APPENDIX B

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-498/88-07 Operating License: NPF-71
50-499/88-07 Construction Permit: CPPR-129

Dockets: 50-498
50-499

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

Facility Name: South Texas Project (STP), Units 1 and 2

Inspection AT: STP, Matagorda County, Texas

Inspection Conducted: January 25-29 and February 8-12, 1988

Inspectors: R. C. Stewart — 3/16/88
R. C. Stewart, Reactor Inspector, Materials and Quality Programs Section, Division of Reactor Safety — Date

L. D. Gilbert — 3/17/88
L. D. Gilbert, Reactor Inspector, Materials and Quality Programs Section, Division of Reactor Safety — Date

D M Hunnicutt — 3/16/88
D. M. Hunnicutt, Project Engineer, Reactor Project Section D, Division of Reactor Projects — Date

Other Accompanying Personnel:

R. V. Azua, Test Programs Section, Division of Reactor Safety
R. C. Haag, Materials & Quality Programs Section, Division of Reactor Safety

Approved: I. Barnes — 3/17/88
I. Barnes, Chief, Materials and Quality Programs Section, Division Reactor Safety — Date

Inspection Summary

Inspection Conducted January 25-29 and February 8-12, 1988 (Report 50-498/88-07)

Areas Inspected: Routine, unannounced inspection of licensee action on previously identified inspection findings, nuclear welding, and manual reactor trip circuits.

Results: Within the three areas inspected, two violations were identified (failure to test welding material for different postweld heat treatment applications, paragraph 4; and failure to provide adequate control of quality assurance records, paragraph 2).

Inspection Conducted January 25-29 and February 8-12, 1988 (Report 50-499/88-07)

Areas Inspected: Routine, unannounced inspection of structural steel welding, pipe supports and restraints, nuclear welding, preoperational testing, onsite design changes, safety-related components, and manual reactor trip circuits.

Results: Within the seven areas inspected, a violation was identified (failure to test welding material for different postweld heat treatment applications, paragraph 4).

DETAILS

1. Persons Contacted

HL&P

*G. Vaughn, Vice President, Nuclear Operations
+*J. T. Westermeier, Project Manager
+*J. E. Geiger, General Manager, Nuclear Assurance
+*J. S. Phelps, Supervisor, Project Compliance
+*T. J. Jordan, Project Quality Assurance (QA) Manager
+J. N. Bailey, Engineering & Licensing Manager, Unit 2
+*D. C. King, Construction Manager
*M. A. McBurnett, Operations Support & Licensing Manager

Bechtel Energy Corporation

+*R. W. Miller, Project QA Manager
+*R. D. Bryan, Field Construction Manager
+K. P. McNeal, Project QA Engineer
+E. B. Luder, Lead QA Engineer

Ebasco Services Inc.

+*D. D. White, Construction Manager
+*A. M. Cutrona, Quality Program Site Manager
F. G. Miller, Welding Superintendent
+R. E. Abel, Quality Control Site Supervisor

NRC

+*D. M. Hunnicutt, Project Engineer
+*R. C. Stewart, Reactor Inspector
+*L. D. Gilbert, Reactor Inspector
+*R. C. Haag, Reactor Inspector
+R. V. Azua, Reactor Inspector
*C. E. Johnson, Senior Resident Inspector (Construction)
*D. R. Carpenter, Senior Resident Inspector (Operations)
*D. L. Garrison, Resident Inspector
*A. B. Beach, Deputy Director, Division of Reactor Projects
*J. P. Jaudon, Deputy Director, Division of Reactor Safety

The NRC inspectors also interviewed other licensee and contractor employees during the inspection.

*Denotes those attending exit interview on January 29, 1988.
+Denotes those attending exit interview on February 12, 1988

2. Licensee Action on Previously Identified Inspection Findings (92702)

(Closed) Deviation (498/8726-06): Inadequate Control of Records. This deviation identified the failure to properly control quality assurance (QA) records while in temporary working files and the failure to comply with transmittal requirements when transferring QA records. This deviation, as reported in NRC Inspection Report 50-498/87-26, dated June 25, 1987, was discovered prior to issuance of the operating license and the technical specifications (TSs).

In the corrective action portion of the licensee's response letter, dated July 27, 1987, to the above noted deviation, the licensee committed to the following action:

° Each division that maintains QA records will assign a designated record custodian who will be responsible for filing records, maintaining accountability, controlling access, indexing stored record, and transmitting those records to the record retention area.

° Promulgation of the procedural requirement to use transmittal form.

In the response letter, the licensee also reported that some divisions had determined that QA records will no longer be maintained outside of the Operations Document Control Center (ODCC) storage facility; therefore, QA records will be transmitted to ODCC in an expeditious manner. The licensee stated "STP is in full compliance at this time" in the response letter.

During this inspection, the NRC inspector reviewed approximately 30 recent QA record transmittals to the ODCC. All these transmittals were accompanied by the required transmittal forms. The ODCC supervisor stated that ODCC personnel will not accept QA records without a transmittal form. While inspecting reactor operations (RO) division for compliance with the corrective action, the NRC inspector learned that no QA records are being stored in division QA record files. Present policy requires that all QA records generated by RO division be transmitted to ODCC upon completion with copies of selected records being retained in the division file.

The chemistry department maintains QA records in their division records file for 60 days. A records custodian has been assigned with responsibility for maintaining control of QA records while in the division files. Access to the files is limited to the custodian and selected supervisors with any additional access being controlled by a checkout card system. The NRC inspector verified that recent transmittals of chemistry department QA records to ODCC were completed with transmittal forms.

During this inspection, the NRC inspector noted that the engineering department also maintains QA records. Selected surveillance tests require a trending review by system engineering upon completion of the test. Those completed surveillance tests, that are awaiting engineering review, makeup the QA records in the divisional file reviewed by the NRC inspector. A record custodian had not been assigned for this divisional QA records file. Access to the records is not limited. Also, no system exists for maintaining accountability, indexing, or ensuring that the maximum retention time is not exceeded for records being maintained in the division file.

10 CFR Part 50, Appendix B, Criterion XVII, "Quality Assurance Records," states, in part, ". . . . Consistent with applicable regulatory requirements, the applicant shall establish requirements concerning record retention, such as duration, location and assigned responsibility." Section 6.10, "Record Retention," of the TSs requires "Records of surveillance activities, inspections, and calibrations required by these TSs" be retained for at least 5 years. Station Procedure No. OPGP03-ZA-0042, "Operations Quality Records," dated December 31, 1987, describes the requirements and responsibilities for the control of QA records. Paragraph 4.3.4 of this procedure states, "Retention time in Division QA Record files shall not exceed 90 days", while paragraph 4.3.5 states, "A Record Custodian, who will be responsible for filing records, maintaining accountability, controlling access, index stored record and transmitting those records to the Record Retention Area, should be designated in writing."

This deviation is considered closed based on the overall licensee compliance with committed corrective actions. The licensee's failure to properly maintain a divisional QA record file in the engineering department is contrary to the requirements established in Station Procedure OPGP-03-ZA-0042, paragraphs 4.3.4 and 4.3.5 and constitutes an apparent violation (498/8807-01).

3. Followup Inspection of Welding of Structural Steel (55100)

During the period January 25-29 and February 8-12, 1988, the NRC inspector conducted a followup inspection to determine through direct observations and records review, whether the structural welding activities performed at the site are performed in accordance with specifications, procedures, and Safety Analysis Report (SAR) commitments to the American Welding Society D1.1 Code.

a. Observation of Work

The NRC inspector made a random selection of 118, Category B, field welds for direct visual examination. The field welds selected were composed of electrical raceway hanger welds (25), pipe support welds (56), and HVAC welds (37). In conducting the visual examinations, the NRC inspector utilized the 11 attributes prescribed by the AWS D1.1-85 Code, "Visual Weld Acceptance Criteria" and licensee Procedure SSP-16, "General Structural Welding Requirements," Revision 3. The criteria are applicable to structural systems subject to static loading (seismic loads included) for which fracture resistance and fatigue resistance are not governing design considerations.

No violations or deviations were identified.

b. Review of Weld Records

In conjunction with the visual examination of field welds, the NRC inspector selected the corresponding 118 weld records for the welds examined and an additional 50 weld records for review.

The records consisted of 29 travelers and drawing and weld maps for the selected welds. In addition to the AWS Code requirements, the NRC inspector utilized the licensee's Procedures SSP-11, "Fabrication, Erection, and Bolt-up of Structural and Miscellaneous Steel," Revision 2 and SSP-16, "General Structural Welding Requirements," Revision 3, which delineate requirements for recording and documenting field welding activities.

The NRC inspector observed that the entries on the weld records were consistent with AWS D1.1 Code and Procedure SSP-11 requirements, which include welder identification, weld process used, electrode traceability, preheat temperature, weld identification traceable to specific component, QC inspector signoffs, and approvals.

No violations or deviations were identified.

c. Records Review - Welders and Weld Inspectors Qualifications

The NRC inspector selected eight welders and five QC weld inspectors training and qualification records for the period March 11, 1986, through September 23, 1987. The NRC inspector observed that welder records reflect that all welders were qualified in accordance with established licensee Procedure SSP-31, "Welder Qualification," and in accordance with Section 5 of AWS D1.1 Code requirements. In addition, the licensee maintains a continuous computer data record system establishing the qualification status of all welders.

During the review of QC weld inspector records, the NRC inspector observed that individual inspector training and certification records were well documented in the specific training and certifications received, including ASME and AWS Code requirements. In addition, each inspector's records indicated specific training in the visual acceptance criteria of SSP-16 for structural welds.

No violations or deviations were identified.

4. Nuclear Welding (55050)

As a followup inspection to NRC Inspection Report 50-499/88-02, the NRC inspector reviewed the records associated with the postweld heat treatment of welds SB1101-FW0009, SB1201-FW0009, SB1301-FW0009, and SB1401-FW0010 for Unit 1, and SB2101-FW0009, SB2201-FW0009, SB2301-FW0009, and SB2401-FW0010 for Unit 2. These welds are the penetration assembly to penetration sleeve welds in the steam generator blowdown piping system. The penetration assembly and sleeve material specifications are SA182 Grade F22 and SA333 Grade 6, respectively. Bechtel Specification 5A010PS002, Revision 13, specifies that the penetration assembly to penetration sleeve weld shall be in accordance with ASME III, Division 1, 1974 Edition through Winter 1975 Addenda, Subsection NE or Subsection NC when criteria for welding postweld heat treatment, or material is not provided in Subsection NE. The penetration assembly material is classified as a P-Number 5 material in ASME Section IX. P-Number 5 materials are included in Subsection NC, but are not included in Subsection NE; therefore, the requirements of Subsection NC are applicable to the above penetration weld. Paragraph NC-4600 of Subsection NC specifies that P-Number 5 materials shall be postweld heat treated at 1250 to 1400°F. Paragraph NC-2400 of Subsection NC specifies testing of all welding material used in construction and that the test coupons shall be postweld heat treated to the specified temperature indicated in the welding procedure specification. The welding procedures, WP-129 and WP-69, specify postweld heat treatment temperatures of 1300 to 1400°F and 1325 to 1375°F, respectively. The welding materials used for making these welds were tested using coupons postweld heat treated at temperatures of 1100 to 1200°F. The failure to test welding materials in accordance with the postweld heat treatment requirments of the applicable welding procedure specif··ations is an apparent violation. (498/8807-02; 499/8807-01)

5. Unit 2 Safety-Related Pipe Support and Restraint Systems (50090)

a. Observation of Work and Work activities

The NRC inspector observed the following small bore pipe supports:

Support Drawing	Class
CV 9141-HS 5003	1
RC 9419-HS 5001	1
RC 9419-HS 5006	1
CV 2142-HF 5039	2
CV 2142-HF 5041	2
CV 2142-HF 5044	2
SI 2306-HF 5006	2
SI 2306-HF 5005	2
SI 2306-HF 5004	2
SI 2319-HF 5001	2
SI 2319-HF 5002	2
CC 9129-HS 5001	3
CC 9229-HS 5001	3

The NRC inspector also observed the crossbracing weldments for the 2A, 2B, and 2C residual heat removal pump supports.

In the areas inspected, the supports and weldments were consistent with the requirements of the drawings. The supports were inspected for type, location, dimensions, orientation, clamps, bolting, and clearances. The weldments were inspected for size and appearance.

b. Records

The NRC inspector reviewed the quality control records for the pipe and component supports identified above.

In the areas reviewed, the records were complete, accurate, and retrievable.

No violations or deviations were identified.

6. Unit 2 Safety-Related Components (50073 and 50075)

An inspection was conducted of activities related to selected safety-related components other than reactor pressure vessel and piping. This inspection was performed to determine whether specific activities associated with the reviewed components were being controlled and performed according to NRC requirements, FSAR commitments, and licensee procedures.

a. Work Observations

The NRC inspector examined the following equipment for which work had been completed or was in progress to determine conformance with the applicable procedural requirements:

° Pressurizer Power Operated Relief Valve (PORV) Nos. 2RC-PVC-655A and -656A

° Pressurizer Spray Valve Nos. 2RC-PVC-655B and -655C

° Pressurizer Safety and Relief Valve Discharge Header Serial No. 39813

° Steam Generator PORV No. A2MS-PV-7411

° Regenerative Heat Exchanger No. 2R172NHX201A

° Three-Way Power Operated Valve No. B2CU-FV-3123

All attributes that could be visually inspected were examined for adequacy of design and completeness of construction. Particular areas examined included welding, bolting identification, switches, restraints, locking devices, flow direction, and cleanliness. No problems were identified by the NRC inspector in this area of inspection.

b. Records Review

The NRC inspector reviewed the work packages associated with the installation of the previously listed equipment. Documents reviewed included installation drawings, mechanical equipment installation travelers, valve checklists, nondestructive examination reports, nonconformance reports (NCRs), material list, and process data checklists. The manufacturer's ASME Code Data Reports were reviewed for the steam generator PORV and the 3-way power operated valve.

Also included in the records review was the examination of the fabrication package for the pressurizer safety and relief valve piping and support assembly. The package contained data associated with the actual fabrication of the assembly and the material tests and certification reports.

The records were reviewed for attributes required by the codes or specification from which they were fabricated to and also for retrievability, completeness, and legibility. No problems or discrepancies were identified by the NRC inspector during review of the installation and fabrication records.

No violations or deviations were identified.

7. Review of Unit 2 Preoperational Test Procedures

The NRC inspector reviewed the following preoperational test procedures:

a. Procedure 2-RC-P-01, "Reactor Coolant System Cold Hydrostatic Test," Revision 0, dated October 23, 1987. (70362)

The objective of this procedure was to verify the integrity and leak-tightness of the reactor coolant system (RCS) and the associated systems that form the RCS boundary. This scheduled hydrostatic test of the primary system is performed to meet the requirements of the FSAR, Section 14.2.12.2 (73), and the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Class 1 requirements. This primary system hydrostatic test will be performed at a test pressure of 3107 (+20, -0) psig and a system temperature greater than 150°F, but less than 250°F.

The NRC inspectors identified several potential problem areas (e.g., resolve relief valve pressure relief setting of 3185 (+65, -0) psig versus lower weld on tube sheet to tube maximum pressure of 3121 psig; establish provisions for completing hydrostatic test if 1 of the 2 calibrated gauges should fail; and appropriate protection for the suction side of the positive displacement pump from overpressure) and discussed these areas with cognizant Unit 2 licensee personnel. These identified areas and other areas of lesser concern will be reviewed and evaluated by the licensee's technical staff prior to initiation of the scheduled primary system hydrostatic test. The NRC inspectors will "re-review" the procedure, if revisions are made prior to performance of the scheduled cold hydrostatic test of the RCS. These potential problem areas would not have resulted in a reduction in the safe operation capabilities of the plant or the scheduled performance of this test, if these items had not been identified.

b. Procedure 2-SP-P-01, "Solid State Protection System Reactor Protection Logic Test," Revision 0, dated November 13, 1987. (70305)

The objective of this test is to verify that the reactor protection logic functions as designed and that the solid state protection system (SSPS) internal logic, excluding inputs and outputs, is in the correct configuration using the installed test equipment. This test will partially satisfy the requirements stated in Chapter 14, Section 14.2.12.2 (42.b.2) of the FSAR by verifying the combinational logic internal to the SSPS logic trains functions, as designed, and by verifying that the combinational logic associated with the SSPS inputs is correct. The proper SSPS internal logic configurations must demonstrate functions as designed, using the installed test equipment to meet acceptance criteria.

The NRC inspector did not identify any technical, safety, or operational problems in this test procedure.

c. Procedure 2-SP-P-02, "Solid State Protection System," Revision 0, dated September 25, 1987. (70305)

The purpose of this SSPS test is to verify the master relay - output relay configuration is installed and will function as designed. This test is to verify continuity from the master relays to the respective output relays, using the installed test system and the approved procedure. This test will partially satisfy the requirements stated in Chapter 14, Section 14.2.12.2 (42.b.1) of the FSAR. The test is designed to verify that the output relays test panel functions as designed and that the setpoints associated with the SSPS inputs are within the specified tolerances. The test must demonstrate proper relationship between master relay and the output relays by verifying continuity from the mater relays to the appropriate output relays, using the installed test equipment to meet acceptance criteria.

The NRC inspector did not identify any technical, safety, or operational problems in this test procedure.

d. Procedure 2-RC-P-02, "Hot Functional Test," Revision 0, dated November 24, 1987.

The objectives of this procedure are:

(1) Provide a guideline for the sequence of event and testing that is to be performed during the initial primary system heatup, testing at normal primary coolant system temperatures, and plant cooldown at completion of the testing.

(2) Provide a record that the required reactor coolant pump (RCP) operating time is at least 240 hours at full flow operation with at least 120 hours of this operating time at a temperature of at least 525°F. Two hundred forty hours of RCP running time is adequate to provide the required hours at temperature on the reactor vessel internal components.

(3) Verification of plant operating procedures during the scheduled evolutions involved during the hot functional testing (HFT), including:

(a) heatup of RCS to normal operating temperature,
(b) solid plant control,
(c) cooldown of RCS to ambient conditions,
(d) degassing of the RCS,
(e) RCS temperature control,
(f) RCP operation,
(g) pressurizer steam bubble formation and collapse of the steam bubble (plant going into solid RCS status),
(h) charging and letdown operation, and
(i) cooldown from hot no load from outside the control room.

Acceptance criteria included satisfactory performance of a solid plant pressure control test, opening of the PORVs in two seconds or less with the RCS at hot no-load conditions, and the specific acceptance criteria stated in the various procedures that are scheduled to be performed during the HFT.

The tests scheduled to be performed during the HFT should demonstrate that the HFT will meet the requirements stated in Section 14.2.12.2 (98), "Reactor Coolant System Hot Functional Preoperational Test Summary," of the FSAR.

The NRC inspector did not identify any technical, safety, or operational problems in procedure.

e. Procedure 2-SI-P-02, "Safety Injection Accumulators," Revision 0, dated November 18, 1987.

The objective of this test procedure was to verify that the safety injection (SI) accumulator system discharge and operational performances in the cold unpressurized RCS meet design requirements.

The NRC inspector discussed some specific comments that could improve the overall comprehension of this procedure. The comments, if implemented by the licensee, would not change the safety considerations of this procedure.

f. Procedures 2-SI-P-01, "Safety Injection System Train A," Revision 0, dated November 5, 1987, and 2-SI-P-04, "Safety Injection System Train B," Revision 0, dated November 5, 1987. (70301)

The objectives of these two test procedures were to demonstrate proper operation of:

(1) High head safety injection (HHSI) and low head safety injection (LHSI) pump controls and interlocks, including response to SI signals and load sequencer start and stop signals.

(2) SI system valves to an SI signal and to a containment isolation phase A signal.

(3) Containment sump isolation valves interlock with reactor water storage tank outlet valves.

The tests scheduled to be performed in accordance with each of these two test procedures should demonstrate that each train of the SI system functions as required by design and will meet the requirements stated in Section 6.3, "Emergency Core Cooling System," and Section 14.2.12.2 (76) of the FSAR.

The NRC inspector requested clarification on three groups of wording on these two procedures. The clarifications were related to general topics and would not affect the safety significance of these procedures.

g. Procedure 2-RS-P-01, "Rod Control System," Revision 0, dated December 17, 1987. (70332)

The objectives of this preoperational test were to demonstrate that the control rod drive system (CRDS) functions in automatic and manual (programmed/individual bank) modes, under no-load (control rod drive mechanisms (CRDMs) not connected) conditions and also to demonstrate that the CRDS protective and control functions operate correctly, providing rod stop capabilities. The scheduled performance of this test of the CRDS should meet the requirements stated in Section 7.7.1.1, "Control Systems Not Required for Safety; Rod Control System, Amendment 61," and Section 14.2.13.2 (47), "Control Rod Drive System Preoperational Test Summary, Amendment 61," of the FSAR.

The acceptance criteria were established to assure that: (1) the CRDS functions as designed in the various modes; and (2) the CRDS protective, control and alarm functions operate as designed to verify the various CRDS functions during testing of stop rod motion on a control rod stop or plant interlock.

The NRC inspector did not identify any technical, safety, or operational problems in this test procedure.

No violations or deviations were identified.

8. Inspection of the Location of the Manual Reactor Trip Circuit in the South Texas Project, Units 1 & 2, SSPS (92703)

The inspection of the STP (Units 1 & 2) SSPS was prompted by IE Bulletin 85-18.

The objective of this inspection was to determine whether the licensee's controlled drawings of the SSPS correctly depict the actual location of the manual trip circuit and confirm that the manual trip circuits are located downstream of the output transistors, Q3 and Q4, in the undervoltage (UV) output circuit.

The NRC inspector reviewed the controlled drawings of the STP, Units 1 & 2, SSPS. In addition, the NRC inspector examined the installed card containing the manual reactor trip circuit for Unit 2.

The NRC inspector had no further questions concerning the drawings and the subsequent hardware.

9. Onsite Design Activities (37055)

An inspection was conducted to determine whether onsite design activities, including controls for engineering and construction initiated field changes, are being conducted in compliance with the technical and QA requirements described in the FSAR. The licensee has the overall responsibility for the design and engineering of the facility. The programmatic description for the implementation of this responsibility and the requirements imposed on contractors authorized to performed design work is provided in Section 3.0, "Design Control," of HL&P's Quality Assurance Program Description (QAPD). The NRC inspector reviewed Revision 19 of the QAPD and found it to satisfy the design control criteria of 10 CFR Part 50, Appendix B.

The licensee has assigned the authority to Bechtel and Westinghouse to perform the design, engineering, and design verification at STP. Presently both of these organizations perform design control functions with onsite personnel. In 1987, Bechtel expanded the engineering effort located onsite to include the project engineering personnel formerly assigned to the Houston field office. This newly formed project engineering team now performs virtually all the Bechtel-related design work. Section 3.0, "Design Control," of Bechtel's Project Quality Program Manual contains the policies and requirements for Bechtel design work. Westinghouse has an onsite support engineering team (SET) which performs limited design work. The SET is a multidisciplined group of engineers operating onsite as an extension of the design engineering functions in Pittsburgh. The STP Interface Document for Westinghouse Support Engineering Team provides the scope and responsibilities for onsite design-related work. The NRC inspector found both of these documents to be in compliance with the design control requirements imposed by the licensee in the QAPD.

The following implementing procedures utilized in design-related work at STP were reviewed by the NRC inspector:

a. Bechtel Engineering Department Procedures (EDPs).

- No. 2.13, "Project Engineering Team Organization and Responsibilities," Revision 5
- No. 4.1, "Design Criteria and Project Q-List," Revision 6
- No. 4.26, "Interdisciplinary Design Review," Revision 0
- No. 4.27, "Design Verification," Revision 3
- No. 4.37, "Design Calculations," Revision 6
- No. 4.46, "Project Drawings," Revision 10
- No. 4.47, "Drawing Change Notice," Revision 6
- No. 4.61, "Nonconformance Reports," Revision 2
- No. 4.62, "Field Change Request (FCRs)/Field Change Notices (FCNs)," Revision 8
- No. 4.72, "Configuration Control Package," Revision 6

b. Bechtel Engineering Directive (PED) No. 041, "Design Checklist (DCL)," Revision 2

c. Bechtel Work Plan Procedure (WPP) No. 201, "Field Change Notice," Revision 8

d. STP Standard Site Procedures (SSPs)

- No. 8, "Nonconformance Reporting," Revision 4
- No. 37, "Configuration Control Package," Revision 3
- No. 49, "Field Change Requests," Revision 2
- No. 68, "Change Approval Request (CAR)," Revision 0

The reviewed procedures provide control of engineering design work to ensure technical and regulatory requirements are met. Also included in these procedures is the assignment of responsibility and listing of requirements to assure existing design criteria are not affected by design-related changes and if so, that design requirements are properly reviewed. One observation the NRC inspector had concerned the incomplete incorporation of Bechtel's onsite project engineering team and associated responsibilities into applicable EDPs. Several EDPs still have the distinction of a Houston based project engineering group and a site engineering office. The current engineering structure should be properly depicted in EDPs to ensure assigned responsibilities concerning design work is fulfilled by the engineering staff.

During the present stage of construction at STP, the majority of design-related activities involve changes to the existing design of the facility (design changes). The NRC inspector reviewed the following engineering initiated documents to determine if established design controls were being observed.

- Westinghouse FCN Nos. THXM-10633 (Addition of Limit Switches on the Refueling Machine) and THXM-10630 (Modification to Reactor Coolant Water Purity Panel)

- Bechtel Configuration Control Package (CCP) Nos. 2-M-FST-244 (Replace Concentrates Transfer Pump 1B and Add Flow Orifice), 2-M-ST-0235 (Relocate Valve WS 051), and 2-M-FST-0220 (Modification of Diesel Generator Air Filter Basket Lifting Device)

- FCR Nos. SM-00519, EM-00603, EM-00605, and EM-00084.

The NRC inspector discussed with the Westinghouse cognizant engineer the design-related details of the two FCNs and observed the completed work on the Unit 1 refueling machine. The engineer was very knowledgeable of the existing design criteria governing these two jobs and the applicable design control procedures. Proper design review and approval was apparent during the preparation of these documents.

The NRC inspector also discussed with Bechtel engineers the design considerations associated with the three CCPs. While familiar with procedural design control and the application of these controls to the CCPs, the Bechtel engineers were not as knowledgeable in the overall content of the jobs as the Westinghouse engineer. The Design Checklist (DCL) is used by Bechtel project engineering to document considerations made during the preparation of documents which change or impact other project documentation. The DCL is an effective method of ensuring responsible engineers make conscious decisions concerning the proposed change and how existing plant design may be affected. During review of the CCP that modifies the lifting device on the diesel generator air filter basket, the NRC inspector noted the calculation for sizing the lifting lug attachment weld was not in the work package. The Bechtel engineers informed the NRC inspector that an informal calculation concerning weld size was performed during CCP preparation. Considering the addition of lifting lugs was a new design for the filter basket, this type documentation should be included in the job package.

The reviewed FCRs received the proper engineering review of the proposed change. The affected drawings, while not currently revised to reflect the FCR change, were posted with the appropriate FCR. The NRC inspector did note that one of the affected drawings exceeded the maximum number of amendments that are allowed to be posted for a drawing. EDP No. 4.46, "Project Drawings," allows an aggregate total of ten amendments, before a drawing must be updated to incorporate the changes. Greater attention is required to updating drawings to prevent excessive number of amendments from overly complicating drawings used in project construction and modification.

The design-related activities at STP effectively control changes that can impact design criteria and requirements. STP procedures provide the required controls to ensure design consideration are addressed. While the NRC inspector noted some minor concerns during the review of design change documents, the engineering organizations with design responsibility do appear to be properly controlling design requirements.

No violations or deviations were identified.

10. Exit Interview

The NRC inspectors met with the licensee representatives denoted in paragraph 1 on January 29 and February 12, 1988, respectively, and summarized the inspection scope and findings. The licensee commented that the testing of welding material at different postweld heat treatment temperatures should be considered an ASME Code interpretation problem and not a violation.