation are discussed in Section 4OA7. This LER is closed.

.3 <u>(Closed) LER 0500499/2005-005</u>: 125 Vdc Switchboard De-energized Without Placing Fuel Handling Building Heating, Ventilation, and Air Conditioning (HVAC) in Emergency Recirculation Mode

On October 22, 2005, during an outage work review, the licensee identified that Technical Specification 3.3.2, Action 30, was not performed as required. Technical Specification 3.3.2, Action 30, states, in part, that with irradiated fuel in the spent fuel pool and with the number of operable engineered safety feature actuation channels less than the minimum channels operable requirement, fuel movement within the spent fuel pool or crane operation with loads over the spent fuel pool may proceed provided the fuel handling building exhaust air filtration system is in operation and discharging through at least one train of HEPA filters and charcoal absorbers (emergency mode). On October 19, 2005, during fuel movement, Train C fuel handling building HVAC engineered safety feature actuation relays became inoperable due to Electrical Bus E2C11 being de-energized for planned maintenance and the fuel handling building exhaust air filtration system not being placed in emergency mode. These relays require direct current power to automatically place the fuel handling building HVAC in the emergency mode and would not have actuated with Bus E2C11 de-energized. The bus was de-energized for approximately 3 hours. This finding was of very low safety significance due to the short duration of the condition and that the ability to manually align to the emergency mode was not compromised. The details of this event and the NRC's subsequent issuance of a licensee-identified Green NCV are further discussed in Section 4OA7 of NRC Inspection Report 05000498;05000499/2005005. The corrective actions implemented in response to this event were documented in accordance with the licensee's corrective action program in CR 05-13732. No additional issues were identified by the inspectors. This LER is closed.

.4 Nonvital Inverter Fire in Vital Switchgear Room B on March 23, 2006

The inspectors reviewed the licensee's response to a fire in a nonvital inverter in vital switchgear Room B. The first indication of an inverter problem was a control board alarm followed by a switchgear room fire alarm. The licensee initially sent a control room operator and the fire brigade leader to the room. Two additional fire brigade members responded when they heard the radio conversation between the control room and fire brigade leader.

When the fire brigade leader identified that there was an actual fire in inverter Transformer T1, he reported it into the control room. The shift supervisor activated the fire brigade and requested notification of offsite fire departments and the local law enforcement agency. Two fire departments and a sheriff's deputy responded but did not enter the protected area.

The fire brigade applied CO_2 to the fire. However, since the transformer was extremely hot, the fire would reflash when the CO_2 was not being applied. Following deenergization of the inverter, the fire brigade applied water to the transformer to cool it off. The fire did not reflash again.

The total time from identification to extinguishing the fire was approximately 12 minutes. The licensee was not required to declare an unusual event since the fire lasted less than 15 minutes. The inspectors did not identify any issues or concerns with the licensee's response.

4OA5 Other

.1 Temporary Instruction 2515/163, Operational Readiness of Offsite Power

a. Inspection Scope

Temporary Instruction 2515/163 requires confirmation of the operational readiness of offsite power systems in accordance with NRC requirements prescribed in General Design Criteria 17, “Electric power systems”; Plant Technical Specifications for offsite power systems; 10 CFR 50.63, “Loss of all alternating current power;” and 10 CFR 50.65(a)(4), “Requirements for monitoring the effectiveness of maintenance at nuclear power plants.” The inspectors reviewed licensee response to the latest additional inspection requirements and the following documents were reviewed:

- OPOP04-AE-0005, “Offsite Power System Degraded Voltage,” Revision 0.
- Technical Specification Limiting Condition for Operation Action 3.8.1.1.e
- OPOP01-ZO-0006, “Extended Allowed Outage Time,” Revision 13
- OPOP01-ZG-002, “STP Coordinator,” Revision 1
- OPOP04-AE-0003, “Loss of Power to one or more 13.8kV Standby Bus,” Revision 6
- OPOP04-AE-0004, “Loss of Power to one or more 4160 ESF Bus,” Revision 10
- ERCOT Operating Guide Section 2.10, “System Voltage Profile,” May 1, 2005
- ERCOT Operating Guide Section 4, “Emergency Operation,” September 1, 2004
- NRR Safety Evaluation of Revised Blackout Position, dated July 24, 1995
- OPGP03-ZO-0045, “CenterPoint Energy Real Time Operations Emergency Operations Plan,” Revision 1
- South Texas Project Interconnection Agreement (Reliant Energy, CP&L, San Antonio, and Austin /STP), dated August 15, 2002

b. Findings

No findings of significance were identified.

.2 (Closed) URI 05000498;499/2005006-02: Inadequate Motor-Operated Valve Operation Method

Introduction: The inspectors identified an NCV of 10 CFR Part 50, Appendix R, Section III.L.3, in that the method used to position motor-operated valves (“hot-sticking”) following a fire in the control room was not independent of the fire area. Specifically, a portion of each valve control circuit was located in the control room. A fire affecting those circuits could result in maloperation or overthrusting of the valves.

Description: The licensee utilized a motor-operated valve repositioning method called "hot-sticking." Operators repositioned or checked the position of a given valve by pushing in either the open or close contactor at the motor control center breaker for the valve. Operators were trained that a contactor will "suck in" if the valve is out of its required position. Then the valve will travel to the requested position and the contactor will "pop out." For valves in the required position, the contactor will immediately "pop out." The inspectors identified a concern with this method because the method utilizes circuits that are not independent of the fire area. If the control room circuit fails or hot shorts, the method will not work as intended. The reliance on circuits that are within the fire area is not consistent with 10 CFR Part 50, Appendix R, Section III.L.3, which states, in part, ". . . the alternative shutdown capability shall be independent of the specific fire area(s)." The failure modes are described below:

Open Circuit: A control circuit failure (open circuit) could result in improper indication to operators when hot-sticking valves. In this instance, following a hot-sticking attempt, the MOV contractor would immediately "pop out." Operators were trained that this response indicates that: "the valve is already in the required position." In reality, the valve should be in the opposite position.

Hot Short: A hot short could cause a valve to start repositioning to an inappropriate position after operators had performed the hot-sticking method for the valve. Since an operator opens the breaker after repositioning the valve, the valve would still be able to reposition during the few seconds before the operator opened the breaker. Some of the valves had very short stroke times (about 10 seconds).

Overthrust/Torque: In the case where the necessary control room circuits are undamaged and a valve is in its required position, valves and actuators can be overthrust and overtorqued well in excess of manufacturers ratings. The hot-sticking method drives the valves into its seats with locked rotor torque. The licensee performed a detailed analysis to evaluate the impact on the MOVs. The licensee concluded that, while catastrophic valve/actuator failure is not expected, the stress to some valve components would exceed the yield point.

As an initial corrective action, the licensee trained the plant operators to ensure that they understood the vulnerabilities associated with the hot-sticking method. The licensee verified that adequate indication was available in all cases to ensure that maloperation of valves could be promptly identified and corrected.

Analysis: The failure to ensure that all circuits relied on for safe shutdown in response to a control room fire was free of the fire area was a performance deficiency. The issue was more than minor because it affected the reliability objective of the Equipment Performance attribute under the Mitigating Systems Cornerstone. Specifically, MOVs that are relied upon to achieve postfire safe shutdown were less reliable because parts of their control circuits could be damaged by the fire. A Senior Reactor Analyst evaluated the safety significance of this issue. The frequency of fires that require evacuation of the control room and use of the remote shutdown panels at South Texas Project are 7E-6/yr. In the base case, it is assumed that the conditional core damage probability of a control room evacuation is 0.1. That is, approximately 10 percent of control room evacuations are assumed to result in core damage. Therefore, to obtain a delta-core-damage-frequency increase of greater than 1E-6/yr, the conditional core damage probability of a control room evacuation, given the performance deficiency, would have to increase to at least 0.24.

The Senior Reactor Analyst noted that, in response to a control room fire, the licensee would attempt to recover all three trains of safe shutdown equipment rather than just the one credited train. Further, most valves were already in their required positions, which negated the need for repositioning. Finally, the licensee had valve position indication at the remote shutdown panel for all affected valves. Accordingly, the operators could quickly troubleshoot and address a mispositioned valve. Therefore, the Senior Reactor Analyst concluded that the increased risk from the hot-sticking method was insufficient to cause more than a very small change in the conditional core damage probability. Also, the implications of large early release would not be relevant to a risk increase in this instance. Considering the above, the Senior Reactor Analyst evaluated the safety significance of this finding using Manual Chapter 0609, “Significance Determination Process,” Appendix F, and determined that the finding constituted a low level of degradation for postfire safe shutdown equipment. As such, the finding was of very low safety significance.

Enforcement: 10 CFR Part 50, Appendix R, Section III.L.3, states, in part, ". . . the alternative shutdown capability shall be independent of the specific fire area(s).” Contrary to the above, approximately 25 MOVs utilized for mitigation of a fire in the control room, the licensee specified a valve repositioning method that relied upon circuits that were not independent of the fire area. Because this issue is of very low safety significance and has been entered into the corrective action program as CR 05-8004, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000498;499/2006002-04).

.3 (Closed) URI 05000498;499/2005006-03: Inadequate Fire Protection Alternate Shutdown Analysis

Introduction: The inspectors identified a noncited violation of 10 CFR Part 50, Appendix R, Section III.L.1, because the thermohydraulic analysis was inconsistent with actions allowed in the South Texas Project licensing basis for a control room evacuation. The inconsistencies affected the control room fire timeline that operators must meet in order to successfully accomplish safe shutdown. Specifically, the analysis inappropriately allowed four additional manual actions to be performed from the control room while the license basis allowed only one manual action to be performed prior to evacuating the control room. The other four manual actions are required to be performed in the field.

Description: The inspectors performed a walkdown with plant operators to verify that alternate shutdown actions could be performed within the time limits derived by the thermohydraulic analysis.

The inspectors identified that Calculation NC-7079, “Fire Hazards Analysis,” Revision 1 (thermohydraulic analysis), contained inappropriate assumptions. For alternate shutdown outside the control room (control room fire), Fire Hazards Analysis Report, Section 2.4.4, “Alternate Shutdown,” credits tripping the reactor from the control room and nothing more. Licensee Calculation NC-7079 assumed that the following additional actions would be accomplished prior to exiting the control room:

- Isolating main steam isolation valves
- Isolating feedwater
- Securing charging
- Isolating letdown

Since the licensee had not obtained NRC approval for the deviations to the licensing basis, crediting performance of the additional actions and using the timeline for completion of the actions from the control room was inappropriate. Further, if the licensee completed the actions outside the control room, there would be an impact on the thermohydraulic analysis results and the associated timeline. Specifically, performing these actions inside the control room ensured that the reactor coolant system process variables remained within those values predicted for a loss of normal ac power, as required by 10 CFR Part 50, Appendix R, Section III.L.1. For a loss of normal ac power at South Texas Project, the predicted pressurizer level (a reactor coolant system process variable) remains well within the indicating range.

The NRC Enforcement Manual, Section 8.1.7, states, in part: “Failure to have an adequate written evaluation available for an area in which Appendix R compliance is not apparent will be taken as an indication that the area does not comply with NRC requirements . . . .”

Analysis. The failure to have an adequate written evaluation available for a control room fire scenario was a performance deficiency. This issue was more than minor because it affected the Mitigating Systems Cornerstone attributes of protection from external factors (fire). The inadequate analysis overestimated the amount of time available when accomplishing shutdown actions and, during walkdowns, the inspectors could not verify compliance with the requirements. An NRC Senior Reactor Analyst evaluated this issue. The frequency of fires that require evacuation of the control room and use of the remote shutdown panels at South Texas Project is 7E-6/yr. In the base case, it is assumed that the conditional core damage probability of a control room evacuation is 0.1. That is, approximately 10 percent of control room evacuations are assumed to result in core damage. Therefore, to obtain a delta-core-damage-frequency increase of greater than 1E-6/yr., the conditional core damage probability of a control room evacuation, given the performance deficiency, would have to increase to at least approximately 0.24.

The Senior Reactor Analyst noted that the additional actions that operators would take prior to evacuation of the control room would cause a delay of approximately 90 seconds beyond the times assumed in the analyzed recovery action timeline. The Senior Reactor Analyst determined that a delay of this magnitude was insufficient to cause more than a very small change in the conditional core damage probability. Also, the implications of large early release would not be relevant to a risk increase in this instance. Based on the above, the Senior Reactor Analyst evaluated the safety significance of this finding using Manual Chapter 0609, "Significance Determination Process," Appendix F, and determined that the finding constituted a low level of degradation for postfire safe shutdown analysis. Accordingly, the finding was of very low safety significance.

Enforcement: 10 CFR Part 50, Appendix R, Section III.L.1, requires that reactor coolant system process variables be maintained within those predicted for a loss of normal ac power. License Condition 2.E specifies, "STPNOC shall implement and maintain in effect all provisions of the approved fire protection program as described in the . . . Fire Hazards Analysis Report." The Fire Hazards Analysis Report, Section 2.4.4, "Alternate Shutdown," credits tripping the reactor from the control room and nothing more. The licensee used Calculation NC-7079 to demonstrate compliance with 10 CFR Part 50, Appendix R, Section III.L.1. Contrary to the above, Calculation NC-7079 inappropriately credited several additional actions from the control room, including main steam isolation valve closure, feedwater isolation, securing charging, and isolating letdown. Because this issue is of very low safety significance and has been entered into the corrective action program as CR 05-8507, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000498;499/2006002-05).

4OA6 Meetings, Including Exit

The results of the ALARA inspection were presented to Mr. Gary Parkey, Vice President of Generation, and Mr. Thomas Jordan, Vice President of Engineering, and other members of the staff, on February 9, 2006.

On February 27, 2006, the inspector conducted a telephonic exit meeting to present the emergency preparedness inspection results to Mr. A. Morgan, Supervisor, Emergency Planning, who acknowledged the findings.

On April 10, 2006, the inspector conducted a telephonic exit meeting to present the electrical URI inspection results to Mr. K. Taplett, Licensing Staff Engineer, who acknowledged the findings.

The results of the resident inspection were presented to Mr. Gary Parkey, Vice President of Generation, and other members of licensee management on April 13, 2006.

During each exit meeting, the inspectors asked the licensee representatives 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 safety 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 NCVs.

Technical Specification 3.5.2 requires, in part, that, with only two of three required essential cooling water loops operable, three loops be restored to operable within 7 days or be in at least hot standby within 6 hours. Contrary to this, Unit 2 continued to operate at 100 percent power while ECCS Train 2C was inoperable for an indeterminate time greater than 7 days due to missing T-drains in MOV Motor 2-SI-0019C. The licensee entered the performance deficiency into their corrective action program as CR 05-13606 for resolution. This finding is of very low safety significance because of the availability of two other trains.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

T. Bowman, Manager, Operations
W. Bullard, Manager, Health Physics
K. Coates, Manager, Maintenance
J. Crenshaw, Manager, Work Management
T. Frawley, Manager, Performance Improvement
R. Gangluff, Manager Chemistry
E. Halpin, VP Oversight
S. Head, Manager, Licensing
D. Hubenak, Supervisor ALARA/Planning, Health Physics
T. Jordan, Vice President, Engineering and Technical
J. Jump, Manager, Process Improvement Leadership Team
D. Leazar, Manager, Nuclear Fuels and Analysis
W. Mookhoek, Senior Engineer, Quality and Licensing
J. Myers, ALARA Specialist, Health Physics
G. Parkey, Vice President, Generation
D. Rencurrel, Manager, Plant Engineering
D. Stillwell, Supervisor, Configuration Control and Analysis
D. Swett, Supervisor CAP & Assessments, Health Physics
D. Towler, Manager, Quality
T. Walker, Manager, Quality

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Open

05000498/2006002-01	URI	DGs Potentially Inoperable for Greater Than the TS Allowed Outage Time (AOT) (Section 1R22)
05000499/2006002-02	NCV	Failure to Correctly Install Temporary Shielding (Section 2OS2)
05000499/2006002-03	NCV	Inadvertent Boration (Section 4OA2)
05000498;499/2006002-04	NCV	Inadequate Motor-Operated Valve Operation Method (Section 4OA5)
05000498;499/2006002-05	NCV	Inadequate Alternate Shutdown Analysis (Section 4OA5)

Closed

05000499/2006002-02	NCV	Failure to Correctly Install Temporary Shielding (Section 2OS2)
05000499/2006002-03	NCV	Inadvertent Boration (Section 4OA3)
05000498;499/2006002-04	NCV	Inadequate Motor-Operated Valve Operation Method (Section 4OA5)
05000498;499/2006002-05	NCV	Inadequate Alternate Shutdown Analysis (Section 4OA5)
05000499/2005-006	LER	Unit 2 Missing Motor Operated Valve T-Drains (Section 4OA3)
05000499/2005-005	LER	125 Vdc Switchboard De-energized Without Placing FHB HVAC in Emergency Recirculation Mode (Section 4OA3)
05000498;499/2005006-02	URI	Inadequate Motor-Operated Valve Operation Method (Section 4OA5)
05000498;499/2005006-03	URI	Inadequate Alternate Shutdown Analysis (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

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

Section 71111.12: Maintenance Implementation (Condition Records reviewed)

95-1718	96-10465	97-12499	97-13259	99-12426	00-792
00-3152	02-5184	03-18541	04-4820	04-5159	04-5328
04-5425	05-3268	05-6342			

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

Corrective Action Documents (Condition Reports)

05-12074, 05-12202, 05-12704, 05-14212, 05-14506, 05-15598

Procedures

0PGP03-ZR-0051, Radiological Access and Work Controls, Revision 21
0PRP04-ZR-0015, Radiological Posting and Warning Devices, Revision 20
0PRP01-ZR-0005, Access Control Point Management, Revision 12

Section 2OS2: ALARA Planning and Controls (71121.02)

Corrective Action Documents (Condition Reports)

05-12304, 05-13034, 05-13524, 05-13527, 05-13899, 05-14200, 05-14289, 05-14302, 05-14343, 05-15049, 06-217, 06-867

Procedures

0PGP03-ZR-0050, Radiation Protection Program, Revision 8
0PGP03-ZR-0052, ALARA Program, Revision 9
0PQP01-ZA-0001, Internal Audits, Revision 7
0PRP07-ZR-0001, ALARA Engineering and Procedure Review, Revision 2
0PRP07-ZR-0004, Shielding, Revision 10
0PRP07-ZR-0007, Radiation Shielding Verification and Trending Surveys, Revision 5
0PRP07-ZR-0009, Performance of High Exposure Work, Revision 25
0PRP07-ZR-0010, Radiation Work Permits, Revision 16
0PRP07-ZR-0011, Radiological Work ALARA Reviews, Revision 7

Audits and Self-Assessments

Radiation Protection Corrective Action Program and Human Performance Monitoring Reports, June-November 2005

Quality Monitoring Reports: MN-05-2-9969, MN-05-2-10470, and MN-05-2-10553

ALARA Review Packages

05-0055-02	2RE11 Non-Rapid Refueling
05-0055-04	2RE11 In-Service Inspections and Flow Accelerated Corrosion Inspections
05-0055-06	2RE11 Replace Body to Bonnet Gaskets, Rework Valves, and Seal Weld
05-0055-12	2RE11 Reactor Head Vent Valves
05-0055-17	2RE11 Replace Packing 2R162XRH0019B (B2SIMOV0019B) in Mode 2

Radiation Work Permits

2006-0-0018 Perform Spent Resin Movement, Revision 0

2005-2-0249 2RE11-Maintenance, Walkdowns, Inspections and Tours, Revision 0

2005-2-0251 2RE11-Radioactive Material Control, Revision 0

2005-2-0253	2RE11-Install/Remove Scaffolding and Insulation, Revision 0
2005-2-0273	Decon of Reactor Cavity, Revision 0
2005-2-0290	Reactor Head Disassembly/Reassembly Mechanical Support-For Work In and Around Reactor Cavity, Revision 1
2005-2-0291	Under Reactor Head Mechanical Support, Inspections, and Walkdowns, Revision 1
2005-2-0292	O-Ring Groove Inspections, Revision 0
2005-2-0293	Reactor Head Disassembly/Reassembly-Clean Lower O-Ring Groove and Seating Surfaces, Revision 1
2005-2-0294	Reactor Head Disassembly/Reassembly-Work Inside Reactor Head Shroud Doors, Revision 0
2005-2-0296	Reactor Head Disassembly/Reassembly-Guide CRDs to Funnels, Revision 1
2005-2-0297	Reactor Head Disassembly/Reassembly-Clean Upper O-Ring Groove and Seating Surfaces, Revision 0

Work Authorization Numbers

4312	Perform decontamination of reactor cavity, lower internals storage area, in containment storage area and Reactor Containment Building transfer canal.
260528	Install and remove scaffolding to support Reactor Coolant Pump Preventative Maintenance
288102	Reactor Head Disassembly/Reassembly
290960	Scaffolding Support for Flow Accelerated Corrosion
290964	Scaffolding Support for In-Service Inspections
290966	Insulation Support for In-Service Inspections
292119	Install and remove scaffolding Reactor Coolant Pump Platform Access Modification

Section 4OA1: Performance Indicator Verification (71151)

Corrective Action Documents (Condition Reports)

05-12240

Procedures

0PGP05-ZN-0007, Preparation and Submittal of NRC Performance Indicators, Revision 2

LIST OF ACRONYMS

ALARA	as low as is reasonably achievable
CFR	*Code of Federal Regulations*
CR	condition report
ECCS	emergency core cooling system
HEPA	high-efficiency particulate air
HVAC	heating, ventilation, and air conditioning
LER	licensee event report
MOV	motor operator valve
NCV	noncited violation
NEI	Nuclear Energy Institute
SBDG	standby diesel generator
TSR	temporary shielding request
URI	unresolved item