APPENDIX B

U. S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-445/85-07 Permit: CPPR-126
50-446/85-05 CPPR-127

Docket: 50-445; 50-446

Applicant: Texas Utilities Electric Company (TUEC)
Skyway Tower
400 North Olive Street
Lock Box 81
Dallas, Texas 75201

Facility Name: Comanche Peak Steam Electric Station (CPSES)
Units 1 and 2

Inspection At: Glen Rose, Texas
Inspection Conducted: April 1, 1985, through June 21, 1985

Inspectors: H. S. Phillips 10/1/85
H. S. Phillips, Senior Resident Reactor Inspector Construction (pars. 1, 2, 3, 8, 9, 10, 11, 15, 16, 17, 18, and 19) Date

for D M Hunnicutt 10/2/85
J. E. Cummins, Senior Resident Reactor Inspector Construction (April 1 - May 10, 1985) (pars. 1, 3, and 19) Date

H. S. Phillips for 10/1/85
D. E. Norman, Reactor Inspector (pars. 1, 12, 13, 14, and 19) Date

D M Hunnicutt 10/2/85
D. M. Hunnicutt, Section Chief Reactor Projects Branch 2 (pars. 1, 4, 5, 6, 7, and 19) Date

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Approved: D M Hunnicutt 1/28/86
D. M. Hunnicutt, Section Chief, Date
Reactor Project Section B

Inspection Summary

Inspection Conducted April 1, 1985, through June 21, 1985(Report 50-445/85-07)

Areas Inspected: Routine, announced and unannounced inspections of Unit 1 which included plant tours and review of plant status, action on previous NRC inspection findings (violations/unresolved items), review of documentation for site dams, and review of 10 CFR Part 21 and 10 CFR Part 50.55(e) construction deficiency status, The inspection involved 77 inspector-hours onsite by four NRC inspectors.

Results: Within the areas inspected, five violations were identified: failure to promptly correct an identified problem with RTE - Delta Potential Transformer Tiltout Subassemblies, paragraph 3.a.; commercial non-shrink grout was used to grout the Unit 1 reactor coolant pump and steam generator supports in lieu of Class "E" concrete, paragraph 3.b.; hydrogen recombiners out-of-specification voltage recorded on quality release document but QC receipt inspector accepted, paragraph 3.c; failure to provide objective evidence to show that central and truck mixer blades were inspected, paragraph 8; and failure to issue a deficiency report on cement scales that were out-of-calibration, paragraph 9.c.

Inspection Summary

Inspection Conducted April 1, 1985, through June 21, 1985 (Report 446/85-05)

Areas Inspected: Routine, announced and unannounced inspections of Unit 2 which included plant tours and review of plant status, action on previous NRC inspection findings (violations/unresolved items), review of documentation for site dams, review of documentation for voids behind the stainless steel cavity liner of reactor building, observation of NDE on liner plates, inspection of concrete batch plant, review of calibration laboratory records for batch plant, review of concrete laboratory testing, inspection of level C and D storage, review of reactor pressure vessel (RPV) and piping records/completed work, and review of 10 CFR Part 21 and 10 CFR Part 50.55(e) construction deficiency status, and review of violation and unresolved items status. The inspection involved 335 inspector-hours onsite by four NRC inspectors.

Results: Within the sixteen areas inspected three violations were identified: failure to correct RTE-Delta transformer problem, paragraph 3.a; failure to provide objective evidence to show that concrete central and truck mixer blades were inspected, paragraph 8; and failure to issue a deficiency report on cement scales that were out-of-calibration, paragraph 9c.

DETAILS

1. Persons Contacted

Applicant Personnel

M. McBay, Unit 2 Reactor Building Manager
B. Ward, General Superintendent, Civil
D. Chandler, QA/QC Civil Inspector
W. Cromeans, QA/QC, TUGCO Laboratory/Civil Supervisor
*#J. Merritt, Assistant Project General Manager
*#P. Halstead, Construction Site QA Manager
#C. Welch, QA Supervisor TUGCO (Construction)
J. Walters, TUGCO Mechanical Engineer
K. Norman, TUGCO Mechanical Engineer
J. Hite, B&R Materials Engineer
G. Purdy, B&R CPSES QA Manager

*Denotes those present at May 10, 1985 exit interview.
#Denotes those present at June 10, 1985 exit interview.

The NRC inspectors also interviewed other applicant employees during this inspection period.

2. Plant Status

Unit 1

At the time of this inspection, construction of Unit 1 was 99 percent complete. The fuel loading date for Unit 1 is pending the results of ongoing NRC reviews.

Unit 2

At the time of this inspection, construction of Unit 2 was approximately 74 percent complete. Fuel loading is scheduled for approximately 18 months after Unit 1 fuel loading.

3. Applicant Action on Previous NRC Inspection Findings

a. (Closed) Unresolved Item 445/8440-02: Potential Problem with Potential Transformer Tiltout Subassemblies.

By letter dated June 15, 1983, Transamerica Delaval notified the applicant of an RTE - Delta 10 CFR Part 21 report to the NRC reporting a potential problem with the primary disconnect clips of the potential transformer tiltout assembly used in the emergency diesel generator control panels at CPSES. The Transamerica Delaval

letter also provided instructions for correcting the problem. However, the NRC inspector could not determine if the problem had been corrected at CPSES and made this an unresolved item. The applicant determined that the problem had not been corrected and subsequently performed the recommended corrective action. The Unit 1 corrective action work activities were documented on startup work permits Z-2912 (train A) and Z-2914 (train B). The Unit 2 work activities are being tracked as master data base (MDB) item 3003-31. The failure to promptly correct this identified problem is an apparent violation (445/8507-01; 446/8505-01).

b. (Closed) Unresolved Item 445/8416-03: Commercial Grout Used in Lieu of Class "E" Concrete

The applicant determined that the use of nonshrink commercial grout in lieu of the Class "E" concrete specified on drawing 2323-S1-0550 was acceptable. Design Change Authorization 21179 was issued to drawing 2323-S1-0550 accepting the use of the commercial non-shrink grout. However, the failure to grout with Class "E" concrete as specified on the drawing at the time the work was accomplished is an apparent violation (445/8507-02).

c. (Closed) Unresolved Item 445/8416-04: Hydrogen Recombiners - Out-of-Specification Voltage Recorded on Westinghouse Quality Release Document

Quality Release N-41424 was revised by Westinghouse changing the specified voltage from 10+-2V to 12+-2V which put the questionable voltage within specification limits. However, the failure of receipt inspection to verify that the QRN-41424 was filled out accurately as required by Procedure QI-QAP7.2-8 is an apparent violation (445/8507-03).

d. (Open) Unresolved Item 445/8432-06; 446/8411-06; Lobbin Report Described Site Surveillance Program Weaknesses

During this reporting period the NRC inspector reviewed the status of this open item several times and interviewed TUEC management and site surveillance personnel. The Lobbin report stated that the scope and objectives of the site surveillance program were unclear, lacking both purpose and direction.

There is no specific regulatory requirement to have a surveillance program; however, TUEC committed to have a surveillance program and has established procedures to implement such a program as a part of the 10 CFR Part 50, Appendix B, QA program. This extra effort is a strength; however, the NRC inspector also observed, as did the Lobbin Report, that the surveillance program lacks both purpose and direction to be effective and complimentary to the audit and

inspection programs. Since the TUEC audit group is not located on site, the TUEC surveillance program on site takes on added significance.

This item was discussed with the TUEC site QC manager who described a reorganized site surveillance function and changes that have occurred. New procedures which describe this organization's duties and responsibilities are forthcoming.

TUEC has elected to defer responding to the violations pertaining to the audit function in NRC Inspection Report 445/84-32; 446/84-11, but rather to have the Comanche Peak Response Team (CPRT) respond to this report and other QA matters. The surveillance issue is closely tied to the audit deficiencies in NRC Inspection Report No. 445/84-32; 446/84-11. This item will remain open pending the review and implementation of the CPRT action plan. A special point of interest will be how audits and surveillance work together to evaluate the control of all safety-related activities on site to assure quality, especially the overview of quality control effectiveness.

4. Document Inspection of Site Dams

The NRC inspector reviewed documents describing the inspection activities performed on the Squaw Creek Dam (SCD) and the safe shutdown impoundment (SSI) for impounding cooling water for the two units at CPSES. The purpose of the SCD is to impound a cooling lake for CPSES. A secondary reservoir (SSI) is formed by a channel connecting the SCD impoundment to the SSI.

Three documented inspections have been performed since 1980. The inspections were:

a. Relevant data for SCD is contained in Phase I Inspection, National Dam Safety Program, Squaw Creek Dam, Somervell County, Texas, Brazos River Basin, inspection by Texas Department of Water Resources. Date of Inspection: June 10, 1980.

b. Inspection on August 25, 1982, by registered professional engineers from Mason-Johnston & Associates, Inc., and Freese & Nichols, Inc.

c. Inspection on September 19, 1984, by a registered professional engineer from Mason-Johnston & Associates, Inc.

The inspection activities consisted of visual inspections by inspection teams that included accompanying Texas Utilities Service, Inc. (TUSI), and Texas Utilities Generating Company (TUGCO) representatives. Photographs were taken as a part of the documentation. The data for the

piezometer observations and the data for the surface reference monuments were reviewed by applicant personnel and Mason-Johnston engineers. No items of significance were observed or reported by these inspection teams. Slight erosion areas were observed and reported. A cracked area on the service spillway upstream right bridge seat was observed by the inspection teams and continued monitoring of this area was recommended by Mason-Johnston and Associates. No signs of cracks, settlements, or horizontal movement at any location within the SCD or the SSI were reported.

The NRC inspector reviewed the applicant's records and the Mason-Johnston inspection reports. These documents indicated that the SCD and SSI were structurally stable and that the applicant was performing inspection activities to maintain the structural integrity of these dams.

The state of Texas requires periodic inspections of these dams (principally the SCD) due to inhabited dwellings downstream. The applicant has met these inspection requirements.

No violations or deviations were identified.

5. Voids Behind the Stainless Steel Cavity Liner in Unit 2 Reactor Building

In review of previous related TRT concerns, the NRC inspector reviewed applicant records, including NCR C-82-01202; NCR C-1784, Rev. 1; NCR C-1784, Rev. 2; NCR C-1766, Rev. 1; NCR C 1791, Rev. 1; NCR C-1824, Rev. 1; NCR C-1824, Rev. 2; Significant Deficiency Analysis Report (SDAR) - 26, dated December 12, 1979; DCA-20856; and Gibbs and Hill Specification 2323-SS-18. The review of records and documentation and discussions with various applicant personnel indicated the following:

> Structural concrete was placed in Unit 2 reactor building at elevation 819 feet 6-3/4 inches to 846 feet 6 inches on June 21, 1979. This concrete was placed adjacent to the stainless steel liner walls. The concrete forms for this pour were not removed until October 1979 due to subsequent concrete placements for the walls to elevation 860 feet 0 inches. When the forms were removed, honeycombs and voids were observed by applicant personnel. The applicant's review of the extent of unconsolidated concrete resulted in the issuance of SDAR-26 on December 12, 1979. Investigations were begun and Meunow and Associates (M&A) of Charlotte, North Carolina, were contracted to perform nondestructive testing on in-place concrete. M&A performed these tests on a two foot grid pattern on the compartment and liner sides of all four steam generator (SG) compartment walls. The selected test locations did not include the locations where the voids were later found to be located.

In August 1982, preparations were made to pour the concrete annulus around the reactor vessel. When the expanded metal formwork was removed from the reactor side of the compartment walls, voids were observed and NCR C-82-01202 was prepared. DCA 20856 was prepared as a procedure to repair the void area. DCA 20856 indicated that the voids were not extensive (a surface area of about 28 square feet by 8 inches maximum depth) and that the repair procedure assured that the total extent of voids had been identified. One half (0.5) of a cubic yard of concrete was used to complete the repairs as indicated on grout pour card 261.

The applicant's review and evaluation of the gird pattern and a comparison of SG compartments 2 and 3 to 1 and 4 indicated that voids did not exist in SG compartments 2 and 3. The review of test girds extended down to elevation 834 feet, which is the floor elevation of the liner. The liner walls of SG compartments 1 and 4 were not tested at elevation 834 feet, but at elevation 836 feet which is above the area of the identified voids. No testing was done on the liner side of the area of the voids below elevation 836 feet. The program also included removal of 2 inch x 2 inch plugs from the stainless steel liner at locations where test indications raised questions concerning the concrete. The inspections of the concrete by applicant personnel after the plugs were removed confirmed that there were no additional unconsolidated concrete areas (voids).

In accordance with OCA 20856, the applicant removed stainless steel liner plates from three areas (one area about 1 foot by 1 1/2 feet and two areas about 3 feet by 1 foot, excavated or chipped to sound concrete, and cleaned the concrete surface area. One and one-quarter inch (1 1/4) diameter probe holes and grout access holes were drilled in the liner plates to determine the extent of and to assure full definition of the void area. Air access holes were drilled in the stainless steel liner plates to assure that grouting would be accomplished in accordance with the procedure.

The procedure (DCA-20856) specifed that the grout was to be cured for 28 days or until the grout reached a compressive strength of 4000 psi. Repairs to the liner plates were specified in DCA-20856 and G&H Procedure 2323-SS-18.

DCA-20856 required that under no circumstances was cutting of the liner across weld seams, across embedded weld plates, or into leak chase seal welds or drilling through the liner at leak chase channels, embeds, or weld seams permitted. Documentation review indicated that DCA-20856 was adhered to and that no cutting or drilling occurred in prohibited locations.

No violations or deviations were identified.

6. Nondestructive Testing Observations of Liner Plates in Fuel Transfer Canal

The NRC inspector observed portions of non-Q liquid penetrant examinations (PT) being performed on liner plate welds following re-installation of the liner plates in the areas of the fuel transfer canal removed for inspection and repair of the concrete. The inspector performed the PT on the welds as required by the repair package and the procedure (QI-QP-11.18-1, "Liquid Penetrant Examination"). Scattered weld porosity was identified by the inspection. The porosity was ground out and a repeat PT was performed. The final inspection is scheduled to be performed by QC inspection personnel. The liner plate areas to be inspected by PT were identified in DCA 20856.

No violations or deviations were identified.

7. Cadweld Splice Observations and Records

a. Calibration of Tensile Tester

The NRC inspector observed the calibration of the Tinus-Olson Universal Testing Machine (Model Number 600-12 Identification Number M&TE-784) on April 2 and May 7, 1985. The machine was calibrated just prior to performing tensile testing of cadweld splices and subsequent to completion of tensile testing each day that tensile testing was performed. The machine calibration date for April 2, 1985, prior to start of tensile testing was observed by the NRC inspector and recorded as follows:

Nominal load (lbs)	Calibration Reading (lbs)	Error (lbs)	Error %	Remarks
0	0	0	0	0 machine on 4/2/85
100,000	99,750	+250	+0.25	
200,000	199,600	+400	+0.2	
300,00	299,450	+550	+0.18	
350,000	350,300	-300	-0.08	
400,000	401,200	-1200	-0.03	
500,000	501,350	-1350	-0.27	
600,000	602,450	-2450	-0.40	

The NRC inspector reviewed calibration data for March 4, March 8, April 2, April 3, April 30, and May 7, 1985. All calibration data met within the +/- 1% accuracy requirement specified by Calibration Procedure 35-1195-IEI-37, Revision 3, dated March 11, 1982. The reference standards were identified as follows:

ID No.	Manufacturer	Calibration Due Date
RS-75	BLH Electronics	January 27, 1987
RS-75.3	BLH Electronics	January 27, 1987

b. Observation of Cadweld Splice Tensile Testing

(1) Qualification Tensile Testing

On April 2, 1985, the NRC inspector observed the following tensile testing of cadweld splices for cadwelder qualification: EBD Q8, GBH Q1, GBH Q2, GBV Q1, BFD Q4, BFD Q3, BFH Q4, GAH Q1, GAV Q1, and GBV Q2.

Each of the above qualification cadweld splices was tensile tested to 400,000 pounds (100,000 psi) and met the requirements stated in the procedure.

(2) Production Tensile Testing

The NRC inspector observed the tensile tester calibrations and the following production cadweld splices tensile testing on May 7, 1985: FXD 3P, FYD 4P, FYD 8P, FRD 87P, and FUD 6P.

Each of the above production cadweld splices was tested to 400,000 pounds (100,000 psi)and met the requirements stated in the procedure.

(3) Installation of Production Cadweld Splices

The NRC inspector observed installation of rebar and cadweld splices at frequent intervals (five or more observations per week during the weeks of April 8 and 15; May 6, 13, 20, and 27; and June 3, 1985). The rebar installation for the Unit 2 closure was performed in the area identified as elevation 805 feet to elevation 875 feet and azimuth 300 degrees to 335 degrees. The installation activities observed included rebar spacing, location of cadwelds, observation of selection and removal for testing of cadweld splices for testing, and determination of location of rebars and cadwelds for the as-built drawings.

(4) Documentation Reviewed

The NRC inspector reviewed the following documentation for the rebar placement and cadwelding for the Unit 2 containment (reactor building) closure area:

Drawings	DCAs	NCRs
2323-S-0785, Rev.7	22616, Rev. 1	C85-200294
2323-S-0786, Rev.9	22728	C85-200339, Rev.1
2323-S1-500, Rev.5	22737	C85-200355, Rev.1
2323-S1-506, Rev.5	22836	
2323-S2-505, Rev.5	22878 (Sheets 1-7)	
2323-S2-508, Rev.2	22772	
2323-S2-506, Rev.3		

No violations or deviations were identified.

8. Concrete Batch Plant Inspection, Unit 1 and 2

The NRC inspector inspected the concrete production facilities for the following specific characteristics for the following areas: (1) material storage and handling of cement, aggregate, water and admixture, (2) batching equipment scales, weighing systems, admixture dispenser, and recorders, (3) central mixer (not applicable because it had been dismantled), (4) ticketing system, and (5) delivery system.

The current batching is a manual operation since almost all concrete has been placed. The central mixer was dismantled and removed from site two or three years ago when concrete placement was virtually completed. Presently, the backup batch plant (which was a backup system for the central mixer) is in operation to complete the remaining concrete placements. This batch plant is in good condition and complied with the subject checklist except for one area.

The NRC inspector inspected the inside of one of three trucks used for mixing concrete (that is, the batch plant dispenses the correct weight of materials as required by the specific design mix numbers and the truck then mixes the batch to be placed.) The blades inside the truck are subject to wear and should be checked at a reasonable frequency. The Brown & Root (B&R) representative responsible for checking the blades in accordance with B&R Procedure 35-1195-CCP-10, Revision 5, dated December 4, 1978, was asked for evidence that the blades had been checked for wear on a quarterly basis as required by procedures and it was found that there was no record of such checks dating back to 1977 when they were initially checked.

In the FSAR Volume V, Section 3.8.1.2.3, the applicant commits to ACI 304-73. In ACI 304, the maintenance of mixer blades is required.

Procedure CCP-10, paragraph 3.10 "Truck Mixing," is silent on blade wear but Section 3.11 infers that the blades should be checked for both central and truck mixing. The inspection of both central and truck mixing blades

was not documented, although the B&R representative stated that the mixing blades were periodically inspected and laboratory testing would have probably indicated if there was a problem with the mixing blades. Strength and uniformity tests have consistently been within the acceptable range indicating that concrete production was acceptable even though mixing blade inspection was not documented.

Otherwise, the condition of the inside of the truck was satisfactory as the drum and charging/discharging were clean. The water gage and drum counter were in good condition.

This failure to follow procedures is a violation of 10 CFR 50, Appendix B, Criterion V. Subsequent to the identification of this violation, the blades were checked for wear and blade wear was presently within allowable limits (445/8507-04; 446/8505-02).

No other violations or deviations were identified.

9. Calibration Laboratory for Batch Plant Unit 1 and 2

The NRC inspector obtained batch plant scale numbers from tags which indicated that the scales had been calibrated and were within the calibration frequency. Cement (MTE 779), Water (MTE 766), admixture scale (MTE 764), and aggregate (MTE 780) were reviewed. The scales had been periodically calibrated since the batch plant was activated. The records were adequate except as follows:

a. Scales MTE 766 records do not differentiate between the required accuracy of the scale and the digital readout.

b. Scales MTE 779 and 780 records show various accuracy ranges for the same scale; i.e., MTE 779 (SN749687) records the following: report dated January 1976 gives 1%; report dated July 1976 gives 1% while the report dated October 1976 gives +/- 0.2%.

The calibration appeared to be proper, however, the above items are unresolved pending further review of the applicant's actions regarding the correction of these records (445/8507-05; 446/8505-03).

c. Records for scales MTE 779 records contained B&R memo IM-1108 dated July 16, 1975, which described a nonconforming condition. This condition affected the water and cement scales causing a 24-48 pound deviation (7,000 pound scale) during the calibration test. The memo stated that the condition was corrected and the scales were then calibrated; however, no deficiency report was written as required by B&R Procedure CP-QCP-1.3, "Tool and Equipment Calibration and Tool Control" dated July 14, 1975, and CP-QAP-15.1, "Field Control of Nonconforming Items," dated July 14, 1975. As a result there is no evidence that

corrective action included an evaluation to determine if concrete production was adversely affected.

This failure to assure that a nonconforming condition was evaluated is a violation of Criterion XV of 10 CFR Part 50, Appendix B, (445/8507-06; 446/8505-04).

10. Concrete Laboratory Testing Units 1, and 2

TUGCO Procedure QI-QP-11.1-1, Revision 6, was compared with ASME Section III, Division 2, Subsections 5222, 5223 and 5224 to assure that each ASTM testing requirement was incorporated into the procedure.

The NRC inspector inspected the testing laboratory equipment and found the test area and equipment were in good condition and each piece of equipment was tagged with a calibration sticker which showed it to be within the required calibration frequency. Test personnel were knowledgeable of test requirements and equipment.

The NRC inspector witnessed field tests performed by laboratory personnel as follows:

Date	Truck No.	Mix No. Temp(°F)	Ticket No.	Air Content(%)	Slump (in.)	
6/3/85	RT-41	925	64013	Req 8.2-10.3	NA	70 max
				Mea 8.7-9.1	NA	57
6/3/85	RT-35	128	64014	Req 5.0-7.0	5 max	70 max
				Mea 6.6	6.25*	57

*Initial slump was high; however, after additional truck rotations the slump was found acceptable.

The following laboratory equipment was checked and found to be within calibration: Forney Compression Tester, MTE 3031; Temperature Recorder MTE 3013 and 3014; Unit Volume Scale, MTE 1053; Pressure Meters MTE 3000B, 3002 and 3004; Sieves MTE 1286, 1239, 1272, 1274, 1136A, 1156, 1094, 1093, 1095, 1178, 1179, 1300 and 1180; Aggregate scales, MET 1058 and 1067; and 2" grout mold MTE 1111.

The following test records for placement number 201-5805-034 were reviewed: (1) concrete placement inspection, (2) concrete placement summary and, (3) unit weight of fresh concrete.

No violations or deviations were identified.

11. Inspection of Level C and D Storage Unit 1 and 2

The NRC inspector inspected all laydown areas where piping, electrical conduit, cable, and structural reinforcing steel were stored. These materials were neatly stored outside on cribbing in well drained areas which allowed air circulation and avoided trapping water. This met the Level "D" storage requirements of ANSI N45.2.2.

The electrical warehouse contained miscellaneous electrical hardware. This building was required to be fire resistant, weathertight, and well ventilated in order to meet Level "C" storage requirements. This warehouse was well kept and met all requirements except for a lock storage area located upstairs at the rear of this building (electrical termination tool room). Two minor problems were identified and the warehouse personnel initiated action to correct them.

The first problem noted was that a box of nuclear grade cement was marked "shelf life out of date" but it had no hold tag. The box was subsequently tagged inaccordance with TUGCo nonconformance Procedure CP-QAP-16.1, Revision 24 (Nonconformance Report (NCR) E85-200453) after being identified by the NRC. During discussions with the warehouseman, the NRC determined that engineering told the warehouseman to mark the material and lock it up, but did not tell him to apply an NCR or hold tag. Also, the NRC inspector noted a very small leak in the roof above the electrical termination tool room. This leak was in an area that did not expose hardware to moisture. The roof is currently being repaired.

The millwright warehouse storage area was inspected; however, only a small number of items or materials were stored in this area. The overall storage conditions in this area met or exceeded Level "C" storage requirements.

No violations or deviations were identified.

12. Reactor Pressure Vessel and Internals Installation - Unit 2

This inspection was performed by an NRC inspector to verify final placement of the reactor pressure vessel (RPV) and internals by examining the completed installation and inspection records.

a. Requirements for Placement of RPV

Requirements for placement of the RPV to ensure proper fit-up of all other major NSSS equipment are in Westinghouse Nuclear Services Division (WNSD) "Procedure for Setting of Major NSSS Components", Revision 2, dated February 13, 1979, and "General Reactor Vessel Setting Procedure" Revision 2, dated August 30, 1974. The NRC

inspector reviewed the following drawings, which were referenced in the RPV operation traveler, to verify implementation of WNSD recommendations:

- o WNSD drawing 1210E59 "Standard - Loop Plant RV Support Hardware Details and Assembly"
- o WNSD drawing 1457F27 "Comanche Peak SES RCS Equipment Supports - Reactor Vessel Supports"
- o CE drawing 10773-171-004 "General Arrangement Elevation"
- o CE drawing 10773-171-005 "General Arrangement Plan"

Neither site prepared installation drawings nor specifications (which implemented the WNSD recommended procedures) were available and the drawings examined did not show certain specific installation criterion such as centering tolerances, levelness tolerances and clearance between support brackets and support shoes.

The inspector considers this matter unresolved. (446/8505-05)

b. Document Review

The NRC inspector reviewed B&R Construction and Operation Traveler No. ME79-248-5500 which described the field instructions for installation of the Unit 2 RPV. Requirements recommended by WNSD procedures were implemented in the traveler. Worksheets attached to the traveler showed the RPV to be centered and leveled within the established tolerances. Traveler operation 19 required verification of a 0.020 to 0.005 inch clearance between the support bracket and support shoe, after applying the shim plates. Change 5 subsequently changed the clearance to a 0.015 to .025 inch clearance. The installation data reflected in attachment 3B of the traveler indicated an as-built clearance of 0.012 to 0.026 inch which exceeds both the original and revised tolerances. This condition was accepted on the traveler based on Westinghouse concurrence, and there was no documented engineering evaluation onsite justifying the final tolerances. This matter is considered unresolved pending documentation validating the final installation tolerances. (446/8505-06)

The NRC inspector reviewed the following receiving records for the RPV hardware and found them to be in order:

- o Report No. 14322 for 54 each closure studs, closure nuts, and closure washers
- o Report No. 09507 for vessel S/N 11713, Closure Head 11713 and 26 O-Rings

o Deviation notices and corrective action statements

The NRC inspector reviewed the following completed travelers for internals installation and found them to be satisfactory:

o ME-84-4641-5500, "Assemble Upper Internals"

o ME-84-4503-4000, "Install and Adjust Roto Locks"

o ME-81-2145-5500, "Retorque UI Column Extension"

o RI-80-385-5500, "Transport and Install Lower Internals"

o ME-84-4617-5500, "Repair Lower Internals"

o ME-84-4640-5500, "Assemble Lower Internals"

c. Visual Inspection

At this time, visual inspection of the internals by the NRC inspector was not possible, and inspection was limited on the vessel placement to a walk-around beneath the vessel to inspect the azimuth markings and for construction debris between the vessel and cavity. No problems were identified in this area.

d. Records of QA Audits or Surveillances

The NRC inspector requested TUGCO QA audits or surveillances performed by TUGCO of the Unit 2 RPV installation. TUGCO did not make available any documentation of an audit or surveillance which evaluated specified placement criteria, placement procedures, hardware placement, or as-built records. This item is unresolved pending a more comprehensive review of these activities (446/8505-07).

No deviations were identified; however, two unresolved items were identified and are described in the above paragraphs. (11.a and d)

13. Reactor Vessel Misorientation

On February 20, 1979, the applicant reported to the NRC Resident Inspector that a design error had resulted in the reactor support structures being placed in the wrong position on the reactor support pedestal such that the reactor would be out of position by 45 degrees. Initially, Unit 2 was to be a mirror image of Unit 1, however, a design change was initiated to permit identical components for both units. The design change was implemented for the reactor vessel, but not for the pedestal support locations. The problem was not considered by the applicant to be

reportable under provisions of 10 CFR Part 50.55(e) since the error could not have gone undetected.

The deficiency was reported to the NRC Office of Inspection and Enforcement on February 22, 1979 and during a March 27, 1979 meeting in Bethesda, Maryland, the applicant presented the proposed redesign and rework procedures for relocating the pedestal supports. No unresolved safety concerns with the repair were identified at the meeting.

During this inspection the NRC inspector reviewed various documentation relative to the misorientation problem, including design changes and the construction traveler which implemented the repair.

The following documents were reviewed:

- o NRC Inspection Reports 50-446/79-03; 50-446/79-07; 50-446/79-13
- o TUSI Conference Memo, dated March 1, 1979, H. C. Schmidt to S. Burwell (NRC Licensing PM)
- o TUGCO letter TXX-2980, dated April 30, 1979, to W. C. Seidle
- o NRC letter to TUGCO dated May 29, 1979
- o DCA 3872, Revision 1, dated February 28, 1979, Subject: Rework of Structure for Placement of the RPV Support Shoes
- o DCA 4122, dated March 22, 1979, Subject: Replacement of Rebar for RPV Supports
- o Construction Traveler CE79-018-5505, dated March 14, 1979, Subject: Rework of Reactor No. 2 Cavity - New RPV Support Locations
- o Grout Replacement Cards No. 007, 008, 009, 010, 014, and 015, various dates, Subject: Replacement of Grout around Rebar for Repair of RPV Support Shoes
- o Various Inspection Reports for Grout Properties and Application for RPV Support Shoes

No violations or deviations were identified.

14. Reactor Coolant Pressure Boundary (RCPB) Systems

The inspection was performed to verify: the applicants system for preparing, reviewing, and maintaining records for the RCPB piping and components; that selected records reflected compliance with NRC requirements and SAR commitments for manufacture, test and installation of

items; and as-built hardware was adequately marked and traceable to records. The following items were randomly selected and inspected:

a. Pressurizer Safety Valve - This item was inspected to the commitment stated in FSAR, Table 5.2-1 which includes ASME Section III, 1971 Edition through Winter 1972 Addenda. Valve S/N N56964-00-007, which is installed in the B position, was inspected. The following records were reviewed:

   - QA Receiving Inspection Report No. 21211
   - Code Data Report Form NV-1
   - Valve Body Certified Material Test Reports (CMTRs)

   The valve was in place, however, installation had not been completed; therefore, the hardware installation inspection consisted of verifying that the item was traceable to the records.

b. CVCS Spool Piece 3Q1 - Requirements for this item are stated in ASME, Section III, 1974 Edition through Summer 1974 Addenda, which is the commitment from the FSAR, Table 5.2-1. The item was field fabricated from bulk piping and purchased elbows and installed in the CVCS with field welds number 1 and 6 (ref. BRP-CS-2-RB-076). The following records were reviewed:

   - B&R Code Data Report
   - Field Weld Data Card
   - NDE Reports
   - QA Receiving Reports for piping and elbows
   - CMTRs

   The installed spool piece was inspected for weld quality and to verify that marking and traceability requirements had been met. The item had been marked with the spool piece number (3Q1) and the B&R drawing number which provided traceability to the material certifications.

c. Loop 3 RC Cold Leg - Requirements for this item are stated in ASME, Section III, 1974 Edition through Summer 1974 Addenda, which is the commitment from the FSAR, Table 5.2-1. This piping subassembly consists of a 27.5 inch cast pipe with a 22 degree elbow on the reactor end, a 10 inch 45 degree nozzle, a 3 inch nozzle, and three 2

1/2 inch thermowell installation bosses. The following records were reviewed for the subassembly:

- o QA Receiving Inspection Report No. 12389
- o Westinghouse Quality Release (QRN 47523)
- o Code Data Report Form NPP-1
- o 27 1/2 inch line CMTR
- o 3 inch nozzle CMTR
- o Field Weld Data Cards
- o NDE Reports

(1) Sandusky Foundry and Machine Company test report for the cold leg pipe certifies that material meets requirements of ASME Section II, 1974 editions through winter 1975. Southwest Fabrication and Welding Company code data report NPP-1 Form certified that the cold leg subassembly met requirements of ASME Section III, 1974 edition through winter 1975.

(2) The NRC inspector reviewed the procedures and hydro test data applicable to Unit 1, since Unit 2 hydro had not been completed. Requirements for the tests were presented in Procedures CP-QAP-12.2, "Inspection Procedure and Acceptance Criteria for ASME Pressure Testing" and CP-QAP-12.1, "ASME Section III Installation, Verification, and N-5 Certification." Procedure CP-QAP-12.1 requires that a data package to be used in the test, be prepared with the test boundary and the additional following data shown:

- o Base metal defects in which filler material has been added, and the depth of the base metal defect exceeds 3/8 inch or 10% of the actual thickness, whichever is less.
- o Untested vendor performed piping circumferential welds.
- o Approximate location and material identification and description for permanent pressure boundary attachment with applicable support number referenced.
- o Weld history, which shall reflect weld removal and/or weld repair.

The completed hydro data package (PT-5501) for Unit 1, loop 3 cold leg was reviewed for compliance with the above requirements. Drawing No. BRP-RC-1-520-001 had been used to annotate the test boundary. A handwritten statement on the drawing indicated: "No major base metal repairs could be located" and "No hangers with weld attachments could be located." Welds performed by the pipe subassembly vendor, including the 22 degree circumferential weld and the penetration fittings had not been identified. The following items are unresolved pending further review to determine:

- If the statement "no major base metal repairs" was based on a visual inspection or on a review of vendor and site inspection and repair records.
- If the shop circumferential weld attaching the 22 degree elbow to the pipe assembly was inspected during the test.
- If welds for penetrations into pipe assembly were inspected since Procedure CP-QAP-12.1 does not require identification of such welds and they were not identified on the drawing.

The above issues will remain unresolved pending further evaluation by the applicant (445/8507-07; 446/8505-09).

d. Personnel Qualifications - Personnel who had performed selected tasks were identified during inspection of installation records. Training and experience records for the personnel were reviewed to verify that employee qualifications and maintenance of records were current and met requirements. Names or codes for five welders and two NDE examiners, who had performed tasks during installation of the items being inspected, were identified and their qualification records reviewed. There were no questions in this area of the inspection.

No violations or deviations were identified.

15. Special Plant Tours (Unit 1 and Unit 2)

On May 23, 1985, the NRC inspector conducted a tour of selected areas of Unit 1 and Unit 2. The group consisted of one NRC inspector, two NRC Technical Review Team (TRT) representatives, two allegers, and several TUEC representatives. The TUEC representatives tagged each area where a deficiency was alleged. With the alleger's consent, a tape recorder was also used to note locations and describe any alleged deficiencies. The allegers indicated that they had identified all deficiencies during the

tour and all other deficiencies that they had knowledge. The NRC TRT is analyzing this information and will decide what action, if any, should be taken.

During this tour the NRC inspector independently identified a questionable practice in that the top of the the pipe chase at the north end of room 88 in Unit 1, safeguards building had two large stickers which stated that areas on the wall were reserved for pipe hangers GHH-S1-1-SB-038-006 and R1(?)1-087-X11. These stickers were dated 1980. It was not evident whether hangers were missing or none were needed in these locations and the reserve tags were not removed. TUEC representatives were unable to answer the question immediately. This item is unresolved pending further review during a subsequent inspection. (445/8507-08).

No violations or deviations were identified.

16. Routine Plant Tours (Units 1 and 2)

At various times during the inspection period NRC inspectors conducted general tours of the reactor building, fuel building, safeguards building, electrical and control building, and the turbine building. During the tours, the NRC inspector observed housekeeping practices, preventive maintenance on installed equipment, ongoing construction work, and discussed various subjects with personnel engaged in work activities.

No violations or deviations were identified.

17. Review of Part 21 and 10 CFR 50.55(e) Construction Reports Status

The NRC inspector reviewed all reports issued to date to assure that NRC and TUEC status logs were complete and up to date. A total of 183 reports have been submitted to date. This inspection period one Part 21 report on Diesel Generator Oil Plugs and two 10 CFR 50.55(e) reports on the Equipment Hatch Cover and SA106 Piping (light wall) were submitted.

No violations or deviations were identified.

18. Exit Interviews

The NRC inspectors met with members of the TUEC staff (denoted in paragraph 1) on May 10 and June 10, 1985. The scope and findings of the inspection were discussed. The applicant acknowledged the findings.