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

August 5, 2004

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 05000498/2004003 AND 05000499/2004003

Dear Mr. Sheppard:

On June 26, 2004, 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 1, 2004, with Mr. Tom Jordan, Vice President, Engineering & Technical Services, and other 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 three findings of very low safety significance (Green), evaluated under the risk significance determination process (SDP), two of which 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 (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. If you contest any NCVs 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/

William D. Johnson, Chief
Project Branch A
Division of Reactor Projects

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

Enclosure:
NRC Inspection Report 05000498/2004003 and 05000499/2004003
w/Attachment: Supplemental Information

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RIV:SRI:DRP/A	SPE:DRP/A	C:DRS/PSB	C:DRS/PEB	C:DRS/EB
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E - WDJohnson	***/RA/***	***/RA/***	***RLNease for***	***/RA/***
7/30/04	7/30/04	8/3/04	8/2/04	8/2/04

C:DRS/OB	C:DPR/A			
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8/5/04	8/5/04			

ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	50-498, 50-499
Licenses:	NPF-76 NPF-80
Report No:	05000498/2004003 05000499/2004003
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, 2004
Inspectors:	G. L. Guerra, Senior Resident Inspector J. Taylor, Resident Inspector T. Farnholtz, Senior Project Engineer L. Ricketson, P.E., Senior Health Physicist M. Murphy, Senior Operations Engineer G. Johnston, Senior Operations Engineer D. Allen, Senior Resident Inspector, Comanche Peak A. Barett, Project Engineer W. McNeill, Reactor Inspector N. O'Keefe, Senior Reactor Inspector
Approved By:	W. D. Johnson, Chief Project Branch A Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000498/2004003, 05000499/2004003; 04/08/04 - 06/26/04; South Texas Project Electric Generating Station; Units 1 & 2; Integrated Resident Report, Occupational Radiation Safety, Other Activities

This report covered a 3-month period of inspection by the resident inspectors and Region IV inspectors. Three green findings were identified, two of which involved noncited violations. 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: Barrier Integrity

- Green. A Green noncited violation of Technical Specification 6.8.1.a and Regulatory Guide 1.33, Appendix A, was identified for an inadequate procedure that resulted in a letdown pressure relief valve opening during a letdown orifice swap. Operators failed to manipulate the letdown orifice isolation valve in a manner that properly controlled pressure in the chemical and volume control system. As a result, the letdown line relief valve opened, diverting reactor coolant system inventory to the primary relief tank. Corrective actions for this event included enhancing the procedure by adding notes and precautions and holding lessons learned sessions with operators.

  This finding is greater than minor because the opening of the letdown relief valve increased the risk of an initiating event of an interfacing system small loss of coolant accident and degraded the reactor coolant system barrier integrity and therefore could be reasonably viewed as a precursor to a significant event. A Phase 1 screening passed to a Phase 2 evaluation because the letdown line relief that lifted could have failed to reseat or could have continually blown down if not isolated. The Phase 2 evaluation resulted in a Green determination. However, the result was unreliable because the tool did not accurately model the event. Under the Phase 3 analysis, a Region IV Senior Reactor Analyst evaluated several scenarios involving mechanical and human error failures that could result in the failure of the safety relief to close and/or failure of letdown isolation contributing to the continued draining the reactor coolant system. The result indicated that the risk significance of the performance deficiency that caused the event was very low (Section 4OA5).

Cornerstone: Mitigating Systems

- Green. A finding was identified associated with the Fire Safe Shutdown Analysis because the licensee had not accounted for the reactor coolant inventory loss due to expected reactor coolant pump seal leakage. The licensee's Fire Safe Shutdown Analysis credited the charging system to provide the reactor coolant inventory control functions. However, in a number of fire areas charging was procedurally stopped to avoid damaging the charging pumps as a result of a spurious closing of either of the motor-operated volume control tank suction valves for up to two hours. The inspector determined that there was no analytical basis for allowing charging to be secured this long. Because the licensee was able to re-perform the safe shutdown analyses and demonstrate that the plant could meet its fire safe shutdown design without makeup for two hours, no violation of NRC requirements existed.

This issue was determined to be more than minor because it was similar to Example 3.i of Manual Chapter 0612, Appendix E in that the Fire Safe Shutdown Analysis had to be re-performed to assure that the acceptance criteria were met. This issue affected the Mitigating Systems Cornerstone because it related to the availability of charging when it was required to mitigate the effects of a fire. This issue was determined to have very low safety significance because it involved a design deficiency confirmed not to result in a loss of function (Section 4OA5).

Cornerstone: Occupational Radiation Safety

Green. The inspectors reviewed two examples of a noncited violation of Technical Specification 6.12.1, in which the licensee failed to control high radiation areas. On May 3, 2003, the licensee identified, during routine surveys, an uncontrolled high radiation area in Unit 1, Room 108C. The licensee initially concluded that the apparent cause was a plant system that introduced unpredictable dose rates. However, as a result of the inspector's questions, the licensee reviewed the matter further and concluded the cause was a lack of plant system knowledge on the part of some radiation protection personnel. The licensee reopened the original condition report and re-entered it to the corrective action program. The licensee was alerted to a second example when a worker's electronic dosimeter alarmed on April 6, 2004, as the individual worked on scaffolding under Unit 2 Steam Generators B and C. The dose rates were not identified before the worker entered the area because the responsible radiation protection technician was unaware of the existence of drain lines from Steam Generators B and C. The licensee placed the finding into its corrective action program.

The failures to correctly control high radiation areas were performance deficiencies. These examples of a finding were greater than minor because they were associated with one of the cornerstone attributes and affected the cornerstone objective, in that, inadequate exposure controls of high radiation areas affected the licensee's ability to ensure adequate protection of worker health and safety from exposure to radiation. Because the examples of a finding involved the potential for workers to receive significant, unplanned, unintended dose as a result of conditions contrary to technical specification requirements, the inspector used the Occupational Radiation Safety Significance Determination Process described in Manual Chapter 0609, Appendix C, to analyze the significance of the examples. The inspector determined that the examples were of very low safety significance because they did not involve (1) ALARA planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. The first example of this finding also had cross-cutting aspects associated with problem identification and resolution. The original cause determination was inadequate (Section 2OS1).

B. <u>Licensee-Identified Violations</u>

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 operated at essentially 100 percent power throughout the inspection period.

Unit 2 began the inspection period shutdown for scheduled refueling outage. The unit was restarted on April 26 and achieved full power shortly thereafter. The unit operated at essentially 100 percent power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

During the week of, June 7 the inspectors completed a detailed review of the site's adverse weather procedures and adverse weather preparations for the 2004 hurricane season. The inspection included a review of the following licensee procedures:

- 0PGP03-ZV-0001, "Severe Weather Plan," Revision 10
- 0POP04-ZO-0002, "Natural or Destructive Phenomena Guidelines," Revision 27

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors conducted partial and complete walkdowns of the following risk-significant systems to verify that they were in their proper standby alignment as defined by system operating procedures and system drawings. During the walkdowns, inspectors examined system components for materiel conditions that could degrade system performance. In addition, the inspectors evaluated the effectiveness of the licensee’s problem identification and resolution program in resolving issues which could increase event initiation frequency or impact mitigating system availability.

The following two partial system walkdowns were performed:

- On April 22 and 23 the inspectors verified the condition of the Unit 1 Train A essential chilled water lineup. This walkdown was performed while the Train B essential chilled water was out of service due to a temperature controller failure. The inspectors compared system equipment and control board lineups to Plant Operating Procedure 0POP02-CH-0001, "Essential Chilled Water System," Revision 30.

- On May 19 and 20 the inspectors verified the condition of the Unit 2 essential cooling water system Train 2B. No maintenance activities were being conducted at the time of the walkdown, however, Train 2A had been out of service for planned maintenance earlier in the week. The inspectors visually observed the condition of the equipment. The control board lineup was also inspected. The inspectors compared system equipment and control board lineups to Plant Operating Procedure 0POP02-EW-0001, "Essential Cooling Water Operations," Revision 32.

The following complete system walkdown was performed:

- On June 3 and 4 the inspectors verified the condition of the Unit 2 safety injection system which includes three trains of low head and high head pumps. No maintenance activities were being conducted at the time of the walkdown. The inspectors visually observed the condition of the equipment. The control board lineup was also inspected. The inspectors compared system equipment and control board lineups to Plant Operating Procedure 0POP02-SI-0002, “Safety Injection System Initial Lineup,” Revision 16.

b. Findings

No findings of significance were identified

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors toured eight plant areas to assess the licensee’s control of transient combustible materials, the material condition and lineup of fire detection and suppression systems, and the material condition of manual fire equipment and passive fire barriers. The licensee’s fire preplans and fire hazards analysis report were used to identify important plant equipment, fire loading, detection and suppression equipment locations, and planned actions to respond to a fire in each of the plant areas selected. Compensatory measures for degraded equipment were evaluated for effectiveness. The following plant areas were inspected:

- (Unit 1) Component cooling water heat exchanger room on April 22 (Fire Zone Z142)

- (Unit 2) Train A essential cooling water pump rooms on May 19 (Fire Zone Z603)

- (Unit 2) Train B essential cooling water pump rooms on May 19 (Fire Zone Z604)

- (Unit 2) Train C essential cooling water pump rooms on May 20 (Fire Zone Z605)

- (Unit 2) Main Control Room on June 3 (Fire Zone 034)

- (Unit 2) Train A Safety Injection/Containment Spray System Cubicle on June 3 (Fire Zone 307)

- (Unit 2) Train B Safety Injection/Containment Spray System Cubicle on June 3 (Fire Zone 306)

- (Unit 2) Train C Safety Injection/Containment Spray System Cubicle on June 3 (Fire Zone 305)

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

On June 3, the inspectors used the guidance in Inspection Procedure 71111.06 to perform inspections of the Unit 1 emergency diesel generator building to verify that the licensee's flood mitigation plans and equipment were consistent with the licensee's design requirements and risk-analysis assumptions. This inspection was performed for external sources of flooding. The inspection focused on the licensee's design for protecting redundant trains of emergency diesel generators located in this building to verify that adequate mitigation equipment would remain in all flooding scenarios. The inspectors reviewed the Updated Final Safety Analysis Report and the licensee's flooding calculation to evaluate the external flooding design and how current station procedures implemented that design. The inspectors also walked down diesel generator building to identify any missing or degraded flood barriers and flood control features. The following documents were reviewed:

- Calculation MC-5044, ?Flooding Calculation for the Diesel Generator Building," Revision 2

- Diesel Generator Building Plumbing Drawings 9D-06-9-B-0170 and 9D-06-9-B-0170

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07A)

a. Inspection Scope

The inspectors observed the inspection of the Unit 2 Train A essential chiller, an essential chill water/essential cooling water heat exchanger, on May 17, 2004. Review and assessment of the inspection results were performed against the performance criteria in the preventive maintenance work order (WAN 89776) and the previous inspections results maintained by the system engineer.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope

Performance of Nondestructive Examination Activities Other than Steam Generator Tube Inspections

Inspection Procedure 71111.08 specified conducting a review of two or three types of nondestructive examination activities: volumetric (radiographic or ultrasonic); surface (magnetic particle or liquid penetrant); and visual (VT-1 to determine the surface condition of a part or component, VT-2 to locate evidence of leakage, and VT-3 to determine the general mechanical and structural condition of parts or components). The inspectors reviewed multiple examples of two types as noted in column three of the table below. The inspectors observed the performance of the following examinations:

System	Component/Weld Identification	Examination Method
Reactor Coolant	Pipe to Pipe Weld/102300	Ultrasonic
Reactor Coolant	Pipe to Elbow/103360	Ultrasonic
Main Steam	Pipe Lugs/554245	Magnetic Particle
Safety Injection	Pipe Lugs/705810	Penetrant

During the performance of each examination, the inspectors verified that the licensee used the correct nondestructive examination procedure, the licensee met the requirements specified in the procedure, and the licensee used properly calibrated test instrumentation and equipment. The inspectors verified the nondestructive examination certifications of those personnel observed performing examinations. The inspection procedure (71111.08) also specified a review of examinations from the previous outage with recordable indications. The inspectors verified that the licensee compared the recordable indications revealed by the current examinations against the previous outage examination reports.

The inspection procedure (71111.08) further specified that if the licensee completed welding on the pressure boundary for ASME Code Class 1 or 2 systems, then verification should be performed that acceptance and preservice examinations were done in accordance with the ASME Code for one to three welds. The inspectors found the licensee performed only one welding repair under Section III of the ASME Code for Class 1 and 2 components since the last outage. The inspectors reviewed the work order on repair welding of a chemical volume control system letdown isolation Valve 2-CV-02468 body to bonnet seal weld. The licensee performed only visual for leakage (VT-2) type nondestructive examination for this repair. The inspectors reviewed the work order package and verified that the repair activities met ASME Code (Class 1) requirements.

The inspection procedure (71111.08) also specified verification that one or two ASME Code Section XI repairs or replacements met ASME Code requirements. The licensee planned only one repair/replacement activity during the current outage, namely a "like for like" replacement of a reactor coolant pump seal housing gasket. The licensee performed only visual for leakage (VT-2) type nondestructive examination for this repair. The inspectors verified that the repair/replacement activity met ASME Code (Class 1) requirements.

<u>Steam Generator (SG) Tube Inspection Activities</u>

The licensee had replaced the steam generators during the last outage of this unit. The licensee reported they had no appreciable leakage for the steam generators during operations. The licensee and its contractors used properly qualified eddy current probes and equipment for the expected types of tube degradation. The licensee's scope of examination met the technical specification, NRC requirements and industry guidelines. The inspectors observed the collection and analysis of eddy current data by contractor personnel performed to evaluate tubes and a possible loose part in a steam generator. This review included verification of the proper technique sheets, known as ANTS and acquisition specification technique sheets, known as ACTS. The inspectors verified that the licensee compared flaws detected during the current outage against the preservice inspection data. The inspection procedure (71111.08) directs the inspectors to contact the Office of Nuclear Reactor Regulation, if the licensee or its contractors have questionable eddy current data. The inspectors did not identify any questionable analyses. As required by the inspection procedure, the inspectors determined that the licensee identified loose parts. The inspectors verified that the licensee took appropriate corrective action (Condition Report 04-4960). The inspectors observed eddy current analysis and resolution of the following calibration groups:

Calibration Group Number

THX 2 A 1 006	THX 2 A 1 028	THX 2 C 1 037
THX 2 A 1 013	THX 2 A 1 031	THX 2 D 1 089
THX 2 A 1 027	THX 2 A 1 037	THX 2 C 1 103

The inspection procedure (71111.08) addresses steps to be taken when the licensee finds new areas of degradation and the use of industry experience. The inspectors verified that the licensee did not find areas of new degradation. The inspectors verified that the licensee reviewed areas of potential degradation based on site-specific and industry experience. The licensee had incorporated lessons learned from recent inspections at Comanche Peak Steam Electric Station, Diablo Canyon Power Plant, and Palo Verde Nuclear Generating Station. The inspection procedure questions if the licensee expanded the scope of the inspection. The licensee did not plan any expansion of the scope of eddy current examination.

The inspection procedure (71111.08) addresses steps to be taken regarding the previous outage operational assessment. The inspectors found the licensee did not have a previous outage operational assessment to predict flaw type and frequency. Industry guidelines identified in the procedure do not require such a report for the first outage after a steam generator replacement.

Identification and Resolution of Problems

The inspectors reviewed nine condition reports issued since the last outage on inservice inspection and steam generator eddy current testing activities. The inspectors verified that the licensee identified, evaluated, corrected, and trended problems.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

On May 18 the inspectors assessed Crew 2A during licensed operator simulator requalification training. The inspectors observed two control room simulator scenarios that included a loss of coolant accident and an anticipated transient without a scram. One of the scenarios was graded for emergency preparedness performance indicators by licensee personnel. The inspectors observed the performance of Crew 2A for clarity and formality of communications, the correct use of procedures, performance of high risk operator actions, monitoring of critical safety functions, and the oversight and direction provided by the shift supervisor. The inspectors observed the operators' use of emergency action levels and protective action recommendations for accuracy and timeliness, reviewed the scenario sequence and objectives, observed the training critique, and discussed the crew's performance with training instructors. In addition, the inspectors attended the critique held by the operating crew to assess individual performance and training effectiveness.

b. Findings

No findings of significance were identified.

1R12 Maintenance Implementation (71111.12)

a. Inspection Scope

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

- (Unit 2) Pressurizer Spray Valves RC 655B and RC 655C leak by approximately 200 gallons per minute on April 20 (Condition Report (CR) 04-5929)
- (Common) Reactor head vent valve performance on June 1 (CR 04-4472 and 04-4398)

The inspectors reviewed whether the structures, systems, or components were properly characterized in the scope of the Maintenance Rule Program and whether the failure or performance problem was properly characterized. In addition, the inspectors assessed the appropriateness of the established performance criteria. The inspectors independently verified that the corrective actions and responses implemented were appropriate and adequate. Discussions with the responsible system engineer were also held.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors assessed whether the performance of risk assessments for selected planned and emergent maintenance activities was in accordance with 10 CFR 50.65(a)(4). The inspectors assessed the completeness and accuracy of the information considered in the risk assessments and compared the actions taken to manage the resultant risk with the requirements of the licensee's Configuration Risk Management Program. The inspectors reviewed these assessed risk configurations against actual plant conditions and any in-progress evolutions or external events to verify that the assessments were accurate, complete, and appropriate for the conditions. In addition, the inspectors walked down the control room and plant areas to verify that compensatory measures identified by the risk assessments were appropriately performed. The inspectors reviewed the following four activities:

- (Unit 2) Evaluation of differing reactor coolant level indications prior to entering midloop operations on April 18 (CR 04-4498)
- (Unit 1) Evaluation and planning for the repair of the low pressure governor valve on steam generator feedwater Pump 13 on May 20-21 (Work Authorization Number (WAN) 271118)

- (Unit 1) Control room envelope HVAC flow balancing on June 1 through 4 (WAN 271935, 271938, and 272033)

- (Unit 2) Evaluation of loss of coolant to technical support diesel and effect of risk on work week of June 7 through 11 (WAN 275203)

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions (71111.14)

a. Inspection Scope

The inspectors observed the nonroutine evolution described below to verify that they were conducted in accordance with licensee procedures and Technical Specification requirements. The inspectors reviewed the licensee's procedures, attended pre-job briefs, and observed personnel performance in the control room.

- (Unit 2) Reactor startup to 100 percent after outage on April 26 and 27

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors selected six operability evaluations conducted by licensee personnel during the report period involving risk-significant systems or components. The inspectors evaluated the technical adequacy of the licensee's operability determination, determined whether appropriate compensatory measures were implemented, and determined whether or not other pre-existing conditions were considered, as applicable. Additionally, the inspectors evaluated the adequacy of the licensee's problem identification and resolution program as it applied to operability evaluations. Specific operability evaluations reviewed are listed below:

- (Unit 1) Gas used to calibrate the containment hydrogen analyzers had expired manufacturer's use sticker (CR 04-5246-1) on April 17

- (Unit 2) Evaluation of electrical containment penetration that indicated an open circuit (CR 04-5234) on April 21

- (Unit 2) Distribution panel 1202 lost power while swapping battery chargers (CR 04-5388-5) on April 23

- (Common) Evaluation for departure from nucleate boiling ratio for more than one feedwater regulating valve failing open (CR 04-5019) on May 18

- (Unit 2) Evaluation for valve performance after packing torque on safety injection Valve SI-MOV-31A (CR 04-7494) on June 9

- (Common) Evaluation for lighting strike close to site and effect on plant equipment (CR 04-8572 and 04-8577) on June 15

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors reviewed the following licensee-identified operator workaround on an existing condition to verify that it had been identified and assessed in accordance with STP's Total Impact Assessment document and to determine if the functional capability of the system or human reliability in responding to initiating events had been affected. The ability of operators to implement normal and emergency operating procedures with the existing equipment issues was specifically evaluated.

- (Unit 2) Emergent - Train A accumulator gas leakage on May 10 requiring operators to pressurize each shift (CR 04-6533)

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17A)

a. Inspection Scope

The inspectors, in conjunction with the Office of Nuclear Reactor Regulation Mechanical Engineering Branch, reviewed one permanent plant modification package involving a design change to equipment in the plant. The inspectors reviewed the licensee's modification package associated with the use of solenoid valves. Additionally, the inspectors interviewed the cognizant engineers as to their understanding of the modification package. Also, a telephonic conference was held between licensee and NRC representatives on June 22, 2004. The following specific documents were reviewed:

- Design Change Package (DCP) 01-19793-10, Supplement 0

- 10 CFR 50.59 screening for DCP 01-19793-10, Supplement 0

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed post maintenance test procedures and associated testing activities for four risk-significant mitigating systems. In each case, the associated work orders and test procedures were reviewed against the attributes in Inspection Procedure 71111.19, to determine the scope of the maintenance activity and determine if the testing was adequate to verify equipment operability. The Updated Final Safety Analysis Report, Technical Specifications, and design basis documents were also reviewed, as applicable, to determine the adequacy of the acceptance criteria listed in the test procedures. The inspectors witnessed or reviewed the results of post maintenance testing for the following maintenance activities:

- (Unit 2) Temporary Engineering Procedure 2TEP07-DG-0005, ?Standby Diesel Generator 22 Return to Service Testing," Revision 0, after rebuild due to thrown rod event on April 8 through 22 (WAN 269653)
- (Unit 2) Essential cooling water system Train 2A after planned maintenance on April 19 (WAN 239361)
- (Unit 2) Component cooling water system Train 2B after planned maintenance on June 24 (WAN 168868)
- (Unit 1) Standby diesel generator Train 1C after output breaker planned maintenance on May 26 (WAN175215)

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the major work and weekly outage risk assessments on an ongoing basis to assess completeness, accuracy, and adequacy of risk management for Refueling Outage 2RE10. The inspectors used Inspection Procedure 71111.20 to perform the following inspection activities.

Refueling

The inspectors observed refueling activities from the control room, radiation protection control center, and during containment tours to determine if these activities were conducted in accordance with the Technical Specifications and administrative procedures.

Maintaining Plant Conditions

The inspectors conducted frequent plant walkdowns to assess the availability of instrumentation, electrical power, decay heat removal, inventory control, reactivity control, and containment integrity. The inspectors reviewed plant conditions and observed selected outage activities throughout the outage to verify that the licensee maintained the plant in a configuration consistent with the requirements of Technical Specifications and with the assumptions of the outage risk assessment. Control room operators were also observed and interviewed on the status of plant conditions. The inspectors verified that emergent issues were properly assessed for their impact on plant risk.

Monitoring of Heatup and Startup Activities

The inspectors observed control room operations and reviewed control room logs to verify that the Unit 2 operational mode changes, including heatup and startup activities, were conducted in compliance with the applicable Technical Specifications and administrative procedures. Additionally, Plant Operating Procedures 0POP03-ZG-0004, ?Reactor Startup," Revision 25, and 0POP03-ZG-0005, ?Plant Startup to 100%," Revision 46, were reviewed.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors evaluated the adequacy of six periodic tests of important nuclear plant equipment. This review included aspects such as preconditioning, the impacts of testing during plant operations, the adequacy of acceptance criteria, test frequency, procedure adherence, record keeping, the restoration of standby equipment, test equipment and the effectiveness of the licensee's problem identification and resolution program. The inspectors observed or reviewed the following tests:

- (Unit 2) 0PSP03-DG-0014, ?Standby Diesel 22 LOOP - ESF Actuation Test," Revision 13, on April 21

- (Unit 2) 0PEP07-SG-0005, ?Steam Generator Water Level Control Test," Revision 2, on April 29

- (Unit 1) 0PSP03-DG-0003, "Standby Diesel 13 Operability Test," Revision 26, on April 28

- (Unit 1) 0PSP03-AF-0007, "Auxiliary Feedwater Pump 14 Inservice Test", Revision 28, on May 6

- (Unit 1) 0POP07-DB-0005, ?Technical Support Center Diesel Generator Performance Test," Revision 9, on May 3

- (Unit 2) 0PSP03-DG-0002, "Standby Diesel 22 Operability Test," Revision 25, on May 27

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the temporary modification of the 120 volt vital AC distribution system for critical loads to the main turbine, feedwater pump turbine controls, and radiation monitoring. The inspectors looked at the following attributes in reviewing this temporary modification: (1) the adequacy of the safety evaluation; (2) the consistency of the installation with the modification documentation; (3) the updating of drawings and procedures, as applicable; and (4) the adequacy of the post-installation testing. The inspectors also walked down the temporary modifications. The following document was reviewed

- T2-04-8385-2, ?Temporary 15KVA regulating transformer for DP048A," on June 17

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness (EP)

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

On May 5, the inspectors used the guidance in Inspection Procedure 71114.06 to assess a licensee evaluated emergency drill. The inspectors attended the briefing for the drill players on May 4. Observation of the drill activities commenced on May 5 at 7:00 a.m. at the Training Center Simulator, upon declaration of an 'alert' the emergency support facilities were manned. The inspectors proceeded to the Unit 1 Technical Support Center and completed drill observations. The inspectors evaluated operators and licensee emergency response staff for clarity and formality of communications, the correct use of procedures, and the oversight and direction provided by the shift supervisor and emergency director. The inspectors also observed the licensee's use of emergency action levels for proper emergency classification and reporting timeliness, reviewed the scenario sequence and objectives, and observed the post drill critique in the Technical Support Center.

On May 18 the inspectors assessed operating Crew 2A during a licensee evaluated emergency drill. The inspectors observed and reviewed drill activities in the control room simulator. The inspectors evaluated operators and licensee emergency response staff (security, health physics, and plant operators) for clarity and formality of communications, the correct use of procedures, and the oversight and direction provided by the shift supervisor/emergency director. The inspectors also observed the licensee's use of emergency action levels for proper emergency classification and reporting timeliness, reviewed the scenario sequence and objectives, and observed the licensee's critique.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

This area was inspected to assess the licensee's performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, high radiation areas, and worker adherence to these controls. The inspector 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 inspector interviewed the radiation protection manager, radiation protection supervisors, and radiation workers. The inspector performed independent radiation dose rate measurements and reviewed the following items:

- Controls (surveys, posting, and barricades) of three radiation, high radiation, or airborne radioactivity areas
- Radiation work permit, procedure, and engineering controls and air sampler locations

- Conformity of electronic personal dosimeter alarm set points with survey indications and plant policy; workers' knowledge of required actions when their electronic personnel dosimeter noticeably malfunctions or alarms

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

- Self-assessments and audits related to the access control program since the last inspection

- Corrective action documents related to access controls

- Licensee actions in cases of repetitive deficiencies or significant individual deficiencies

- Radiation work permit briefings and worker instructions

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

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

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

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

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

Either because the conditions did not exist or an event had not occurred, no opportunities were available to review the following items:

- Performance indicator events and associated documentation packages reported by the licensee in the Occupational Radiation Safety Cornerstone

- Barrier integrity and performance of engineering controls in airborne radioactivity areas

- Adequacy of the licensee's internal dose assessment for any actual internal exposure greater than 50 millirem CEDE

- Licensee event reports, and special reports related to the access control program since the last inspection

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

The inspector completed 21 of the required 21 samples.

b. Findings

Introduction. The inspector reviewed two examples of a Green, noncited violation of Technical Specification 6.12.1 resulting from the licensee's failure to post and control high radiation areas.

Description. The first example was identified by the licensee in Condition Report 03-7554. During a routine survey on May 3, 2003, a radiation protection technician identified an uncontrolled high radiation area in Room 108C of the Unit 1 mechanical auxiliary building, near Valve 1-WL-0636. The dose rates were 3800 millirems per hour on contact with the valve and 120 millirems per hour at 30 centimeters.

The licensee stated in its apparent cause determination, ?This valve has a high potential for changing dose rates associated with it as the reactor containment building normal sump is automatically pumped down. This can lead to a change in room conditions without awareness of the radiation protection staff." The resulting corrective action was to expand the high radiation area posting in the room similar to that in Unit 2 to ensure that changing conditions do not result in unposted high radiation areas.

The inspectors interviewed radiation protection representatives to determine if they should have been aware of increasing dose rates and, therefore, should have identified and controlled the high radiation area sooner. In response to the inspector's questions, radiation protection personnel reviewed their logs and identified facts that were missed during the licensee's original investigation. They determined that the radiation protection group was notified on April 11, 2004, that the cavity was being drained. A radiation protection technician was dispatched to Room 108C to perform a survey of the area. However, the survey was performed at the start of the cavity drain and radiological conditions had not yet changed. Water typically does not pass through Room 108C until approximately 3 to 5 hours after the start of the cavity draining process. The area was not surveyed again until May 3, 2004, when the high radiation area was identified and controlled. The licensee subsequently reopened Condition Report 03-7554, revised the apparent cause to address ?a lack of understanding of the draining sequence by (radiation protection) personnel working in the mechanical auxiliary building, and initiated additional corrective actions.

The second example of a failure to control a high radiation area was identified as the result of an individual receiving an electronic alarming dosimeter dose rate alarm and, therefore, is self-revealing. On April 6, 2004, an instruments and controls technician was working on a scaffold under Steam Generators B and C in Unit 2 and received an electronic alarming dosimeter dose rate alarm. In response, radiation protection personnel performed a survey and identified an uncontrolled high radiation area. A hot spot was identified with dose rates of 6 rem per hour on contact and 350 millirems per hour at 30 centimeters from the source of radiation. The instruments and controls technician had been informed to expect a general area dose rate of 0.8 millirems per hour based on a radiation survey conducted

April 1, 2004. Radiation protection personnel determined that the higher dose rates were caused by the flushing and draining of collection drums associated with steam generator eddy current testing equipment. The drain path for Steam Generators B and C passed over the scaffolding.

A similar drain path for Steam Generators A and D had been surveyed by a radiation protection technician who identified and controlled a high radiation area. However, the technician failed to take the same actions for the drain path of Steam Generators B and C. Consequently, the associated high radiation area was not identified and controlled. According to the licensee's review, the reason was that the responsible radiation protection technician was unaware of the existence of the Steam Generators B and C drain pathway. The licensee determined that the high radiation area was uncontrolled for at least six hours.

Analysis. The failures to correctly control high radiation areas were performance deficiencies. These examples of a finding were greater than minor because they were associated with one of the cornerstone attributes and affected the cornerstone objective, in that, inadequate controls of high radiation areas affected the licensee's ability to ensure adequate protection of worker health and safety from exposure to radiation. Because the examples of a finding involved the potential for workers to receive significant, unplanned, unintended dose as a result of conditions contrary to technical specification requirements, the inspector used the Occupational Radiation Safety Significance Determination Process described in Manual Chapter 0609, Appendix C, to analyze the significance of the examples. The inspector determined that the examples were of very low safety significance because they did not involve; (1) ALARA planning and controls; (2) an overexposure; (3) a substantial potential for overexposure; or (4) an impaired ability to assess dose. The first example of this finding also had crosscutting aspects associated with problem identification and resolution. The original cause determination was inadequate.

Enforcement. Technical Specification 6.12.1 states, ?Pursuant to 10 CFR 20.1601(c), in lieu of the requirements of 10 CFR 20.1601(a), each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is greater than 100 millirem per hour, but equal to or less than 1000 millirem per hour at 30 centimeters from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and the entrance thereto shall be controlled by radiation work permit." The licensee violated this requirement when it failed to barricade, post, and control high radiation areas it should have identified sooner on two separate occasions. Because the failure to correctly control high radiation areas was determined to be of very low safety significance and has been entered into the licensee's corrective action program as Condition Reports 03-7554 and 04-4787, this violation is being treated as a noncited violation (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000498;499/2004003-01, Two examples of failure to control high radiation areas.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspector sampled licensee submittals for the performance indicators listed below for the period from April 1, 2003, through March 31, 2004. 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 Indicator

Licensee records reviewed included corrective action documentation that identified occurrences of locked high radiation areas (as defined in Technical Specification 6.12.2) very high radiation areas (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in NEI 99-02). Additional items reviewed included radiological control area entry and electronic dosimeter alarm setpoints. 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, locked 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 inspector interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data.

The inspector completed both of the required samples.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Occupational Radiation Safety

Section 2OS1 describes a licensee-identified finding which was addressed by an inadequate apparent cause determination. This subsequently resulted in corrective actions that were not comprehensive.

.2 Semi-Annual Trend Review

a. Inspection Scope

On June 8, 2004, the inspectors completed a semi-annual review of licensee internal documents, reports, audits, and performance indicators to identify trends that might indicate the existence of more significant safety issues. This review was to identify any repetitive equipment failures or human performance issues for further followup. The inspectors reviewed the following:

- Corrective Action Document Summaries
- System Health Reports
- Open Temporary Modifications
- Quality Audit Executive Summaries
- Control Room Operability Assessment Logs
- Total Impact Assessments (Operator Workarounds)

b. Findings and Observations

No findings in the area of identification and resolution of problems were identified.

4OA3 Event Followup (71153)

.1 (Closed) Licensee Event Report 05000499/200303, Revisions 0 and 1: Standby Diesel Generator 22 Failure.

This licensee event report discussed the failure of a connecting rod during a surveillance test run of the engine, which resulted in catastrophic failure of the engine. Revision 1 was issued to discuss the safety significance associated with the latent condition, which existed and could have impacted a design basis event during a period of time prior to the observed failure. This event and its causes were documented in Special Inspection Report 05000499/2004006. No additional issues were identified as a result of the review of these reports.

4OA5 Other Activities

.1 Unit 2 Reactor Pressure Vessel (RPV) Lower Head Penetration Nozzles (TI 2515/152)

a. Inspection Scope

On April 5-9, 2004, the inspector reviewed the licensee’s response to NRC Bulletin 2003-02, “Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity.” The response described the South Texas Project RPV lower head penetration nozzle inspection program along with the inspection history at Units 1 and 2. The inspector reviewed the licensee’s procedures for the inspection of the Unit 2 lower head penetrations.

The inspector reviewed photographs taken of the nozzle penetrations that covered a full 360 degrees of all 58 penetrations on the Unit 2 RPV. Photographs taken in October and December 2002 were reviewed along with photographs taken in April 2004. Access to the penetrations was gained by the removal of three panels approximately 120 degrees apart.

b. Findings

The licensee conducted a lower head penetration inspection on the Unit 2 RPV at the beginning of the 2RE10 refueling outage in April 2004. Based on photographs and inspections conducted by engineering personnel, the licensee did not identify any penetrations with evidence of boric acid deposits or indications of reactor coolant system leakage.

During the April 2004 inspection, some deposits were noted on some penetrations. The licensee considered these deposits to be similar to a deposit identified in December 2002. Samples of that deposit were taken at that time and analyzed to determine their origin. The inspector reviewed the report detailing the results of these tests. It was determined that the source of the deposit identified in December 2002 was not due to reactor coolant system leakage. Based on laboratory results, it was determined that the most likely source was leakage of the reactor cavity liner during refueling operations when the cavity is full of water and/or from pressure washing of the cavity for decontamination.

The licensee considered the deposits identified in April 2004 to be similar in appearance to the deposit noted in December 2002. Based on this similarity of the appearance and the laboratory analysis results, the licensee did not take samples of the deposits identified in April 2004. The inspector considered this action to be not well supported based on the photographs of Penetration 54 taken in April 2004. This penetration appeared to have a white deposit at the annulus between the nozzle and the reactor vessel. The nature and source of this deposit was not apparent in the photograph. Based on the inspector's concerns, the licensee committed to taking and analyzing samples from the Unit 2 RPV, including Penetration 54, when the area became radiologically accessible before plant startup from Refueling Outage 2RE10.

Results from the laboratory analysis of the samples taken from four penetrations on the Unit 2 RPV were reviewed and compared to samples taken from the Unit 1 RPV in 2003. The Unit 1 RPV Penetrations 1 and 46 deposits were known to be due to reactor coolant system leakage. These two penetrations were successfully repaired and the unit returned to service. Comparing the samples taken from Unit 1 and Unit 2 confirmed that the source of the deposits on the penetrations on Unit 2 were not from the reactor coolant system. The inspector considered this analysis to be adequate to provide assurance of the reactor coolant system integrity in the area of the Unit 2 RPV lower penetrations.

.2 Unit 2 Reactor Containment Sump Blockage (TI 2515/153)

a. Inspection Scope

On April 5-9, 2004, the inspector reviewed the licensee's response to NRC Bulletin 2003-01, ?Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors." The response included details of procedural guidance to delay refueling water storage tank (RWST) depletion, availability of alternative water sources to refill the RWST, ensuring containment drainage paths are unblocked and the sump screens are free of adverse gaps and breaches, and operator training issues.

Unit 2 was in a refueling outage at the time of the inspection. The inspector conducted a complete Unit 2 containment walkdown to ensure drainage paths were clear and to examine the sump screens for adverse gaps and breaches. The inspector reviewed the licensee's procedures regarding potential containment sump blockage and containment walkdown requirements along with the Updated Final Safety Analysis Report Sections 6.1.2.1 (Protective Coatings) and 6.2.2.2.3 (Containment Emergency Sump Description). Procedures in place contained requirements for a containment walkdown to quantify potential debris sources and to inspect for major obstructions in the containment upstream of the sumps. This walkdown is performed at the end of the refueling outage.

The inspectors interviewed licensed operator training personnel, the responsible engineer in charge of the containment sump, and control room licensed operators. The licensee had conducted supplemental training for licensed operators to heighten awareness of the potential for emergency sump clogging by debris and indications of loss of flow due to sump blockage. This information was also included in plant emergency operating procedures.

The inspectors conducted a containment walkdown and sump screen inspection during the last Unit 1 outage to verify that the screens were free from gaps and obstructions.

b. Findings

No findings of significance were identified.

.3 Offsite Power System Operational Readiness (TI 2515/156)

a. Inspection Scope

The inspectors interviewed engineering, maintenance, and operations staff to collect data necessary to complete the Temporary Instruction (TI) 2515/156. This review was conducted to assess the operational readiness of the offsite power systems in accordance with NRC requirements such as Appendix A to 10 CFR Part 50, General Design Criterion (GDC) 17; Criterion XVI of Appendix B to10 CFR Part 50, Plant Technical Specifications (TS) for offsite power systems; 10 CFR 50.63; 10 CFR 50.65 (a)(4), and licensee procedures. Specifically, the inspectors discussed the licensee's processes for ensuring that the grid reliability conditions are appropriately assessed during periods of maintenance in accordance with the maintenance rule 10 CFR 50.65 (a)(4). Documents reviewed for this TI is listed in attachment.

b. Findings

No findings of significance were identified. Based on the inspection, no immediate operability issues were identified. In accordance with TI 2515/156 reporting requirements, the inspectors provided the required data to the headquarters staff for further analysis.

.4 <u>(Closed) URI 05000498/2004002-01: Inadequate Procedure Results in Relief Valve Opening</u>

<u>Introduction</u>. A Green noncited violation of Technical Specification 6.8.1.a and Regulatory Guide 1.33, Appendix A, was identified for an inadequate procedure that resulted in a letdown pressure relief valve opening during a letdown orifice swap.

<u>Description</u>. In order to reduce the risk of letdown isolation during a Train A emergency safety features (ESF) sequencer post maintenance test, the shift supervisor decided to place the "C" powered letdown orifice in service. The plant has three letdown orifices in the chemical and volume control system. The operators used Plant Operating Procedure 0POP03-CV-0004, ?Chemical and Volume Control System," Revision 34, to perform the evolution. Operators established that, to account for the difference in valve stroke times and to ensure that letdown would not be isolated, a full open indication on the middle sized orifice was needed before the large orifice was taken to close. This was believed to meet the intent of the procedure. The procedure outlined the following steps:

11.5 PERFORM the following simultaneously:
11.5.1 Open the letdown orifice isolation valve selected to be placed in service.
11.5.2 Close the letdown orifice isolation valve selected to be secured.

While the medium size orifice was going open, system pressure began to rise and operators attempted to control pressure using Pressure Control Valve PCV-0135. Pressure Control Valve PCV-0135 was not adjusted appropriately to compensate for its slow response. The valve response was slow and pressure in the system lifted the letdown Pressure Relief Valve PSV 3100 (600 psig lift point) to the primary relief tank. As soon as the medium size orifice indicated full open, the large orifice was closed and the system stabilized. Alternatively, the operators could have used other sections of this procedure to accomplish this task due to the concern of letdown isolation.

The chemical and volume control system is an interfacing system with the reactor coolant system (RCS). The maximum total capacity of these two letdown orifices is 250 gpm, which is approximately equal to the capacity of both centrifugal charging pumps operating in parallel. The maximum capacity of the relief valve is 343 gpm.

Operators failed to manipulate the letdown orifice isolation valve in a manner that properly controlled pressure in the chemical and volume control system. As a result, the letdown line 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 break could be isolated. Corrective actions for this event included enhancing the above procedure by adding notes and precautions and holding lessons learned sessions with operators.

<u>Analysis</u>. This finding is greater than minor because it had the actual impact of lifting a relief valve and therefore could be reasonably viewed as a precursor to a significant event. Because the finding was associated with one of the cornerstone attributes (barrier integrity) and affected the associated cornerstone objective, it surpassed the screening criteria of MC 0612, Appendix B. Under the MC 0609, Appendix A, Phase 1 screening of the

Significance Determination Process, the issue involved the initiating event cornerstone. Question #1, ?Does the finding contribute to the likelihood of a primary system or secondary system LOCA," was answered ?yes," because the letdown line relief valve that lifted could have failed to reseat or could have continually blown down if not isolated, resulting in a draindown path from the RCS to the primary relief tank. Consequently, the finding did not screen out in Phase 1.

A stuck-open relief valve in the letdown system would constitute a small-break LOCA. Therefore, under Phase 2, only the small break LOCA sequences were relevant to the finding. The initiating event frequency for the less-than-3-day (chosen because of the event's short duration) column was increased by 1 in each sequence. Using the counting rule, the Phase 2 resulted in a Green determination, but the analyst concluded the result was unreliable because the tool did not accurately model the event, particularly the probability that a small-break LOCA would occur. Therefore, a Phase 3 analysis was performed.

Under the Phase 3 analysis, the Region IV Senior Reactor Analysts evaluated several scenarios involving mechanical and human error failures that could result in the failure of the safety relief to close and/or letdown isolating contributing to the continued drain down of the RCS. The result indicated that the risk significance of the performance deficiency that caused the event was very low (Green). The event did not cross risk thresholds (as determined by the Phase 3 result) that would require consideration of external events or large early release.

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 0POP03-CV-0004, "Chemical and Volume Control System," Revision 34, was not properly maintained in that it was inadequate in that the guidance it provided allowed the letdown relief valve to open. The opening of the letdown relief valve increased the risk of an initiating event of an interfacing system small loss of coolant accident and degraded the reactor coolant system barrier integrity. Because this finding was entered into the licensee's Corrective Action Program as CR 04-1143 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 05000498/2004003-02, Inadequate Procedure Results in Relief Valve Opening.

.5 (Closed) Unresolved Item 05000498;499/2002003-01: Acceptability of using manual actions to operate equipment necessary for achieving and maintaining hot shutdown in lieu of providing protection for III.G.2 fire areas.

a. Inspection Scope

On March 22 through April 15, 2004, the inspector conducted a followup inspection of the issues related to this unresolved item for the same fire areas used in the original inspection (Fire Areas 1, 2, 3, 7 and 31) using Inspection Procedure 71111.05. Additional in-office reviews were conducted between April 15 and June 25, 2004. The inspector reviewed a third-party assessment of the manual actions being credited by the licensee. The assessment was intended to document the technical and regulatory bases, as well as demonstrate the feasibility of performing these actions within the times required. The inspector reviewed the four plant procedures used to implement fire response actions, including specific changes based on the third-party recommendations and validation packages for the manual actions for each fire area. The inspector reviewed the fire safe shutdown compliance strategies, thermal hydraulic analysis, operator action list, and implementing procedures for consistency and validity of the assumptions and credited equipment. The bases for manual actions were reviewed to determine the need for the action, as well as the technical reason for the time required for completion and the ability to meet that time with the minimum shift manning. The manual actions' intent was assessed to determine that manual actions were not inappropriately being credited in lieu of protecting the equipment from fire damage as required by 10 CFR Part 50, Appendix R. Specific documents reviewed are listed in the Attachment.

b. Findings

This unresolved item was closed. A Green finding was identified because the safe shutdown analysis did not address expected reactor coolant pump (RCP) seal leakage. When the analysis was re-performed, the plant response was determined to be acceptable.

b.1 Acceptability of the Use of Manual Actions in Fires

Introduction. The inspectors concluded that the manual actions being credited were approved during original plant licensing as an acceptable alternative to protecting the associated functions. The manual actions were implemented to prevent or mitigate potential spurious actuations, and did not involve restoring equipment that was required to have been protected.

Description. As part of its license application, the licensee submitted a list of equipment that could be subject to spurious operation as a result of fire damage. The actions needed to avoid or compensate for these potential spurious actuations, along with the times required to complete these actions, were submitted by the licensee as a list in an engineering report (5A019MFP001, ?Post Fire Operator Actions and Equipment Protection Requirements," (common name: Operator Actions List)). During the first triennial fire protection inspection at the site in May 2002 the team attempted to determine whether these actions could be performed as required. This could not be completed because there was insufficient documentation to show the basis of the acceptability of the required action completion times. The licensee agreed to provide additional information to support completing this unresolved item.

The licensee asserted, during the inspection, that the Operator Actions List (OAL) was intended and used as a procedure. The team identified concerns about the effectiveness of this document as a procedure, and whether it was adequate to implement the intended actions. In response, the licensee performed an extensive validation of the actions to verify that the actions were approved in the license basis, were needed to address potential fire damage in the specified fire area, and to establish a basis for the specified completion time. The actions for each of the fire areas reviewed by the NRC inspection team were walked down and timed to verify that the times could be met. The validation and walkdown were conducted by consultants.

The licensee concluded that the OAL was adequate to accomplish the necessary manual actions, but decided that it did not meet their current standards for procedure format and clarity. As a result, the licensee created a new procedure, Plant Operating Procedure 0POP04-ZO-0009, ?Safe Shutdown Fire Response," Revision 0, to incorporate all the actions of the OAL into a normal procedure format. The licensee then validated that the new procedure could also be performed as intended within the times specified in the OAL.

The inspectors reviewed the results of the validation efforts and the walkdowns of both the OAL and the new Plant Operating Procedure 0POP04-ZO-0009. The inspector also considered the results of training fire drills conducted by the licensee to assess the use of the OAL and the new procedure which had been observed by resident inspectors. The inspectors concluded that the actions listed in the OAL were approved by the NRC during original plant licensing, could be adequately understood and implemented by the available operators, and that completion times specified in the OAL could be met.

Analysis. The subject of this URI was determined to involve a substandard procedure format which did not necessarily inhibit the effective implementation, and therefore was considered a minor issue.

Enforcement. No violation of NRC requirements was identified. This URI is closed.

b.2 Safe Shutdown Analysis Did Not Account for Reactor Coolant Pump Seal Leakage

Introduction. A Green finding was identified because the fire safe shutdown analysis had not accounted for the impact of reactor coolant pump (RCP) seal leakage. The licensee re-performed the analysis, and was able to demonstrate that the plant response was acceptable.

Description. The licensee's safe shutdown analysis credited charging borated water for maintaining both reactivity control and reactor coolant inventory control functions. However, in a number of fire areas, charging was procedurally stopped to avoid damaging the charging pumps as a result of a spurious closing of either of the motor-operated volume control tank suction valves (CV-MOV-112B and CV-MOV-113C). Loss of the suction source would damage a running charging pump. The operator action list directed establishing charging within two hours. The inspector determined that there was no analytical basis for allowing charging to be secured this long. In particular, the analysis treated the reactor coolant system as a closed system without losses, and did not account for the RCP seal leakage expected under those conditions. Also, in Fire Area 3, RCP seal cooling was only crediting seal injection by charging, but the inspector identified that charging was manually secured by procedure. Loss of thermal barrier cooling coincident with no seal injection would result in rapid seal overheating and significantly increased seal leakage. This loss of inventory could be significant to the reactor coolant system during a two hour period without charging available.

In response to this issue, engineering performed a new analysis of the availability of thermal barrier cooling, and determined that one train of component cooling water was available to cool thermal barriers in all fire areas. Therefore, charging was not needed to perform the seal injection function. Engineering also re-analyzed the safe shutdown thermal hydraulic calculation, NC-7079, ?Fire Hazards Analysis," Revision 1, to demonstrate that the equipment and manual actions credited for fire safe shutdown would satisfy regulatory requirements for plant safety. The new analysis adequately demonstrated that the pressurizer would not empty until after two hours without charging.

Analysis. This issue was determined to be more than minor because it was similar to Example 3.i of Manual Chapter 0612, Appendix E in that the analyses had to be re-performed to assure that the acceptance criteria were met. This issue affected the Mitigating Systems Cornerstone because it related to the availability of charging when it was required to mitigate the effects of a fire. This issue was determined to have very low safety significance because it involved a design deficiency confirmed not to result in a loss of function.

Enforcement. Because the licensee was able to re-perform the safe shutdown analyses and demonstrate that the plant could meet its fire safe shutdown design without charging or seal injection for two hours, no violation of NRC requirements existed.
FIN 05000498;499/2004003-3 ; Fire safe shutdown analysis did not account for the impact of reactor coolant pump seal leakage.

.6 (Closed) Unresolved Item 05000498;499/2002003-02: Availability of diagnostic instrumentation during a fire to allow operators to identify mal-operation of fire-affected plant components and take manual actions to mitigate the consequences.

This URI was intended to assess the licensee's assertion that operators were expected to identify fire-induced mal-operations and respond to them, rather than to implement all the manual actions assigned to a given fire area preemptively. However, the licensee subsequently reconsidered their fire safe shutdown design and concluded that the intent of the design was to attempt to avoid spurious operations if possible, and mitigate the rest. Training was conducted for operators to reinforce this point. Therefore, the inspector reviewed the OAL and new Plant Operating Procedure 0POP04-ZO-0009 to determine whether the indications needed to perform this new procedure were protected from fire damage for Fire Areas 1, 2, 3, 7, and 31. In some instances, instrumentation was required to assure proper equipment operation, so the inspector verified, on a sample basis, that this instrumentation was free of fire damage and clearly specified as reliable in the above documents. In some cases, Plant Operating Procedure 0POP04-ZO-0009 specified that operators should obtain information from instruments which were not assured to be free of fire damage, but in each case the information was available from an alternate source. The licensee was addressing these examples in Condition Report 04-4702. However, these examples did not affect the ability to reach and maintain a safe shutdown condition. No issues or violations were identified. This item is closed.

4OA6 Meetings, Including Exit

The results of the inservice inspection activities inspection were presented to Mr. Tom Jordan, Vice President Engineering and Technical Services, and Mr. Gary Parkey, Vice President of Generation, and other members of licensee management on April 15, 2004.

The results of the radiation safety inspection were presented to Mr. James Sheppard, President and Chief Executive Officer, and other members of his staff who acknowledged the findings on April 16, 2004.

The results of the resident inspection were presented to Mr. Tom Jordan, Vice President Engineering and Technical Services, and other members of licensee management on July 1, 2004.

The results of the fire protection followup inspection results were presented to Mr. James Sheppard and members of his staff, who acknowledged the findings on April 15, 2004. On July 7, 2004, an additional exit meeting was conducted with Mr. Tom Jordan and members of his staff by telephone.

In each case, the inspectors asked the licensee representatives whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

Other Meetings

On April 22, 2004, Mr. Mark Satorious, Deputy Director, Division of Reactor Projects, Region IV, toured the plant and visited with licensee management.

40A7 Licensee-identified Violations

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

> 10 CFR 26.24.(a).(3) specified, in part, in order to deter and detect substance abuse, the licensee shall implement a for-cause chemical testing program that includes testing after receiving credible information that an individual is abusing drugs or alcohol. 10 CFR 26.20 specified, in part, each licensee shall implement written procedures to meet the specific performance objectives of this part. Plant General Procedure 0PGP09-ZA-0002, ?Fitness For Duty Program Procedure," Revision 13, Section 6.7.2, required supervision to request an immediate drug or alcohol test for any individual exhibiting behavior suggesting a lack of ?fit for duty." Section 6.7.2.2 specified, in part, the immediate supervisor, or designee, is responsible for escorting the individual to the fitness-for-duty center. The licensee provided information in fitness-for-duty training that the above actions shall be taken when a report is received that an individual may be unfit for duty. However, on April 2, 2004, a violation occurred when a supervisor failed to escort an individual to

the fitness-for-duty center after receiving multiple reports of an alcohol smell.

A licensee investigation determined that poor communication among supervisors resulted in the wrong individual being tested. The finding was entered into the corrective action program as Condition Report 04-4746. The licensee took appropriate corrective actions against the individual. Using the Interim Physical Protection Significance Determination Process, the violation was determined to be of very low safety significance. Although a supervisor failed to perform the required actions related to fitness-for-duty testing under the behavior observation program, there was no malevolent intent and no greater than two similar findings in the last 4 quarters.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

R. Aguilera, Supervisor, Health Physics Division
C. Albury, Supervisor, Design Engineering
M. Berg, Manager Testing/Programs Engineering
W. Bullard, Manager, Health Physics
K. Coates, Manager, Maintenance
J. Conly, Engineer Licensing Staff
F. Cox, Design Engineer
J. Crenshaw, Manager, Plant Engineering
L. Earls, Instrument Health Physicist, Health Physics Division
R. Gangluff, Manager, Chemistry
C. Grantom, Manager, PRA
E. Halpin, Manager, Plant General
J. Haning, Test Engineer
E. Heacock, Senior Engineer
S. Head, Manager, Licensing
K. House, Supervisor, Plant Design Engineering
M. Johnson, Specialist, Senior Licensing
T. Jordan, Vice President, Engineering and Technical
J. Loya, Engineer Licensing
M. Ludwig, Supervisor, Operations Quality
C. McIntyre, Steam Generator Engineer
A. Mikus, Supervisor, Communication and Public Affairs
A. Moldenhauer, Staff, PRA Engineer
W. Mookhoek, Senior, Engineer Licensing Staff
G. Parkey, Vice President, Generation
U. Patel, Senior Engineer
M. Polishak, Manager, SED
G. Powell, Manager, Operating Experience Group
D. Rohan, Licensed Operator, Operations Support
W. Russell, Procedure Supervisor
R. Savage, Senior Staff Specialist
J. Sheppard, President and CEO
L. Spiess, Nondestructive Examination Level III
J. Stauber, Repair and Replacement Engineer
D. Swett, Supervisor, Health Physics Division
K. Taplett, Licensing Engineer
S. Thomas, Manager, Engineering Projects
D. Towler, Manager, Quality
T. Walker, Manager, Quality
C. Younger, Inservice Inspection Engineer

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Open

05000498;499/2004003-01	NCV	Two examples of failure to control high radiation areas (Section 2OS1)
05000498/2004003-02	NCV	Inadequate procedure results in relief valve opening (Section 4OA5.4)
05000498;499/2004003-03	FIN	Fire safe shutdown analysis did not account for the impact of reactor coolant pump seal leakage. (Section 4OA5.5.b.2).

Closed

05000498;499/2004003-01	NCV	Two examples of failure to control high radiation areas (Section 2OS1)
05000498/2004003-02	NCV	Inadequate procedure results in relief valve opening (Section 4OA5.4)
05000498;499/2004003-03	FIN	Fire safe shutdown analysis did not account for the impact of reactor coolant pump seal leakage (Section 4OA5.5.b.2)
05000498;499/2002003-01	URI	Acceptability of Using Manual Actions to Operate Equipment Necessary for Achieving and Maintaining Hot Shutdown in Lieu of Providing Protection for III.G.2 Fire Areas (Section 4OA5.5)
05000498;499/2002003-02	URI	Availability of diagnostic instrumentation during a fire to allow operators to identify mal-operation of fire-affected plant components and take manual actions to mitigate the consequences (Section 4OA5.6)
05000499/200303-00	LER	Standby Diesel Generator 22 Failure (Section 4OA3.1)
05000499/200303-01	LER	Standby Diesel Generator 22 Failure (Section 4OA3.1)
05000498/2004002-01	URI	Inadequate Procedure Results in Relief Valve Opening (Section 4OA5.4)

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:

Documents Reviewed for Section 1R08

Certification Reports

Magnaflux Certification Reports for Batches 94E03K. 96C06K, and 01E02K
Krautkramer Transducer Serial Number 45168 Certification
Ultragel Certification Report for Lots 01243 and 95343

Condition Reports

02-03544	03-05358	03-15381	03-17436	04-02458
02-16758	03-14364	03-17194	04-03604	04-4960

Magnetic Particle Examination Reports

97-0054
2004-00144
2004-00145

Procedures

Number	Procedure	Revision
MRS-2.4.2 GEN-35	Eddy Current Inspection of Preservice and Inservice Heat Exchanger Tubing	11
MRS-GEN-1072-ANSER	ANSER™ Auto Data Screening (ADS) User Manual	1
MRS-GEN-1127	Guidelines for Steam Generator Eddy Current Data Quality Requirements	1
MRS-SSP-1619-TGX/THX	Steam Generator Eddy Current Data Analysis Guidelines for Inservice Inspection at South Texas Units 1 and 2	0
OPEP10-ZA-0004	General Ultrasonic Examination	2
OPEP10-ZA-0012	Color Contrast Solvent Removable Liquid Penetrant Examination for ASME XI Preservice Inspection/Inservice Inspection	

Number	Procedure	Revision
OPEP10-ZA-0018	Dry Power Magnetic Particle Examination for ASME XI Preservice Inspection/Inservice Inspection	1
SGO-01-62	ANSER Auto Analysis Supplemental Training	0
UTI-PDI-UT-2	Performance Demonstration Initiative Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds	1
WEC 9.2	Qualification, Training and Certification of Nondestructive Testing Personnel	2

Qualification Reports

Certificates of Personnel Qualification No 0002406, 0004408, 0004922, 0007040, and 0007041
Performance Demonstration Initiative Program Qualification Nos. 499 and 510

Penetrant Examination Reports

1993-0158
2004-036

Reports

"Preservice Inspection Summary Report for Tubing in the Replacement Steam Generators," dated February 2003

Ultrasonic Examinations Test Reports

110138	93-0586	2004-017	2004-022
181003	93-0587	2004-021	2004-026

Work Orders

WRWO RC-2-428879, Replacement of Number One Seal Housing Gasket on Number Two ?C" Reactor Coolant Pump

WRWO CV-2-424506, 2-CV-0468 Letdown Isolation Valve Body to Bonnet Leak Welding Repair

Miscellaneous

Licensee Letter NOC-AE-000689, dated December 30, 1999, Relief Request RR-ENG-2-16

Licensee Letter NOC-AE-000823, dated April 17, 2000, Supplement to Relief Request

RR-ENG-2-16

Licensee Letter NOC-AE-01001181, dated September 18, 2001, Relief Request RR-ENG-38

NRC Letter ST-AE-NOC-01000769, dated February 5, 2003, Safety Evaluation associated with Relief Request RR-ENG-2-16

NRC Letter ST-AE-NOC-02000915, dated January 24, 2002, Safety Evaluation associated with Relief Request RR-ENG-38

Electronic Data Base for Site Specific Exam Results

Documents Reviewed for Section 2OS1

Radiation Work Permits

2004-2-0051	Decon of Reactor Cavity, LISA, ICSA, and Tilt Pit in Support of 2RE10
2004-2-0110	Room 003 Replacement of Thimbles and EGS Seal Fittings
2004-2-0112	Install Freeze Seals on Guide Tubes to Support Replacement of Thimbles and Incore Fittings - 2RE10

Procedures

0PEP02-ZM-0009	Spent Fuel Pool Storage and Work, Revision 4
0PGP03-ZR-0051	Radiological Access and Work Controls, Revision 18
0PRP04-ZR-0013	Radiological Survey Program, Revision 15
0PRP04-ZR-0015	Radiological Posting and Warning Devices, Revision 16
0PRP07-ZR-0009	Performance of High Exposure Work, Region 19

Condition Reports

03-7554, 03-8102, 03-9247, 03-11713, 03-11889, 03-12031, 03-13269, 03-14511, 04-2830,04-4787, 04-5020

Audits

Quality Audit Report 03-13, Radiological Controls/Radwaste Program

Documents Reviewed for Sections 4OA5.5 and 4OA5.6:

Condition Reports:

	02-17831	02-18434
02-9239	02-17837	02-18951
02-5491		03-3998
02-7771		03-16606
02-9239		03-16608
02-11442		03-17841

04-3341	04-4663
04-3808	04-4670
	04-4702
	04-5487
	04-5490
	04-5491
	04-5493
	04-6033

Procedures:

0POP04-ZO-0001, Control Room Evacuation, Revisions 17 and 23

0POP04-ZO-0008, Fire/Explosion, Revisions 6 and 9

0POP04-ZO-0009, Safe Shutdown Fire Response, Revisions 0 and 1

0PGP03-FP-0001, Safe Shutdown Fire Methodology and Operations, Revision 0

0POP01-ZA-0017, Emergency Operating Procedure Revision and Implementation, Revision 10

Calculations:

5A011MC6023, Appendix R Evaluation, Revision 9

NC-7079, Fire Hazards Analysis, Revisions 0 and 1

Framatome Report 00474.00.0006-01, Manual Action Validation and Timeline Analysis, Revision 0

5A019MFP001, Post Fire Operator Actions and Equipment Protection Requirements, Revisions 10 and 12 (common name: Operator Actions List)

Instruments Reviewed for Section 4OA5.6:

A1NINE0045, extended range nuclear instrument
A1RCMOV0001A, valve position indication
N1CVTI0216, reactor coolant pump seal injection temperature
A1RCPCV-655A, pressurizer power operated relief valve position indication
C1RCLT0468, pressurizer level indication
B1MSPT7421, main steam pressure indication
B1RCHCV0602, B1RCHV3657B, and B1RCHV3658B, reactor head vent valve position indications
C1AFFT7523, auxiliary feedwater flow indication
C1SILI0931, refueling water storage tank water level indication
C1RCPT0407, reactor coolant wide range pressure indication

LIST OF ACRONYMS

ALARA	As Low As is Reasonably Achieved
CFR	*Code of Federal Regulations*
CR	condition report
ESF	emergency safety features
LER	licensee event report
NCV	noncited violation
OAL	Operator Actions List
PORV	power operated relief valve
RCP	reactor coolant pump
RCS	reactor coolant system
RPV	reactor pressure vessel
RWST	refueling water storage tank
URI	unresolved item
WAN	work authorization number