UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

January 14, 2002

William T. Cottle, President and
Chief Executive Officer
STP Nuclear Operating Company
P.O. Box 289
Wadsworth, Texas 77483

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 - NRC INTEGRATED INSPECTION REPORT 50-498/01-04; 50-499/01-04

Dear Mr. Cottle:

On December 6, 2001, the NRC completed an inspection at your South Texas Project, Units 1 and 2. The enclosed report documents the inspection findings, which were discussed on December 6, 2001, with Mr. J. Sheppard and other members of your staff. An additional telephonic exit was conducted January 14, 2002, to provide you our resolution for unresolved issues.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC has identified two findings that were evaluated under the risk significance determination process as having very low safety significance. One finding involved the failure to identify and implement corrective action to address excessive moisture in the hydraulic operating system for a steam generator power-operated relief valve in Unit 2. The second finding involved failure to establish measures to assure that the design and licensing bases for the auxiliary feedwater systems would support the safety analysis for both units. The NRC has also determined that violations are associated with this issue. Because of the very low safety significance, the violations are being treated as noncited violations, consistent with Section VI.A.1 of the Enforcement Policy. If you deny the noncited violations, 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; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the South Texas Project facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be 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**/

Charles S. Marschall, Chief
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Dockets: 50-498; 50-499
Licenses: NPF-76; NPF-80

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	50-498; 50-499
Licenses:	NPF-76; NPF-80
Report Nos:	50-498/01-04; 50-499/01-04
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:	November 5 through December 7, 2001
Team Leader:	J. E. Whittemore, Senior Reactor Inspector, Engineering Maintenance Branch
Inspectors:	L.E. Ellershaw, Senior Reactor Inspector, Engineering Maintenance Branch
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Accompanying Personnel:	Michael Shlyamberg, Consultant, Nuenergy, Inc.
Approved By:	C. S. Marschall, Chief Engineering Maintenance Branch

Attachment: Supplemental Information

SUMMARY OF FINDINGS

IR 05000498-01-04; IR 05000499-01-04; on 11/05-12/07/2001; STP Nuclear Operating Company; South Texas Project; Units 1 and 2 safety system design and performance capability, evaluation of changes, tests, or experiments.

The inspections were conducted by five regional inspectors and one contractor. The inspection identified two green findings, which were characterized as noncited violations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) and determined by using IMC 0609, "Significance Determination Process (SDP)." Findings for which the significance determination process does not apply are indicated by "No Color" or by the severity level of the applicable violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

Cornerstone: Mitigating Systems

- Green. The team determined that the licensee failed to identify the need for and implement corrective action to address the degraded condition of Steam Generator Power-Operated Relief Valves, 2MSPV7411, 2MSPV7431, and 2MSPV7441 for a period of four weeks, until prompted by the inspection team. The licensees corrective action program did not promptly evaluate the out-of-specification condition of the electrohydraulic fluid for the steam generator power operated relief valves. Analysis results received in early November for oil samples drawn in late October 2001 were not reviewed and assessed by the licensee's engineering staff until December 6, 2001, when questioned by the inspection team. Three sample results exceeded the licensee's criteria. This was a violation of Criterion XVI of Appendix B to 10CFR50, Corrective Action, which requires that conditions adverse to quality be promptly identified and corrected.

  The safety significance of this condition is very low as the licensee performed an evaluation to determine that the valves were operable, and the evaluation was accepted by the team. Since the licensee entered this finding into their corrective action program in Condition Reports 2001-19637,-19641, and -19642, this violation is being treated as a noncited violation in accordance with Section VI.A.1 of the Enforcement Policy. (Section 1R21.3.b)

- Green. The measures established by the licensee to assure that the current design would support the safety analysis were not adequate. The team identified a failure to verify the adequacy of the plant design in both units to support the current safety analysis for a loss of normal feedwater event. The failure of the A Train Emergency Diesel Generator with a loss of offsite power could result in the loss of two of the four auxiliary feedwater pumps. The safety analysis for loss of normal feedwater assumes that three pumps will be available. The Train D (Turbine-Driven) Pump cannot be assumed to be available as the essential power for the Train D pump room cooling is supplied from Train A essential power which also supplies the Train A (Electric-Driven) pump. This was identified as a violation of Criterion III of Appendix B to 10 CFR Part 50.

The licensee performed an evaluation which concluded that the Train D Pump would perform its safety function at the predicted elevated room temperature for the required mission time. The licensee had previously installed administrative requirements to assure that three pumps would be operable when required. Because of the very low safety significance, and because the licensee has included the item in their corrective action program as Condition Reports 2001-19586 and 2000-19700, this design control violation is a noncited violation (NCV 50-498/01-04-02; 50-499/01-04-02). in accordance with Section VI.A of the Enforcement Policy. (Section 1R21.5b)

Report Details

1 REACTOR SAFETY

Introduction

A team inspection was performed to verify that facility safety system design and performance capability was adequate and that the initial design and subsequent modifications have preserved the current design basis of the systems selected for review. The scope of the review also included any necessary nonsafety-related structures, systems, and components that provided functions to support safety functions. The inspection effort also reviewed the licensee's programs and methods for monitoring the capability of the selected systems to perform the current design basis functions. This inspection verified through sampling, the inspectable aspects of the initiating events, mitigating systems, and barrier cornerstones.

The probabilistic risk assessment model for South Texas Project units is based on the capability of the as-built safety systems to perform their intended safety functions successfully. The area and scope of the inspection were determined by reviewing the licensee's probabilistic risk analysis models to identify the most risk significant systems, structures, and components according to their ranking and potential contribution to dominant accident sequences and/or initiators. Deterministic effort was also applied in the selection process by considering recent inspection history, recent problem area history, and all modifications developed and implemented. The team reviewed in detail the auxiliary feedwater system and the steam generator power-operated relief valves in both units. The primary review prompted parallel review and examination of support systems, such as, electrical power, instrumentation, room cooling systems, and related structures and components.

The objective of this inspection was to assess the adequacy of calculations, analyses, engineering processes, and engineering and operating practices that were used to support the performance of the safety systems selected for review and the necessary support systems during normal, abnormal, and accident conditions. Acceptance criteria utilized by the NRC inspection team included NRC Regulations, the technical specifications, applicable sections of the Updated Final Safety Analysis Report, applicable industry codes and standards, as well as industry initiatives implemented by the licensee's programs.

The annual inspection to assess the continuing performance of the licensee's program to meet the regulatory requirements of 10 CFR Part 50.59, "Changes, Tests, And Experiments," was also conducted by one member of the team, during the first week of the inspection.

1R02 Evaluation of Changes, Tests, and Experiments (71111.02)

a. Inspection Scope

The team reviewed a selected sample of 10 safety evaluations to verify that the licensee had appropriately considered the conditions under which the licensee may make changes to the facility or procedures or conduct tests or experiments without prior NRC approval. In addition, the team reviewed Procedure 0PGP05-ZA-0002, "10 CFR 50.59 Evaluations," Revision 11, which implemented the new safety evaluation program and was effective August 1, 2001.

The team reviewed a selected sample of 12 safety evaluation screenings, in which the licensee determined that safety evaluations were not required, to ensure that the licensee's exclusion of a full evaluation was consistent with the requirements of 10 CFR 50.59, "Evaluations of Changes, Tests, or Experiments."

The team reviewed 10 condition reports initiated by the licensee that addressed problems or deficiencies associated with 10 CFR 50.59 requirements to ensure that appropriate corrective actions were being taken. The team also reviewed the licensees on going self-assessment to ensure that problems or deficiencies were appropriately addressed.

b Findings

No findings of significance were identified.

1R21 Safety System Design and Performance Capability (71111.21)

.1 System Requirements

a. Inspection Scope

The team reviewed the following attributes for the auxiliary feedwater system and the steam generator power operated relief valves: (1) process medium (water, steam, and air), (2) energy sources, (3) control systems, and (4) equipment protection. The team then verified that procedural instructions to operators were consistent with operator actions required to meet, prevent, and/or mitigate design basis accidents. The review also considered requirements and commitments identified in the Updated Final Safety Analysis Report, technical specifications, design basis documents, and plant drawings. These reviews further verified that required support functions for the auxiliary feedwater system and the steam generator power-operated relief valves would be available.

The team verified that the system needs for the auxiliary feedwater system and the steam generator power-operated relief valves were met. The supply of air, water, steam, and electrical power required by the technical specifications were verified through a review of the design of the motor driven and steam driven auxiliary feedwater pumps, the steam generator power-operated relief valves, and systems providing support functions.

The team verified equipment for the steam generator power-operated relief valves and auxiliary feedwater systems required to operate and/or change state during accidents and events would have control power available. The team further reviewed the adequacy of alarm setpoints and verified that necessary instrumentation and alarms were available to operators for making necessary decisions in coping with postulated accident conditions. In addition, the team verified that systems' standby alignments were consistent with assumptions in the operating procedures as well as design and licensing basis assumptions.

b. Findings

No findings of significance were identified.

.2 System Condition and Capability

a. Inspection Scope

The team reviewed the periodic testing procedures for the auxiliary feedwater system and the steam generator power-operated relief valves to verify that the design requirements were adequately demonstrated. The team reviewed the environmental qualification of a sample of system components to verify the capability to operate under design environmental conditions and the assumed operating parameters including: voltage, speed, power, flow, temperature, and pressure. The team also reviewed recent instrument setpoint changes to verify that the design basis or capability for the selected systems had not been affected by the setpoint change process.

The team also reviewed the systems' operations by conducting system walkdowns; reviewing normal, abnormal, and emergency operating procedures; and reviewing the Updated Final Safety Analysis Report, technical specifications, design calculations, drawings, and procedures. In addition, the team then reviewed the list of active and closed standing orders and operator work-arounds to ensure no design assumptions were invalidated by past or current operator daily practices.

b. Findings

No findings of significance were identified.

.3 Identification and Resolution of Problems

a. Inspection Scope

The team reviewed a sample of problems identified by the licensee in the corrective action program to evaluate the effectiveness of corrective actions related to design issues. The sample included open and closed condition reports going back three years that identified issues related to or affecting the systems and safety-related setpoint issues. The team also reviewed Procedure OPGP03-ZX-0002, "Corrective Action Program," Revision 4. The specific corrective action documents that were sampled and reviewed by the team are listed in the attachment to this report. Inspection

Procedure 71152, "Identification and Resolution of Problems," was used as guidance to perform this part of the inspection.

The issues addressed by the condition reports reviewed included:

- The disposition of technical specification interpretations to address system and component operability,
- The identification and correction of configuration control events and errors,
- The identification and correction of issues related to testing failures,
- The identification and corrective action associated with personnel errors, primarily in the operations area,
- The identification and correction of safety-related setpoint issues, and
- The identification and correction of apparently degraded equipment.

b. Findings

Oil Sampling Program

The team identified a noncited violation of Criterion XVI of Appendix B to 10 CFR 50, with very low safety significance (Green) for the licensee's failure to fully implement the requirements of the oil sampling program.

The licensee implemented a hydraulic oil sampling program for the steam generator power-operated relief valve hydraulic oil actuating system, due in part to problems identified in Licensee Event Reports 50-499/90-011-00 and 50-499/91-007-01. The team reviewed the latest test results for samples analyzed from October 29 to November 8, 2001 and determined that the licensee's sampling program for the steam generator power-operates relief valves did not promptly identify conditions that were out of specification. The oil sample results were not reviewed by the licensee until questioned by the team on December 6, 2001. There team noted that results of three oil samples exceeded the licensee's criteria, specifically:

Steam Generator 2A Power-Operated Relief Valve 2MSPV7411 slightly exceeded particulate count specification,

Steam Generator 2C Power-Operated Relief Valve 2MSPV7431 exceeded entrained moisture specifications and viscosity specifications, and

Steam Generator 2D Power-Operated Relief Valve 2MSPV7441 sightly exceeded the acidity specification.

The team requested the licensee to assess the operability of Steam Generator Power-Operated Relief Valve 2MSPV7431, as this sample had significantly exceeded the allowable moisture specification of 1500 ppm to a value of approximately 2800 ppm. The licensee's Condition Report Engineering Evaluation(CREE) 01-19641-2 determined that steam generator Power-Operated Relief Valve 2MSPV7431 was degraded, but operable. The basis for this operability conclusion was information from the valve vendor that this moisture concentration would not result in performance degradation unless the oil acidity was out of specification. The sample analysis results showed that the acidity was only 10 percent of the specification. The other two valves were just slightly out of specification for particulate count and acidity as noted above. The team agreed that the three power-operated relief valves were degraded, but operable. The licensee initiated corrective action to resample and monitor the condition of the oil until it could be brought into specification or changed.

The team assessed this condition in accordance with Appendix B of NRC Inspection Manual Chapter 0610*, "Power Reactor Inspection Reports." The team determined that this condition had a credible impact on safety and the issue was more than minor since contaminants in the power-operated relief valve hydraulic fluid previously resulted in test failures. The team also concluded that the issue affected the mitigating system cornerstone since the steam generator power-operated relief valves are safety-related equipment required to mitigate a design basis event, and the safety function could have been impacted. Therefore, the Significance Determination Process, as described in NRC Inspection Manual Chapter 0609 was entered.

Using Phase 1 of the Significance Determination Process, the team determined that only the mitigating systems cornerstone was affected and there was no actual loss of safety function as the power-operated relief valves remained operable. Therefore, the problem had very low safety significance (Green).

The licensee's procedure, 0PGP03-ZM-004, "Lubrication Program," revision 13 states that the licensee engineering staff is responsible for initiating corrective actions for out-of-specification lubricant analysis results.

Criterion XVI of Appendix B to 10 CFR Part 50 states, in part, that "[m]easures shall be established to assure that conditions adverse to quality . . . are promptly identified and corrected." Contrary to the above, the team determined that the licensee failed to promptly identify the need for and implement corrective action to address the degraded condition of Power-Operated Relief Valve 2MSPV7431 until prompted by the inspection team, for a period of four weeks. However, because of low safety significance and because the licensee has included the item in their corrective action program as Condition Reports 2001-19637, 2001-19641 and 2001-19642, this violation is being treated as a noncited violation (50-499/0104-01) in accordance with Section VI.A.1 of the Enforcement Policy.

.4 System Walkdowns

a. Inspection Scope

The team performed walkdowns of the accessible portions of the auxiliary feedwater system and the steam generator power-operated relief valves, as well as the required support systems. The walkdowns focused on the installation and configuration of power supplies, piping, components, and instruments. During the walkdowns, the team assessed:

- The placement of protective barriers and systems,
- The susceptibility to flooding, fire, or environmental conditions,
- The physical separation of trains and the provisions for seismic concerns,
- Accessibility and lighting for any required local operator action,
- The materiel condition and preservation of systems and equipment, and

Finally, the team assessed the conformance of the currently installed system configurations to the current design and licensing bases.

b. Findings

No findings of significance were identified.

.5 Design Review

a. Inspection Scope

The team reviewed the current as-built instrument and control, electrical, and mechanical design of the auxiliary feedwater systems and the power-operated relief valves. These reviews included a review of design assumptions, calculations, required system thermal-hydraulic performance, electrical power system performance, protective relaying, and instrument setpoints and uncertainties. The team also performed a single failure review of individual components to determine the effects of such failures on the capability of the systems to perform their design safety functions.

The inspectors reviewed the steam generator power-operated relief valves and the auxiliary feedwater systems for both units, including a review of calculations, drawings, specifications, vendor documents, Updated Final Safety Analysis Report, technical specifications, emergency operating procedures, and temporary and permanent modifications. The team specifically reviewed the auxiliary feedwater control valves and the instrumentation and control aspects for the steam generator power-operated relief valve.

b. Findings

The design of the South Texas Project incorporated four auxiliary feedwater pumps in each unit, consisting of one steam turbine-driven pump and three electric motor-driven pumps using safety-related essential power. Trains A, B, and C are electric driven and all electrical power is supplied from the respective train. Train D pump is steam turbine-driven and all required instrument and control power and other support power, such as pump room cooling is supplied from A Train power. The team’s review of the design basis identified a condition where the auxiliary feedwater system current design would not support the safety analysis for the loss of normal feedwater flow concurrent with a loss of offsite power. The team determined that the associated failure to verify the adequacy of design was a noncited violation of Criterion III of Appendix B to 10 CFR 50, with very low safety significance (Green).

Use of Three Auxiliary Feedwater System Pumps

Prior to March 2000, the licensee's safety analyses for all credible events postulated that only two auxiliary feedwater pumps were needed to provide adequate heat removal to preclude the pressurizer from becoming completely filled with water during a loss of normal feedwater with a loss of offsite power. On March 1, 2000, the licensee identified a potential for the pressurizer becoming completely filled with water (over-filled) following a loss of normal feedwater due to the flow from the charging pumps. An assumed limiting condition in the analysis for loss of normal feedwater flow with a loss of offsite power analysis is that the pressurizer will not become over-filled. This condition could pressurize the reactor coolant system such that water relief through the pressurizer safety valves, which are not licensed to pass water, would occur. Flowing water through these valves could result in a stuck open safety valve. This would constitute a small break loss of coolant event, which is not an acceptable consequence for the loss of feedwater event. The pressurizer power operated relief valves are qualified to pass water, which would protect the safety valves, however the plant is not licensed to credit the power-operated relief valves.

The problem was documented in Condition Report 2000-03229. This resulted in the performance of Condition Report Engineering Evaluation CREE 00-3229-1, which further resulted in Unreviewed Safety Question Evaluation (USQE) 00-3229-20. This evaluation quoted the design basis as assuming that two auxiliary feed water pumps would be automatically available and feed two steam generators before reaching dryout conditions in the unfed steam generators in 47 minutes. The licensee considered this to be adequate time to correct any condition that would lead to a loss of heat sink event. Therefore, the licensee consulted with the NSSS vendor and conducted an evaluation that took credit for operator actions not previously assumed and the availability of four steam generators. The licensee made procedure and policy revisions to require the necessary operator actions.

The initial problem related to the pressurizer overfill was addressed by a plant modification to preclude automatic starting of the charging pumps. Subsequently, the NRC issued a noncited violation related to the charging pump pressurizer overfill issue. The failure to properly incorporate the licensing basis into the plant as-built design was identified by the inspectors as a violation of 10 CFR Part 50, Appendix B, Criterion III, (NCV 50-498/00-07-01; 50-499/00-07-01).

In February 2001, subsequent analysis was performed to evaluate the reactor coolant system heatup for the maximum plugged steam generator tube condition in both units. On February 6, 2001, this analysis identified that successful mitigation of the loss of normal feedwater with loss of offsite power required flow from three auxiliary feedwater pumps for 1-2 minutes duration early (15-20 minutes) in the event, to prevent the pressurizer from becoming over-filled, even without the charging pump flow contribution. The additional auxiliary feedwater flow is to remove heat from the reactor coolant system. Insufficient heat removal causes thermal expansion of reactor coolant, which can also overfill the pressurizer.

The current technical specifications required that with two auxiliary feedwater pumps inoperable, the unit be placed in at least hot standby withing 72 hour, and hot shut down within an additional 6 hours. The licensee initiated Condition Report 2001-02103. The immediate solution was to require three operable auxiliary feedwater pumps in Modes 1, 2, and 3 in order to limit reactor coolant system heatup and coolant expansion early in this particular event. This change would support the loss of feedwater/offsite power analysis in that the predicted reactor coolant system heatup and expansion would not occur early in the event, with a third pump available and operating to provide adequate cooling.

Upon the initial discovery of the need for the three auxiliary feedwater pumps the licensee issued Operability Assessment OAS 4982 dated April 2001 for Technical Specification 3.7.1.2 that is tracked under Condition Report 2000-3229. Since the technical specifications were nonconservative, the licensee also initiated Condition Report Corrective Action Item 2001-2103-03 requiring submission of a Technical Specification amendment request to the NRC. Finally, OAS 4982 imposed the following actions for operators:

> Until the required technical specification revision is issued, the following administrative action statements for the auxiliary feedwater pump limiting condition for Operation 3.7.1.2 shall be complied with. These changes support the loss of feedwater and pressurizer overfill accident analysis.
>
> a) With any one auxiliary feedwater pump inoperable, restore the affected auxiliary feedwater pump to operable status within 72 hours or be in at least hot standby within the next 6 hours and in hot shutdown within the following 6 hours. The provisions of TS 3.0.4 are not applicable for entry into mode 3 for the turbine driven auxiliary feedwater pump.
>
> b) With any two or any three auxiliary feedwater pumps inoperable, be in at least hot standby within 6 hours and in hot shutdown within the following 6 hours.
>
> c) With four auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to operable status as soon as possible”.

When the initial pressurizer overfill problem (related to the charging pump flow) was identified the safety evaluation for this discovery was documented in USQE 2000-3229-20 which was approved by the Plant Operations Review Committee on April 19, 2000. The current version of USQE 2000-3229-20 addressed the condition documented in Condition Report 2001-2103 in an attachment to USQE 2000-3229-20, dated February 22, 2001. The team’s review of the Condition Reports and the current USQE 00-3229-20, identified the following.

- Review of the Condition Report 2000-03229 and related records did not indicate that this latest change to the USQE received a subsequent Plant Operations Review Committee review.

- An assigned action in Condition Report 2001-02103 was to prepare and submit a Technical Specification amendment request to the NRC by December 3, 2001.

The team understood that this action had not been completed on December 7, 2001, when the inspection team departed the site. During past onsite review, the team was made aware that the request had been submitted on December 3, 2001.

The licensee's evaluation had identified the requirement for three operable auxiliary feedwater pumps and postulated that with a single failure of the Train A Essential Bus during a loss of offsite power, the Train D pump would be operable because of prescribed operator actions within 47 minutes. During the course of the inspection, the team reviewed licensee’s Calculation MC06455, "AFW Pump D Room Heatup," Revision 0. The current design basis assumes the Train D turbine-driven pump room temperature to be less than 170°F. The maximum room temperature of 170°F. was required to support the environmental qualification for the Train D pump trip solenoid valves. The team’s review and evaluation of the licensee’s calculation MC06455 identified that the room temperature could exceed this value. The calculation contained some invalid assumptions, the most notable being that normal Train D pump room ventilation was available. However, the D Train pump room safety-related cooling and ventilation system is supplied by train A essential power. Therefore, during a loss of normal feedwater concurrent with a loss of offsite power, a failure of the Train A Emergency Diesel Generator would cause two auxiliary feedwater pumps, Train A and D, to become inoperable. Because of the erroneous calculation, the licensee's evaluation had failed to consider the total effect of losing Train A power during the analyzed event.

The licensee did not dispute the team’s conclusions with regard to the calculation or the potential for a degraded mitigation system, and initiated Condition Report 2001-19681. The team asked licensee management to evaluate the operability of the D pump, based on the conditions identified by the team. The licensee conducted an operability assessment to determine if the affected equipment was operable and provided documentation of an operability review in Condition Report Engineering Evaluation 01-19681-1. This document assured the operability of the turbine trip solenoid valves on the basis of information in a vendor test report which verified that the particular design of solenoids had tested satisfactorily after exposure to 225°F, for a period of 50 days. Since the team had determined that the room temperature would not exceed 173 degrees F, the team agreed with the licensee's operability conclusion.

Although the inadequate auxiliary feedwater flow concern documented in Condition Report 2001-02103 was identified on February 6, 2001, the licensee had yet to initiate action to change the Technical Specifications or reflect the changes to the plant licensing and design bases. Furthermore, none of the design engineering personnel with whom the team interfaced, knew of the problem or its implications. The team’s discussion with the plant staff knowledgeable of this condition indicated that they relied on the administrative actions for TS 3.7.1.2 (Operability Assessment OAS 4982) and the Condition Report 2001-02103 to implement the plant licensing and design changes. The licensee issued Condition Report 2001-19700 to reflect that the issue documented in Condition Report 2001-02103 was not documented in the design basis and to determine if the design change process should have been used to track what appears to be a significant modification to the plant.

The team’s finding identified a credible failure scenario (loss of normal feedwater and offsite power with a single active failure of the A train emergency diesel generator) that was outside of the design basis and did not meet the licensing basis. The team agreed with the licensee's evaluation documented in CREE-01-19681-01, which concluded that the D train auxiliary feedwater pump remained operable under the postulated condition for the required mission time. Also, changes to the USQE 00-3229-20 where not reviewed by the PORC, hence, the safety evaluation for the condition described in CR 01-2103, has not been properly reviewed, as required by Technical Specification 6.5.1.6.b. The team evaluated this observation in accordance with NRC Appendix B of Inspection Manual Chapter 0610*, and determined that it was a minor issue because the D train pump remained operable and performance indicators would not be affected.

Criterion III of Appendix B to 10 CFR Part 50 states, in part, that “[d]esign control measures shall provide for verifying or checking the adequacy of design such as by the performance of design reviews or . . . a suitable testing program." Contrary to the requirement, the licensee's process had not resulted in the verification of the adequacy of the design for the auxiliary feedwater system.

The team evaluated this finding in accordance with NRC Appendix B of Inspection Manual Chapter 0610*. The team determined that there was a credible impact on safety because the plant could have experienced an event with a potentially degraded mitigating system. Therefore, the issue was greater than a minor violation.

The team also concluded that during the analyzed event, the single active failure of an emergency diesel generator could affect the operability, availability, reliability, or function of a mitigating system. As a result, there was a potential to initiate a secondary event, loss of coolant due to a stuck open pressurizer power-operated relief valve, which had not been analyzed in conjunction with the primary event. Therefore the significance determination process as described in NRC Inspection Manual Chapter 0609 was entered.

The team determined there was no actual loss of safety function because the licensee's operability evaluation validated that the D auxiliary feedwater pump would perform its safety function at the predicted elevated room temperatures with no loss of capability. The credible impact on safety affected only the mitigating systems cornerstone. Therefore, the finding is considered to be of very low safety significance (Green). The team concluded that the failure to verify the adequacy of the plant design in both units to support the current safety analysis for a loss of normal feedwater event, was a violation of Criterion III of Appendix B to 10 CFR Part 50. However, because of the very low safety significance, and because the licensee has included the item in their corrective action program as Condition Reports 2001-19586 and 2000-19700, this design control violation is a noncited violation (NCV 50-498/01-04-02; 50-499/01-04-02). in accordance with Section VI.A.1 of the Enforcement Policy.

.6 Safety System Inspection and testing

a. Inspection Scope

The team reviewed the program and procedures for testing and inspecting selected valves and the motor and steam driven pumps of the auxiliary feedwater system, and the steam generator power operated relief valves. The review included the results of technical specification required surveillance tests and ASME Code required quarterly in service tests conducted since 1994.

b. Findings

No findings of significance were identified.

4 OTHER ACTIVITIES (ZA)

4OA6 Management Meetings

Exit Meeting Summary

The lead inspector presented the inspection results to Mr. J. Sheppard, Vice-President, Engineering, and other members of licensee management at the conclusion of the onsite inspection on December 6, 2001.

A telephonic re-exit meeting was conducted on January 14, 2002, to provide the licensee with the final results of the inspection.

At the conclusion of this meeting, the team leader asked the licensee's management whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT

Licensee Contacts :

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E. Halpin, Plant Manager
S. Head, Manager, Licensing
T. Jordan, Manager, Engineering
W. Jump, Manager, Projects Department
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M. McBurnett, Director, Quality and Licensing
W. Mookhoek, Senior Licensing Specialist
G. Parkey, Vice-President, Generation
D. Roncurrel, Manager, Operations
J. Sheppard, Vice President, Engineering and Technical Support
S. Thomas, Manager, Plant Design Engineering
T. Walker, Manager, Engineering Support Services

NRC:

N. O'Keefe
G. Guerra

ITEMS OPENED AND CLOSED

Opened and Closed

50-498/0104-01 50-499/0104-01	NCV	The failure to identify the need for and initiate corrective action to address excessive moisture in the hydraulic operating system for a steam generator power-operated relief valve was a violation of Criterion XVI of Appendix B to 10 CFR Part 50 (Section 1R21.).
50-498/0104-02 50-499/0104-02	NCV	The failure to adequately verify the design capability to support the safety analysis was a violation of Criterion III of Appendix B to 10 CFR Part 50, and the untimely corrective actions was a violation of Criterion XVI of Appendix B to 10 CFR Part 50 (Section 1R21.5.b.1.).

DOCUMENTS REVIEWED

SAFETY EVALUATIONS

NUMBER	DESCRIPTION	REVISION
98-0021	Replace existing Woodward model 2301 EDG Speed Control Governors with model 701 Governors on all EDGs	1
98-8993-99	Test methodology for replacement steam generators return to service tests	1
99-13150	Changing room temperature limits for area temperature monitoring switches	1
01-9518-6	Increase in pressurizer water level above program in Mode 3	0
01-8768-1	One time deferral of surveillance test for molded case breaker	0
01-3473-2	Change to TRM and USAR change in turbine valve test frequency from monthly to quarterly	0
00-1417-1	Evaluation to justify our current method of control for FHB truck bay doors, MAB/FHB door and FHB exhaust system	0
00-3229-20	The proposed change revises the analysis for the loss of normal feedwater flow event described in UFSAR Section 15.2.7. The revised analysis takes credit for operator actions not originally credit for this event and the availability of four steam generators.	0
00-10065-16	Review test methodology for unit return to service test Procedure 0PEP07-SG-0005	0
00-13574-1	Revise fire protection program	0
00-3229-20	Revises analysis for the loss of normal feedwater flow event	0

SCREENING REVIEWS

NUMBER	DESCRIPTION	REVISION
DCP 01-2908-1	Radiation monitoring system design basis document correction	0
DCP 99-10940-2	Correct T-drain requirement for D1AS-FV-7526 in EQCP-Limitorque and MOV database	0
DCP 01-9022-3	Replace the existing Limitorque MOV for CWP-11 with a new EIM 7000 series MOV	0
DCP 98-19447-7	SGR pressurizer relief tank alarm setpoint changes	0
DCP 01-5201-6	Design change allows for an increase of the CCP 2A and 2B gear box high vibration setpoint	0
DCP 01-7761-9	Use-as-is disposition for the spare ECW pump upper shaft	0
DCP 01-7761-19	Repair disposition for ECW pump 1C bearing housing	0
DCP 01-7761-15	Change ECW pump impeller lift from 0.120 inches to 0.180 inches	0
DCP 98-19447-10	Steam generator replacement anti-water hammer modifications	0
0PMP08-RH-0867	RHR discharge flow calibration	7
0PMP04-RC-0008	Pressurizer safety valve removal and reinstallation	13
0PEP07-AF-0001	Auxiliary feedwater turbine overspeed trip test	9

CONDITION REPORTS REVIEWED FOR IP 71111.02

2000-08193	2001-09518	2001-13824	2001-15323	2001-17433
2001-02487	2001-11754	2001-14762	2001-15533	2001-17553

CONDITION REPORTS REVIEWED FOR IP 71111.21

1998-10031	2000-00951	2000-14800	2001-00524	2001-08353
1998-12276	2000-01415	2000-15127	2001-00830	2001-08355
1998-15536	2000-02042	2000-15216	2001-02509	2001-10842
1998-16279	2000-03229	2000-15250	2001-03247	2001-11129
1998-17166	2000-04101	2000-16751	2001-03805	2001-14864
1998-17645	2000-06358	2000-17076	2001-05793	2001-17532
1999-00461	2000-08027	2000-17076	2001-06501	2001-17702
1999-08817	2000-10110	2000-18414	2001-07076	2001-17916
1999-15877	2000-11750	2000-18648	2001-08197	2001-18338
1999-15906	2000-13887	2001-00496	2001-08349	2001-18433
1999-17418				

CONDITION REPORTS INITIATED BY THE LICENSEE PRIOR TO THE INSPECTIONS

2001-15136
2001-16788

CONDITION REPORTS INITIATED BY THE LICENSEE DURING THE INSPECTIONS

2001-18332	2001-18434	2001-19637	2001-19502	2001-19588
2001-18338	2001-18438	2001-19641	2001-19531	2001-19605
2001-18426	2001-18458	2001-19642	2001-19586	2001-19700
2001-18433				

PREVENTIVE MAINTENANCE WORK AUTHORIZATIONS

000565	036722	133877	162102	190914
000571	046393	134319	164430	193719
001021	077860	134320	164446	193733
001022	078905	135224	175402	193749
001023	092436	135652	176662,	193751
001024	098756	135917	178268	194944
001025	109428	136399	179736	200582
001026	119166	137154	181097	202540
001027	121173	140769	182381,	205185
001028	122050,	141621	187276	210123
016688	124134,	144166	187277	213324
025574	132763	157162	189239	216195
027278	132769	157165	190023	

MAINTENANCE/WORK ORDERS

31034600	31164641	31201845	31260616	31369263
31083962	31180733	31205644	31260618	31369265
31084955	31190417	31235360	31351325	31379510
31155652	31194317			

ENGINEERING REQUESTS

NUMBER	DESCRIPTION	REVISION
CREE 1998-12276-31	Analysis of Stroke Times for AFW System Valves to Establish Reference Values, Acceptance Criteria, and Limiting Values to be Used in Valve Operability Tests	2
CREE 2000-8027-1	Evaluation of the Condition of the AFW valves (3S141ZAF7525, 2S141TAF0048, 2S141TMS0143, and 3S141XMS0514) for Determining In service Testing Program Actions	3
2001-19641-2	Evaluate Main Steam PORV C2PSPV7431 water content being above STPNOC administrative limit	0
2001-19681-2	Operability Review	0

PROCEDURES

NUMBER	DESCRIPTION	REVISION
0PAP01-ZA-0104	Plant Operations Review Committee	1
OPBP03-ZM-0004	Lubrication Program	13
OPDP01-ZE-0002	Equipment Qualification Program	0
0PGP03-ZE-0021	In service Testing Program for Valves	15
0PGP03-ZE-0022	In service Testing Program for Pumps	17
OPGP03-ZO-0003	Temporary Modifications	19

NUMBER	DESCRIPTION	REVISION
0PAP01-ZA-0104	Plant Operations Review Committee	1
0PGP03-ZX-0002	Condition Reporting Process	22
OPGP03-ZX-0013	Industry Events Analysis	4
OPGP04-ZA-0307	Preparation of Calculations	1
OPGP04-ZE-0309	Design Change Package	10
0PMP05-NA-0002	4160v Gould Breaker Tests	10
0PMP05-NA-0004	Molded Case Breaker Test	19
0PMP05-ZE-0033	Calibration of ITE GR-5 Relays	5
0PMP05-ZE-0037	Calibration of ITE-51 Relays	8
0POP01-ZA-0021	AC Electrical Notes and Precautions	1
0POP02-AF-0001	Auxiliary Feedwater	15
0POP02-AF-0002	Resetting Auxiliary Feedwater Pump (14/24 Mechanical Overspeed Trip Device	0
0POP02-MS-0001	Main Steam System	17
OPSP03-AF-0003	Auxiliary Feedwater Pump 13(23) In service Test	15
0PSP03-AF-0007	Auxiliary Feedwater Pump 14/24 In service Test	17
0PSP03-AF-0010	Auxiliary Feedwater System Valve Operability Test	9,10
0PSP03-MS-0001	Main Steam System Valve Operability Test	12, 13, 14
0PGP03-ZM-0002	Preventive Maintenance Program	31
0POP03-ZG-0007	Plant Cooldown	29
	10 CFR 50.59 Resource Manual	0

CALCULATIONS

NUMBER	DESCRIPTION	REVISION
NC-7068	CVCS malfunction that increases RCS inventory during Mode 3 operation	0
ST-WN-YB-2958	Westinghouse Steam Generator Precautions, Limitations, and Setpoints	0
ST-W2-NOC-000718	South Texas Delta 94 RSG Feedring Design	0
MC00236	Addendum No. 1 to the Calculation of AFW Turbine Inlet And Exhaust Line Sizing	0
MC05051	Maximum Discharge Pressure of Aux FW System	4
MC05057	Max/Min Flow Requirements of AFW System	4
MC05426	Vacuum Breaker/Relief Valve Sizing for AFST	1
MC05694	AFW System FMEA	1
MC05759	Aux Feedwater System Design Data Package	3
MC05824	AFW Storage Tank Loop Seal for Vacuum Protection	1
MC05861	Auxiliary Feedwater Pump Design TDH, Flow Rate and Pump Runout	4
MC05871	Verification of AFW 10 Min Unattended Operation	1
MC05872	Verification that AFW System Can Initiate Flow Within 60 Seconds	1
MC05874	AFW Pump Turbine Steam Bypass Sizing	0
MC05889	Verification of Adequate Steam Pressure to AFW Turbine	0
MC06082	Auxiliary Feedwater Storage Tank Volume and Setpoints Calculation	5
MC06090	AFW Pump Turbine Steam Line Trap Capacity	0

CALCULATIONS

NUMBER	DESCRIPTION	REVISION
NC-7068	CVCS malfunction that increases RCS inventory during Mode 3 operation	0
MC06286	AF-0208 Flowrate Determination	0
MC06426	IVC/AFW Cooling Load and Room Heat-Up	0
MC06455	AFW Pump D Room Heatup	0
ZC07019	AFW Discharge Pressure Uncertainty Calculation	2
ZC07042	Loop Uncertainty Calculation for AFW Flow Monitoring Instrumentation	2

DESIGN CHANGES

NUMBER	DESCRIPTION	REVISION
1992-0037	Change the Auxiliary Feed Water Flow Control Valve Trim	0
1995-7154-1	Install Overspeed Trip and Rapid Blowdown AFW Speed Controller, Unit 2	0
1998-5898-5	Incorporate Loop Uncertainty fro AFW Flow Monitoring	0
1996-2843-2	Large Bore Secondary System Pipe Work (FW,AF,MS)	0
1996-2846-1	Steam Generator Level Setpoint Change	0
1999-8241-3	AMSAC Initiation on Lo-Lo Steam Generator Water Level	0

DRAWINGS

NUMBER	TITLE	REVISION
5SI109F00001 6#1	Piping and Instrument Diagram Main Steam	25

5SI109F00001 6#2	Piping and Instrument Diagram Main Steam	25
5SI141F00024. Sheet 1	Piping and Instrument Diagram, Auxiliary Feedwater	8
5SI141F00024. Sheet 2	Piping and Instrument Diagram, Auxiliary Feedwater	5
5S102Z511002	Main Steam Power Operated Relief Valve-Hydraulic System	6
PD89272	(Vendor Drawing) Hydraulic Schematic	B
9EAF14-01 31	Elementary Diagram, Auxiliary Feedwater Turbine Pump 14 Isolation Valve MOV0019	11
5S149Z40142 #1	AFW to Steam Generating Regulating Valve Logic Diagram	9
5S199F00020 #1	Piping and Instrumentation Diagram Condensate Storage	30
5S109F00016# 1	Main Steam	25
5S102Z51002	Power Operated Relief Valve - Hydraulic System	6
PD89272	Hydraulic Schematic - Power Operated Relief Valve	B
5S141F00024	Auxiliary Feedwater	8
9-E-MS19-02	Steam Generator A, B, C, and D, PORV N2 Control Solenoid Valves	12
5S141F00024, Sht. 1	Piping & Instrumentation Diagram. Auxiliary Feedwater	8
5S141F00024, Sht. 2	Piping & Instrumentation Diagram. Auxiliary Feedwater	5
5S142F00024, Sht. 2	Piping & Instrumentation Diagram. Auxiliary Feedwater	3
5S199F00020 #1	Piping & Instrumentation Diagram. Condensate Storage	30
5S109F00016 #1	Piping & Instrumentation Diagram. Main Steam	25

MISCELLANEOUS DOCUMENTS

NUMBER	DESCRIPTION	REVISION/ DATE
5Z120ZQ1028	Design Criteria for Instrument Loop Uncertainty and Setpoint Methodology	1
5S149MB1016	Design Basis Document - Auxiliary Feedwater System	January 4, 1900
RR-17	Request For Alternative to 10 CFR 50.55a(f)(4)(ii), In service Testing Requirements	February 01, 1999
RR-ENG-2-16	Relief Request for Application of an Alternative to the ASME Boiler and Pressure Vessel Code Section XI Examination Requirements for Class 1 Piping Welds	December 30, 1999
RR-ENG-2-23	Relief Request for Application of an Alternative to the ASME Boiler and Pressure Vessel Code Section XI Examination Requirements for Class 1 Socket-Welded Piping and Class 2 Piping Welds	February 27, 2001
PIE 00-2813	Plant Impact Evaluation on IEN 00-01, “Operations Issues Identified In Boiling Water Reactor Trip and Transient”	0
PIE 00-2813	Plant Impact Evaluation on IEN 00-20, “Potential Loss of Redundant Safety-Related Equipment Because of the Lack of High-Energy Line Break Barriers.”	0
516477	Herguth Laboratories, Inc. test results for 2MSPV7411	October 23, 2001
516478	Herguth Laboratories, Inc. test results for 2MSPV7421	
516479	Herguth Laboratories, Inc. test results for 2MSPV7431	October 23, 2001
516480	Herguth Laboratories, Inc. test results for 2MSPV7441	October 29, 2001
516540	Herguth Laboratories, Inc. test results for 1MSPV7421	November 1, 2001
516541	Herguth Laboratories, Inc. test results for 1MSPV7431	November 1, 2001

516542	Herguth Laboratories, Inc. test results for 1MSPV7441	November 1, 2001
	Particle Counter Worksheet, Specification NAS 1638 (1/69)	
	Lubrication Analysis Specification Sheet, LMON-MS-FYRQ-EHC	
	Auxiliary Feedwater Valve Quarterly Tests	January, April, July, and October 2001
	Auxiliary Feedwater TDAFW Quarterly Pump Tests	January, April, July, and October 2001