APPENDIX

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-498/88-24   Operating License: NPF-76

Docket: 50-498

Licensee: Houston Lighting & Power Company (HL&P)
P.O. Box 1700
Houston, Texas 77001

Facility Name: South Texas Project, Unit 1 (STP)

Inspection At: STP, Matagorda County, Texas

Inspection Conducted: April 5 through May 2, 1988

Inspectors:

for D. R. Carpenter, Senior Resident Inspector, Reactor Project Section D, Division of Reactor Projects — 5/24/88 Date

for D M Hunnicutt
J. E. Bess, Resident Inspector, Reactor Project Section D, Division of Reactor Projects — 5/24/88 Date

D M Hunnicutt
D. M. Hunnicutt, Senior Reactor Inspector, Reactor Project Section D, Division of Reactor Projects — 5/24/88 Date

Accompanying
Personnel: J. P. Clausner, French Commissariar A L'Energie Atomique, Institute De Protection Et De Surete Nucleaire

Approved: 5/24/88

G. L. Constable, Chief, Reactor Project Section D, Division of Reactor Projects — Date

Inspection Summary

Inspection Conducted April 5 through May 2, 1988 (Report 50-498/88-24)

Areas Inspected: Routine, unannounced inspection including startup testing results review, operability of the steam generator power operated relief valves, No. 13 standby diesel generator oil spill essential cooling water leaks, monthly surveillance observations, security observations, radiological protection observation, operational safety verification, and engineered safety feature system walkdown.

Results: Within the areas inspected, four apparent violations were identified (inadequate documentation of tests, paragraph 3a; inadequate review of test results, paragraph 3c; Operation in TS 3.0.3, paragraph 4; and failure to follow procedure for equipment clearance orders, paragraph 5).

DETAILS

1. Persons Contacted

HL&P

*W. P. Evans, Licensing Engineer
*G. L. Jarvela, HP Division Manager
*J. E. Geiger, General Manager, Nuclear Assurance
*S. L. Rosen, General Manager, Operations Support
*S. M. Head, Support Licensing Engineering
*J. J. Nesrsta, Plant Engineering Manager
*T. J. Jordan, Project Quality Assurance Manager
*D. N. Brown, Construction Manager, Unit 1
*W. H. Kinsey, Plant Manager
*J. N. Bailey, Manager Engineering, Licensing

In addition to the above, the NRC inspectors also held discussions with various licensee, architect engineer (AE), constructor and other contractor personnel during this inspection.

*Denotes those individuals attending the exit interview c ...cted on May 2, 1988.

2. Licensee Action on Previous Inspection Findings 9270

(Closed) Violation (498/8807-01): Failure to Maintain Quality Assurance (QA) Records

This item concerned surveillance test records in the engineering department record file that were found to lack required record retention control, a records custodian with the responsibility for the control of records while in the division file had not been designated and procedures did not exist to control access to the division file. The licensee had taken the following actions: revised Plant Procedure OPGP03-ZE-0004, "Plant Surveillance Program" to include provisions that control access to the division file; provided a one hour fire rated, lockable filing cabinet for document storage; established a log to track the removal of Technical Specifications (TS) packages and provide an index of the file contents; and designated a records custodian.

This violation is considered closed.

3. Startup Testing Results Review

During this period of time, the following test results were reviewed:

- 1PEP04-ZX-0001 Test Sequence for Initial Criticality and Low Power Testing
- 1PEP04-ZX-0003 Boron Endpoint Measurement
- 1PEP04-ZX-0004 Isothermal Temperature Coefficient (ITC) Measurement
- 1PEP04-ZX-0006 N-1 Rod Worth Verification
- 1PEP04-ZX-0007 Rod Cluster Control Assembly (RCCA) Pseudo Ejection Test
- 1PEP04-ZX-0010 Natural Circulation Verification

The purpose of this inspection was, through the review of completed procedures, to assess the performance and the overall adequacy of the tests performed during the Hot Zero Power (HZP) plateau.

This program, starting with initial criticality on March 8, 1988, and continuing with the low power testing appears to have been conducted and completed successfully by March 16, 1988. It was verified by reviewing test data and evaluating test results that the acceptance criteria were met. A detailed discussion of the review of each of these procedures is as follows:

a. Test Sequence for Initial Criticality and Low Power Testing (IPEP04-ZX-0001)

During the determination of the Nuclear Heat Test, Step 6.6 of Procedure 1PEP04-ZX-0001, a steam generator power operated relief valve (PORV) opened.

The NRC inspector identified that this information was not noted in the chronological test log, although the test summary mentions that a PORV opened momentarily and that the problem was corrected quickly (the valve was stuck open about 20 percent for about 45 seconds). The information in the test summary, however, does not describe which valve opened, under what circumstances the event occurred, and what corrective actions were taken. Some available information of this failure was found in Maintenance Work Request (MWR) MS-48225, issued soon after the event occurred. It appears that during troubleshooting, signs of hydraulic leakage were found, but the source of the leak was not determined.

The MWR (page 4a paragraph "as found condition") also indicated that the valve (1) would not move from 30 percent open, and (2) later moved from 30 percent to 90 percent open with no signal either from the control room or locally. Reasons to explain these deficiencies were not documented. In addition, a review of the operator log book did not provide any new information, except documentation of entrance into the limiting condition for operation (LCO) corresponding to this problem (TS 3.7.1.6)

The NRC inspector considers that the licensee should have adequately documented this malfunction affecting safety-related equipment.

Therefore, this constitutes of an apparent violation (498/8824-01) of the licensee's procedures.

b. Boron Endpoint Measurement (1PEP04-ZX-0003)

The purpose of this procedure was to determine the boron endpoint concentrations at various configurations as specified in the test sequence for initial criticality and low power testing.

The only specific boron endpoint acceptance criterion given in the procedure is for the all rods out (ARO) condition. The predicted value was 917 ±50 parts per million (ppm). The actual measured value was 953.5 ppm. For the other configurations, the boron endpoint concentrations measures were Control Rod Banks; CD-in; 896.8 ppm; CD+CC-in; 817 ppm; CD+CC+CB-in; 717.9 ppm; CD+CC+CB+CA; 624.4 ppm. All these values appeared to be very close to the predicted value.

The NRC inspector verified by reviewing test data and independently evaluating test results from the Boron Endpoint Measurement test procedure performed on March 8-11, 1988, that the licensee had complied with the test requirements and that the acceptance criteria were met.

c. ITC Measurement (1[illegible]?04-ZX-0004)

The NRC inspector evaluated and verified that the licensee met the requirements and conditions of the facility license by reviewing test data and evaluating test results from the ITC measurement test performed from March 8-11, 1988. The NRC inspector ascertained that the licensee was adhering to station procedures and that test program records were adequate.

The purpose of this test was: (1) to determine the ITC of reactivity at three different control rod configurations:

- ARO
- Control Bank D in
- Control Banks D and C in

(2) to derive the ARO MTC, and (3) to perform the surveillance requirement of TS related to the MTC limit which constitutes the safety criterion.

Reviewing the ARO and Bank D configurations, the NRC inspectors noted that during the beginning of the test, cooldowns had been suspended because of a cooldown rate too fast (the procedure required approximately 10°F/hour). After resetting conditions, the test was resumed at a slower rate.

ITC was determinated by measuring the slope of reactivity change versus change in average RCS temperature during heatup and cooldown of the RCS.

Reviewing the ARO configuration results, the NRC inspectors noticed a discrepancy between the slope of reactivity change recorded during the heatup configuration and the reactivity change value (delta rho) noted on the data sheet. After verification, the licensee agreed to the identified error and consequently committed to correct this error.

The value given by Westinghouse in Procedure T6X/TUX-SU3.3.6, step No. 6.2 specifies that the measured values of the ITC obtained from heatup and cooldown agree within ±1 pcm/degree Fahrenheit. This value was not considered a criterion by the licensee and, consequently was not included in the Procedure 1PEP04-ZX-0004. However, in this particular case, it would have permitted the test director and the verifier to note that the ITC values found were outside of ±1 pcm/degree Fahrenheit criterion.

Although this error does not impact on the acceptance criteria of the test procedure (the MTC value become -1.44 pcm/degree Fahrenheit instead of -2.06 pcm/degree Fahrenneit which is in accordance with the TS requirement), this error was not found during the verification process. The NRC inspector noted that the licensee then took appropriate action to correct this error. A Station Report Problem (SRP) 88-0130 was issued on April 15 that involved the procedure correction and required reverification of all test data recorded during the core physics testing on April 16, 1988.

This constitutes an apparent violation (498/8824-02) the licensee's procedural requirements.

d. N-1 Rod Worth Verification (1PEP04-ZX-0006)

The NRC inspector evaluated and verified that the licensee met the requirements and acceptance criteria of the procedure by reviewing test data and evaluating test results for the N-1 Rod Worth Verification Measurement test performed on March 12 and 13, 1988. The purpose of this test was to verify that the HZP insertion limits defined under TS provide adequate shutdown margin with the most reactive RCCA defined as being RCCA F-14 (Control Bank B) stuck in the withdrawn position.

The total worth of all RCCA Banks, less the most reactive RCCA, was determined to be 8283.8 pcm. This value met the acceptance criterion of 7661 ±766 pcm. The HZP shutdown margin is given by the worth of all rods less the most reactive RCCA minus the worth of all rods above the Rod Insertion Limit (specified in TS 3.1.3.6 and obtained from the Nuclear Design Report). This value (8283.8 - 1429 = 6854.8 pcm) exceeds the minimum value of 1750 pcm required by TS 3.1.1.1.

Since these results met the acceptance criteria, the NRC inspector concluded that the test was satisfactory.

e. RCCA Pseudo Ejection Test (1PEP04-ZX-0007)

The NRC inspector verified by reviewing test data and evaluating test results from the RCCA Pseudo Ejection Test performed on March 13 and 14, 1988, that the licensee complied with the test requirements and the acceptance criteria of the test procedure were met.

The purpose of this procedure was to verify the conservatism of the assumed worth of an ejected rod at zero power and of the assumed hot channel factors for an ejected rod at zero power from the configuration assumed in the Safety Analysis Report (SAR).

The reactivity change caused by the withdrawal of the most reactive rod (Rod D-12) was used to determine the worth of the rod. The equilibrium boron concentration and the reactivity change caused by the withdrawal of the rod was used to determine the boron endpoint concentration. A flux map was performed then evaluated and the hot channel factors were determined.

The NRC inspector noted that two Field Change Requests (FCR) were initiated in this test to correct inadequacies in the test procedure. In particular, FCR 88-0530 was initiated to change measurement uncertainty so as to match the Westinghouse startup procedures.

The NRC inspector verified that licensee technical reviews were performed and confirmed the value as correct from Westinghouse startup Procedure TGX/THX-SU3.1.1, Revision 1.

During this test, a Test Deficiency Record (TDR) 88-028 was generated. The test method consists of determining the core reactivity to endpoint condition with the controlling bank at the zero power insertion limit, the most reactive rod being borated out to approximately 230 steps, then fully withdrawn. At this point, the procedure specifies (Step 6.12.6.3), "Allow the neutron level to increase until the flux signal is greater than 30% of full scale and the indicated reactivity is constant." When the most reactive rod (D-12) was pulled out it was observed that the reactivity addition resulting from withdrawal from 228 to 259 steps was only 2.33 pcm due to the single rod moving through the region of low relative flux in the upper portion of the core. This very small amount of reactivity addition resulted in the flux signal increasing very slowly (period of approximately 4500 seconds). Because the reactivity trace had stabilized and the flux signal was slowly increasing at a rate consistent with the very small reactivity addition, rod D-12 was reinserted and the reactivity change measured prior to the flux increase above 30 percent of full scale as called for in the procedure.

The licensee repeated the test three times and an average reactivity was used in the end point calculation. Although this was not consistent with the guidance in the procedure, the licensee considered the reactivity measurement valid for the following reasons:

(1) the reactivity trace had stabilized prior to rod reinsertion.

(2) the flux trace increased at a rate consistent with a such small reactivity addition.

(3) the reactivity change resulting from the endpoint was so small that the measurements resulting were especially susceptible to the introduction of an additional error due to temperature changes. Thus, by attempting to delay reinsertion of the rod until the flux signal reached 30 percent, the endpoint measurement accuracy would have been affected by the negative reactivity feedback from the ITC.

Westinghouse provided information that the basis for the 30 percent value used in the startup procedure is to allow for margin above the 20 percent minimum value and that the additional error resulting from reactivity measurements with flux level between 20 and 30 percent is negligible. If the reactivity measurement (2.33 pcm) had a 100 percent error, the difference in the total measured worth of RCCA D-12 should be only 0.57 percent and has no effect on the test results.

Therefore, the worth of the ejected RCCA at HZP was determined to be 407.9 pcm. The worth must be less than or equal to 860 pcm (Final Safety Analysis Report (FSAR), 15.4.8) and the peak hot channel factor FQ(z) resulting from the ejected rod at HZP was determined to be 7.269. This value is to be less that or equal to 13 (FSAR, 15.4.8).

Since these results met the acceptance criteria, the NRC inspector concluded that the test was satisfactory.

f. Natural Circulation Verification (1PEP04-ZX-0010)

The NRC inspector verified by reviewing test data and evaluating test results from the Natural Circulation Verification test procedure performed on March 14 and 15, 1988, that the licensee complied with the test requirements and the acceptance criteria of the procedure were met.

The purpose of this procedure was to demonstrate the ability of the RCS to remove decay heat from the reactor core and to obtain coolant flow and temperature distribution data under a condition of loss of forced convective cooling.

During the power ascension while delta-T indicated about 1 percent power, the nuclear power range channels indicated 8 percent. The licensee performed the Procedure 1PEP04-ZY-0040 (Initial Adjustment of Nuclear Instrumentation) twice to adjust the gains of the power range instrumentation to the power shown by loop delta-T. This adjustment was necessary to avoid the P10 interlock actuation.

The natural circulation was established and maintained for one hour. During this period of time, the acceptance criteria were verified as indicated by steam pressure remaining constant, the RCS hot leg temperature approximately equal to core exit thermocouple temperature and remaining constant, and RCS cold leg temperature approximately equal to the saturation temperature for the indicated steam pressure. Thermocouple maps were obtained while natural circulation was maintained.

The NRC inspector noted that the TSAT had to be recalculated because of wrong values of the Loop 1 cold leg temperature. It appears that this problem was found after the completion of the test and a MWR was issued to troubleshooting. However, this problem was not reported on the test summary (Section 5.0 - problems encountered during testing).

Further, in a previous NRC inspection report (50-498/88-12), NRC inspectors noted that the reactor coolant system (RCS) flow measurement test results appeared to lack in documented test information and test data. The above observation is an additional example that indicates the licensee must provide more detailed information in the chronological test log and test summary, especially when problems are experienced during test performance.

These problems are additional examples of inadequate test documentation apparent violation 498/8824-01, noted in paragraph 3.a.

4. Operability of Steam Generator PORVs

On April 12, 1988, the licensee received notification from Paul-Munroe/ENERTECH, the vendor, that seal materials installed in the hydraulic actuators and hydraulic pumps for the steam generator PORVs in both Unit 1 and 2 were not of the correct material and were incompatible with the hydraulic fluid (FYRQUEL-EHC). Subsequently, a letter dated April 18, 1988, was received by the NRC Region IV Administrator from Paul-Munroe/ENERTECH providing the details of this issue. Specifically, the seai kits installed in the PORV pump shaft seals and hydraulic cylinder rod seals were specified as Viton (a fluorcarbon) and were accompanied by a Certificate of Conformance (COC) from the seal vendor (Parker Cylinder Division). However, in fact, the seals installed were BUNA-N, a nitril, that undergoes erosion when in contact with FYRQUEL-EHC hydraulic fluid. Erosion of the seal materials could prevent proper operation of the PORVs.

The PORVs are identified in the TS as the atmosphere steam relief valves. The TS Section 3.7.1.6 requires that all four PORVs be operable in Modes 1, 2, 3, and 4. With one PORV inoperable, the plant is in a 7-day action statement (TS 3.7.1.6.a). With two PORVs inoperable, the plant is in a 72-hour action statement (TS 3.7.1.6.b). Having more than two PORVs inoperable is not defined in the action section of TS 3.7.1.6. Thus, with more than two steam generator PORVs inoperable, the plant would be governed by TS 3.0.3 which would require initiation of a plant shutdown within one hour and be in cold shutdown (Mode 5) within 37 hours or resolve the condition which would allow resumption of operations under an action statement or LCO.

On April 23, 1988, the licensee issued a 10 CFR 50.59 Safety Evaluation (SE) and a justification for continued operation (JCO) which addressed the operability status of the four PORVs with the improper BUNA-N seal material. The PORVs are safety-related and are required to be environmentally qualified (EQ) because they are located in the isolation valve cubicles (ICVs) and as such could be subject to high temperature, high humidity, spray, chemical and radiation effects under certain accident conditions. The NRC inspectors did note, however, that the SE and JCO failed to consider that the BUNA-N seal material would undergo degradation in a radiation field that could be expected to be present under certain accident senarios, for example gross steam generator tube failures. The EQ aspect of this issue will be addressed in NRC Inspection Report 50-498/88-30. The SE and JCO concluded that plant operation could continue as long as at least two PORVs remained operable as defined by an enhanced surveillance program. This program included (1) the manual timing of the hydraulic pumps cycle frequency for the three worst PORVs (A, B, and D) once per shift, (2) the visual examination of the PORV skids once per shift, and (3) a stroke test of all four PORVs once per day. A pump cycle frequency of less than 10 minutes, gross oil leakage, or a failure of the stroke test would result in a declaration of inoperability for the applicable PORV that would lead to the appropriate action statement in TS 3.7.1.6.

On April 23, 1988, at 1:30 a.m. (CST), the "D" PORV was declared inoperable (TS action statement 3.7.1.6.a). The same day, at 7:30 p.m., the "A" PORV was declared inoperable due to excessive hydraulic pump cycling (TS action statement 3.7.1.6.b). On April 24, 1988, at 4:22 p.m. (CST) the licensee exceeded the TS action statements in order to comply with the SE/JCO surveillance requirement and isolated a third PORV ("C" PORV) which by default placed the plant outside the requirements of 3.7.1.6 and into TS 3.0.3. The licensee entered TS 3.0.3 for operational convenience with no stated intent of shutting down. Eight minutes later, the "C" PORV was returned to service and the licensee exited TS 3.0.3 and consequently returned to TS action statement 3.7.1.6.b. At 4:33 p.m., the licensee isolated "B" PORV for stroke testing and, for operational convenience, again entered TS 3.0.3 for 10 minutes. These events were recorded in the reactor operators log book, including the statement on entering and exiting TS 3.0.3. The "A" and "D" PORVs were subsequently repaired that week, returned to service, and TS 3.7.1.6. was exited.

On April 30, 1988, at 6:20 p.m. (CST), the "A" PORV failed its stroke test, was declared inoperable and TS 3.7.1.6.a was entered. At 9 p.m., the same day "B" PORV failed because of a problem that was not associated with the hydraulic seal problem. The plant was steady state at 30 percent reactor power when the "B" PORV was driven full open. The control room dispatched an operator who shut the manual block valve after about six minutes. The "B" PORV remained open about another nine minutes, then closed. The "B" PORV was then declared inoperable and TS 3.7.1.6.b was entered. The NRC inspectors reviewed the plant computer (ERFDADS) printout for the "B" PORV activation. The PORV opened and closed (stroke time and rate) exactly as if a demand signal was present. However, no such signal or need for that signal either manually or automatically from the QDPS computer was indicated on the computer printout. The plant then had two of the four PORVs failed under different failure modes, one of which was undetermined and unanalyzed since it was outside of the scope of the SE and JCO that was written for the BUNA-N hydraulic seal issue.

The NRC inspectors discussed with a shift supervisor the stroke testing of the two remaining operable PORVs as required by the SE and JCO without exceeding the action statement of TS 3.7.1.6 and entering TS 3.0.3, for operational convenience. The shift supervisor indicated that it appeared the licensee would have to enter TS 3.0.3. The NRC inspectors then asked if it was acceptable to voluntarily enter TS 3.0.3 for operational convienence with no intent to shutdown. The shift supervisor indicted he was not sure.

On May 1, 1988, at 9:15 p.m. (CST), the licensee began a normal shutdown of Unit 1, one day ahead of schedule. The licensee indicated that the reason for the early shutdown was that 30 percent plant testing was essentially complete and that the licensee would not be able to meet the enhanced surveillance testing requirements established by the SE and JCO without entering TS 3.0.3.

As a result of this inspection, the NRC inspectors concluded that the SE and JCO was incomplete in that it ignored the effects of radiation; the licensee violated TS 3.7.1.6 on two occasions the day of April 23, 1988, when three PORVs were inoperable and entered TS 3.0.3 for operational convenience; and that the staff and management of the licensee does not have a uniform understanding of the bases for and application of TS 3.0.3.

The two instances of the violation of the requirements of TS 3.7.1.6 is an apparent violation of NRC requirements. (498/8824-03).

5. <u>No. 13 Standby Diesel Generator (SDG) Oil Spill - Unit 1</u>

On April 19, 1988, an Equipment Clearance Order (ECO) was issued per Plant Procedure OPGP03-ZO-0001, Revision 7, "Equipment Clearance," on the safety-related No. 13 SDG lubricating oil (LO) circulation pump. The system was tagged out per the ECO. In preparation for disassembly, the pump required draining. The control room concurred with draining the small amount of LO from the pump into the No. 13 SDG sump. Drain Valve LU-0071 was opened and Valve LU-0119 was unlocked and opened. An entry was made in the Locked Valve Deviation logbook. However, none of these valves were identified as boundary valves requiring clearance tags on the ECO. Subsequently, these valves were left open and a change of plans lead to the deferral of the LO circulation pump work till another time and the ECO was cleared.

The system alignment was reestablished per the ECO. The LO circulation pump and heater was turned on. After about 5 minutes of pump operation, the control room noted No. 13 SDG LO was reading approximately 170°F and 2 psig pressure. The operator went to investigate and noticed smoke coming from the LO heater. He deenergized all power to the SDG. His investigation determined that the LO drain valves were open and the LO sump was about 1100 gallons below normal.

A review of the event indicated there was an inadequate lineup review, no independent verification of system restoration, and no review of the Locked Valve Deviation log prior to energizing the SDG support system. Timely observation and action by the crew prevented a fire and the resultant damage to the SDG.

After the event, the NRC inspector observed the end of the cleanup of the spilled oil. The area was posted "No Smoking" and there was security guards posted to control access and work activities. The events leading to this oil spill involving the failure to follow the appropriate procedures is considered a violation of NRC requirements (498/8824-04).

6. <u>Essential Cooling Water (ECW) Valve Body and Pipe Fitting Leaks</u>

On April 1, 1988, the licensee found evidence of leakage on small bore (less than two inches) valves and pipe fittings of the safety-related ECW system. The ECW is constructed of welded aluminum bronze material. Three of these small bore fittings were initially removed and sent off-site for analysis.

The licensee reported preliminary results of Bechtel metallurgical laboratory analysis on two 1-inch sockets removed from the essential cooling water system. The sockets exhibited corrosion with apparent through wall leaks. Microsections of the aluminum bronze material revealed selective corrosion associated with the dealuminization of one phase of the bi-phase alloy used in castings for small bore socket joints. Selective corrosion has not been observed on the single phase aluminum bronze pipe. The corrosion appears to be associated with the socket crevices. The actual weld and heat affected zone have not exhibited corrosion. The bi-phase alloys were not expected to corrode in this manner; however, improper heat treatment of these fittings could lead to the observed corrosion. The bi-phase alloy normally consists of an alpha and beta phase. If the metal is not cooled quickly during heat treatment then a gamma-2 phase alloy forms which is susceptible to corrosion.

The average deterioration of wall thickness was up to 88 percent. The system has been in use for three years. Evidence of through wall leaks (amounting to a maximum of 8-10 ml per day) were observed on approximate 70 sockets. The corrosion has only been observed on 2- and 1-inch fittings. Larger pipe is not joined with socket welds. The licensee is identifying all applications of the suspect two-phase alloy in the system for evaluation.

The corrosion appears to progress slowly and a monitoring program is being developed to measure the rate of deterioration of the fittings. The licensee's analysis indicates that if 70 percent of the wall thickness were lost, the system would still retain its structural integrity. Therefore, the licensee considers the ECW system to be operable with no immediate safety concerns. The licensee plans to complete the evaluation of the problem and to establish an action plan acceptable to the NRC prior to exceeding the 30 percent power testing plateau. NRC Region IV and NRR are closely monitoring the licensees evaluation and corrective action plan development.

7. Monthly Surveillance Observations - Unit 1 (61726)

The NRC inspector observed selected portions of surveillance testing and reviewed completed data packages to verify that TS requirements are being met for safety-related systems and components. The following surveillance tests were observed:

- 1PSP03-SI-006, Revisions 5, "High Head Safety Injection Pump 1C Inservice Test"
- 1PSP02-SP-0008B, Revision 0, "Train B Diesel Generator Slave Relay Test"
- 1PSP03-DG-0002, Revision 5, "Standby Diesel 12 Operability Test"

The NRC inspectors verified the following items during the inspection:

- Test results were reviewed by personnel other than the persons directing the test.
- The surveillance testing was completed at the required frequency per TS requirements.
- Testing was performed by qualified personnel using approved procedures.

- Removal and restoration of the affected system and or components were accomplished.

- Test instrumentation was calibrated.

No violations or deviations were identified.

8. Security Observations (71881)

The NRC inspectors verified the physical security plan was being implemented by selected observation of the following items:

- Security monitors at the secondary and central alarm stations were functioning properly for assessment of possible intrusions.

- Persons and packages were properly cleared and checked before entry into the protected area (PA) was permitted.

- The security organization was properly staffed.

- The PA barrier was maintained and the isolation zone kept free of transient material.

- Vital area barriers were maintained and not compromised by breaches or weakness.

- Illumination in the PA was adequate to observe all areas during hours of darkness.

No violations or deviations were identified.

9. Radiological Protection Observations (71709)

The NRC inspectors verified that selected activities of the licensee's radiological protection program were implemented in conformance with regulatory requirements. The activities listed below were observed:

- Radiation work permits contained the appropriate information to ensure work was performed in a safe and controlled manner.

- Personnel properly frisked prior to exiting the radiation protected area (RPA).

- Personnel in the RPA were wearing the required personnel monitoring equipment.

- Radiation and contaminated areas were properly posted based on the amount of activity levels within the area.

No violations or deviations were identified.

10. Operational Safety Verification (71707)

The objectives of the inspection were to ensure that the plant is being operated in a safe manner and in conformance with regulatory requirements, the licensee's management controls are effective in discharging their responsibilities, and TS requirements are being met. The NRC inspectors visited the control room on a daily basis to verify the following:

- Operators adherence to approved procedures and TS requirements.
- Control room staffing was proper.
- Management personnel toured the control room on a regular basis.
- Operability of the safety parameter display system.
- MWRs were written for equipment/components in need of repairs.
- Conduct of reactors were professional.

No violations or deviations were identified.

11. Engineered Safety Feature (ESF) System Walkdown (71710)

The NRC inspectors conducted a walkdown of the accessible portions of Train "A" of the ECW system to verify operability of the system. A review was performed to confirm that the licensee's system operating procedure matched plant drawings and the as-built configuration. Equipment conditions, valve and breaker positions, housekeeping, labeling, permanent instrument indications and apparent operability of support systems essential to actuation of the ESF system were all noted as appropriate.

The NRC inspectors identified the following observations to licensee management:

a. MWR Tag No. 02099, which was attached to a conduit junction box was not filled out completely. The tag did not have a date which the tag was hung as required by Step 4.3.9 of Procedure OPGP03-ZM-0003.

b. Permanent identification tags were not installed on Valves EW-0259, EW-0198, EW-0018, EW-0017, and EW-0120.

c. Emergency Diesel Generator ECW Return Isolation Valve EW-0019 is incorrectly identified as a test connection on the attached identification tag.

d. A large diameter overhead conduit in the Train "A" Supplemental Chiller Room was missing a coverplate.

e. An unidentified cable was found hanging from cable trays CIXMIATJAA and CIXMIATSVA in the Train "A" Supplemental Chiller Room.

f. The tubing channel supporting instrument root valves EW-0009 and EW-0010 plus the associated tubing was not attached to its support.

g. Train "C" Return Loop Chlorine Analyzer Panel mounting bolt was not installed correctly. Additionally, the panel did not have a permanent identification tag.

h. ECW system Drawing No. 5R289F05038, Revision 13, shows Valve EW-0117 as locked shut and Valves EW-0020 and FW-0259 as locked open. These valves were found in the open and throttled positions respectively, as required in the system operating Procedure 1POP02-EW-001-1.

i. The vent cap downstream of the common vent valve NI-EW-PSL-6882 and NI-EW-WPI-6882 was not installed.

j. The local control station for the Standby Diesel Generator No. 11 Jacket Water Makeup Valve is missing the valve control switch.

k. Valve EW-0331 had a work traveler from 1985 attached.

The above listed discrepancies do not appear to be safety significant; however, collectively they represent weaknesses on the part of the licensee regarding attention to detail in safety-related activities.

No violations or deviations were identified.

12. Exit Interview

The NRC inspector met with licensee representatives (denoted in paragraph 1) on May 2, 1988, and summarized the scope and findings of the inspection. Other meetings between NRC inspectors and licensee management were held periodically during the inspection to discuss identified concerns. The licensee did not identify as proprietary any of the information provided to or reviewed by the inspectors during this inspection.