ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

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Licensee:	Houston Lighting & Power Company
Facility:	South Texas Project Electric Generating Station, Units 1 and 2
Location:	8 Miles West of Wadsworth on FM 521 Wadsworth, Texas 77483
Dates:	February 23 through April 5, 1997
Inspectors:	D. P. Loveless, Senior Resident Inspector J. M. Keeton, Resident Inspector W. C. Sifre, Resident Inspector F. L. Brush, Resident Inspector, Callaway D. B. Pereira, Reactor Inspector
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EXECUTIVE SUMMARY

South Texas Project, Units 1 and 2
NRC Inspection Report 50-498/97-02; 50-499/97-02

This resident inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 6-week period of resident inspection.

Operations

- Licensed operators in both units were routinely observed performing their duties in a professional, effective manner, continuously aware of existing plant conditions with a good focus on safety (Section O1.1).

- Safety system condition and availability were observed to be excellent. However, the Main Steam to Auxiliary Pressure-Control Valve Outlet Isolation was found misaligned. Other valves in the system maintained the integrity of the system flowpath (Section O2.1).

- The licensed operator response to both Unit 2 reactor trips was considered excellent. Operators' actions to manually trip the reactor in response to loss of feedwater was appropriate and conservative (Sections O1.2 and O1.3).

- One noncited violation was identified because Standby Diesel Generator 11 had been inoperable for greater than the Technical Specification allowed outage time. This was an event-revealed and licensee corrected violation (Section O8.1).

Maintenance

- The plant systems and equipment functioned as designed following both Unit 2 reactor trips and the response was indicative of excellent material condition (Sections O1.2 and O1.3).

- Corrective maintenance and surveillance activities were generally performed by knowledgeable technicians in a quality manner using effective communications and in accordance with licensee procedures and Technical Specifications requirements (Sections M1.1 and M1.2).

- The surveillance procedure enhancement program was effective in addressing procedural problems. Although minor deficiencies were noted, none of the deficiencies affected either the validity of the test nor equipment operability (Section M3.1).

- A noncited violation was identified for the failure to test reactor trip bypass breakers prior to placing in service. This was a licensee-identified and corrected violation (Section M8.1).

- A noncited violation was identified for the failure to test a relay contact in the load shedding circuitry. This was a licensee-identified and corrected violation (Section M8.2).

Engineering

- Engineering personnel reviewing a Unit 2 reactor trip were thorough and supported plant operations (Section O1.2).

- Engineering products reviewed were thorough and, where applicable, contained proper unreviewed safety question determinations (Sections E1.1 and E1.2).

- A noncited violation was identified because licensee engineers failed to ensure that an alternate power source met station blackout rule requirements. This was a licensee-identified and corrected violation (Section E8.1).

Plant Support

- The observed activities involving radiological controls, plant chemistry activities, and the maintenance of emergency response facilities and equipment were well conducted and controlled (Sections R1.1, P2.1, and P2.2).

- On one occasion, a security officer demonstrated weak performance in conducting a search of a package entering the protected area (Section S1.1).

- A violation was identified because licensee personnel failed to adequately perform Technical Specification required containment inspection of the Unit 2 reactor containment building resulting in bags of protective clothing remaining in containment after containment integrity had been established. Several staff members knew of the problem but none demonstrated adequate ownership to document the problem to management. This licensee-identified violation was not considered for enforcement discretion in accordance with Section VII.B.1 of the NRC Enforcement Policy because of the regulatory concerns and potential safety significance associated with the operability of containment emergency sumps. In addition, the corrective actions for Violation 498/96004-01 should have prevented this event, and did not (Section R4.1).

- Inadequate corrective actions taken following the identification of bags in the Unit 2 containment on February 24 resulted in additional loose debris remaining in containment during power operations until March 27 (Section R4.1).

- The fire watch personnel were knowledgeable in the classification of fires and the appropriate type of fire extinguisher to use for each. The licensee's fire protection audit frequency was satisfactory (Section F8.1 and F8.2).

Report Details

Summary of Plant Status

At the beginning of this inspection period, Unit 1 was operating at 100 percent reactor power. On April 4 licensed operators began reducing reactor power and on April 5, the unit was removed from service for the purpose of testing rod cluster control assemblies. Following the completion of testing, the reactor was restarted and the main generator breaker was closed. The reactor was at 15 percent power, and power escalation was continuing at the end of this inspection period.

At the beginning of this inspection period, Unit 2 was in Mode 5, conducting Refueling and Equipment Outage 2RE05. The reactor was restarted on February 24 and on February 25, the main generator output breaker was closed. On March 1, following core physics testing, 100 percent reactor power was achieved.

On March 19, the Unit 2 reactor tripped when the main turbine inadvertently unlatched and the turbine trip and throttle valves closed. This event is further described in Section O1.2 of this inspection report. On March 21, after correcting the cause of the trip, the reactor was restarted. On March 22, the main generator output breaker was closed. On March 23, the unit achieved 100 percent power.

On March 26, licensed operators initiated a manual trip of the Unit 2 reactor following a loss of feedwater to Steam Generator 2B. This event is further described in Section O1.3 of this inspection report. After correcting the cause of the trip, the reactor was restarted on March 28 and the main generator output breaker was closed. On March 29, 100 percent power was achieved. The unit was operating at 100 percent power at the end of this inspection period.

I. Operations

O1 Conduct of Operations

O1.1 Control Room Observations (Units 1 and 2)

a. Inspection Scope (71707)

Using Inspection Procedure 71707, the inspectors routinely observed the conduct of operations in the Units 1 and 2 control rooms. Frequent reviews of control board status, routine attendance at shift turnover meetings, observations of operator performance, and reviews of control room logs and documentation were performed. The inspectors observed portions of the following evolutions in addition to full power operations:

- Unit 1 shutdown and rod cluster control assembly testing (April 5)
- Unit 2 postrefueling outage plant startup (February 23 through 25)
- Unit 2 reactor trip response and recovery (March 19)
- Unit 2 reactor trip response and recovery (March 26 and 27)
- Unit 2 posttrip plant startup (March 28)

b. Observations and Findings

During routine observations and interviews, the inspectors determined that the control room operators were continually aware of existing plant conditions. Operators responded to annunciator alarms in accordance with approved procedures. Annunciator alarms were promptly announced to the control room staff who, in turn, acknowledged by restating the announcement. The unit supervisors remained cognizant of ongoing activities. Communications techniques utilized during radio contacts routinely met management expectations.

The inspectors routinely attended shift turnover meetings. The on-shift operators provided clear and concise information to the oncoming operators. Oncoming operators routinely reviewed the control room logs, discussed current plant conditions, and verified major equipment status. Plant managers and operations department managers were often observed attending shift turnover meetings.

c. Conclusions

The inspectors concluded that licensed operators in the control room performed in a professional manner and were continuously aware of existing plant conditions with a good focus on safety. Shift turnover meetings were thorough and routinely attended by plant management.

O1.2 Reactor Trip During Electrohydraulic Control System Testing (Unit 2)

a. Inspection Scope (93702)

At 3:55 p.m. on March 19, with Unit 2 operating at 100 percent of rated thermal power, the reactor tripped when 2 of 4 turbine trip and throttle valves closed. The inspectors responded to the control room and observed the licensed operators' response to this event. In addition, a portion of the turbine-generator building was toured to verify reports of minimal equipment failures. Following reactor coolant system stabilization, the inspectors reviewed the following licensee documents:

- Plant Operating Procedure 0POP05-E0-E000, Revision 8, "Reactor Trip or Safety Injection"
- Plant Operating Procedure 0POP05-E0-ES01, Revision 12, "Reactor Trip Response"
- Plant Operating Procedure 0POP07-TM-0003, Revision 4, "Main Turbine Emergency Trip System"
- Plant General Procedure 0PGP03-Z0-0022, Revision 4, "Post-Trip Review Report"

- Condition Report 97-5730
- Event Review Team Report
- Event Notification Worksheet

b. Observations and Findings

Prior to the reactor trip, operators were performing Procedure 0POP07-TM-0003, as a postmaintenance test following the calibration of the main turbine electrohydraulic control system fluid auto stop pilot pressure, high pressure switch. During the test performance, a licensed operator tripped one channel of the emergency turbine trip circuitry. During the time that the channel was tripped, a negative spike in electrohydraulic control system fluid pressure was indicated. Licensee engineers stated that this pressure spike most likely caused the turbine to unlatch. Nine seconds after the spike, all four turbine trip and throttle valves shut. The reactor automatically tripped on a main turbine trip signal generated when the first two valves closed.

All control rods fully inserted into the core and all safety systems functioned properly following the trip. The main feedwater system isolated on low average temperature, and the auxiliary feedwater system actuated on low steam generator water levels. The inspectors observed licensed operator response. Annunciator alarms were observed and acknowledged quickly, plant parameters were controlled within normal ranges, and the emergency operating procedures were being followed. The inspectors noted that the few secondary components that failed to operate did not adversely affect plant trip response nor complicate the unit recovery.

During troubleshooting activities, instrumentation and controls technicians identified that the auto stop solenoids in the electrohydraulic control system Channel 2 were chattering. Engineers determined that this may have resulted in the system pressure spike during testing of Channel 1. Further investigation identified that the power supply inverter had failed and was intermittently providing reduced voltages. The inverter was replaced along with all four auto stop solenoids. A review of a modification to the electrohydraulic system conducted during the recent refueling and equipment outage was addressed and documented in Section E1.2 of this inspection report.

The inspectors reviewed the event review team's preliminary report and the licensee's posttrip review prior to the plant restart. No discrepancies nor new information were identified during the review. Engineering personnel effectively supported plant operations in response to the event and in determining its cause. Plant operators restarted the reactor and the unit was returned to 100 percent rated thermal power.

c. Conclusions

Plant operators responded to the Unit 2 reactor trip in an excellent manner. Plant equipment functioned as designed with a few minor exceptions indicating outstanding material condition of primary and secondary systems. The engineering review of the event was thorough and supported plant operations.

O1.3 Manual Reactor Trip Following Feedwater Regulating Valve Failure (Unit 2)

a. Inspection Scope (93702)

At 9:47 p.m. on March 26, with Unit 2 operating at 100 percent of rated thermal power, licensed operators initiated a manual reactor trip when Steam Generator 2B water level reached 36 percent with no indicated feedwater flow. The inspectors responded to the control room and observed the licensed operators' response to this event. Following reactor coolant system stabilization, the inspectors reviewed the following licensee documents:

- Plant Operating Procedure 0POP05-EO-E000, Revision 8, "Reactor Trip or Safety Injection"
- Plant Operating Procedure 0POP05-EO-ES01, Revision 12, "Reactor Trip Response"
- Plant General Procedure 0PGP03-ZO-0022, Revision 4, "Post-Trip Review Report"
- Condition Report 97-6144
- Event Review Team Report
- Event Notification Worksheet

b. Observations and Findings

Prior to the trip, the primary reactor operator responded to an annunciator alarm indicating a mismatch between steam flow and feedwater flow in Steam Generator 2B. The operator observed a reduction in feedwater flow to Generator 2B with an associated drop in water level. The operator placed Feedwater Regulating Valve 2B in manual and attempted to open the valve. However, the valve's controller already indicated a 100 percent demand signal. When the Steam Generator 2B water level decreased to approximately 36 percent on the narrow range level indicators, the shift supervisor directed the reactor operator to manually trip the reactor.

All control rods fully inserted into the core and all safety systems functioned properly following the trip. The main feedwater system isolated on low average temperature and the auxiliary feedwater system actuated on low steam generator water levels. The inspectors observed licensed operator response. Licensed operators followed plant operating procedures throughout the recovery. Controls were manipulated in a careful and methodical manner. Shift supervision provided appropriate levels of oversight in ensuring that plant parameters were being maintained.

Investigation revealed that a relay in the actuation circuitry for Feedwater Regulating Valve 2B had failed. This failure caused the valve to close despite automatic and manual demand for the valve to open. The resulting loss of feedwater to the Steam Generator 2B resulted in an unrecoverable loss in water level. Reactor operators responded appropriately and conservatively in performing a manual reactor trip.

The inspectors reviewed the posttrip review. No deficiencies were noted. The plant responded well following the trip. All plant equipment functioned as expected with the exception of one additional secondary system valve. This was indicative of outstanding plant system and equipment material condition prior to the trip.

c. Conclusions

The licensed operators decision to manually trip the reactor was both appropriate and conservative. Operator response following the trip was considered excellent. The response of plant systems and equipment following the reactor trip was indicative of excellent material condition.

O2 Operational Status of Facilities and Equipment

O2.1 Plant Tours (Units 1 and 2)

a. Inspection Scope (71707)

The inspectors routinely toured the accessible portions of plant areas in Units 1 and 2. Areas of special attention during this inspection period included:

- Units 1 and 2 standby diesel generators
- Units 1 and 2 isolation-valve cubicles
- Units 1 and 2 turbine-generator buildings
- Unit 1 essential cooling water pump cubicles
- Units 1 and 2 fuel handling buildings
- Units 1 and 2 mechanical auxiliary buildings
- Unit 2 reactor containment building

b. Observations and Findings

The inspectors found that plant equipment was maintained in excellent material condition. Plant housekeeping was good. However, several minor deficiencies were noted, including debris located inside the reactor containment building as documented in Section R4.1 of this inspection report. All deficiencies were communicated to the appropriate shift supervisor and were corrected. Licensee management was routinely observed in the plant monitoring plant equipment and work activities for proper implementation of expectations.

On March 26, during a routine tour of the Unit 2 turbine-generator building, the inspector observed that Valve 2-MS-0214, the Main Steam to Auxiliary Steam Pressure-Control Valve Outlet Isolation, was closed. A review of the associated piping and instrumentation diagram and plant operating procedure indicated that the valve was improperly positioned. Operators verified the appropriate alignment and opened the valve. The mispositioning was not considered to be safety significant because additional valves in the system maintained the integrity of the system flowpaths.

Condition Report 97-6118 was written to address the issue. Operations personnel reviewed the event and determined that the valve had been mispositioned during a diagnostic leak search. Management stated that corrective actions would be taken to prevent recurrence.

c. Conclusions

The inspectors concluded that equipment material condition and safety system availability were excellent. Licensee management was actively monitoring plant areas for material condition. One main steam system valve was found to be mispositioned, but other valves in the system maintained integrity of the system flowpath.

O8 Miscellaneous Operations Issues (92901)

O8.1 (Closed) Licensee Event Report (LER) 50-498/94-012: Failure to meet the requirements of Technical Specifications. Standby Diesel Generator 11 was inoperable as a result of an intermittent failure of the K1 contactor for the voltage-regulator/field-flash circuit on February 3, and March 1, 1994. Technical Specification 3.8.1.1.b required restoration of an inoperable diesel generator to an operable condition within 72 hours.

The February 3, 1994 failure was erroneously diagnosed as faulty contacts on the VR1 (voltage release) relay that supplied power to the field-flash relay. The normally closed contacts were found to be intermittently open, discolored, and badly pitted. The VR1 relay was replaced, but other similar relays and the K1

contactor were tested satisfactory. The standby diesel generator was retested and placed back in service. Three subsequent surveillance tests of Standby Diesel Generator 11 were conducted satisfactorily.

On March 1, during the fourth start subsequent to the February 3rd failure, the diesel generator again failed to obtain normal voltage and frequency during its emergency-mode start test. Subsequent troubleshooting determined that the K1 contactor periodically failed to reset. Therefore, the licensee determined that Standby Diesel Generator 11 had been inoperable since February 3, based on a failed K1 contactor. This exceeded the Technical Specification limiting condition for operation allowed outage time of 72 hours. The cause of not meeting the requirements of the Technical Specifications was stated to be the lack of recognition that the intermittent failure of the K1 contactor was the cause of the February 3 event.

The following corrective actions were performed:

- Voltage release relays in voltage regulator circuits and the K1 contactor of Standby Diesel Generator 11 were replaced.
- The preventive maintenance activity for the K1 contactor was enhanced.
- Preventive maintenance procedures for the field-flash circuits of all six standby diesel generators were developed.

The inspector determined that these corrective actions were satisfactory to prevent recurrence. The failure to restore the inoperable diesel generator to an operable condition within 72 hours was a violation of Technical Specification 3.8.1.1.b. However, because the failure was intermittent, the K1 contactor had tested satisfactorily during the troubleshooting activity and subsequent successful starts. This self-disclosing event and licensee corrected violation is being treated as a noncited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (498/97002-02).

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments on Field Maintenance Activities

a. Inspection Scope (62707)

The inspectors observed portions of the following on-going work activities identified by their work authorization numbers:

Unit 1:

- 38632 — Calibration of Train A Essential Chiller Air Handling Unit 11A High Temperature Switch
- 45446 — Internal Inspection and Overhaul of Essential Cooling Water Self-Cleaning Strainer 1A
- 45395 — Inspection and Replacement of Essential Cooling Water 1A Motor Air Filters
- 47382 — Service the Coupling on Essential Cooling Water Screen Wash Booster Pump

Unit 2:

- 105067 — Hydrostatic Test of the Drain Line for Auxiliary Feedwater Pump 24
- 106804 — Replacement of a Main Turbine Electrohydraulic Control System Channel 4 Inverter

b. Observations and Findings

The inspectors found that the work performed during these activities was conducted in a thorough and professional manner. The work was performed by knowledgeable, qualified technicians utilizing approved procedures. Prejob briefings were thorough and prejob work risk assessments were performed and approved in accordance with Plant General Procedure 0PGP03-ZA-0090, Revision 8, "Work Process Program." System engineers were observed providing quality technical support as needed.

On one occasion, during the replacement of an electro-hydraulic control system inverter, the inspectors observed a supervisor performing direct work activities. Licensee management concurred that this was contrary to management's expectations for craft supervision in the field.

c. Conclusions

The activities observed were conducted by qualified technicians in a professional manner. Personnel involved were thorough in the implementation of the related maintenance program requirements. On one occasion, a supervisor performed direct field work, contrary to managements expectations.

M1.2 General Comments on Surveillance Testing

a. Inspection Scope (61726)

The inspectors observed portions of the following surveillance activities:

Unit 1:

- Plant Surveillance Procedure 0PSP02-CM-4105, Revision 0, "Containment Hydrogen Analyzer"
- Plant Surveillance Procedure 0PSP02-PK-0003, Revision 3, "4.16 KV Class 1E Undervoltage Relay Channel Calibration/TADOT - Channel 3"
- Plant Surveillance Procedure 0PSP06-PK-0007, Revision 5, "4.16 KV Class 1E Degraded Voltage Relay Channel Calibration/TADOT - Channel 3"
- Plant Surveillance Procedure 1PSP05-AF-7523, Revision 2, "Aux Feedwater Flow Loop 3 Channel C Calibration (F-7523)"
- Plant Surveillance Procedure 0PSP03-FC-0002, Revision 5, "Spent Fuel Pool Cooling Pump 1A Inservice Test"

Unit 2:

- Plant Surveillance Procedure 0PSP03-DG-0001, Revision 7, "Standby Diesel 21 Operability Test"

b. Observations and Findings

The inspectors found that the observed surveillance activities were performed in accordance with approved procedures. The inspectors noted some minor deficiencies in the procedures. They included inconsistent equipment nomenclature, inconsistent notes, and steps that were designated as containing acceptance criteria that did not. Nevertheless, Technical Specification surveillance requirements were correctly implemented. The inspectors verified that the test equipment calibrations were current. Good communications between the control room operators and personnel performing the tests were noted. Prejob briefings were thorough and comprehensive.

c. Conclusions

The surveillance activities observed were performed in accordance with the applicable Technical Specifications.

M3 Maintenance Procedures and Documentation (92902)

M3.1 (Closed) Violation 498/94010-01: Failure of Operators to Follow Plant Surveillance Procedures

This violation documented several examples of the failure to follow surveillance procedures. The problems were partly attributable to procedural inadequacies. The specific corrective actions in response to the violation were reviewed and documented as being adequate in NRC Inspection Report 50-498/95-23; 50-499/95-23. However, the violation remained open to track the status of the surveillance procedure enhancement program. The NRC's Diagnostic Evaluation Team Report, dated June 10, 1993, documented the licensee's original commitment to establish a surveillance procedure enhancement program. Subsequently, as documented in NRC Inspection Report 50-498/95-15; 50-499/95-15, the inspectors performed a programmatic review of the results from the enhancement program. The inspectors determined that the enhancement program had resulted in a stronger surveillance program. However, the licensee had not completed the program at the time of that inspection.

The inspectors reviewed the surveillance procedure enhancement program including:

- Observations of maintenance personnel performing Technical Specification required surveillance tests in accordance with enhanced procedures.
- Discussions regarding and reviews of the status of the enhancement program with appropriate licensee management personnel.

The specific observations of testing were documented in Section M1.2 of this inspection report. The inspectors noted some minor procedure deficiencies that included: inconsistent equipment nomenclature, inconsistent precautionary notes, and misleading information in steps designated as containing test acceptance criteria. None of the deficiencies affected either the validity of the test nor equipment operability. All deficiencies were discussed with the appropriate licensee contact for evaluation and correction.

As of the date of this inspection, the licensee had completed upgrading all of the operations, mechanical maintenance, and electrical maintenance procedures and approximately 60 percent of the instrumentation and controls procedures. The schedule was prioritized based on the risk associated with procedure performance. Those that impacted high risk systems as defined by the probabilistic safety assessment were upgraded first.

Licensing organization personnel stated that the surveillance procedure enhancement program was closed out on December 20, 1995. This was based on the assessment of low risk for the remaining procedures. Although the completion

status of the remaining procedures was not tracked, enhancements continued to be made using the guidelines developed for the enhancement program.

A large number of surveillance procedures were scheduled to be upgraded in accordance with the Improved Technical Specifications following their approval and issuance. Approximately one third of the operations and instrumentation and controls procedures will require revision as part of the Improved Technical Specifications implementation process.

The inspectors noted that nine LERs were issued in 1994 and two in 1995 related to surveillance procedure technical adequacy. There were no reports issued in 1996 related to procedure adequacy. This indicated that the surveillance procedure enhancement program had been effective in addressing procedure problems.

The inspectors concluded that the licensee's corrective actions to address problems with surveillance procedure technical adequacy were thorough. Current revisions of surveillance procedures reviewed, properly implemented Technical Specifications surveillance requirements. Therefore, no additional tracking or review of the surveillance procedure enhancement program was deemed necessary.

M8 Miscellaneous Maintenance Issues (92902)

M8.1 <u>(Closed) LER 50-498/94-007</u>: Reactor trip bypass breaker testing had not been performed in accordance with Technical Specifications.

On February 22, 1994, licensee engineers discovered that the station procedure for testing the reactor trip bypass breakers did not properly satisfy Technical Specification 4.3-1, Surveillance Requirement 22. This required a test of the bypass breaker local manual shunt trip prior to placing the bypass breaker in service. The surveillance procedure had specified testing the bypass breaker after it had been racked in and closed. Three other station procedures were identified with the same problem. The root cause of this event was determined to be inadequate procedure preparation and review. The Technical Specification requirement had not been properly incorporated into the procedure when it was written and reviewed.

The following corrective actions were performed:

- The surveillance procedure enhancement program was conducted as documented in Section M3.1 of this inspection report.
- A team consisting of a procedure writer, an engineer, and a representative of the procedure users' organization was established for each surveillance procedure to ensure that assigned procedures were administratively and technically correct and able to be performed as written.
- The associated procedures were appropriately revised.

The inspector determined that these corrective actions were satisfactory to prevent recurrence. The failure to test the bypass breaker local manual shunt trip prior to placing the bypass breaker in service was a violation of Technical Specification 4.3-1, Item 22. This licensee-identified and corrected violation is being treated as a noncited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (498;499/97002-03).

M8.2 (Closed) LER 50-498/95-004: Failure to meet the requirements of the Technical Specifications by not testing a contact of a load sequencer relay.

On March 28, 1995, while performing a review of procedures as part of the surveillance procedure enhancement program, engineers determined that Contact 51-52 on Load Sequencer Relay K243 had not been tested during the performance of the Train A loss-of-offsite-power surveillance test. This condition failed to meet the requirements of Technical Specification 4.8.1.1.2.e.4.b, that required the diesel generator to energize the auto-connected shutdown loads through the load sequencer when a simulated loss-of-offsite-power signal was initiated. Contact 51-52 initiated the start of several fan loads after the shedding of bus loads. Therefore, the failure to test this contact resulted in the failure to ensure energization of these auto-connected loads. The root cause for this failure was determined to be less than adequate preparation, review, and revision of the surveillance test procedure.

The following corrective actions were performed:

- The operability of the specific loads in question was verified by review of completed surveillance test data for all trains in both units.

- The associated surveillance procedures were revised appropriately.

The inspector determined that these corrective actions were satisfactory to prevent recurrence. However, the failure to ensure that the diesel generator energized the auto-connected shutdown loads through the load sequencer following shedding of bus loads when a simulated loss-of-offsite-power signal was initiated, was in violation of Technical Specification 4.8.1.1.2.e.4. This licensee-identified and corrected violation is being treated as a noncited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (498;499/97002-04).

M8.3 (Closed) LER 50-498/94-020: Failure to fully meet the surveillance requirements of the Technical Specifications because previous testing of the adjustable molded-case circuit breakers had been inadequate.

On December 8, 1994, an NRC inspector determined that the licensee had misapplied the surveillance test tolerance for the 480 VAC adjustable magnetic molded-case circuit breakers used for containment penetration protection required

by Technical Specification 4.8.4.1.a.2. A combined total of 66 breakers from the two units were determined to be potentially out-of-tolerance. This event was caused by misinterpretation of the acceptance criteria.

The following corrective actions were preformed:

- The surveillance test packages for potentially impacted breakers were reviewed.

- A formal procedure revision was developed incorporating use of the appropriate acceptance criteria.

- The 66 breakers identified as inoperable during the review were tested utilizing the appropriate acceptance criteria band.

The inspector determined that these corrective actions were satisfactory to prevent recurrence. The failure to properly test these circuit breakers was cited as a violation in NRC Inspection Report 50-498/94-35; 50-499/95-35. This violation was closed as documented in Section M8.4 of this inspection report.

M8.4 (Closed) Violation 498/94035-01: Failure to perform functional testing of the instantaneous trip element of molded-case circuit breakers by injecting a test current within a certain tolerance of the element pickup value.

The licensee had acknowledged this violation in a letter dated February 22, 1995. As discussed in Section M8.3 of this inspection report, the corrective actions included a review of the surveillance test packages for potentially impacted breakers and additional testing of a total of 66 breakers identified as being inoperable. These corrective actions were reviewed and found to be acceptable.

M8.5 (Closed) Inspection Followup Item 498;499/94025-02: Inspect the licensee's programs for and performance of infrequently performed evolutions, surveillance test procedures and licensee upgrade efforts, equipment clearance implementation, and the effectiveness of the licensee's corrective actions.

The inspector reviewed selected NRC inspection reports concerning the increased focus on the above issues. The subject areas had been reviewed during core, regional-initiative, and reactive inspections in accordance with the master inspection plan. Therefore, continued tracking of these areas are no longer necessary and this IFI is administratively closed.

III. Engineering

E1 Conduct of Engineering

E1.1 Evaluation of the Replacement of a Standby Diesel Generator K1 Relay (Unit 1)

a. Inspection Scope (37551)

On March 12, 1997, Standby Diesel Generator 13 failed to develop an output voltage during routine surveillance testing. On March 11, maintenance technicians replaced the K1 relay in the generator start circuitry with a new design as part of a program to phase out obsolete equipment. The K1 relay was designed to flash the generator field when the diesel started. Licensee engineers determined that the relay had failed resulting in the valid failure of the diesel generator. The inspectors reviewed the following design documents related to this event:

- Design Change Package 96-4039-8, Revisions 0 and 1
- Replacement Item Equivalency Evaluation 95-10944-1
- Condition Report Work Order, Work Authorization Number 82122

b. Observations and Findings

Following the initial relay replacement, the standby diesel generator successfully passed the postmaintenance and surveillance operability tests. However, on March 12, licensed operators started the diesel a third time for the Technical Specification required monthly surveillance test. During the test, the generator failed to develop an output voltage.

Licensee engineers determined that the new K1 field-flash relay had failed. Technicians reinstalled a relay of the original design and sent the failed relay to a laboratory for failure analysis. The standby diesel generator then passed the monthly surveillance operability test.

The inspectors did not identify any deficiencies in the design change packages or item equivalency evaluation. The equivalency evaluation was thorough. The condition report work order contained the appropriate installation information. An unreviewed safety question evaluation had been performed in accordance with 10 CFR 50.59. This evaluation was complete and properly conducted.

c. Conclusions

The engineers developed the design change packages and item equivalency evaluation associated with a standby diesel generator relay replacement in a thorough manner.

E1.2 Review of Electro-hydraulic Control System Modification (Unit 2)

a. Inspection Scope (37551)

As documented in Section O1.2 of this inspection report, a perturbation in the electrohydraulic control system resulted in a reactor trip. Following that trip, the inspectors reviewed a modification that had been implemented in the system during Refueling and Equipment Outage 2RE05. The inspectors reviewed the following associated documents:

- Design Change Package 95-5755-17
- Updated Final Safety Analysis Report Section 10.2.1
- Change Notice 2117
- 10 CFR 50.59 Evaluation Screening Form
- Westinghouse Operation and Maintenance Memo 085
- Unreviewed Safety Question Evaluation 96-0060
- Plant Operating Procedure 0POP07-TM-0003, Revisions 3 and 4

b. Observations and Findings

The modification reviewed was performed to remove the low hydraulic system pressure trip function from the main turbine protective system. The associated trip relays were removed, and the turbine trip first-out annunciator in the main control room was disabled and the panel removed. This was performed to improve system health by reducing spurious channel trips and improve the ability to latch the main turbine. The vendor provided operational information indicating that the removal of the trip was optional and recommending the method of removal. The inspectors determined that the vendor's recommendations had been followed.

The unreviewed safety question determination was properly performed and documented. Proper safety conclusions were drawn. The inspectors determined that a change to the Updated Final Safety Analysis Report was being processed to revise the appropriate drawings and words. In addition, the inspectors determined that no evidence existed that the modification had resulted in the March 19 reactor trip.

c. Conclusions

The documentation associated with a modification to the electro-hydraulic control system was thorough and complete. The unreviewed safety question determination met the requirements of 10 CFR 50.59.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) LER 50-498/94-013: Failure to fully meet the requirements of the station blackout rule.

On August 4, 1994, the licensee conducted a reportability review that determined that they were not fully meeting the requirements of NUMARC 87-00, which implemented 10 CFR 50.63, "Loss of All Alternating Current Power." A self-assessment of the station blackout commitments had identified two cases where the criteria of 10 CFR 50.63 were not being satisfied.

The cases related to the physical protection and electrical vulnerabilities of alternate AC power sources used for station blackout. Although the South Texas Project design utilized a standby diesel generator for providing electric power, the transformers used to transfer the power to the companion train instrumentation were not protected from likely weather-related events and were a single point of vulnerability for preferred and station blackout power. In addition, 10 CFR 50.63(c)(2) required a demonstration test of the time required to provide power from the alternate AC source and associated equipment. The required test had not been performed. The root cause of this event was determined to be a lack of familiarity by engineering personnel with the history of the station blackout initiative.

The following corrective actions were performed:

- The severe weather guidelines were changed to provide for plant shutdown prior to any predicted hurricane landfall, rather than prior to hurricane winds in excess of 120 mph.
- The staffing of the design engineering-electrical group was doubled to one supervisor and ten engineers.
- A self-assessment of the station blackout program was conducted and corrective actions taken.
- A revised station blackout position was submitted to the NRC staff for review.

The inspector determined that these corrective actions fully met the requirements of the station blackout rule. In addition, the NRC staff had concurred with the revised position and formally issued a safety evaluation report on July 24, 1995. However, the failure to initially ensure that alternate AC power met the station blackout rule criteria was a violation of 10 CFR 50.63. This licensee-identified and corrected violation is being treated as a noncited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (498;499/97002-05).

E8.2 (Closed) IFI 498;499/94025-04: Inspect the licensee's programs for performing: 10 CFR 50.59 evaluations; inservice testing; station blackout rule implementation; engineering backlog reduction; corrective actions associated with inappropriate use of plant change forms, and the new modification process.

The inspector reviewed selected NRC inspection reports concerning the increased inspection on the above issues. The subject areas had been reviewed during core, regional-initiative, and reactive inspection in accordance with the master inspection plan. Therefore, continued tracking of these areas are no longer necessary and this IFI is administratively closed.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 Tours of Radiological Controlled Areas

a. Inspection Scope (71750)

The inspectors routinely toured the mechanical-auxiliary and fuel handling buildings in Units 1 and 2. These tours included observation of work, verification of proper radiological work permits, sampling of locked doors, and observations of personnel entrance and egress from the radiological controlled areas.

b. Observations and Controls

Radiological housekeeping in the areas toured was very good. Doors required to be locked in accordance with Technical Specification 6.12.2 and the licensee's radiological program were properly secured. No entrance/egress discrepancies were identified.

c. Conclusions

Routine radiological controls observed were considered in place and effective.

R1.2 Secondary Chemistry Controls

The inspectors routinely reviewed secondary water chemistry reports and radiation monitor alarm status. Secondary chemical analysis, the calculated primary to secondary leak rate, and indication from the Nitrogen-16 radiation monitors all confirmed steam generator tube integrity. The chemical analysis results provided evidence of management attention and commitment to maintaining chemistry parameters within appropriate limits.

R4 Staff Knowledge and Performance in Radiological Protection and Chemistry Controls

R4.1 Review of the Circumstances Surrounding Items Inadvertently Left Inside the Reactor Containment Building (Unit 2)

a. Inspection Scope

On February 24, 1997 with Unit 2 in Mode 3, a radiological protection technician found three bags containing protective clothing inside the reactor containment building on the 34-foot elevation near the auxiliary airlock. Subsequently, on March 27, during a posttrip walkdown of the Unit 2 reactor containment building, an operator found piping insulation on the floor of the regenerative heat exchanger room. Both of these events occurred after containment integrity had been established and a Technical Specification required performance of Plant Surveillance Procedure 0PSP03-XC-0002, "Containment Inspection," had been completed.

The inspectors reviewed the licensee's response and corrective actions to these events in relation to the corrective actions of a previous similar event. This inspection included a review of the following documents:

- Condition reports, licensee investigation reports, and corrective actions associated with the two events.
- LERs 50-498/96-003 and 50-499/97-003 and the supplement to LER 50-499/97-003.
- Notice of Violation 498/96004-01 and the licensee's response.
- Revisions to Procedure 0PSP03-XC-0002 in effect during these events.

The inspectors discussed these reviews with the appropriate operations, engineering and management personnel. In addition, the inspectors performed a detailed walkdown of several areas in containment.

b. Event Description

On February 23, a licensed operator completed a Unit 2 containment inspection in accordance with Procedure 0PSP03-XC-0002, Revision 11, prior to Mode 4 entry. Several maintenance jobs were performed inside containment during and after the performance of this surveillance procedure. It was management's stated expectation that the maintenance craftsmen working inside containment would perform and complete Form 1, "Partial Containment Inspection For Loose Debris." This form required personnel to inspect the work area and remove all loose debris following work completion.

On the morning of February 24, a radiological protection technician found three partially open bags containing three sets of protective clothing, a radiological control sign, and some plastic booties inside the reactor containment building. The material was immediately removed from containment. Condition Report 97-3989 was issued to document and address this event.

On March 27, during a posttrip Unit 2 containment walkdown, two pieces of insulation and other small items were found on the floor of the regenerative heat exchanger room. The material was removed from containment and Condition Report 97-6147 was written to document and address this issue.

c. History

On May 16, 1996, while Unit 1 was at 100 percent power, a health physics technician had found plastic bags containing prestaged equipment inside the reactor containment building during a containment inspection. As documented in NRC Inspection Report 50-498/96-04; 50-499/96-04, the equipment had been placed in containment in preparation for a refueling outage. Licensee engineers had performed an evaluation for the placement of the equipment inside containment but had not taken into account the potential impact of plastic bags on the emergency sump. Technical Specification 4.5.2.c.2 states that the emergency core cooling system be verified to be operable by a:

> Visual inspection of the affected areas within containment at the completion of each containment entry when containment integrity is established to verify no loose debris is present which could be transported to the containment sump and cause restriction of pump suctions during LOCA conditions.

The inspectors had determined that Procedure 0PSP03-XC-0002, Revision 9, that had been intended to implement Technical Specification 4.5.2.c.2, had been inadequate to ensure that the surveillance requirements were properly performed. The inadequate containment inspection had been cited as a violation of Technical Specification 6.8.1 and documented as Violation 498/94004-01.

d. Observations and Findings

The inspectors reviewed the condition reports and subsequent investigation reports for the February 24 and March 27 events. In each event, licensee personnel removed the loose material from containment. Engineering evaluations were performed to determine safety significance of the events.

The inspectors reviewed the engineering evaluation for the items found inside the reactor containment building. Engineers determined that the amount of containment emergency sump blockage represented by the material found on February 24 was greater than allowed blockage as determined by the minimum emergency core

cooling system net positive suction head stated in Table 6.3-1 of the Updated Final Safety Analysis Repc However, they further determined that it was not likely to reach the emergency sumps because of a torturous path between the two locations. The basis for this assumption was that material on or above the 19-foot elevation could not be transported to the containment sumps. This assumption was contained in Calculation MC-6220, "Safety Injection and Containment Spray Pump Net Positive Suction Head." Assumption 13 in MC-6220 stated that objects at or above the 19-foot elevation and outside of the bioshield would not reach the sumps unless they could fit through the grating spaces. The inspectors noted that this assumption did not account for the stair wells or the spaces between the grating and the containment walls.

In addition to these actions, a containment reinspection was conducted by a team consisting of two senior reactor operators, two reactor plant operators, and two radiological protection technicians on February 24. The shift supervisor stated that approximately 3/4 of a cubic foot of miscellaneous debris was removed from containment during the inspection. This inspection did not identify the debris found on and after March 27.

After removing the material from containment on March 27, five licensee managers conducted a detailed inspection of the reactor containment building. One manager was assigned to each level of the building. The inspectors reviewed the list of items removed from containment during the management inspection and determined that it could not have significantly restricted flow to the emergency sumps. The Unit 1 reactor containment building was also inspected, and a small amount of debris was removed from the building.

The inspectors toured selected, accessible portions of the Unit 2 reactor containment building concurrent with the management inspections. The inspectors found less than one third of a cubic foot of loose debris during the tour that included areas inside the bioshield.

Licensee management determined that a failure to communicate management's expectation for the control of loose debris in the reactor containment building to personnel working inside containment had resulted in these events. The inspectors discussed the events with licensee personnel involved in reviewing these events and determined that a lack of communication during containment decontamination and closeout resulted in the bags left in containment. Each crew working in containment thought that another crew would remove the bags, which demonstrated inadequate ownership of a known problem. The inspectors also concluded that the March 27 event indicated inadequate corrective action for the February 24 event.

The inspectors reviewed the results of the performances of Procedure 0PSP03-XC-0002 conducted on February 23 and 24 to implement Technical Specification 4.5.2.c.2 prior to these events. The inspectors determined

that the performances of Procedure 0PSP03-XC-0002 were not adequate to ensure that the surveillance requirements were properly performed. As a result, loose material was incorrectly left in containment after containment integrity had been established and operating modes entered. The failure to properly implement this safety-related procedure was a violation of Technical Specification 6.8.1 (499/97002-01).

e. Conclusions

The inspectors concluded that the failure to remove the loose materials from containment was caused by the inadequate implementation of a Technical Specification required surveillance procedure. This licensee-identified violation was not considered for enforcement discretion in accordance with Section VII.B.1 of the NRC Enforcement Policy because of the regulatory concerns and potential safety significance associated with the operability of containment emergency sumps. In addition, the corrective actions for Violation 498/96004-01 should have prevented this event, and did not. The inspectors also concluded that corrective actions taken following the February 24, 1997 event had been ineffective in ensuring that the material identified on March 27 was removed from primary containment.

P2 Status of EP Facilities, Equipment, and Resources

P2.1 Emergency Response Facilities (71750)

The inspectors observed that the Technical Support Centers and Operations Support Centers in both units were readily available and maintained for emergency operation.

P2.2 Meteorological Towers (71750)

The inspectors routinely observed indication of meteorological conditions in the main control rooms of both units. The data obtained indicated that both the 10-meter and the 60-meter towers remained operable.

S1 Conduct of Security and Safeguards Activities

S1.1 Daily Physical Security Activity Observations (71750)

a. Inspection Scope (71750)

The inspectors observed the practices of security force personnel and the condition of security equipment on a daily basis. On one occasion, the inspector reviewed package search practices at the protected area entrance.

b. Observations and Findings

Protected and vital area barriers were in good condition. Personnel access measures and equipment searches for contraband were observed on a daily basis and were well performed. One exception was noted.

On February 26, the inspectors observed an officer operating an x-ray search machine in the East Gate personnel access point. The officer requested to search a lunch box that had been processed through the x-ray machine. During the search, the officer's attention was taken away from the machine. However, the belt was allowed to continue to move in the forward direction. During this time, the inspector observed an individual remove a bag that had just finished processing through the machine, and proceed to take the bag into the protected area. The inspector noted the weak performance by the officer observing the bag process through the x-ray machine.

This observation was discussed with the Security Force Supervisor who initiated an investigation as documented in Condition Report 97-4353. Statements indicated that neither the officer nor an assisting officer in the area had observed the bag process through the machine. Security force management informed the inspector that a new position identified as Position 7 had been recently implemented. This post was established inside the badge issuance area, and was provided with monitors to observe the X-ray signature of packages processing into the protected area. Statements by this individual indicated that he did not observe contraband entering the protected area.

During the inspection period, the inspectors interviewed several officers posted at Position 7. All officers indicated that the post did not require continuous observation of the monitors. Additionally, several officers were observed to spend a considerable period of time performing alternate duties that took their attention away from the monitors. The inspectors concluded that Position 7 was not an acceptable alternative to the search officer's job performance.

The collected data and interviews were not conclusive in determining that the package observed by the inspector was properly searched prior to entering the protected area. However, management's expectations for the search officers duties were clearly not met. The Manager of Nuclear Plant Protection stated that the search officer was expected to stop the x-ray machine belt prior to performing any hand search. A training bulletin was sent to all officers indicating the responsibilities of operating the x-ray machine.

c. Conclusions

In general, daily security force activities were conducted in an appropriate manner. However, on one occasion, an officer operating the x-ray machine failed to meet

management's expectations regarding proper procedure for hand searching articles entering the protected area.

F8 Miscellaneous Fire Protection Issues (92904)

F8.1 (Closed) IFI 498;499/95001-01: Ineffective training in the fire watch program identified when fire watch personnel could not describe a Class C fire.

The inspector reviewed the following training documents for certifying fire watch personnel:

- Lesson Plan Fire Brigade Training 006.01, Revision 5, "Fire Watch"
- Fire Watch Training Handout FBT006, Revision 5
- Fire Watch Certification Report, dated March 4, 1997

These training documents correctly defined the four classifications of fires. The fires were classified according to the type of material involved and the proper type of fire extinguisher to control each classification. The inspector determined through interviews with three fire watch personnel that they were able to correctly describe a Class C fire and the correct type of fire extinguisher to be used for each classification. In addition, the examination for fire watch personnel required knowledge of fire classifications and the appropriate fire extinguisher for each type.

The inspector concluded that the fire watch personnel were knowledgeable in the classification of fires and what type of fire extinguisher to use for each.

F8.2 (Closed) IFI 498;499/95001-03: Inappropriate audit criteria for fire protection program.

NRC Inspection Report 50-498/95-01; 50-499/95-01 documented that the periodicity requirements for quality assurance audits had been removed from Technical Specification 6.5.2.8 to the quality assurance program in accordance with NUREG-1431, "Standard Technical Specifications-Westinghouse Plants," dated September 1992. However, the inspection report noted that the Quality Assurance Plan, Chapter 15, Revision 5, "Quality Assurance and Audit," did not specify definite frequencies for fire protection audits.

The inspector reviewed the licensee's oversight planning and scheduling process document that defined the audit frequency for quality assurance audits. The audit requirement included a nominal frequency of 24 months for fire protection audits. The inspector concluded that the audit frequency of 24 months was satisfactory.

ATTACHMENT

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

T. Cloninger, Vice President, Nuclear Engineering
W. Cottle, Executive Vice President and General Manager Nuclear
B. Dowdy, Manager, Operations, Unit 2
J. Groth, Vice President Nuclear Generation
E. Halpin, Manager, Maintenance, Unit 2
S. Head, Licensing Supervisor
K. House, Supervising Engineer, Design Engineering Department
M. Kanavos, Manager, Mechanical/Civil Design Engineering
D. LeGrand, Performance Assessment Supervisor, Operations Support
B. Logan, Manager, Health Physics
R. Lovell, Manager, Operations, Unit 1
B. Masse, Plant Manager, Unit 2
G. Parkey, Plant Manager, Unit 1
M. Sicard, Acting Assistant to the Manager, Operations, Unit 2
F. Timmons, Manager, Nuclear Plant Protection
T. Waddell, Manager, Maintenance, Unit 1

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support
IP 92700:	Onsite Followup of Written Reports at Power Reactor Facilities
IP 92901:	Followup - Operations
IP 92902:	Followup - Maintenance
IP 92903:	Followup - Engineering
IP 92904:	Followup - Plant Support

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

499/97002-01	VIO	Inadequate Containment Inspection regarding debris not removed.
498/97002-02	NCV	Standby Diesel Generator 11 inoperable for greater than the Technical Specification allowed outage time

498;499/97002-03	NCV	Failure to test the reactor trip bypass breakers prior to placing in service
498;499/97002-04	NCV	Failure to test contact of vital power load shedding relay
498;499/97002-05	NCV	Failure to ensure that alternate power source met station blackout rule requirements

Closed

50-498/94-012	LER	Valid Failure of Standby Diesel Generator 11 upon K1 Relay Failure
498/97002-02	NCV	Standby Diesel Generator 11 inoperable for greater than the Technical Specification allowed outage time
498/94010-01	VIO	Failure of Operators to Follow Plant Surveillance Procedures
50-498/94-007	LER	Reactor trip bypass breaker testing had not been performed in accordance with Technical Specifications.
498;499/97002-03	NCV	Failure to test the reactor trip bypass breakers prior to placing in service
50-498/95-004	LER	Failure to test a contact of a load sequencer relay.
498;499/97002-04	NCV	Failure to test contact of vital power load shedding relay
50-498/94-020	LER	Testing of the adjustable molded-case circuit breakers had been inadequate.
498/94035-01	VIO	Failure to perform functional testing of the instantaneous trip element of molded-case circuit breakers
498;499/94025-02	IFI	Inspect the licensee's programs for and performance of infrequently performed evolutions, surveillance test procedures and licensee upgrade efforts, equipment clearance implementation, and the effectiveness of the licensee's corrective actions.
50-498/94-013	LER	Failure to fully meet the requirements of the station blackout rule.
498;499/97002-05	NCV	Failure to ensure that alternate power source met station blackout rule requirements

498;499/94025-04 IFI	Inspect the licensee's programs for performing: 10 CFR 50.59 evaluations; inservice testing; station blackout rule implementation; engineering backlog reduction; corrective actions associated with inappropriate use of plant change forms, and the new modification process.
498;499/95001-01 IFI	Ineffective training in the fire watch training program identified when fire watch personnel could not describe a Class C fire.
498;499/95001-03 IFI	Inappropriate audit criteria for fire protection program.