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

July 27, 2005

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

SUBJECT: SOUTH TEXAS PROJECT ELECTRIC GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 05000498/2005003 AND 05000499/2005003

Dear Mr. Sheppard:

On June 26, 2005, the US 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 July 7, 2005, 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 four 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. 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. Additionally, one licensee-identified violation which was determined to be of very low safety significance is listed in this report. 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/

Thomas R. Farnholtz, Chief
Project Branch A
Division of Reactor Projects

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

Enclosure:
NRC Inspection Report 05000498/2005003
and 05000499/2005003
w/Attachment: Supplemental Information

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RIV:RI:DRP/A	SRI:DRP/A	SRI:DRP/A	PE:DRP/A	C:DRS/PSB
GLGuerra	JLTaylor	JCruz	MABrown	MPShannon
E - TRFarnholtz	**T-TRFarnholtz**	**E-TRFarnholtz**	**/RA/**	**/RA/**
7/27/05	7/26/05	7/26/05	7/27/05	7/26/05

C:DRS/OB	C:DRS/EB1	C:DRS/EB2	C:DRP/A
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7/26/05	7/26/05	7/26/05	7/27/05

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	50-498, 50-499
Licenses:	NPF-76 NPF-80
Report No:	05000498/2005003 05000499/2005003
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:	April 8 through June 26, 2005
Inspectors:	J. Cruz, Senior Resident Inspector J. Taylor, Resident Inspector G. Guerra, Resident Inspector T. Brown, Project Engineer B. Baca, Health Physicist P. Elkmann, Emergency Preparedness Inspector J. Larsen, Physical Security Inspector J. Keeton, Consultant Inspector
Approved By:	T. Farnholtz, Chief Project Branch A Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000498/2005003, 05000499/2005003; 04/08/05 - 06/26/05; South Texas Project Electric Generating Station; Units 1 & 2; Integrated Resident Report, Problem Identification and Resolution, Fitness for Duty.

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 are indicated by the severity level of the applicable violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a noncited violation of Technical Specification 6.8.1.a and Regulatory Guide 1.33, Appendix A, Item 8.b.(1).i, "Emergency Core Cooling Tests," for inadequate procedures that resulted in a letdown pressure relief valve opening during the performance of Plant Surveillance Procedure 0PSP03-RH-0009, "Residual Heat Removal System Valve Operability Test," Revision 5, on March 16, 2004, and again during performance of preventive maintenance procedure PM IC-2-89001568 on May 2, 2005.

  This finding was a performance deficiency because it had the actual impact of lifting a relief valve and, therefore, is associated with an increase in the likelihood of an initiating event. The finding was of greater than minor significance since it was associated with the cornerstone attribute of Initiating Events and affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. The finding was only of very low safety significance because, assuming worst case degradation, the lifted relief valve would not have resulted in exceeding the Technical Specification limit for identified reactor coolant system leakage. This issue also involved problem identification and resolution crosscutting aspects in the area of prioritization and evaluation. Additionally, the event had crosscutting aspects in the area of human performance related to procedural adequacy and equipment knowledge (Section 4OA2.1).

Cornerstone: Barrier Integrity

- Green. A self-revealing noncited violation of License Condition 2.C(1) of Facility Operating License NPF-76 was identified. License Condition 2.C(1) of Facility Operating License NPF-76 requires, in part, that South Texas Project Unit 1 operate at reactor core power levels not in excess of 3,853 megawatts thermal. It was determined that the reactor thermal output instruments provided nonconservative data to the reactor power calculation. This resulted in the 8-hour power average routinely being in excess of the licensed thermal power limit of 3,853 megawatts thermal between April 15 and May 19, 2005.

This finding was a performance deficiency because the facility was not operated in accordance with the conditions of the South Texas Project license. The finding was more than minor because it was associated with the Barrier Integrity cornerstone and the protection of the fuel cladding barrier attribute. The finding was only of very low safety significance because the small increase in power above the licensed limit could be accommodated by the available margins in the safety analysis and, therefore, did not significantly degrade plant safety. This issue involved problem identification and resolution crosscutting aspects associated with identifying and evaluating conditions adverse to quality (Section 4OA2.2).

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, associated with the licensee's failure to assure that applicable regulatory requirements and the design basis for the containment radiation gas monitors were correctly translated into the reactor containment building radiation monitor setpoints. This deficiency resulted in the radiation monitors being incapable of performing the design basis function to detecting a one gallon per minute reactor coolant system leak within one hour in accordance with the licensee's commitment to Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

  This finding was a performance deficiency because the reactor containment building radiation monitor was not capable of performing the design basis function for an extended period of time. The finding was of greater than minor significance because the failure to alarm by the containment radiation monitor resulted in potential impact on reactor safety and adversely affected the reactor coolant leakage performance attribute of the Barrier Integrity cornerstone. The finding was only of very low safety significance because other methods of reactor coolant system leak detection were available to the licensee and operators responded to the trending in the volume control tank level and then noted the rising trend recorded with the particulate radiation monitor. The failure of the radiation monitor to alarm within one hour did not contribute to an increase in core damage sequences when evaluated using the Significance Determination Process Phase 2 worksheets (Section 4OA5).

Cornerstone: Physical Protection

- Green. The inspectors identified a noncited violation of 10 CFR Parts 26.20 and 26.27 (b)(1) and South Texas Project Policy 502. Specifically, an individual whose fitness was in question was allowed to return to duty prior to determining whether he was fit to safely and competently perform his job function. The licensee initiated a corrective action document to address this failure.

This finding is greater than minor because it affects the Physical Protection cornerstone attribute associated with access authorization systems. When this finding was processed through the interim physical protection significance determination process, it was determined to be a finding of very low significance, although there was no malevolent act and there were no greater than two similar findings in four quarters. This finding had crosscutting aspects associated with human performance, because of the licensee's failure to follow their procedures (Section 3PP8).

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 action tracking numbers are listed in Section 4OA7.

REPORT DETAILS

Summary of Plant Status

Unit 1 began the inspection period shut down for scheduled Refueling Outage 1RE12. The unit was restarted on April 15 and achieved full power shortly thereafter. The unit operated at essentially 100 percent power for the remainder of 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 hurricanes. 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, 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.

- (Unit 1 and 2) Essential Cooling Water Systems, June 9
- (Unit 1 and 2) Standby Diesel Generators, June 9

The inspectors completed two samples.

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdown

a. Inspection Scope

The inspectors: (1) walked down portions of the two 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 2) The inspectors verified the condition of the mechanical auxiliary building heating, ventilation, and air conditioning systems. The walkdown was performed with the system engineer to access cooling coils, inspect tornado dampers, and verify the proper equipment lineup. The inspectors also examined component condition, April 20

- (Unit 1) The inspectors verified the alignment and condition of Emergency Diesel Generator 11 while Emergency Diesel Generator 12 was out of service. The inspectors verified that the diesel generator system equipment and control board were aligned in accordance with Plant Operating Procedure 0POP02-DG-0001, "Emergency Diesel Generator 11," Revision 39, Checklist 1 and Lineup 2, May 17

The inspectors completed two samples.

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.

- (Common) Essential cooling water intake structure (Fire Zones Z600-605), May 11

- (Unit 1) Trains A and D, auxiliary feedwater pump rooms (Fire Zones Z400, Z401, Z404, Z405, Z408 and Z409), May 12

- (Unit 1) Train B cable spreading Room 302 (Fire Zone ZO47), May 18

- (Unit 2) Train B cable spreading Room 302 (Fire Zone ZO47), May 18
- (Common) Main switchyard, June 9
- (Common) Central alarm station and secondary alarm station (Fire Area 004, zone 055), June 16

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 June 23 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 Unit 2 turbine lube oil filtration skid. 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 seasonal susceptibilities involving external flooding; (2) reviewed the corrective action program to determine if the licensee identified and corrected flooding problems; (3) inspected underground bunkers/manholes to verify the adequacy of: (a) sump pumps, (b) level alarm circuits, (c) cable splices subject to submergence, and (d) drainage for bunkers/manholes; (4) verified that operator actions for coping with flooding can reasonably achieve the desired outcomes; and (5) 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.

- C (Common) the inspectors reviewed the flood analysis documentation and calculations completed with the installation of the vehicle barrier around the protected area. The inspectors also conducted a walkdown of the external doors and penetrations identified in Plant Operations Procedure 0POP04-ZO-0002, "Natural or Destructive Phenomena Guidelines," Revision 30, May 9

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 May 16 to identify deficiencies and discrepancies in the training and to assess operator performance and the evaluator's critique. The training scenario involved a failure of Incore Instrument NI-41, the failure of a steam generator feedwater pump, a steam generator tube rupture, and a loss of offsite power.

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 three 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.

- (Unit 2) Power-operated relief Valve (PORV) 2D corrective maintenance issues resulted in PORV 2D being inoperable in excess of 7 hours, as documented in Condition Report 04-9508, May 10

- (Common) main steam isolation valves evaluated as operable with stuck open internal pilot in accordance with Condition Report Engineering Evaluation 02-19149-3 when main steam isolation valves are closed, as discussed in Condition Report 04-7135, May 11

- (Common) Maintenance associated with the “tin whisker” phenomena identified in the solid state protection system, June 16 (Condition reports reviewed are listed in the attachment to this report.)

The inspectors completed three 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 five 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 recognized, and/or entered as applicable, the appropriate licensee-established risk category according to the risk assessment results and licensee procedures, and (4) that the licensee identified and corrected problems related to maintenance risk assessments.

- (Common) Evaluation of risk for main switchyard maintenance during Refueling Outage 1RE12 (Evaluation 1355), April 15

- (Common) Evaluation of risk for main switch yard maintenance during extended Refueling Outage 1RE12 (Evaluation 1358), April 18

- (Unit 1) Evaluation of risk for troubleshooting and calibration of reactor coolant reference temperature (Evaluation 1364), April 18

- (Unit 1) Evaluation of risk for replacing a reverse power relay on main control board Panel 10 (Evaluation 1368), April 28

- (Unit 1) Evaluation of risk for replacing degraded relay wiring on 4160 volt engineered safety features Switchgear E1B (Evaluation 1366), April 28

The inspectors completed five 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 Risk Assessment and Emergent Work Control problems.

- (Unit 1) Condition Report Engineering Evaluation 05-3589-8 for operability/reportability of Steam Generators 1A, 1B, and 1C following a foreign materials inspection, April 13

- (Unit 1) Condition Report Engineering Evaluation 05-4114-8 for operability/reportability of Steam Generator 1D following a foreign materials inspection, April 14

The inspectors completed two 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; (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) Preparation and conduct of midloop operation during the Refueling Outage 1RE12 extension for steam generator foreign material inspection, April 8

- (Unit 1) Turbine testing in preparation for power ascension at the conclusion of Refueling Outage 1RE12, April 15

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors: (1) reviewed plants 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 1) Loop 3 average temperature fluctuation less than the other loops (CR 05-6247), April 26

- (Unit 1) Pressurizer pressure Channel P-456 material condition requiring a change in calibration procedure (CR 05-6511), May 3

- (Unit 2) An inappropriate spare relay wired into the circuit for a main steam isolation valve (CR 05-6588), May 4

- (Unit 1) Steam generator PORV 1A would not close in manual during a loss of power test (CR 05-6596), May 4

- (Unit 1) Essential cooling water Train B localized erosion damage on the inside surface of aluminum bronze piping located on the discharge side of throttle Valve EW-0064 (CR 05-7071-1), June 14

- (Unit 1) Essential cooling water Train C localized erosion damage on inside surface and through-wall cracks identified in the aluminum bronze piping located on the discharge side of throttle Valve EW-0101 (CR 05-7303-1), June 14

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

Selected Operator Workarounds

a. Inspection Scope

The inspectors reviewed the two below listed operator workarounds to: (1) determine if the functional capability of the system or human reliability in responding to an initiating event is affected; (2) evaluate the effect of the operator workaround on the operator's ability to implement abnormal or emergency operating procedures; and (3) verify that the licensee has identified and implemented appropriate corrective actions associated with operator workarounds.

- (Unit 2) Intersystem leakage between Train C component cooling water and reactor containment building chilled water systems, April 12

- (Unit 2) Effect on plant operator response times during events as impacted by the required dispatch of a plant operator to verify acceptable tank level in response to spurious essential chillwater expansion tank level alarms received in the control room (CR 05-4813), June 15

The inspectors completed two 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, 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, modification design assumptions were appropriate, 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 the replacement of Class 1E 7.5 kV inverters. Additionally, the inspectors reviewed the specific documents associated with the modification. These included Design Change Package 04-1238–40, Supplement 0-5; the 10 CFR 50.59 screening documentation for Design Change Package 04-1238–40, Supplement 1; Calculation EC0-5002, Load Calculation; and Work Order 438021, April 26

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 seven 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 1) Plant Maintenance Procedure 0PMP08-RS-0001, "Control Rod Drive Mechanism Timing Test," Revision 4, review of postmaintenance testing following cable replacement on control Rod B6, April 19

- (Unit 2) Work Order 440093, "ECW Pump 2A Traveling Screen," postmaintenance testing performed in accordance with work instructions associated with planned maintenance, April 20

- (Unit 1) Plant Operating Procedure 0POP02-FW-0002, “S.G.F.P. Turbine,” Revision 31, postmaintenance testing associated with the start of steam generator feedwater Pump 12, May 14

- (Unit 1) Work Order 449641, "Generator Circuit Breaker Air Compressor A," postmaintenance testing performed in accordance with work instructions following compressor replacement, June 9

- (Unit 1) Plant Maintenance Procedure 0PMP04-ZH-0002, “Prefilter Removal and Replacement,” Revision 6, review of postmaintenance testing following control room air handling unit filter replacement, June 13

- (Unit 1) Plant Maintenance Procedure 0PMP08-ZI-0025, “Pneumatic/Spring Control Valve or Damper Calibration,” Revision 20, review of postmaintenance testing following electrical auxiliary building main area return Fan 11B discharge to exhaust riser smoke purge damper maintenance, June 15

- (Unit 2) Plant Operations Procedure 0POP02-CR-0001, “Main Condenser Air Removal,” Revision 24, review of postmaintenance testing following sight glass and lubricator belt replacement on condenser air removal system Pump 23, June 16

The inspectors completed seven samples.

b. Findings

No findings of significance were identified.

1R20 Refueling Outage (71111.20)

a. Inspection Scope

During April 8-17, the inspectors reviewed the following risk significant refueling items or outage activities to verify defense in depth commensurate with the outage risk control plan and compliance with the Technical Specifications: (1) the risk control plan; (2) tagging/clearance activities; (3) reactor coolant system (RCS) instrumentation; (4) electrical power; (5) decay heat removal; (6) spent fuel pool cooling; (6) inventory control; (7) reactivity control; (8) containment closure; (9) reduced inventory or midloop conditions; (10) refueling activities; (11) heatup and coldown activities; and (12) licensee identification and implementation of appropriate corrective actions associated with refueling and outage activities.

The inspectors completed one sample.

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 four 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) engineering evaluations, root causes, and bases for returning tested SSC's 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-SP-0010B, "Train B ESF Load Sequencer Manual Local Test," Revision 16, May 14
- (Unit 1) Plant Surveillance Procedure 0PSP03-SP-0008B, "SSPS Train B Slave Relay Test (Outputs Blocked)," Revision 14, May 15
- (Unit 1) Plant Surveillance Procedure 0PSP03-SP-0007B, "SSPS Actuation Train B Master Relay Test," Revision 16, May 16
- (Unit 1) Plant Surveillance Procedure 0PSP03-AF-0007, "Auxiliary Feedwater Pump 14 Inservice Test," Revision 30, June 6

The inspectors completed four samples.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report, plant drawings, procedure requirements, and Technical Specifications to ensure that the two 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 SSC's was 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 1) T1-04-12214-1, "Place digital Ashcroft vacuum gauge in parallel with existing analog gauge on Train A control room air sampler," April 11

- (Unit 1) T1-04-12214-2, "Place digital CECOMP vacuum gauge in parallel with existing analog gauge on Train C control room air sampler," April 11

The inspectors completed two samples.

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 an in-office review of Revision ICN 20-3 to the South Texas Project Electric Generating Station Emergency Plan, submitted June 1, 2005. This revision:

- Added initiating conditions to security emergency action levels,
- Corrected the titles of offsite agencies,
- Updated the description of available communications systems,
- Added “shelter in place” to permissible protective action recommendations,
- Updated the description of installed seismic monitors and,
- Added information regarding audits of the emergency preparedness department.

The revision was compared to its previous revision, 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; 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).

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

For the below listed two drills and simulator-based training evolutions 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 Document NEI 99-02 acceptance criteria.

- On May 16, during Crew 1D simulator-based training, the inspectors evaluated the use of emergency action levels and protective action recommendations for accuracy and timeliness, reviewed the scenario sequence and objectives, observed the evaluation critique, and discussed the crew's performance with training instructors and evaluators.

- On June 8, the inspectors observed a drill in which the scenario consisted of a sheered reactor coolant pump shaft causing fuel damage and a steam generator tube rupture. The scenario progressed in a manner which required the emergency response organization to declare and respond to a General Emergency.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

3. SAFEGUARDS

Cornerstone: Physical Protection (PP)

3PP8 Fitness For Duty Program (71130.08)

a. Inspection Scope

The inspector reviewed an investigation conducted by the licensee to determine if management removed from duty individuals who were impaired or whose fitness may have been questionable. This review also included whether these individuals were returned to duty prior to their fitness being determined as fit and safe to perform their job.

b. Findings

Introduction. The inspector identified a violation of 10 CFR Parts 26.20 and 26.27 (b)(1) and South Texas Project Policy 502. Specifically, an individual whose fitness was in question was allowed to return to duty prior to determining whether he was fit to safely and competently perform his job function. The licensee initiated a corrective action document to address this failure.

Description. On February 14, 2005, an individual's protected area access was withdrawn and an interview was conducted regarding fitness for duty issues. During the interview, the individual reported having a problem with a controlled medication and needed to seek help for that problem. The individual was administered a urinalysis test, with "negative" results for substances, provided with information about the site's Employee Assistance Program, and allowed to return to work on February 15, 2005. South Texas Nuclear Operating Company's administrative policy manual (STP-502) required initiating an administrative referral to the Employee Assistance Program prior to the individual returning to work.

Analysis. The failure to administratively refer the individual to the Employee Assistance Program prior to completing a fitness evaluation is a performance deficiency. This finding is greater than minor because it affects the Physical Protection cornerstone attribute associated with access authorization systems. When this finding is processed through the interim physical protection significance determination process, it was determined to be a finding of very low significance, although there was no malevolent act and there were no greater than two similar findings in four quarters. This finding had crosscutting aspects associated with human performance, because of the licensee's failure to follow their procedures.

Enforcement: 10 CFR 26.20 states in part, "Each licensee subject to this part shall establish and implement written policies and procedures designed to meet the general performance objectives and specific requirements of this part." Title 10 CFR 26.27 (b)(1), states, "Impaired workers, or those whose fitness may be questionable, shall be removed from activities within the scope of this part, and may be returned only after determined to be fit to safely and competently perform activities within the scope of this part." Paragraph 6.6.1 of STP Nuclear Operating Company Administrative Policy Manual, Policy: Drugs and Alcohol - Fitness for Duty, STP - 502, Revision 3, stated in part: "An administrative referral is a mandatory referral made by a member of management or Access Program Director to the Employee Assistance Program." Contrary to the above, the licensee failed to follow procedures by not initiating an administrative referral of the employee to the Employee Assistance Program for determining if the individual was fit to safely and competently perform his duty prior to reinstatement of his protected area access. Because the failure to refer the individual to the Employee Assistance Program is of very low safety significance and has been entered into the corrective action program as Condition Report 05-4590, this violation is being treated as a noncited violation (NCV) consistent with Section VI of the NRC Enforcement policy: NCV 05000498;499/2005003-04, Failure to refer an employee to the Employee Assistance Program.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152)

.1 Review of Events Involving the Lifting of a Letdown Pressure Relief Valve

a. Inspection Scope

The inspectors evaluated the effectiveness of the licensee's problem identification and resolution processes regarding events that lifted letdown pressure relief Valve PSV-3100. The events were documented by the licensee in several condition reports. 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 reports against the requirements in the licensee's Corrective Action Program and 10 CFR Part 50, Appendix B.

The inspectors reviewed three condition reports involving the lifting of a chemical and volume control system (CVCS) relief valve. On January 21, 2004, while performing a letdown orifice swap, operators failed to control system pressure and lifted relief Valve PSV-3100 (CR 04-1143). The procedure utilized for orifice swaps contained insufficient information on the operation characteristics of letdown pressure control Valve PCV-135. On March 16, 2004, operators again failed to control system pressure while stroking the low pressure letdown valves on the Train B residual heat removal system, which resulted in relief Valve PSV-3100 opening. Again, the procedure in use, in this case a surveillance procedure to test the residual heat removal system, contained insufficient information on the operation characteristics of letdown pressure control Valve PCV-135. On May 2, 2005, while using the charging bypass control valve for a calibration of charging flow Loop FE-205, operators again failed to control system pressure, resulting in relief Valve PSV-3100 opening. Again, the procedure in use was not appropriate for the circumstances and did not account for the operational characteristics of pressure control Valve PCV-135.

b. Findings and Observations

Introduction. A Green NCV of Technical Specification 6.8.1.a and Regulatory Guide 1.33, Appendix A, Item 8.b.(1).i, "Emergency Core Cooling Tests," was identified for inadequate procedures that resulted in a letdown pressure relief valve opening during the performance of Plant Surveillance Procedure 0PSP03-RH-0009, "Residual Heat Removal System Valve Operability Test," Revision 5, on March 16, 2004, and again during performance of preventive maintenance Procedure PM IC-2-89001568 on May 2, 2005.

Description. On March 16, 2004, CVCS pressure control Valve PCV-135 in Unit 2 was unable to respond quickly enough to prevent letdown pressure safety Valve PSV-3100 from lifting while stroking low pressure letdown Valve RH-MOV-66B. Valve PCV-135 response was slow and pressure in the system lifted the letdown safety Valve PSV-3100, which had a 600 psig lift point, and resulted in a volume of reactor coolant being diverted to the primary relief tank. A caution statement in Plant Surveillance Procedure 0PSP03-RH-0009, "Residual Heat Removal Valve Operability Test," Revision 5, did not give specific actions to take in the event this occurs nor did it mention any effects on the letdown header. The licensee attributed the root cause to the absence of procedural guidance.

On May 2, 2005, during performance of preventive maintenance Procedure PM IC-2-89001568, and while using the charging bypass control valve for a calibration of charging flow Loop FE-205, operators again failed to control system pressure, resulting in relief Valve PSV-3100 opening. Again, the procedure in use did not give specific instructions and did not account for the operational characteristics of pressure control Valve PCV-135. This procedure had been changed from outage to operational performance without an adequate review.

The CVCS is an interfacing system with the RCS. Operators failed to account for the slow operation of control Valve PCV-135 during these events and did not properly control pressure in the chemical and volume control system. As a result, the letdown line safety relief valve opened, diverting RCS inventory to the primary relief tank. This event created the possibility of the relief valve sticking open and causing a small-break loss of coolant condition until the valve could be isolated.

Analysis. This finding was greater than minor because it had the actual impact of lifting a relief valve and, therefore, was associated with an increase in the likelihood of an initiating event. As the finding was associated with the cornerstone attribute of Initiating Events and affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations, it surpassed 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 "Assuming worst case degradation, would the finding result in exceeding the Tech Spec limit for identified RCS leakage?" was determined to be "No" because there was no degradation of the CVCS letdown line relief valve or system that could have either decreased the probability of the relief valve to reseat nor prevent the system isolation valves from functioning, thereby precluding a draindown from the RCS to the primary relief tank. Therefore, the finding was determined to be of very low safety significance.

This finding had crosscutting aspects in the area of prioritization and evaluation in that the extent of condition of the health of the licensee's procedures involving the CVCS letdown system were not evaluated. Also, this event had crosscutting aspects in the area of human performance related to procedural adequacy and equipment knowledge.

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 8.b.(1).i, requires procedures be maintained for testing the emergency core cooling system. Plant Surveillance Procedure 0PSP03-RH-0009, "Residual Heat Removal System Valve Operability Test," Revision 5, was not properly maintained in that it was inadequate because the guidance it provided allowed the letdown relief valve to open. Similarly, Procedure PM IC-2-89001568 was not properly maintained in that it was inadequate because the guidance it provided did not account for the operation aspect of changing from outage to operational performance. The opening of the letdown relief valve increased the risk of an initiating event of an interfacing system small loss of coolant accident. Because this finding was entered into the licensee's Corrective Action Program as Condition Reports 04-3554 and 05-6471 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 05000499/2005003-01, Inadequate Procedures Result in Relief Valve Openings During the Performance of Surveillance Tests.

.2 Unit 1 Exceeding Licensed Thermal Power Limits.

Introduction. A Green self-revealing NCV of License Condition 2.C(1) of Facility Operating License NPF-76 was identified. License Condition 2.C(1) of Facility Operating License NPF-76 requires, in part, that South Texas Project Unit 1 operate at reactor core power levels not in excess of 3,853 megawatts thermal. It was determined that the reactor thermal output instruments provided nonconservative data to the reactor power calculation. This resulted in the 8-hour power average routinely being in excess of the licensed thermal power limit of 3,853 megawatts thermal between April 15 and May 19, 2005.

Description. On May 16, 2005, during maintenance on a motor-operated valve in the component cooling water system, component cooling water was isolated to the excess letdown heat exchanger. The control room received an alarm indicating high excess letdown heat exchanger temperature. Since excess letdown was not in service at the time, a condition report was initiated to determine the cause of the elevated temperature. The licensee’s review identified that check Valve CV-0739 was not seated correctly and RCS leak-by past the check valve was causing the elevated temperature condition. Check Valve CV-0739 was properly seated on May 25 and temperatures returned to the expected value of approximately 100EF.

On May 19, 2005, the licensee determined that the leak-by rate of approximately 4.4 gallons per minute represented approximately 0.4 megawatts thermal that was not being considered in the calculation of thermal power using the reactor thermal output instruments. This unaccounted leak-by resulted in a nonconservative calculation of thermal power. A review of hourly power data for Unit 1 from April 15 to May 19, 2005, demonstrated that, when the additional 0.4 megawatts is added to the calculated power, the 8-hour power average regularly exceeded the licensed thermal power limit of 3,853 megawatts thermal. This small increase in power above the licensed limit could be accommodated by the available margins in the safety analysis and, therefore, did not significantly degrade plant safety.

Analysis. The inspectors determined that operating above the licensed power limits was a performance deficiency warranting a significance evaluation in accordance with Inspection Manual Chapter 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Disposition Screening,” issued on January 14, 2004. The inspectors determined that the finding was more than minor because it impacted the Barrier Integrity Cornerstone attribute of reactor safety and affected the design control attribute of core design analysis. The finding was associated with actions that challenged fuel cladding thermal limits. By operating the reactor at a power level greater than 100 percent thermal power, the licensee failed to provide reasonable assurance that the cladding barrier would protect the public from radionuclide releases caused by accidents or events.

This issue involved problem identification and resolution crosscutting aspects associated with identifying and evaluating conditions adverse to quality. Although the initial indications of the possibility of an overpower condition were available on the control room panel for 31 days (indications of elevated excess letdown heat exchanger temperature while the excess letdown system was not in service), the licensee missed several opportunities to conclude that the excess power condition existed.

The inspectors determined that the finding could be evaluated in accordance with Inspection Manual Chapter 0609, “Significance Determination Process.” The finding was associated with the Barrier Integrity cornerstone and protection of the fuel cladding barrier. Therefore, Phase 1 screening under “RCS Barrier or Fuel Barrier” screened as Green for fuel barrier issues.

Enforcement. The licensed maximum thermal power output for a reactor is defined in each licensee's operating license. License Condition 2.C(1) of Facility Operating License NPF-76 requires, in part, that South Texas Project Unit 1 operate at reactor core power levels not in excess of 3,853 megawatts thermal.

Contrary to the above, South Texas Project Unit 1 failed to operate at reactor core power levels not in excess of 3,853 megawatts thermal during various periods between April 15 and May 19, 2005. However, since the finding was determined to be of very low safety significance and was entered into the licensee's Corrective Action Program as Condition Report 056963, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000498/2005003-02, Unit 1 Exceeding Licensed Thermal Power Limits.

.3 Semiannual Sample Review

a. Inspection Scope

On June 15, 2005, the inspectors completed a semiannual review of licensee internal documents, reports, and performance indicators to identify trends that might indicate the existence of more significant safety issues. The inspector's review nominally considered the 6-month period of January through June 2005, although some examples expanded beyond those dates when the scope of the trend warranted. Corrective actions associated with a sample of the issues identified in the licensee's trend reports were reviewed for adequacy. The inspectors evaluated the licensee's implementation of the corrective action program as specified in licensee Procedure 0PGP03-ZX-0002, "Condition Reporting Process," and 10 CFR Part 50, Appendix B. Documents reviewed by the inspectors included:

- Condition Report Daily Monitor
- System Performance Indicators
- System Health Reports
- Systems Engineering Quick Hitter List
- Quality Assurance Audit Reports
- Selected Work Orders from the 1st and 2nd Quarters of 2005
- South Texas Project Internal Performance Summary Reports

b. Findings and Observations

No findings of significance were identified. However, the inspectors did make the following observations which were shared with licensee management.

- The number of condition reports detailing issues with equipment clearance order creation and implementation appear to be trending toward negative performance.

- A number of reactivity management issues were documented in condition reports during the months of May and June.

.4 Daily Condition Report 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 followup, 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 condition report, 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.

4OA3 Event Followup (71153)

.1 (Closed) Licensee Event Report 0500498/2004-005: Standby Diesel Generator 12 Not Being Declared Inoperable While the Auxiliary Feedwater Pump 12 Breaker Cell Switch Was Being Replaced

On August 18, 2004, the licensee identified that Standby Diesel Generator 12 had been inoperable for approximately 12 hours without entering Technical Specification 3.8.1.1.b or performing the associated engineered safety feature power availability surveillance. The details of this event and the NRC's subsequent issuance of a licensee-identified Green NCV are further discussed in Section 4OA7 of this report. The corrective actions implemented in response to this event were documented in accordance with the licensee's Corrective Action Program in Condition Report 04-11428. No additional issues were identified by the inspectors. This licensee event report is closed.

4OA4 Crosscutting Aspects of Findings

.1 A finding described in Section 4OA2.1 of this report affected the crosscutting area of problem identification and resolution associated with prioritization and evaluation in that the extent of condition of the health of the licensee's procedures involving the CVCS letdown system were not evaluated. Also, this event had crosscutting aspects in the area of human performance related to procedural adequacy and equipment knowledge.

.2 A finding described in Section 4OA2.2 of this report affected the crosscutting area of problem identification and resolution associated with identifying and evaluating conditions adverse to quality. Specifically, following the initial indications of a possible overpower condition, the licensee missed several opportunities to conclude that the overpower condition existed.

.3 A finding described in Section 3PP8 of this report affected the crosscutting area of human performance associated with the licensee's failure to follow procedures.

4OA5 Other

.1 (Closed) Unresolved Item 05000498;499/2004002-02: Reactor Coolant Leakage Detection System Calibration.

a. Inspection Scope

On December 17, 2003, Unit 1 experienced an unidentified RCS leakage in excess of 1 gpm for a duration of 2 hours and 40 minutes, with no indication or alarm from the containment noble gas radioactivity monitor. The containment particulate radioactivity monitor indicated an increased activity, but did not alarm. The inspectors reviewed the applicable licensing documents. The South Texas Project Updated Final Safety Analysis Report stated, in part, that "The particulate and noble gas channels are used as part of the reactor coolant pressure boundary leakage detection system. The sensitivity and response time of this part of the leakage detection system, which is used for monitoring unidentified leakage to the containment, are sufficient to detect an increase in leakage rate of the equivalent of one gal/min within one hour." Table 11.5.1 provided the setpoints as: Particulate; Alert Alarm 8.4 (-7) μCi/cm³, High Alarm 1.0 (-6) μCi/cm³; Noble Gas; Alert Alarm 5.8 (-3) μCi/cm³, and High Alarm 5.8 (-3) μCi/cm³. The systems are consistent with recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Systems Leakage Detection Systems," May 1973. The inspectors reviewed Calculation NC-9012, Revision 7, "Process and Effluent Radiation Monitor Setpoints" to verify Table 11.5.1 values. The inspectors also reviewed Calculation NC-9028, Revision 2, "RCS Leak Detection by Particulate Monitor," which had the purpose of ensuring that reactor containment building detector particulate Monitor RT-8011 responds to a one gallon per minute leak from the RCS in one hour or less.

b. Findings

Introduction. The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion III, associated with the licensee's failure to assure that applicable regulatory requirements and the design basis for the containment radiation monitors were correctly translated into radiation monitor alarm setpoints.

Description. Calculations included modeling of radionuclide transport from a postulated RCS leak to the reactor containment building radiation monitors. The RCS source term apparently used was equivalent to 0.1 percent failed fuel. This source term value was much higher than the actual RCS source at any point in the current or previous fuel cycle. Regulatory Guide 1.45 stated that a "realistic" primary coolant radioactivity concentration should be used when demonstrating leak detection capability. The licensee estimated that greater than 230 hours of leakage would have been needed before the particulate channel monitor would have alarmed with the setpoints established at the time of the RCS leak on December 17, 2003. The noble gas concentration would have reached equilibrium well below the gas channel alarm setpoint and, therefore, would never have alarmed. The inadequate setpoint was demonstrated by the event.

Analysis. This finding was greater than minor because the reactor containment building radiation monitor was not capable of performing the design basis function for an extended period of time. The failure to alarm by the containment radiation monitor resulted in potential impact on reactor safety and adversely affected the reactor coolant leakage performance attribute of the Barrier Integrity/Reactor Safety Cornerstone. A Phase I screening resulted in continuing to Phase II due to the RCS barrier being affected. A Phase II evaluation determined that the finding was of very low safety significance because other methods of RCS leak detection were available to the licensee, and operators responded to the trending in the volume control tank level and then noted the rising trend recorded with the particulate radiation monitor. The failure of the radiation monitors to alarm within one hour did not contribute to an increase in core damage sequences when evaluated using the Significance Determination Process Phase 2 worksheets.

Enforcement. Title 10 of the Code of Federal Regulations, Part 50, Appendix B, Criterion III, Design Control, required that measures be established to assure that applicable regulatory requirements and the design basis were correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. Contrary to the above, applicable regulatory requirements, established in Updated Final Safety Analysis Report Section 11.5.2.3.2, were not correctly translated into specifications as applied to setpoints for the radiation monitor alarms. Specifically, Final Safety Analysis Report Table 11.5.1 established containment airborne gaseous radioactivity monitor setpoints that did not detect a one gallon per minute RCS leak within one hour as specified by Regulatory Guide 1.45. However, since the finding was determined to be of very low safety significance and was entered into the licensee's Corrective Action Program as Condition Reports 03-7771 and 03-18538, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000498/2005003-03, Reactor Coolant Leakage Detection System Calibration.

.2 Operational Readiness of Offsite Power (Temporary Instruction (TI) 2515/163)

The inspectors collected data pursuant to TI 2515/163, "Operational Readiness of Offsite Power." The inspectors reviewed the licensee's procedures related to General Design Criteria 17, "Electric Power Systems"; 10 CFR 50.63, "Loss of All Alternating Current Power"; 10 CFR 50.65(a)(4), "Requirements for Monitoring the Effectiveness of

Maintenance at Nuclear Power Plants"; and the Technical Specifications for the offsite power system. The data was provided to the Office of Nuclear Reactor Regulation for further review. The following documents were reviewed:

- 0POP04-AE-0005, "Offsite Power System Degraded Voltage," Revision 0.
- Technical Specification Limiting Condition for Operation Action 3.8.1.1.e
- 0POP01-ZO-0006, "Extended Allowed Outage Time," Revision 13
- 0POP01-ZG-002, "STP Coordinator," Revision 1
- 0POP04-AE-0003, "Loss of Power to one or more 13.8kV Standby Bus," Revision 6
- 0POP04-AE-0004, "Loss of Power to one or more 4160 ESF Bus," Revision 10
- ERCOT Operating Guide Section 2.10, "System Voltage Profile," 1 May 2005
- ERCOT Operating Guide Section 4, "Emergency Operation," 1 September 2004.
- NRR Safety Evaluation of Revised Blackout Position dated 24 July 1995
- 0PGP03-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), August 15, 2002

.3 TI 2515/161 - Transportation of Reactor Control Rod Drives in Type A Packages

a. Inspection Scope

This area was inspected to verify that the licensee's radioactive material transportation program complies with specific requirements of 10 CFR Parts 20 and 71 and Department of Transportation regulations contained in 49 CFR Part 173. The inspector interviewed licensee personnel and determined the licensee had undergone refueling/defueling activities between January 1, 2002, and present, but it had not shipped irradiated control rod drives in Department of Transportation Specification 7A Type A packages.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

The results of the TI 2515/161 inspection were presented to Mr. W. Bullard, Manager, Radiation Protection on June 2, 2005.

The results of a telephonic exit meeting to present the Emergency Preparedness inspection results were presented to Mr. A. Morgan, Supervisor, Emergency Response, on June 22, 2005.

The results of a telephonic exit meeting to present the Physical Protection inspection results were presented to Robyn Savage, Licensing Senior Staff Specialist, on June 23, 2005.

The results of the resident inspection were presented to Mr. James J. Sheppard, President and Chief Executive Officer, and other members of licensee management on July 7, 2005.

Other Meetings

Mr. Steve West, Deputy Director, Division of Reactor Safety, visited the site and toured selected areas of the facility on May 18.

Mr. Tony Vegel, Deputy Director, Division of Reactor Projects, visited the site and toured selected areas of the facility on June 10.

40A7 Licensee-identified Violations

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

- Technical Specification 3.8.1 requires the licensee, in part, with one standby diesel generator inoperable, to demonstrate the operability of the required ac offsite sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. On August 18, 2004, during a work review, the licensee identified the failure to declare Standby Diesel Generator 12 as inoperable on November 4, 2003, while Auxiliary Feedwater Pump 12 was inoperable for scheduled maintenance to replace the pump breaker cell switch. During this maintenance, the diesel output breaker was unable to recognize that the auxiliary feedwater pump breaker was not closed and this prevented the auto closure capability of Standby Diesel Generator 12 output breaker during an emergency start. Standby diesel Generator 12 was inoperable from approximately 8:15 a.m. until 8:05 p.m. on November 4, 2003. During this time, the licensee also failed to perform the associated engineered safety feature power availability surveillance. This item was documented in the licensee's

Corrective Action Program as Condition Report 04-11428. This finding is of very low safety significance because two redundant standby diesel generators were available.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

T. Bowman, Manager, Operations
W. Bullard, Manager, Health Physics
K. Coates, Manager, Maintenance
E. Halpin, VP Oversight
W. Harrison, Senior Engineer, Quality and Licensing
S. Head, Manager, Licensing
R. Gangluff, Manager, Chemistry
R. Grantum, Manager, PRA
B. Jenewel, Supervisor, Engineering
M. McBurnett, Manager, Quality and Licensing
M. Meier, General Manager, Station Support
W. Mookhoek, Senior Engineer, Quality and Licensing
A. Morgan, Supervisor, Emergency Response
M. Ruvalcaba, Supervisor, Systems Engineering
R. Savage, Senior Staff Specialist
J. Sheppard, President and CEO
D. Stillwell, Supervisor, Configuration Control and Analysis
J. Winters, SED

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Open

05000498/2005003-01	NCV	Inadequate Procedures Result in Relief Valve Openings During the Performance of Surveillance Tests (Section 4OA2)
05000498/2005003-02	NCV	Unit 1 Exceeding Licensed Thermal Power Limits (Section 4OA2)
05000498/2005003-03	NCV	Reactor Coolant Leakage Detection System Calibration (Section 4OA5)
05000498;499/2005003-04	NCV	Failure to Refer an Employee to the Employee Assistance Program (Section 3PP8)

Closed

05000498/2005003-01	NCV	Inadequate Procedures Result in Relief Valve Openings During the Performance of Surveillance Tests (Section 4OA2)

05000498/2005003-02	NCV	Unit 1 Exceeding Licensed Thermal Power Limits (Section 4OA2)
05000498/2005003-03	NCV	Reactor Coolant Leakage Detection System Calibration (Section 4OA5)
05000498;499/2005003-04	NCV	Failure to Refer an Employee to the Employee Assistance Program (Section 3PP8)
05000498;499/2004002-02	URI	Reactor Coolant Leakage Detection System Calibration (Section 4OA5)
0500498/2004-005-00	LER	Standby Diesel Generator 12 Not Being Declared Inoperable While the Auxiliary Feedwater Pump 12 Breaker Cell Switch Was Being Replaced (Section 4OA3)

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 reports reviewed)

96-10465	97-12499	97-13259	95-1718	99-12426	00-792
00-3152	02-5184	03-18541	05-6342		

Section 3PP8: Fitness For Duty Program

OPGP09-A-0001, Plant Access Authorization Program, Revision 17
OPGP09-A-0002, Fitness For Duty Program, Revision 14
OPGP09-A-0003, Behavior Observation Program, Revision 9
OPGP09-A-0004, Employee Assistance Program, Revision 5
OPGP09-A-0007, Unescorted Access Evaluation Process, Revision 3

STP-313, Employee Assistance Program, Revision 1
STP 502, Drugs and Alcohol-Fitness for Duty, Revision 3

Condition Reports

CR-04-4820	CR-04-5159	CR-04-5328	CR-04-5425	CR-05-3268

LIST OF ACRONYMS

CFR	*Code of Federal Regulations*
CVCS	chemical and volume control system
ECW	essential cooling water
NCV	noncited violation
NEI	Nuclear Energy Institute
PORV	power-operated relief valve
RCS	reactor coolant system
SSC	structures, systems, and components
TI	temporary instruction