ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket Nos.:	50-498 50-499
License Nos.:	NPF-76 NPF-80
Report No.:	50-498/99-20 50-499/99-20
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 7 through December 25, 1999
Inspectors:	Neil F. O'Keefe, Senior Resident Inspector Wayne C. Sifre, Resident Inspector Gilbert L. Guerra, Resident Inspector
Approved By:	Joseph I. Tapia, Chief, Projects Branch A

ATTACHMENT: Supplemental Information

EXECUTIVE SUMMARY

South Texas Project Electric Generating Station, Units 1 and 2
NRC Inspection Report No. 50-498/99-20; 50-499/99-20

This inspection report included aspects of licensee operations, maintenance, engineering, and plant support. The report covers a 7-week period of resident inspection.

Operations

- Operators attempted to isolate the normal level control valve for a low pressure feedwater heater without a procedure. This action resulted in sequentially isolating all three strings of low pressure feedwater heaters, in part due to existing but unrecognized material deficiencies. A rapid power reduction was necessary due to reduced condensate system flow, but control room operators were slow to initiate a power reduction and boration, and then did so in a poorly coordinated manner. This event was complicated by several automatic valve failures. A reactor trip criterion intended to protect equipment was exceeded but not recognized because the requirement was not included in any of the procedures in use (Section O1.2).

- Reactor reactivity manipulations were not properly balanced between borations and rod insertion and, as a result, the rod insertion limit was closely approached. Operators chose to override automatic control rod insertion in order to preserve shutdown margin. In doing so, the reactor coolant system temperature and pressure transient was made more severe and Technical Specification action statements for exceeding the minimum temperature for criticality and departure from nucleate boiling minimum pressure were entered for brief periods (Section O1.2).

- Annunciator response procedures that indicated reduced condensate flow did not direct entry into the abnormal operating procedure for rapid load reduction. In addition, adequate procedural guidance was not provided for timing and flow rate of borations during a rapid load reduction to avoid loss of shutdown margin. These procedural inadequacies constitute multiple examples of procedures inappropriate to the circumstances and are a violation of 10 CFR Part 50, Appendix B, Criterion V. This issue was entered in the licensee's corrective action program as Condition Report 99-17296. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy (Section O1.2).

- Corrective actions for a previous uncontrolled power increase caused by improper operation without a procedure of a reheater drain tank level control system were too narrowly focused. Procedural guidance was only created for the reheater drain tank, even though the same guidance was needed for all feedwater heaters. The inadequate corrective actions were a violation of 10 CFR Part 50, Appendix B, Criterion XVI. This issue was entered in the licensee's corrective action program as Condition Report 99-17296. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy (Section O1.2).

- An inadvertent dilution of the reactor coolant system boron concentration caused a small increase in reactor power. The dilution resulted from an improper valve lineup while refilling the boron concentration monitor tank without a procedure. Operators quickly recognized the power increase and borated to restore power below 100 percent. The significance of the overpower transient was small due to the brief duration and small magnitude. The failure to utilize and follow the procedure for refilling the tank was a violation of 10 CFR Part 50, Appendix B, Criterion V.. This issue was entered in the licensee's corrective action program as Condition Report 99-17762. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy (Section O1.3).

- Unit 2 operators identified a charcoal leak in the Train B fuel handling building emergency ventilation filter on December 3, 1999. It was erroneously considered to have no impact on system operability based on incomplete knowledge. Three and a half days later, the system engineer determined that the leak rendered the system inoperable. On August 19, 1999, a similar leak in Unit 1 was not recognized as rendering the system inoperable until evaluated by the system engineer on August 23. In both cases, no Technical Specification Limiting Condition for Operation was exceeded because the leaks were quickly repaired. The licensee addressed the poor initial operability determinations in Condition Report 99-17218 (Section O4.1).

- The inspectors observed that the Plant Operations Review Committee and the Nuclear Safety Review Board were effective in identifying and resolving problems and improving plant operations. Committee members actively challenged the plant staff with questions focused on safety while reviewing plant procedure changes, safety evaluations, and modifications. Technical Specification requirements governing these committees were satisfied (Section O7.1).

Maintenance

- The maintenance and surveillance activities observed were well controlled and carefully performed. High quality prejob briefings were consistently observed. Operators and technicians were very knowledgeable of their assigned tasks. The inspectors observed that the preparation and maintenance activities for repairing a hydraulic leak on a main turbine throttle valve on line were carefully coordinated. The necessary plant conditions were established and practiced on the simulator, and the repair work was practiced on a mock-up. Troubleshooting efforts for load instabilities on Standby Diesel Generator 23 were thorough and prompt, and the potential for a common mode failure was promptly determined not to exist (Section M1.1).

Engineering

- The licencee's engineering evaluations for the movement and storage of replacement steam generators were thorough and appropriately detailed. Replacement steam generator transport was performed in accordance with the licensee's plan without incident or damage (Section E2.1).

Plant Support

- The inspectors determined that the licensee's initial assessment of the dose received while refilling a shield tank around a neutron source utilized electronic dosimetry which did not register neutron dose. A technician had refilled a shield tank around a 3.88 Curie neutron source in response to a low level alarm. Although some loss of shielding resulted from the low level, the licensee subsequently performed a conservative estimate and determined that the dose received was small (Section R1.1).

- The inspectors observed that the licensee was implementing the compensatory hourly fire watch program within regulatory requirements. However, the inspectors found that fire watch personnel were, in some instances, performing fire watch inspections at the end of one hour and the beginning of the following hour. In one case, the area inspection was performed twice within 10 minutes, with 1 hour 47 minutes elapsing since the earlier inspection. Licensee management stated that this practice did not meet their expectations and promptly conducted training to clarify expectations and eliminate this practice (Section F1.1).

Report Details

Summary of Plant Status

Unit 1 operated at full power throughout this inspection period.

At the start of this inspection, Unit 2 was operating at low power following the completion of a scheduled refueling outage. The unit was synchronized to the grid on November 9. The unit remained at 90 percent power until vibration problems with Steam Generator Feed Pump 21 were resolved and then reached 100 percent power on November 23. On December 7, power was lowered to 90 percent to facilitate repairs to one of the main turbine throttle valves. The unit was returned to 100 percent power the same day. On December 8, operators performed a rapid power reduction to 40 percent in response to the sequential isolation of all three low pressure feedwater heater strings. The unit was returned to 100 percent power on December 9.

I. Operations

O1 Conduct of Operations

O1. 1 Conduct of Operations (71707)

The inspectors used Inspection Procedure 71707 to conduct frequent reviews of ongoing plant operations. In general, the conduct of operations was focused and safety conscious. Specific comments and noteworthy events are discussed below.

On December 14, a Unit 2 licensed operator inadvertently actuated the Train A control room emergency ventilation system while performing Plant Surveillance Procedure 0PSP03-HE-0001, Revision 8, "Control Room Emergency Ventilation System" for Train B. The procedure was written so that a preparation section was performed and then the procedure section that corresponds with the train being tested. After completing the preparation section, the operator proceeded with the next section of the procedure which pertained to Train A rather than skipping to the section for Train B. After completing the Train A test, the operator recognized that it was the wrong train. The operator and the shift supervisor verified that the system had remained operable throughout the test. Condition Report 99-17578 was written to address this human performance error and the surveillance test was then performed on Train B.

O1.2 Rapid Load Reduction Performed in Response to Sequential Isolation of All Low Pressure Feedwater Heaters

a. Inspection Scope (93702, 71707)

The inspectors responded to the site when notified that Unit 2 had experienced a loss of low pressure feedwater heating and was conducting a rapid load reduction. The inspectors verified that the plant was stable and that no safety equipment actuations had been required. Plant logs, computer data, and applicable station procedures were reviewed, and control room operators were interviewed. The inspectors subsequently observed a simulator session conducted to model the event. The licensee's event review report findings were discussed with senior licensee management.

b. Observations and Findings

Sequence of Events

On December 8, Unit 2 operators attempted to place Low Pressure Feedwater Heater 25C level control on the high level dump valve to perform corrective maintenance on the normal level control valve. The operators' actions caused the associated heater drip pump to trip and then each of the low pressure feedwater heater strings isolated in succession over a period of 10 minutes. With more than a single string of low pressure feedwater heaters isolated, a 50-percent heater bypass valve opened in the condensate flow path. With all strings isolated, condensate flow of 65 percent to the de-aerator and 100 percent feedwater flow out of the de-aerator caused de-aerator level to lower. Five minutes after the last heater string isolated, operators began decreasing load at a rate of 1 percent per minute. A rapid load reduction at 5 percent per minute was begun 10 minutes later when attempts to restore condensate flow were unsuccessful. The first reactor coolant system boration was started 16 minutes after load was first reduced.

To accomplish the rapid load reduction, operators estimated, in accordance with abnormal operating Procedure 0POP04-TM-0005, Revision 2, "Fast Load Reduction," that they would need to add about 600 gallons of boric acid. The 600 gallons were added in several increments while the power reduction was performed. However, a mismatch between turbine power reference temperature (Tref) and average coolant temperature (Tave) developed. The rod control system responded as designed by inserting rods at a high rate. When an alarm was received indicating that the minimum rod insertion limit was being approached, operators took the rod control system out of automatic to stop inward rod motion. This action was taken to preserve the Technical Specification required rod height; however, less than half of the calculated boric acid had been added. Without the rod insertion, Tave stopped lowering and increased rapidly. A few minutes later, rods were manually inserted to reduce Tave and again a minute later, while boration continued. This rapidly lowered Tave until it was well below Tref. Operators observed that the rod control system indicated a withdraw demand and manually withdrew rods three times in small increments to successfully turn Tave back upward. Operators also diluted the reactor coolant system with 170 gallons of water. The transient caused the reactor coolant system to go below the Technical Specification required minimum temperature for criticality (561°F) for 4 minutes and pressure to drop below the departure from nucleate boiling minimum pressure limit of 2219 psig for 27 minutes.

During the time Tave was decreasing rapidly, de-aerator level stopped lowering so operators stopped reducing turbine load near 56 percent power. However, the boric acid addition caused reactor power to continue to lower. As a result, the balance of plant was expending more energy than the reactor was producing and this accentuated the drop in Tave. Operators recognized this and reduced turbine load in an attempt to match Tave and Tref. The net effect of their actions resulted in ending the transient at a stable power level of 40 percent. Throughout the transient, no engineered safety features equipment was called upon to actuate.

The licensee conducted a prompt, thorough review of the event. Initial findings and lessons learned were promptly presented to all operating crews. A material condition assessment of affected systems was also conducted as part of the investigation.

Reactivity Manipulation Issues

During the initial part of the rapid load reduction, operators did not properly match heat loads between the primary and secondary plants. A large load reduction required operators to add negative reactivity, primarily by borating the reactor coolant system. However, this was not started until 16 minutes after the load reduction was started. The rod control system compensated by automatically inserting control rods from 242 steps to 125 steps and gave a Bank Insert Low alarm. This warned operators that rods were approaching the point, called the rod insertion limit, where adequate reactor shutdown margin might not be assured. Although Tave was not within the required 3°F of Tref, operators decided to take rods out of automatic and stop the insertion demanded by the system. Operators did this in order to avoid the need to emergency borate if further inward rod motion occurred and the rod insertion limit was exceeded.

The automatic feature of the rod control system was designed to allow the reactor to handle a rapid load reduction such as the one that was in progress. When operators overrode the automatic operation of rod control in order to preserve the Technical Specification required rod insertion limit, the ability to expediently handle the existing primary to secondary thermal mismatch was made worse by removing the fastest available reactivity control mechanism. The impact of this action was accentuated by core characteristics that included a relatively small temperature coefficient of reactivity which caused the reactor core's natural stability to behave more sluggishly. The direct result of operators' actions was a temperature increase and subsequent lowering that caused both temperature and pressure to exceed Technical Specification limits. The inspectors noted that Technical Specifications permitted the licensee to have rods inserted further than the rod insertion limit for 2 hours and that the rod insertion limit would have been restored in a short time if the system were left in automatic.

The licensee performed a number of simulations of the event with and without automatic rod motion to evaluate the effectiveness of different reactivity control strategies. The results demonstrated that:

- The reactor coolant system temperature and pressure transient would have been less severe with automatic rod control. The minimum temperature for criticality and departure from nucleate boiling pressure limit would have remained satisfied.

- The rod insertion limit could have been satisfied with an earlier initiation of boration.

- For the same sequence of events as the actual transient, but with rod control in automatic, rods would have been below the Technical Specification rod insertion limit for about 2.5 minutes (of a 2-hour allowed action statement).

- Operator concerns about having to emergency borate if rods had been left in automatic were shown to be unjustified because they were already effectively emergency borating at the time.

Based on the simulations, placing rod control in manual unnecessarily complicated the event and directly contributed to entering Technical Specification action statements for exceeding the minimum temperature for criticality and minimum pressure for departure from nucleate boiling.

The licensee's event review team identified that abnormal operating Procedure 0POP04-TM-0005, "Fast Load Reduction," guidance for borating required excessive values. In this event, the planned reduction of 50 percent power would result in the procedure indicating 675 gallons was the required boration. Operators knew this was usually too much, so they concluded they should add 600 gallons. The licensee later calculated that 400 gallons was actually the appropriate value. The procedural guidance was planned to be revised to provide more realistic values for load reductions.

Operator Experience and Procedure Inadequacies Complicated the Event

The licensee identified that the control room crew responding to this transient included individuals who were newly licensed and supervisors who were relatively inexperienced in their roles during the event.

Throughout this transient, numerous control room annunciators actuated which were intended to alert operators to abnormal feedwater and condensate system alignments. While operators understood the condition of these systems throughout the event, they felt no sense of urgency to initiate a prompt or rapid power reduction until well after all low pressure feedwater heaters were isolated. This was apparently due to a false sense that the water volume in the de-aerator would provide sufficient response time to restore feedwater heater flow. When a power reduction was begun, it was initially started at too slow a rate. Boration of the reactor coolant system was not coordinated with the turbine load reduction, which significantly complicated the event. The inspectors noted that no procedural or policy guidance was provided by the licensee on when and how fast to borate during a rapid power reduction to avoid loss of shutdown margin.

The inspectors also determined that multiple annunciator response procedures for alarms received during this event did not clearly direct the need for entry into an abnormal operating procedure, nor did they identify conditions indicative of the need to conduct a rapid power reduction due to the partial loss of condensate flow. This contributed to the relatively inexperienced crew's delay in deciding to reduce power and subsequent urgency of their actions. Additionally, the abnormal operating procedure for rapid power reduction did not provide adequate guidance for reactivity manipulations, which complicated this event. These were considered to be examples of procedures inappropriate to the circumstances in that the combination of operator inexperience and procedural guidance was inadequate to ensure that a controlled power reduction was promptly initiated and are therefore a violation of 10 CFR Part 50, Appendix B, Criterion V. These issues were addressed in Condition Report 99-17296. This violation will not be cited in accordance with Section VII.B.1.a of the NRC Enforcement Policy (NCV 499/99020-01).

Operators Exceeded a Plant Trip Criteria Without Recognizing It

Operators were closely monitoring de-aerator water level during the period of condensate-feedwater flow mismatch. The de-aerator is composed of a steam tank and two storage tanks that provide net positive suction head to the feedwater booster pumps. Annunciator Response Procedure 0POP09-AN-09M1-A-1, "Condensate Pump Trip," included the requirement to trip the plant if de-aerator storage tank level went below 30 percent in anticipation of an impending loss of feedwater for equipment protection. During this event, one licensed operator recalled this requirement but the crew was unable to locate the requirement in any of the procedures they were using. De-aerator water level dropped as low as 12 percent before being recovered.

The criterion intended to designate the need to trip the plant was not placed into each procedure which could be expected to result in the limit being exceeded. As a result, operators were unable to locate this information and were hesitant to trip the plant based on memory alone. However, this was not a violation because there was no regulatory requirement to have such a trip criterion. The licensee was addressing this issue in their corrective action program.

Narrow Corrective Actions from a Previous Event Contributed to Initiating This Event

A similar event occurred in February 1999. Unit 2 operators attempted to place a moisture separator reheater drain tank water level control on the high level dump valve without a procedure. Improper valve response and inadequate monitoring of system response resulted in draining of the tank and an uncontrolled power increase. Corrective action for the February event (Condition Report 99-2103) included creating a new section in Operating Procedure 0POP02-HV-0001, Revision 12, "Feedwater Heater Drains and Vents," to place a moisture separator reheater drain tank water level control on the high level dump valve. This corrective action was too narrow because only one type of tank was addressed and the procedure should have been made applicable to other similar tanks.

As a result, no procedure existed for placing Feedwater Heater 25C level control on the high level dump valve. For convenience, operators decided to use a method different than had been previously used. However, the method selected, shutting a manual valve to isolate the normal level control valve, did not allow for operator response if level did not control as expected. Having no procedure to place Feedwater Heater 25C level control on the high level dump valve directly contributed to initiating this event. Failure to implement adequate corrective actions following the February event was considered a violation of 10 CFR Part 50, Appendix B, Criterion XVI. This licensee identified and corrected violation will not be cited in accordance with Section VII.B.1.a of the NRC Enforcement Policy (NCV 499/99020-02).

Material Condition Issues

The normal level control valve for Feedwater Heater 25C was identified as requiring maintenance in June 1999; however, the valve was not repaired during the recently completed refueling outage. If all components had been performing properly, conducting the maintenance on line should not have triggered a plant transient. The licensee determined that the unexpected system response was caused by improper tuning of the feedwater heater level controllers causing slow response to level transients. The response was further complicated by several automatic valves which stuck or otherwise failed to operate as intended. One feedwater heater high level dump valve stuck shut, resulting in a high water level isolation of the associated feedwater heater, and two condensate isolation valves to feedwater heaters shut but would not reopen, delaying restoration of condensate flow.

The licensee initiated a prompt evaluation of the calibration and functionality of the feedwater heating control system. Additional material condition deficiencies were identified and corrected. The slow response of the system was identified. Corrective action was planned to improve system response and to formalize control of the controller settings.

c. Conclusions

Operators attempted to isolate the normal level control valve for a low pressure feedwater heater without a procedure. This action resulted in sequentially isolating all three strings of low pressure feedwater heaters, in part due to existing but unrecognized material deficiencies. A rapid power reduction was necessary due to reduced condensate system flow, but control room operators were slow to initiate a power reduction and boration and did so in a less than coordinated manner. This event was complicated by several automatic valve failures. A reactor trip criterion intended to protect plant equipment was exceeded but not recognized because the requirement was not included in any of the procedures in use.

Reactor reactivity manipulations were not properly balanced between borations and rod insertion and, as a result, the rod insertion limit was closely approached. Operators chose to override automatic control rod insertion in order to preserve shutdown margin. In doing so, the reactor coolant system temperature and pressure transient was made more severe and Technical Specification action statements for exceeding the minimum temperature for criticality and departure from nucleate boiling minimum pressure were entered for brief periods.

Annunciator response procedures that indicated reduced condensate flow did not direct entry into the abnormal operating procedure for rapid load reduction. In addition, adequate procedural guidance was not provided for timing and flow rate of borations during a rapid load reduction to avoid loss of shutdown margin. These procedural inadequacies constitute multiple examples of procedures inappropriate to the circumstances and are a violation on 10 CFR Part 50, Appendix B, Criterion V. This issue was entered in the licensee's corrective action program as Condition Report 99-17296. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy.

A violation of 10 CFR Part 50, Appendix B, Criterion XVI, was identified for inadequate corrective actions for a previous similar event. Corrective actions for an uncontrolled power increase caused by improper operation of a reheater drain tank level control system without a procedure were too narrowly focused. Procedural guidance was created only for the reheater drain tank, even though all feedwater heaters needed the same guidance. This issue was entered in the licensee's corrective action program under Condition Report 99-17296. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy.

O1.3 Inadvertent Dilution Caused Unit 2 Power Increase

a. Inspection Scope (71707)

The inspectors performed a followup inspection into the circumstances and corrective actions involving an inadvertent dilution of the reactor coolant system boron concentration. Inspectors interviewed control room operators and the chemistry technician involved in the event. The equipment involved was observed in company with the chemistry technician. Radiological surveys of the area involved were reviewed and discussed with health physics personnel.

b. Observations and Findings

On December 20, 1999, Unit 2 control room operators noted that the reactor coolant system temperature and steam plant pressure were slowly increasing. The increases were recognized to be consistent with a dilution of boron concentration in the reactor coolant system. Operators began to look for potential sources of dilution while borating to restore proper boric acid concentration. When no source of dilution was identified, recent plant manipulations were reviewed.

An hour before the dilution was recognized, a chemistry technician had refilled a shield tank for the boron concentration monitor. This tank was normally full of pure water which shielded a 3.88 curie radioactive neutron source used to measure the boron concentration of reactor coolant system letdown flow to the volume control tank. When the chemistry technician was unable to clear a tank low level alarm, operators were requested to refill the tank. After a discussion to determine which group had responsibility to operate that portion of the system, a chemistry technician refilled the tank. This was done using a valve sequence from memory, since no procedure was known to exist. After the tank was full, the valves were returned to what was thought to be their original positions. This erroneously resulted in two valves being throttled open when they should have been shut. The misaligned valves allowed pure water to flow into the volume control tank, diluting the water that was subsequently charged back into the reactor coolant system as part of normal makeup.

Over the one hour period of the uncontrolled dilution, the level in the volume control tank increased 3 percent, or about 135 gallons. This caused a power increase of 10 MW, with a peak power of 3805.8 MW or 100.15 percent reactor power. This power level did not exceed licensed limits or result in any time averaged power alarms. The event was therefore not a significant overpower event.

During the investigation of this event, operators identified that a procedure existed to perform the evolution. The licensee also concluded that it was inappropriate for chemistry personnel to operate the equipment since operations personnel had responsibility for the system. The licensee was evaluating whether enhancements were needed to improve the administrative controls for this unrecognized dilution path. Plant Operating Procedure 0POP02-CV-0004, "Chemical and Volume Control System Subsystem," Revision 18, provided steps to refill the boron concentration monitor tank. This was a procedure required by Technical Specification 6.8 and Regulatory Guide 1.33. Appendix B to 10 CFR Part 50, Criterion V, requires that activities affecting quality shall be prescribed in instructions, procedures, and drawings, and shall be accomplished in accordance with these instructions, procedures, and drawings. Failure to follow Procedure 0POP02-CV-0004 while refilling the boron concentration monitor tank was a violation. This issue was reported in Condition Report 99-17762. This licensee identified and corrected violation will not be cited in accordance with Section VII.B.1.a of the NRC Enforcement Policy (NCV 499-99020-03).

The radiological aspects of this event are discussed in Section R1.1 below.

c. Conclusions

An inadvertent dilution of reactor coolant system boron concentration and a small increase in reactor power occurred as a result of using an improper valve lineup while refilling the boron concentration monitor tank without a procedure. Operators quickly recognized the power increase and borated to restore power below 100 percent. The significance of the overpower transient was small due to the brief duration and small magnitude. This issue was reported in Condition Report 99-17762. This is a violation of 10 CFR Part 50, Appendix B, Criterion V, for failure to follow procedure. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy.

O2 Operational Status of Facilities and Equipment

O2.1 Engineered Safety Feature (ESF) Systems Walked Down (71707)

The inspectors used Inspection Procedure 71707 to walk down accessible portions of the following ESF systems:

- Control Room Ventilation System (Unit 1)
- Standby Diesel Generator 23 (Unit 2)

Equipment operability and material condition were acceptable in each case. The inspectors verified that the systems were aligned properly for the existing mode of operation. The inspectors conducted daily control board walkdowns to verify that ESF systems were aligned as required by Technical Specification for the existing operating mode, that instrumentation was operating correctly, and that power was available.

O4 Operator Knowledge and Performance

O4.1 Repeat Failure to Recognize Inoperable Fuel Handling Building Ventilation System Filter Promptly

a. Inspection Scope (71707)

The inspectors reviewed the circumstances surrounding the discovery of leaking charcoal from the Unit 2 Train B fuel handling building emergency ventilation filter. Interviews and discussions were held with reactor operators and licensee operations management.

b. Observations and Findings

On December 3, 1999, a Unit 2 plant operator identified that charcoal filter media had leaked out of the Train B fuel handling building emergency filter. A senior reactor operator evaluated the report and concluded that this condition was not an operability concern. On December 6, the system engineer concluded that this condition rendered the system inoperable.

The same situation had previously occurred in Unit 1 on August 19, 1999, and the system was not declared inoperable until August 23, also about 3.5 days later. In both cases no Technical Specification Limiting Conditions for Operation were exceeded because the conditions were quickly repaired.

The inspectors determined that the Unit 2 operators were aware that Unit 1 had experienced a charcoal leak in the fuel handling building ventilation system and that it had been determined not to render the system inoperable. Although this was the initial evaluation when the condition was first discovered, it was subsequently determined that it was in error and the operators had not been informed of this.

c. Conclusions

Unit 2 operators identified a charcoal leak in the Train B fuel handling building emergency ventilation filter on December 3, 1999. It was erroneously considered to have no impact on system operability based on incomplete knowledge. Three and a half days later, the system engineer determined that the leak rendered the system inoperable. On August 19, 1999, a similar leak in Unit 1 was not recognized as rendering the system inoperable until evaluated by the system engineer on August 23. In both cases, no Technical Specification Limiting Condition for Operation was exceeded because the leaks were quickly repaired. The licensee addressed the poor initial operability determinations in Condition Report 99-17218.

O7 Quality Assurance in Operations

O7.1 Offsite and Onsite Review Committee Observations (71707)

On December 1, 1999, the inspectors observed a scheduled meeting of the Plant Operations Review Committee, which functioned as the onsite review committee required by Technical Specification 6.5.1. The inspectors reviewed the meeting minutes and administrative Procedure 0PAP01-ZA-0104, Revision 1, "Plant Operations Review Committee." On December 2, the inspectors observed a scheduled meeting of the Nuclear Safety Review Board, which functioned as the offsite review committee required by Technical Specification 6.5.2.

The inspectors observed that both the onsite and offsite review committees were effective in identifying and resolving problems and improving plant operations. Committee members actively raised questions focused on safety while reviewing plant procedure changes, safety evaluations, and modifications as required by Technical Specifications. High standards for committee approval were demonstrated when a number of cases were observed where approval was tabled pending additional information to resolve member concerns. The inspectors verified that quorum and qualifications requirements were satisfied. Both committees met much more frequently than required by Technical Specifications in order to support ongoing plant operations.

O8 Miscellaneous Operations Issues (92700)

O8.1 (Closed) License Event Report (LER) 50-498/99009-00: Insertion of incore flux thimbles with Technical Specification requirements for core alterations not satisfied. This event was discussed in NRC Inspection Report 50-498;499/99-18, and a noncited violation was issued. No new issues were revealed by the LER.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments: Maintenance and Surveillance Observed

a. Inspection Scope (62707, 61726)

The inspectors observed all or portions of the following maintenance and surveillance activities. For surveillance tests, the procedures were reviewed and compared to the Technical Specification surveillance requirements and bases to ensure that the procedures satisfied the requirements. Maintenance work was reviewed to ensure that adequate work instructions were provided, that the work performed was within the scope of the authorized work, and that it was adequately documented. Work practices were also observed. In each case, the impact to equipment operability and applicable Technical Specifications actions were independently verified.

Surveillances Observed:

- OPSP10-ZG-0004, "End of Life Moderator Temperature Coefficient Measurement" (Unit 1)
- 0PSP07-NI-0043A, "NIS Axial Flux Difference Calibration" (Unit 1)
- 0PSP03-DG-0003, "Standby Diesel Generator 22 Operability Test" (Unit 2)

Maintenance activities observed:

- Circuit Breaker Maintenance (Unit 1)
- SDG 23 Troubleshooting (Unit 2)
- Unit 2 Main Turbine Throttle Valve Repair (Unit 2)

b. Observations and Findings

The inspectors observed that surveillance tests were performed utilizing the proper procedures. Prejob briefings were consistently of good quality. Personnel performing surveillance activities had experience with the task. Equipment manipulations during tests were well controlled by operators. Where required, independent verification techniques were properly conducted. Communications were precise and sufficiently detailed. The inspectors verified that surveillance activities satisfied Technical Specifications requirements.

The inspectors noted that, during a nuclear instrument calibration in Unit 1, the operators logged that the channel was considered inoperable and that the channel must be placed in a tripped condition within 6 hours per Technical Specifications. The procedure directed the tripping of the channel before beginning the calibration. However, the completion of this action was not logged nor was restoring the channel to an untripped condition. This was also found to be the case with calibrations of the remaining three channels. While not considered a violation, this was considered to be a poor practice because the control room log did not accurately reflect the status of Technical Specification required actions.

Standby Diesel Generator 23 exhibited swings in real and reactive load as well as frequency during a surveillance test at full load on December 21. Operators appropriately declared the diesel inoperable and initiated troubleshooting. The diesels in the other two trains were started in accordance with Technical Specifications because a potential common mode failure could not be quickly ruled out. Condition Report 99-17778 was written to conduct troubleshooting. The inspectors observed that the licensee aggressively addressed the problem and obtained vendor assistance despite the holiday leave schedule. Operability determination efforts were prompt and thorough. The diesel was returned to service following gain adjustments recommended by the vendor.

The inspectors observed a good prejob brief in support of the "End of Life Moderator Temperature Coefficient Measurement" surveillance conducted to measure the reactivity change associated with operator-controlled reactor coolant system temperature changes in Unit 1. Reactor engineering prepared a package describing the test with a performance schedule and participant responsibilities. Good coordination was observed from the engineering, chemistry, and operations departments. Reactivity manipulations and reactor coolant sample collections were performed without problem.

On December 7, 1999, the licensee performed an on line repair of a hydraulic leak on Unit 2 main turbine throttle Valve TV-1. The licensee decided to perform the repair earlier than planned in response to the degrading condition of the leak because a continual degradation of the leak could cause the valve to shut. Preparations for the repair were thorough. Key control room operators were given just-in-time training on the simulator. Power was reduced to 90 percent to allow the remaining throttle valves to control steam flow consistent with load if required. Maintenance personnel practiced the work in a mock-up, were thoroughly briefed, and took necessary industrial safety precautions. The repair was completed as planned without incident. Plant equipment performed well, with one exception. A steam dump valve opened as expected while shutting Valve TV-1 but did not reclose fully. Operators were unable to close it remotely and it was manually isolated.

c. Conclusions

The maintenance and surveillance activities observed were well controlled and carefully performed. High quality prejob briefings were consistently observed. Operators and technicians were very knowledgeable of their assigned tasks. The inspectors observed that the preparation and maintenance activities for repairing a hydraulic leak on a main turbine throttle valve on line were carefully coordinated. The necessary plant conditions were established and practiced on the simulator, and the repair work was practiced on a mock-up. Troubleshooting efforts for load instabilities on Standby Diesel Generator 23 were thorough and prompt, and the potential for a common mode failure was promptly determined not to exist.

M8 Miscellaneous Maintenance Issues (92700)

M8.1 (Closed) LER 498/99008-00: Turbine trip while performing main turbine emergency trip testing

On September 12, 1999, Unit 1 tripped automatically from 100 percent power during main turbine emergency trip system test. The trip was attributed to dust and lint contamination found on the turbine trip test selector switch. While the licensee was unable to reproduce the problem, available indications led to the conclusion that the turbine protection circuit sensed a false Channel 2 overspeed trip condition at the same time operators were testing Channel 1.

Corrective action for this event included cleaning the turbine trip test switches and inspecting the main control room panels for other critical switches that could be affected by dust and lint in both units and evaluating the preventive maintenance frequencies of existing main control board cleaning. This event was discussed in NRC Inspection Report 50-498/99-16. No additional issues were identified during the review of this LER. This item is closed.

M8.2 (Closed) LER 50-499/98004-00 and -01: Unit 2 shutdown required by technical specifications due to failure in the solid state protection system (SSPS) test circuitry. This event was discussed in NRC Inspection Report 50-498;499/98-11. In the original LER, the licensee stated that troubleshooting of the circuit would be performed during the Unit 2 refueling outage to determine the cause of the test circuit failure. On November 5, 1998, during the Unit 2 refueling outage, the licensee determined that the test circuit failure was caused by a 16 kHz noise induced to the test circuit by a wire from the multiplexing circuit that was bundled with the test circuit wires. The licensee unbundled and physically separated the noise-inducing wire from the test circuit wires. The licensee also revised the surveillance test procedure to incorporate manual testing to prevent a test circuit failure from resulting in an inoperable SSPS. The licensee concluded that the system remained operable and the shutdown was not required. The inspectors identified no issues during this review.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Transportation and Storage of Replacement Steam Generators

a. Inspection Scope (50001)

Using Inspection Procedure 50001, Section 02.03.b.4, the inspectors reviewed the licensee's plan and associated engineering evaluations for the receipt, transport, and storage of the Unit 1 replacement steam generators. The inspectors also observed portions of the receipt and transport activities.

b. Observations and Findings

The inspectors reviewed the following Condition Report Engineering Evaluations (CREEs):

- CREE 96-2845-3, "STP Unit 1 Steam Generator Replacement Haul Route Evaluation Report"
- CREE 96-2845-12, "STP Unit 1 Steam Generator Replacement Rigging."

The inspectors found that the evaluations were appropriately detailed and thorough. The haul route evaluation included calculations for underground commodities, including safety-related systems. Based on the depth and configurations, the licensee determined that these systems were capable of withstanding loads equal to or exceeding the resultant loads from the transport of the steam generators.

The inspectors observed portions of the steam generator lifting and transport activities. The work was performed as planned and there were no adverse effects to the plant or the new steam generators. The steam generators were transported and stored with a protective coating and positive nitrogen pressure in both primary and secondary compartments to maintain the material condition of the steam generators. The steam generators were guarded while outside the protected area and were promptly moved within the protected area.

c. Conclusion

The licencee's engineering evaluations for the movement and storage of the replacement steam generators were thorough and appropriately detailed. Replacement steam generator transport was performed in accordance with the licensee's plan without incident or damage.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 Poor Dose Assessment Identified

a. Inspection Scope (71750)

The inspectors performed a followup inspection into the circumstances and corrective actions for an inadvertent dilution of the reactor coolant system boron concentration. Inspectors interviewed control room operators and the chemistry technician involved in the event. The equipment involved was observed in company with the chemistry technician. Radiological surveys of the area involved were reviewed and discussed with health physics personnel. A bounding dose rate calculation for the source was reviewed with a health physics supervisor.

b. Observations and Findings

The inspectors determined that the licensee performed a poor preliminary dose assessment for a chemistry technician who refilled the boron concentration monitor shield tank on December 20. The tank contained a 3.88 Curie neutron source which was surrounded by water for personnel shielding. Because a low tank level existed, some of the normal shielding from the water was not present. This was particularly important because the worker had to lean closely over the top of the tank during the refilling evolution. The worker's dose was evaluated by the licensee by checking the reading on his electronic dosimeter; however, it indicated that no dose was received because the electronic dosimeter did not register neutron dose.

In response to the inspector's concerns, the licensee performed a bounding calculation which demonstrated that the neutron dose rate without any shielding from water in the tank was about 10 mrem/hour at 1 meter. Since most of the shielding was available during this event, and the duration of exposure was only a few minutes, the dose was expected to be small. The actual dose will be determined during the regularly scheduled processing of the chemistry technician's thermoluminescent dosimeter.

The licensee posted the boron concentration monitor rooms so that personnel must contact health physics prior to entry to ensure the radiological conditions were known. Health physics personnel were instructed to perform neutron surveys in the room if a low level condition in the shield tank existed. Condition Report 99-17903 was written to track corrective actions for this issue.

c. Conclusions

The inspectors determined that the licensee performed a poor initial dose assessment of the neutron dose received while refilling a shield tank around a neutron source. Electronic dosimetry which did not register neutron dose was the only thing used to evaluate dose. The technician had refilled a shield tank around a 3.88 Curie neutron source in response to a low level alarm, so some loss of shielding should be presumed to exist. In response to the inspectors' observations, the licensee performed a conservative estimate that the dose received was small.

F1 Control of Fire Protection Activities

F1.1 Fire Watch Checks

a. Inspection Scope (71750)

The inspectors reviewed fire watch logs and examined the conditions necessitating the compensatory actions. A number of security guards who performed fire watches were questioned to determine their familiarity with the conditions being compensated for, knowledge of their duties, and fire watch requirements.

b. Observations and Findings

The inspectors observed that the licensee was implementing the compensatory fire watch program satisfactorily. Security force members were knowledgeable of their duties and responsibilities as fire watches and were aware of the conditions requiring the fire watches.

The inspectors noted that all required hourly fire watches were completed as specified in the licensee's program. Specifically, each area requiring an hourly watch must be inspected once within the clock hour. However, the inspectors noted that some hourly fire watch log entries were signed toward the end of one hour and near the start of the next. In one case, an area was inspected twice within 10 minutes, such that the area was not inspected for 1 hour 47 minutes between inspections. The inspectors concluded that no violation had occurred since the inspections were conducted within the bounds of the procedure. However, the intent of conducting hourly fire watches was not always met.

These findings were discussed with fire protection and security supervisors. The licensee stated that this practice was not within management expectations for fire watches and that a procedure clarification would be considered and Condition Report 99-17878 was written. Prompt training was conducted with security officers who perform fire watch duties in order to clarify licensee management expectations regarding more even intervals between fire watch rounds.

c. Conclusions

The inspectors observed that the licensee was implementing the compensatory hourly fire watch program within regulatory requirements. However, the inspectors found that fire watch personnel were in some instances performing fire watch inspections at the end of one hour and the beginning of the following hour. In one case, the area inspection was performed twice within 10 minutes, with 1 hour 47 minutes elapsing since the earlier inspection. Licensee management stated that this practice was not per management expectations and conducted prompt training to correct this practice.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management on December 28, 1999. Management personnel acknowledged the findings presented. The inspector asked whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT

PARTIAL LIST OF PERSONS CONTACTED

Licensee

P. Arrington, Licensing Specialist
K. Coates, Manager, Maintenance
L. DeLa Garza, Senior Reactor Operator, Operations Quality Assurance
J. Drymiller, Supervisor, Nuclear Plant Security
E. Halpin, Manager, Unit 1 Operations
B. Humble, Supervisor, Systems Engineering Department
J. Johnson, Manager, Engineering Quality Assurance
A. Kent, Manager, Electrical/Instrumentation and Controls, Systems Engineering
M. McBurnett, Director, Quality and Licensing
R. Lovell, Manager, Training
J. Palen, Supervisor, Chemistry
T. Powell, Manager, Health Physics
J. Sheppard, Vice President, Engineering and Technical Services

INSPECTION PROCEDURES USED

IP 50001: Steam Generator Replacement Inspection
IP 61726: Surveillance Observations
IP 62707: Maintenance Observations
IP 71707: Plant Operations
IP 71750: Plant Support Activities
IP 92700: Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 93702: Prompt Onsite Response to Events at Operating Power Reactors

ITEMS OPENED AND CLOSED

Opened

50-499 /99020-01	NCV	Multiple examples of procedures inappropriate to the circumstances which complicated loss of feedwater heaters event (Section O1.2)
50-499 /99020-02	NCV	Inadequate corrective actions contributed to loss of low pressure feedwater heater event (Section O1.2)
50-499 /99020-03	NCV	Failure to follow procedure caused dilution and overpower event (Section O1.3)

<u>Closed</u>

50-498/99009-00	LER	Insertion of incore flux thimbles with Technical Specification requirements for core alteration not satisfied (Section O8.1)
50-498/99008-00:	LER	Turbine trip while performing main turbine emergency trip testing (Section M8.1)
50-499/98004-00 and -01	LER	Unit 2 shutdown required by Technical Specifications due to failure in solid state protection system (SSPS) test circuitry (Section M8.2)
50-499/99020-01	NCV	Multiple examples of procedures inappropriate to the circumstances which complicated loss of feedwater heaters event (Section O1.2)
50-499/99020-02	NCV	Inadequate corrective actions contributed to loss of low pressure feedwater heater event (Section O1.2)
50-499/99020-03	NCV	Failure to follow procedure caused dilution and overpower event (Section O1.3)