APPENDIX

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-498/88-50 Operating License: NPF-76
50-499/88-50 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: July 11-15 and 25-29, 1988

Inspectors: D M Hunnicutt 9/1/88
D. M. Hunnicutt, Senior Project Engineer Date
Project Section D, Division of Reactor
Projects

T. O. McKernon 9/1/88
T. O. McKernon, Reactor Inspector, Test Date
Programs Section, Division of Reactor Safety

for D M Hunnicutt 9/1/88
H. F. Bundy, Reactor Inspector, Test Programs Date
Section, Division of Reactor Safety

Approved: E. J. Holler 9/1/88
E. J. Holler, Chief, Project Section D Date
Division of Reactor Projects

Inspection Summary

Inspection Conducted July 11-15 and 25-29, 1988 (Report 50-498/88-50)

Areas Inspected: Routine, unannounced inspection of licensee action on previously identified inspection findings, review of licensee event reports, review and evaluation of 10 CFR Part 21 Reports, Generic Letter Action Item Followup, Three Mile Island Action Item Followup, and observation of planned main generator trip from 100 percent of rated thermal power.

Results: Within the six areas inspected, no violations or deviations were identified.

Inspection Conducted July 11-15 and 25-29, 1988 (Report 50-499/88-50)

Areas Inspected: Routine, unannounced inspection of licensee action on previously identified inspection findings, review of 10 CFR Part 50.55(e) reports (IRCs), Generic Letter Action Item Followup, Three Mile Island Action Item Followup, preoperational procedure review, preoperational test results evaluation, and preoperational test program implementation.

Results: Within the seven areas inspected, no violations or deviations were identified.

DETAILS

1. Persons Contacted

HL&P

J. Westermeier, STP General Manager
M. Wisenburg, Plant Superintendent, Unit 1
W. Wellborn, Supervising Project Engineer
S. Head, Supervising Project Engineer
K. O'Gara, Project Compliance Engineer
A. Mikus, General Supervisor, Construction
M. Polishak, Lead Engineer, Project Compliance Group
D. Parker, Startup Engineer
G. Parkey, Plant Superintendent, Unit 2
A. Harrison, Supervising Project Engineer
J. Bailey, Manager, Engineering and Licensing
J. Slabinski, Operations Quality Control (QC) Supervisor
T. Jordan, Project Quality Assurance (QA) Manager
M. Duke, Staff Engineer
J. Geiger, General Manager, Nuclear Assurance
S. Rosen, General Manager, Operations Support

Bechtel

K. McNeal, Project Quality Assurance Engineer

Ebasco

R. Moore, Assistant Quality Control (QC) Site Supervisor
E. Rosoc, Site Manager
R. Sisson, Senior Resident Engineer
P. Phelon, Quality Control (QC) Supervisor

NRC

D. Hunnicutt, Senior Project Engineer
J. Bess, Resident Inspector
D. Garrison, Resident Inspector
R. Vickrey, Reactor Inspector
L. Ellershaw, Reactor Inspector
W. McNeill, Reactor Inspector
T. McKernon, Reactor Inspector
H. Bundy, Reactor Inspector
P. Wagner, Reactor Inspector

All persons listed above attended the exit interview on July 29, 1988.

2. Licensee Action on Previously Identified Inspection Findings (92701 and 92702)

a. (Closed) Violation 498/8801-02: Temporary Modifications

The requirements for a temporary modification had not been met in that no markup on a drawing had been made.

The NRC inspector reviewed the cause and corrective actions related to temporary modifications and markup on drawings. The licensee determined that the causes for this violation were an engineer's lack of attention to detail in following the procedure, lack of positive controls on the Temporary Modification Request (TMR), and a lack of clarity in the procedure with regard to blank and blind flanges. The licensee updated Drawing 5R289F05038 in compliance with Procedure OPGP03-ZO-003, "Temporary Modifications and Alterations," Revision 8, dated February 18, 1988. The licensee revised Procedure OPGP03-ZO-003 to include four additional System Engineer responsibilities. The licensee revised Procedure OPGP03-ZE-0031, "Design Change Implementation After Turnover," Revision 4, to include requirements that Plant Operations and Nuclear Training receive copies of design changes upon approval and that these departments receive notification of design change implementation. This item is closed.

b. (Closed) Violation 498/8801-03: Locked Valves

Inadequately locked and/or unlocked valves were not identified and corrected in accordance with the licensee's locked valve program.

The NRC inspector reviewed the apparent causes for this violation (failure to adequately lock and/or control valves in accordance with the Locked Valve Program) and the licensee's corrective actions related to Procedure OPGP03-ZO-0027, "Locked Valve Program," Revision 4.

The licensee appropriately identified the locked and/or unlocked valves in accordance with the Locked Valve Program. Valves required to be locked were inspected for proper position and locking devices or implementation of administrative controls. The valves were observed to be in the required positions and either properly locked or administrative controls were implemented through the South Texas Project Electric Generating Station (STPEGS) clearance process. This item is closed.

c. (Closed) Violation 498/8801-05: Surveillance Procedure Discrepancies

The licensee failed to follow procedures associated with the Surveillance Program and/or failed to provide adequate procedures to control the activities affecting the quality of surveillance as identified in this violation.

The NRC inspector reviewed the documentation related to the licensee's investigation and corrective actions. The licensee conducted a special investigation to review the Plant Operating Review Committee (PORC) process; evaluated procedures, personnel decisions and actions; revised existing procedures, (Procedure OPGP03-ZA-0002, "Plant Procedures," Rrevision 11, dated February 29, 1988, and Procedure OPGP03-ZA-0004, "Plant Operations Review Committee," Revision 9, dated June 20, 1988); and trained personnel in use of procedures. The licensee's corrective actions adequately resolved the potential surveillance procedure discrepancies. This item is closed.

d. (Closed) Violation 498/8809-06: The Licensee Failed to Reestablish the Proper Testing Configuration

This violation stated that the licensee failed to reestablish the proper testing configuration as required by Station Procedure 1TEP07-AF-0010 and that this resulted in obtaining improper test results. The licensee took exception to the violation with regard to the performance of Procedure 1TEP07-AF-0010, "Auxiliary Feedwater Proof Test," Revision 0. The licensee repeated the test (Procedure 1TEP07-AF-0010) to preclude the possibility that the test results would be subject to questions from the NRC. The licensee agreed that a procedural violation associated with the performance of Procedure 1EP04-ZL-0024, "Rod Drop Time Measurement" had occurred.

The NRC inspector reviewed the documentation and corrective actions related to the lack of control associated with the reactor trip breakers in Procedure 1PEP04-ZL-0024. The lack of control was caused by an inadequate review during the preparation and review of the procedure. The licensee's corrective actions included suspension of testing under Procedure 1TEP07-AF-0010 when questioned by an NRC inspector and reperforming this test, taking into consideration comments regarding the NRC inspector's interpretation of the Note preceeding Step 4.7 in this procedure. The licensee revised Procedure 1PEP04-ZL-0024 to incorporate specific steps to control the position of the reactor trip breakers. This revision provided consistency among various STP Procedures. This item is closed.

e. (Closed) Open Item 498/8812-01: RCS Flow Measurement Results

The NRC considered the Reactor Coolant System (RCS) flow measurement results to be an open item pending completion of documented test results and resolution of the actual RCS flow rates with the flow rates measured by the short radius elbow differential pressure taps.

Test Deficiency Record 1PEP04-ZA-0001-2, Revision 2, documented the investigation of the differences between flow rate measurements using elbow tap transmitters and flow data from other four-loop Westinghouse plants and South Texas Project (STP), Unit 1. This investigation determined that the measured results were acceptable in consideration of the expected accuracy of this type of flow measurement. FSAR Chapter 14.2.12.3 committed that the RCS flow would be measured prior to exceeding 75 percent of rated thermal power using calorimetric data. The licensee agreed to compare this measured flow rate to the mechanical design flow rate. Subsequently Field Change No. 88-0213, dated February 24, 1988, was issued to Procedure 1PEP04-ZL-0054, "Reactor Coolant System Flow Measurement at Hot Standby," to change the differential pressure (d/p) to flow rate conversion equation derivation in Addendum 1 because of to "non-standard" design of the STP Unit 1 RCS elbows.

The STP Unit 1 eblow taps were designed to ASME B&PV Code, 1974 Edition - Winter 1975, Code Case 1423-2, Section III, NB-3690, which required compliance with NB-3640 for pressure design, NB-3650 for piping analysis, and ANSI B16.11-1966 for socket welded half couplings. Functionally, the elbow taps are consistent with the recommendations of ASME B&PV Code as outlined in the text, "Fluid Meters."

Standard Westinghouse designed RCS elbows have a radius of 51 inches. STP Units 1 and 2 RCS elbows have a radius of 37 inches. The flow rate coefficient correction between the standard elbow radius and STP Units 1 and 2 elbow radius was made. Calculations were completed using the relationship between flow rate and elbow tap d/p obtained from ASME B&PV Code "Fluid Meters," Sixth Edition.

The RCS flow rate measurements and calculations using the correct elbow radius values at 75 percent of full power and the agreement of the flow rates with TS requirements and FSAR commitments verified that the RCS flow rates are acceptable. This item is closed.

f. (Closed) Open Item 498/8812-02: RCS Flow Test Requirements

The documented test results and test report for the RCS flow measurements at hot standby lacked the substance necessary to assure that test requirements have been fully satisfied.

The licensee completed the RCS flow measurements at 75 percent of full power on July 10, 1988. Station Procedure 1PEP04-ZG-0007, "Reactor Coolant System Flow Measurement At Power," Revision 3, dated May 16, 1988, verified thermal design RCS flow rate using plant calorimetric data. The calorimetric data, steam table properties, and resultant flow rate calculations are documented on Data Sheet (pages 1 and 2) of Procedure 1PEP04-ZG-0007-1, "Reactor Coolant System Flow Measurement At Power," Revision 3, dated July 10, 1988. The RCS flow rate test measured the RCS flow rate and verified design RCS flow rate using plant calorimetric data prior to operating above 75 percent of full thermal power in accordance with commitments stated in FSAR, Chapter 14.2.12.3, Test Description 6, Amendment 58, page 14.2.131.

Calorimetric data was obtained using Data Sheet Summary, 1PEP04-ZY-0015, "Statepoint Data Collection," Revision 4, dated July 10, 1988, pages 1 through 4. Data included hot and cold leg temperatures in each of the four loops for the measurements at 75 percent of rated thermal power and subsequent calculations.

The relationship between flow rate and elbow tap differential pressure (d/p) is documented in Addendum 1 to Procedure 1PEP04-ZL-0054, Reactor Coolant System Flow Measurement at Hot Standby," Revision 2, dated October 22, 1987.

The RCS flow rate measurements at 75 percent of rated thermal power and the resultant calculations were compared with acceptance criteria stated in the TS and FSAR. The flow rates in gallons per minute (gpm) were as follows:

Loop	Calculated Flow Rate in GPM at 75% Power	TS 3.2.5 Minimum Acceptable (GPM)	FSAR Table 5.1-1 Minimum (GPM)
1	103,134	98,750	94,100
2	100,486	98,750	94,100
3	101,178	98,750	94,100
4	100,157	98,750	94,100
TOTAL	404,955	395,000	382,000

The RCS flow rate measurements, calculations, and agreement between flow rates stated in the TS requirements and FSAR commitments verified that the RCS flow rates are acceptable. This item is closed.

g. (Closed) Violation 498/8833-01: Unexpected Loss of Reference Temperature Signal

Trouble shooting of the steam dump controller using Maintenance Work Request (MWR) MS-55308 resulted in the unexpected loss of the reactor plant Reference Temperature (Tref) control signal. The signal was lost unexpectedly when instrument card TY-660A was pulled as required by Step 5.2.5 of the procedure. The Tref signal controls the Rod Control (RC) system when the RC is in automatic control.

The NRC inspector reviewed the causes and the licensee's corrective actions following the loss of Tref. The alarm in the control room was unexpected because personnel were unaware of the system response that would occur during implementation of the MWR. MWR MW-55308 did not specify the system responses that would occur. The loss of Tref signal had no impact on operation of Unit 1. Card TY-660A was replaced, MWR MS-55308 was completed and Unit 1 plant equipment was restored to pre-test conditions.

The licensee issued a memorandum on May 14, 1988, that stressed the responsibilities of personnel regarding the need for detailed planning and understanding of the actions to be performed and the necessity for adequate communication of information to Operations personnel. The affected Maintenance and Plant Engineering personnel have been trained (course attendance records verified training of 158 personnel).

General Plant Procedure OPGP03-ZM-0021, "Control of Configuration Changes During Maintenance or Troubleshooting," Revision 0, dated June 15, 1988, was prepared, approved, and implemented. This procedure provided instructions for the control of configuration changes performed on permanent plant equipment during the implementation of MWRs, preventive maintenance forms or construction work request activities. This item is closed.

h. (Closed) Violation 499/8816-01: Failure to Follow Procedures

This violation involved a failure to follow procedures for installing temporary modifications. During the followup inspection, the NRC inspector reviewed the licensee's response to the violation, ST-HL-AE-2587 dated April 25, 1988, for corrective actions taken. In addition, the NRC inspector conducted a walkdown inspection of applicable relay rooms and logic cabinets. The walkdown inspection verified the licensee's program for control of electrical jumpers and other temporary modifications is effective and adequate. This item is considered closed.

No violations or deviations were identified.

3. Review of Licensee Event Reports (LERs) (90712)

a. LER 88-02: Failure to Perform Post Maintenance Local Leakage Rate Testing on Containment Isolation Valves

On January 5, 1988, with Unit 1 in Mode 5 prior to initial criticality a TS required post maintenance test (PMT) had not been performed on two Containment Isolation Valves (CIV) before entering Mode 4. Unit 1 had been operated in Mode 4 after the maintenance work and prior to discovery of the inadequate PMT. The Licensee tested these two CIVs (1-inch ball valves located on the reactor containment building radiation monitor sample exhaust line) and found that one CIV exceeded its local leakage rate requirements. The CIV was reworked and retesting verified that the local leakage rate was within the specified limits. The licensee revised Procedure OPGP03-ZM-0003, "Maintenance Work Request Program, " Revision 16, dated February 26, 1988. This procedure established a program for reporting and correcting material deficiencies, satisfying the guidelines stated in Regulatory Guide 1.33, "QA Program Requirements (Operations)", Revision 2; FSAR Chapters 13.5, "Maintenance Control" and 3.2, "Classification of Structures, Systems, and Components;"

Generic Letter 83-028, Item 2.1, (required actions based on Generic Implications of Salem ATWS Events); and SER 84-056 (Mispositioning of valves and controls disabled safety systems) and provided instructions for processing a Maintenance Work Request (MWR). The licensee conducted training on the MWR program for shift supervisors and support personnel. This LER is closed.

b. LER 88-13: Failure to Perform a Test of the Reactor Coolant System Low Flow Timers as Required by TS

On February 4, 1988, Unit 1 was in mode 3 prior to initial criticality. The licensee identified two time delay relays in the Solid State Protection System (SSPS) which had not been tested under the surveillance program as required by the TS. This test was omitted due to a procedural deficiency. Licensee actions to prevent recurrence included a review of other surveillance procedures for similar omission, revision to plant procedures on surveillance test procedure preparation and review, and revision of surveillance procedures to include low flow timer testing. This LER is closed.

c. LER 88-20: Essential Cooling Water Screen Wash Booster Pump (ECWSWBP) Mistakenly Declared Operable

On February 15, 1988, Unit 1 was in Mode 5 prior to initial criticality. A review of the inservice test on ECWSWBP 1A revealed that the pump test data was outside the acceptable limits. The pump was not declared inoperable on February 11, 1988, when the test was performed. A new Reference Values Measurement Test (RVMT) was performed on February 15, 1988. This RVMT verified that the pump was within the acceptance limits and was operable. Corrective actions included training of licensed personnel using this LER as an example and requiring a second independent review of TS surveillance test results prior to submittal to the Shift Supervisor. This LER is closed.

d. LER 88-25: Control Room Ventilation Recirculation Actuation Due To a Radiation Monitor Actuation

On March 23, 1988 with Unit 1 in Mode 2, an Engineered Safety Feature (ESF) actuation of the control room ventilation system to the recirculation mode occurred. The licensee performed diagnostic tests on the monitor and attempts were made to duplicate the event. The most probable cause was an inadvertent actuation of a control room ventilation radiation monitor during maintenance activities. No specific cause was determined; however, the licensee will continue to perform surveillance testing. The licensee completed training for maintenance personnel responsible for testing of radiation monitoring equipment to provide additional assure that maintenance activities are not a cause of future inadvertent actuations. This LER is closed.

No violations or deviations were identified.

4. <u>10 CFR Part 50.55(e) Reports (IRCs) (92700)</u>

a. (Closed) IRC No. 341 (10 CFR Part 50.55(e)): <u>Flooding in Portions of Unit 2 During Heavy Rainfall</u>

The NRC inspector reviewed the licensee's corrective actions related to the flooding in the STP Unit 2 Isolation Valve Cubicle (IVC), Mechanical Electrical Auxiliary Building (MEAB), and Fuel Handling Building (FHB). This flooding occurred in Unit 2 only as a result of heavy rainfall (approximately 9.5 inches of rain) at STP on October 22 and 23, 1986. Mechanical equipment, electrical equipment, instrumentation, stainless steel piping, and motor operated valves were affected by the flooding.

Fifty-one nonconformance reports (NCRs) were dispositioned during the cleaning and meggering (if required) of equipment in accordance with the various manufacturers' recommendations or requirements. Equipment was evaluated by the licensee's engineering staff and the manufacturer (if required) and replaced, refurbished, or repaired, if required. Equipment qualification records and documents were validated or recert ied by the manufacturers' for the affected components. This I is closed.

b. (Closed) IRC-394 (10 CFR Part 50.55(e)): <u>Essential Cooling Water (ECW) Pump Damage</u>

The NRC inspector reviewed the licensee's corrective actions related to damage to the Unit 1 B-train (1B) ECW pump. The 1B pump bearing lubrication flow was greater than the flow rate for either the 1A or 1C pump. The licensee declared the 1B ECW pump inoperable and removed the pump from service. The 1B ECW pump was subsequently disassembled. The upper and two intermediate bearings and associated sleeves were extensively damaged. The lower and upper pump half shafts were found to be bent. A 2-inch plastic pipe cap was found lodged against the upstream side of the lubrication line orifice plate.

The 1B ECW pump was returned to the pump manufacturer for refurbishment, including replacement of bearings, shaft sleeves and shafts. The manufacturer issued a certificate of compliance that certified the 1B ECW pump, Serial Number 804402, was repaired in accordance with the requirements of Purchase Order No. 14926-BF-38055.

The licensee completed performance and endurance testing for the 1B ECW pump in accordance with NRC IE Bulletin No. 83-05, "ASME Nuclear Code Pumps and Spare Parts Manufactured by the Hayward Tyler Pump Company" and in accordance with Prerequisite Test Change Notice, "Specific Prerequisite Test Procedure for ECW System Pumps," dated June 12, 1987. This item is closed.

No violations or deviations were identified.

5. 10 CFR Part 21 Report Inspections (36100)

a. (Closed) 10 CFR Part 21 Report (P21-87-074): Direct Current Motors Not Qualified by Vendors Program

Direct current motors used to operate motor operated valves (MOVs) were supplied to some valve manufacturers from Limitorque Corporation as part of their valve actuator assembly and were to be qualified as part of Limitorque Corporation's Qualification Report B-0009, dated April 30, 1976. These motors were manufacturered between December 1984 and December 1985 by H. K. Porter (now Peerless-Winsmith). These motors were not part of the Limitorque Corporation's Qualification Report because H. K. Porter, with Limitorque Corporation's concurrence, had changed design without a formal analysis, including the potential effect on environmental qualification. Subsequently, the NRC learned of two significant failures directly attributable to the use of Nomex-Kapton leads on these motors.

As a result of the Part 21 report and NRC Information Notice 87-08, the licensee initiated an investigation. During the licensee's investigation, one of the motors (D1AFMOV-0143) on Unit 1 failed to megger properly and was replaced with a similar motor from Unit 2. This deficiency was documented on NCR SE-05724. STP found a total of six MOVs with Nomex-Kapton leads. Three of these MOVs had been installed in each unit to perform similar applications.

The licensee reviewed all Unit 1 and Unit 2 safety-related valves with Limitorque Corporation's motor actuators for the specific nameplate serial number data codes identified in Limitorque Corporation's letter of December 19, 1986, to the NRC. The licensee determined that all three DC motors in each Unit were associated with the turbine driven Auxiliary Feedwater (AFW) pump (Train D) of the AFW System. These motors are identified as follows:

Unit 1	Unit 2
D1AFMOV-0019	D2AFMOV-0019
D1AFMOV-0143	D2AFMOV-0143
D1AF-FV-7526	D2AF-FV-7526

Corrective action has been completed. The licensee has completed a Deficiency Evaluation Report (DER) 87-033 to evaluate the reportability of these potential deficiencies. This DER indicated that a safety hazard would not exist and that this deficiency would not be reportable under either 10 CFR Part 21 or 10 CFR Part 50.55(e). This item is closed.

b. (Closed) 10 CFR Part 21 Report (P21-87-080): Containment Hydrogen Analyzer Systems

This Part 21 Report stated that a design deficiency in the containment hydrogen analyzer systems could permit a loss of calibration gas before the scheduled replacement interval of the storage bottles is reached. Loss of the calibration gas would render the system inoperable. The hydrogen analyzers were originally provided by Exo-Sensor, Inc., which has since been purchased by Whitaker, Inc.

An investigation by the licensee verified that the hydrogen analyzers at STP are Model K-III manufacturered by Comsip Delphi, Inc. At STP each train's gas supply downstream of the storage bottle consists of a pressure regulator, a bubble tight solenoid shutoff valve, and a flow control/isolation valve. Pressure of the gas upstream of the solenoid valve is 25 psig. The shutoff valve is rated to 500 psig. The licensee's procedure includes a check every 12 hours to ensure equipment operability. This item is closed.

c. (Closed) 10 CFR Part 21 Report (P21-87-081): Eastern Testing and Inspection Records Pertaining to NDE Services

This Part 21 report indicated that quality assurance controls including thermometer serial numbers and surface temperatures of the items examined, calibration records, and technicians' eye examination records pertaining to nondestructive examination (NDE) records utilized by Eastern Testing and Inspection, Inc., (ETI) during performance of NDE services at Peach Bottom Units 2 and 3 were not performed as required.

The licensee's investigation determined that Westinghouse had never directly used the services of ETI; however, Westinghouse determined that one supplier, Joseph Oat Corporation, had utilized ETI for Level I work, specifically shooting radiographs under Joseph Oats Corporation supervision and its Quality Assurance Program. Actual interpretation of the radiographs was conducted by Level II or Level III Joseph Oat Corporation employees. The density and correctness of the radiographs were verified by Jospeh Oats Corporation personnel and found acceptable. Westinghouse concluded that the use of ETI had not affected the safety and/or quality of the components identified by its investigation. The investigation determined that ETI was not on Bechtel's Evaluated Supplier List. Bechtel contacted 16 vendors and determined that only Joseph Oats

Corporation had used ETI for radiographic services on STP orders. The results of the licensee's investigation demonstrated that ETI had not performed quality or safety-related work related to the reported Part 21 deficiencies. This item is closed.

d. (Closed) 10 CFR Part 21 Report (P21-87-083): Basler Electric Transformers

A saturable core transformer (part No. BE12173-001) manufacturered by Basler Electric in Highland, Illinois, failed in service. An inspection by the licensee (TVA) of the failed transformer revealed that the insulation between the windings was inadequate.

The licensee performed an investigation and determined that Basler Electric Transformer No. BE12173-01 was not used in any of the standby emergency diesel generators at STP. This item is closed.

e. (Closed) 10 CFR Part 21 Report (P21-87-084): Borg Warner Gate Valves

A potential substantial safety hazard related to fasteners installed in motor operated 16" x 12" x 16" gate valves manufacturered by Borg Warner and supplied by Combustion Engineering and installed in the Shutdown Cooling System (SCS) at Palo Verde Nuclear Generating Station Unit 3 was reported as a Part 21 Report.

The licensee's investigation determined that Bechtel had purchased only 2-inch and smaller gate, globe, and check valves from Borg Warner. Westinghouse had no purchase orders with Borg Warner and suppled no Borg Warner valves to STP. Therefore, there are no Borg Warner valves at STP that should be evaluated for potential problems reported in this Part 21 report. This item is closed.

No violations or deviations were iden[illegible]

6. Generic Letter Action Item Followup

(Closed) Open Item 498/8739-04: Gene[illegible] Letter (GL) 83-23, Item 2.2, "Equipment Classification and Vendor Interface"

The NRC inspector reviewed the licensee's response to NRR letter dated May 4, 1987. The licensee response discussed the implementation of the Nuclear Utility Task Action Committee/Vendor Equipment Technical Information Program (NUTEC/VETIP) at STP and the quality assurance controls over vendor-supplied service on safety-related equipment.

a. The licensee prepared seven procedures to provide HL&P (STP) with a method of communicating with NRC, INPO, other utilities, and vendors regarding equipment technical information.

b. The licensee completed Revision 5 to IP-1.8Q, "Control of Vendor Documents," on June 30, 1988. This program included a periodic contact (interface) with vendors of safety-related components. The licensee identified and classified the vendor manuals for the key components referenced in the NRR letter dated May 4, 1987. Completion of Revision 5 to IP-1.8Q on June 30, 1988, met the commitment the licensee made to the NRC and closed reference to revision of this procedure in paragraph 4.b of NRC Inspection Report 50-498/88-10.

c. The licensee's QA program required vendors performing services on safety-related equipment to be listed on the Approved Vendors List (AVL). Vendor's performing maintenance services under an MWR are under the direct responsibility of the HL&P Maintenance Department. The MWR requires that maintenance activities on quality-related equipment or systems be performed in accordance with existing procedures and requirements, instruction and procedure controls, and related quality requirements, inclduing specifications, as necessary.

   These approved procedures establish a single program for the receipt, review, status determination, and distribution of vendor supplied design and technical documents and verify periodically that quality-related vendor manuals are current and can serve as a reliable basis upon which the licensee's operation and maintenance may be based. This item is considered closed for Units 1 and 2.

No violations or deviations were identified.

7. <u>Three Mile Island Action Item Followup (25565)</u>

(Closed) Open Item (498/8739-03): <u>TMI Item II.E.4.2, "Containment Isolation Dependability"</u>

The licensee transmitted FSAR changes (HL&P Letter to NRC, ST-HL-AE-2182, dated May 29, 1987) describing containment isolation on a Phase B isolation signal of the component cooling water (CCW) supply and return to the reactor coolant pump heat exchangers, reactor coolant drain tank heat exchanger, and the excess letdown heat exchanger. The CCW flow to the components share common containment inlet and outlet penetrations.

Additional information concerning containment isolation, CCW supply/return to reactor coolant pumps, reactor coolant drain tank heat exchanger and excess letdown heat exchanger was submitted to the NRC by HL&P Letter ST-HL-AE-2237, dated June 11, 1987. HL&P determined that this shared system arrangement with common containment inlet and outlet penetrations does not meet the requirements of NUREG-0737, Item II.E.4.2(3) and Standard Review Plan (SRP) 6.2.4, Section III, which require nonessential systems to be isolated at the containment on a Phase A containment isolation signal (safety injection). HL&P requested a deviation from NUREG-0737, Item II.E.4.2(3) and SRP 6.2.4, Section III. The NRC staff considered this deviation from the requirements of NUREG-0737, Item II.4.2, and judged that the deviation is acceptable on the basis that adequate isolation capability exists in the form of redundant valves and the piping system itself. The NRC position is documented in Safety Evaluation Report (SER), NUREG-0781, Supplement No. 4, STP Units 1 and 2, Section 6, "Engineered Safety Features," paragraph 6.2.4, "Containment Isolation System," July 1987. The NRC staff position closed this item.

(Closed) Open Item (498/8708-19): TMI Item I.G.1.3, "Training Requirements During Low Power Testing"

The NRC inspector reviewed the following licensee training records to verify that training was provided prior to "hands on" expereince/training during low power testing at Unit 1:

a. Procedure 1, "Control Room Evacuating," POP04-ZO-0001, Revision 3, dated February 1, 1988. This procedure evaluated each operating crew's performance (licensed and nonlicensed) in the execution of the control room evacuation and each individual's performance on the assigned watchstations. Drills for each shift were completed during February 1988.

b. Summary of Training for Requalification Cycle 6 (January 1 through February 5, 1988). The [illegible] of this requalification cycle was to prepare operators for the scheduled annual simulator examinations (during the period between February 15 and March 18, 1988) and make available 4 hours per day for self-study for the scheduled written examinations to be administered during this scheduled requalification cycle.

c. Course Summary for Licensed Operator Requalification 701 (conducted during the period between April 20 and May 22, 1987). This requalification training included training in off-normal procedures, including control room evacuation, relation to postulated fires (i.e., control room and relay room fires), and natural circulation mode of reactor control.

An NRC inspector observed a planned shutdown of Unit 1 from a thermal power level of 75 percent and subsequent control of the shutdown reactor from the Unit 1 Auxiliary Shutdown Panel (ASP). The shutdown reactor was in natural circulation mode while being controlled from the Unit 1 ASP. This scheduled Unit 1 test verified that operators had received the required training. The operators demonstrated their proficiency by performing the required functions at the ASP. The operators demonstrated that they understood the procedures and that they could implement these procedures, establish and maintain control of Unit 1, and maintain the plant in a safe shutdown condition from outside the control room using the equipment and instrumentation located at the ASP. This test was performed in accordance with specified procedures and licensee commitments.

The licensee has committed to complete training of Unit 2 operators (licensed and nonlicensed) on the ASP by November 23, 1988. This item is closed for Unit 1 and Unit 2.

No violations or deviations were identified.

8. Generator Trip From 100 Percent of Full Power (72580)

The NRC inspectors observed the scheduled Unit 1 main generator trip from a steady state power level of 100 percent of rated thermal power. The purpose of this plant trip from 100 percent was to verify the ability of Unit 1 plant to sustain a trip of the main generator and to determine the overall response time of the reactor coolant hot leg resistance temperature detectors (RTDs).

The NRC inspectors observed the tripping (opening) of the main generator output breaker from the Unit 1 control room at 10:33 a.m. (CDT) on July 28, 1988. The NRC inspectors observed the plant responses and licensee personnel actions to this net loss of electrical load (loss of the electrical load results in the maximum credible overspeed condition for the main turbine) from vantage positions in the control room and on the main turbine/generator deck.

The NRC inspectors determined that this test was conducted in accordance with approved Procedure 1PEP04-ZY-0102, "Plant Trip for 100% Power," Revision 3, dated July 27, 1988. During the test period pertinent plant parameters were recorded by licensee personnel. The recorded data and observed equipment responses verified that associated plant equipment, instrumentation, and components performed in accordance with design requirements and within the anticipated limits. Observations and review of data indicated no major problems or potential nuclear safety concerns. Observations verified that no excessive vibrations in piping or components occurred during this test. The main steam valves closed smoothly and with anticipated force. No unusual noises or equipment malfunctions were observed. The minimum reactor coolant system average temperature (Tavg) was 558°F Tavg immediately prior to the scheduled plant trip was 567°F.

The NRC inspectors verified that the following acceptance criteria stated in Procedure 1PEP04-ZY-0102 was met:

a. The safety limits stated in TS 2.1.1 and 2.1.2 were not exceeded.

b. The neutron flux dropped to less than 15 percent of full power value in less than two seconds.

c. All control and shutdown rod cluster control assemblies (RCCAs) dropped into the reactor core.

d. Safety injection actuation did not occur.

e. Pressurizer safety valves did not lift.

f. Main Steam safety valves did not lift.

g. The overall hot leg RTD response time was less than 7.531 seconds.

The NRC inspectors observed the following safety-related conditions and licensee actions:

a. Plant electrical loads transferred as designed. Conditions of TS 3.8.1.1 were met during the test.

b. The turbine bypass valves operated to maintain the RCS within established limits.

c. Licensee personnel performed necessary manual functions to maintain safe plant limits, including manually tripping the reactor. In accordance with Procedure 1PEP04-ZY-0102, a manual scram was initiated when an automatic scram signal was not received within 2 seconds after opening the main generator breaker.

d. Licensee personnel placed and maintained the plant in normal shutdown condition (Hot Standby - Mode 3) following the plant trip.

The NRC inspectors determined through observations, discussions with licensee personnel, and review of procedures, TS, and FSAR Chapter 14.2.12.3.23, Amendment 56, that the licensee met the requirements and commitments related to the plant trip from 100 percent of rated thermal power. The successful completion of this test verified the ability of the plant and licensee personnel to sustain a trip of the main generator from 100 percent of rated thermal power and maintain the plant in a safe condition.

No violations or deviations were identified.

9. Preoperational Procedure Review (70354)

During the inspection, the NRC inspector reviewed the following preoperational test procedure:

° 2NI-P-01, Revision 0, "Nuclear Instruments System," dated July 6, 1988

The objectives of this procedure are to demonstrate operability of the Nuclear Instrumentation system, verify each channel functions properly and provides the as-designed output permissive and reactor trip system interlocks. Within the scope of this inspection, the NRC inspector verified that the procedure was satisfactory and the stated objectives were delineated in the test procedure.

No violations or deviations were identified.

10. Preoperational Test Results Evaluation

The NRC inspectors verified that the results of the following tests were within the stated acceptance criteria and that any deviations were properly dispositioned. The licensee effectively identified those areas requiring test exception and completed the necessary test procedures and retesting changes The licensee's compliance with administrative controls for test execution and test results review and evaluation was complete and adequate. Changes made to test procedures were properly reviewed, approved, and the procedures annotated. Tests were properly conducted with the appropriate individuals initialing and dating the procedural steps along with the necessary quality assurance signoffs. The NRC inspectors verified that the licensee was meeting the commitments of Regulatory Guide 1.68 and the FSAR.

The licensee's compliance with administrative practices of Startup Administrative Instructions (SAI) 18, Revision 7, "Preoperational Testing," and SAI-19, Revision 6, "Acceptance Testing" was both evident and adequate. Furthermore, the NRC inspectors verified that all test data met the stated acceptance criteria.

a. Engineered Safety Features System (70322)

- 2-SI-P-02, Revision 1, "Safety Injection Accumulators"
- 2-SI-P-04, Revision 1, "Safety Injection System Train B"
- 2-SF-P-03, Revision 2, "Safeguard Test Cabinet Train A"
- 2-SF-P-04, Revision 1, "Safeguard Test Cabinet Train B"
- 2-SF-P-05, Revision 1, "Safeguard Test Cabinet Train C"

b. Reactor Protection System (70325)

- 2-SP-P-01, Revision 0, "Solid State Protection System (SSPS)-Reactor Protection Logic Test"
- 2-SP-P-02, Revision 0, "SSPS - Reactor Protection Master Relay Test"
- 2-HM-P-01, Revision 1, "MAB HVAC System"
- 2-HE-P-02, Revision 0, "Electrical Space HVAC System"
- 2-CH-A-03, Revision 1, "MAB Chilled Water System"

No violations or deviations were identified.

11. Preoperational Test Program Implementation (70302)

During this portion of the inspection, the NRC inspectors verified that the licensee has implemented and complied with written administrative controls over the preoperational testing program. The NRC inspectors conducted interviews with the test program director and other testing personnel, reviewed a sampling of tests from the test program index, and reviewed qualification records of key test personnel. The NRC inspectors verified that the test program director was familiar with the responsibilities of key test personnel, lines of authority and responsibility and interfaces amongst those organizations involved in the test program. Test procedures were reviewed and approved in accordance with the applicable administrative procedures. Furthermore, procedures contained references to the most current issues of drawings and vendor's manuals. Component configuration packages (CCPs) were reviewed, processed, and implemented in accordance with procedural controls. In those instances that warranted retest, the test procedures were properly revised to incorporate test of the design changed system. The NRC inspectors conducted interviews with key test personnel to verify their familiarity with administrative controls covering the conduct of corrective and preventive maintenance during preoperational testing. Furthermore, the NRC inspectors reviewed training records to verify appropriate certification of key test personnel, training had been conducted covering administrative controls for testing, and other applicable quality assurance/quality control indoctrination. Within the scope of this inspection, the NRC inspectors confirmed the licensee's compliance with Regulatory Guide 1.68, FSAR commitments, and guidances provided in ANSI N18.7-1976 and Regulatory Guide 1.58.

No violations or deviations were identified.

12. Exit Interview

The NRC inspectors met with the licensee personnel (denoted in paragraph 1) on July 29, 1988. The NRC inspectors summarized the scope and findings of the inspection. The licensee did not identify as proprietary any of the information provided to, or reviewed by, the NRC inspectors.