

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

DOCKETED
USNRC

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In the Matter of)	
)	
TEXAS UTILITIES GENERATING)	Docket Nos. 50-445 and
COMPANY, <u>et al.</u>)	50-446
)	
(Comanche Peak Steam Electric)	(Application for
Station, Units 1 and 2))	Operating Licenses)

AFFIDAVIT OF SUSAN L. SPENCER
REGARDING DISPOSITION OF NRC
I&E REPORTS

I, Susan L. Spencer, being first duly sworn, do
depose and state: I am employed by Texas Utilities Generating
Company in the position of Quality Assurance Auditor. As
such, I am responsible for monitoring and assuring resolution
of issues raised by the NRC in Inspection & Enforcement
Reports ("I&E Reports") for the Comanche Peak Steam Electric
Station. Thus, I am familiar with the actions taken by the
Applicants in response to those Reports, and the ultimate
disposition made by the NRC. The purpose of this affidavit
is to demonstrate that all matters raised in NRC I&E Reports
(including unresolved items and Notices of Deviation and
Notices of Violation) cited by Intervenor CASE as pertinent
to Contention 5 have been resolved to the satisfaction of
the NRC Staff or Applicants have taken appropriate corrective
action subject to verification by the NRC Office of Inspection

and Enforcement. In addition, I will discuss four specific topics raised in I&E Reports in order to detail the measures taken to resolve the issues, and the status of acceptance by the NRC of those measures. A statement of my educational and professional qualifications is attached as Attachment 1.

I. NRC I&E REPORTS

The NRC retains a Resident Reactor Inspector ("RRI") on site at the Comanche Peak facility. This individual and other Region IV staff personnel perform, on a routine basis, inspections at the plant of the activities authorized by the Construction Permits for Comanche Peak. Inspections generally consist of a selected examination of procedures and representative records, interviews with licensee construction personnel and management and observation of ongoing activities and completed work at the plant.

Following each inspection, the inspector prepares a report which details the findings during the inspection. These findings may include instances which require further examination or information to determine acceptability, but which do not constitute existing variations from NRC requirements. These are identified in the report as "unresolved items." Where apparent variations from NRC requirements (including apparent variations from procedures or instructions established to assure compliance with NRC requirements) are discovered, the Inspector will recommend

and the NRC Office of Inspection and Enforcement will transmit to the licensee either a Notice of Violation (involving either a violation, infraction or deficiency) or a Notice of Deviation, depending upon the severity of the variation. These notices are generally transmitted to the Applicants before the I&E Reports are issued and thus are also referenced in the I&E Reports.

To resolve issues identified as "unresolved items," Applicants generally meet with the Inspector to present the results of the Applicants' investigations of the matter. Additional documentation is provided, as necessary. The RRI will review the information presented and recommend further action such as closing the item, seeking more information or issuing a Notice of Deviation or Violation. When an item is closed out, a notation to that effect generally is made in the first I&E Report issued after such action is taken.

If a Notice of Deviation or Violation is issued, the Applicants transmit a formal written reply within a specified period of time responding to each issue raised. The Applicants provide the information necessary to respond to the discrepancy as requested. The Applicants' response describes any corrective steps already taken and the results achieved, corrective steps to be taken to avoid further noncompliance and the date of full compliance. When the NRC is satisfied that appropriate action has been taken by the Applicants, the Notice is closed out in a future inspection by the NRC.

II. STATUS OF I&E REPORTS AT COMANCHE PEAK

Since 1973 the NRC has performed routine inspections of preparations for and actual construction activities at Comanche Peak, as documented by I&E Reports. During this period over 150 NRC I&E Reports have been issued. Over 8000 NRC inspector hours have been spent on site conducting the inspections and investigations which are the subject of those reports. Significant additional NRC resources have been spent off site on inspection activity as well. The results of these inspections and Applicants' response to each, when required, are a matter of public record. In every case, the findings of the NRC Inspectors were reviewed by Applicants and actions were taken to correct any matters raised therein.

All but two issues raised in I&E Reports which are cited by CASE as pertinent to Contention 5 1/ have been resolved to the satisfaction of the NRC Staff. Such resolution, in all

1/ See CASE's Answers to Applicants' First, Third and Fifth Sets of Interrogatories, dated September 3, 1980 (Supplemented December 1, 1980), March 16, 1982 and April 20, 1982, respectively; and CASE's Answers to the NRC Staff's First and Fourth Sets of Interrogatories, dated February 17, 1981 (Supplemented April 6, 1981) and March 15, 1982, respectively. See also ACORN's Offer Proof, served August 29, 1980 which cited I&E Reports ruled by the Board to come within the scope of Contention 5. Rulings on Objections, October 31, 1980. I&E Report 80-09, regarding excessive groundwater withdrawal rates has also not been formally closed-out but is not addressed here because the issue does not involve matters of Quality Assurance.

but one instance, has been by a formal close-out in a subsequent I&E Report. 2/ The unresolved issues involve a failure to follow certain procedures for the inspection of coatings, 3/ and a concrete pour on the Unit 1 dome. 4/ In addition to these two unresolved issues, two other issues have been raised by CASE. Those issues are honeycombing in the Unit 2 steam generator compartment, and placement of Unit 2 Reactor Vessel. The details and status of the resolution for each item are discussed below.

A. Honeycombing In The Unit 2
Steam Generator Compartment

In October 1979, routine Quality Control inspections identified and documented areas in the concrete placement of the Unit 2 steam generator compartment walls where exposed concrete contained honeycombed conditions. Following engineering review of the condition, repair work was authorized to begin in early December, 1979. During

2/ Applicants have completed work pursuant to a commitment for resolution of an unresolved item raised in I&E Report 80-20 (Attachment 2), involving the spacing of, and circuit breakers for, safety and non-safety cables in the AC Instrument Distribution Panels. See FSAR § 8.3.2.1., Paragraph 7.c. This commitment was accepted by the NRC Staff in SER Supplement No. 1, § 8.4.4, p. 8-1, although a formal close-out has not been made in a subsequent I&E Report.

3/ I&E Report 81-15, transmitted by cover letter dated November 6, 1981 from W.C. Seidle (NRC) to R.J. Gary (TUGCO). Attachment 3.

4/ I&E Report 79-11, transmitted by cover letter dated May 14, 1979, from W.C. Seidle (NRC) to R.J. Gary (TUGCO). Attachment 4.

repairs, Engineering and Senior Quality Control personnel recommended that the integrity of inaccessible portions of the placement be investigated further. It was determined on the basis of that recommendation that further investigation was required. On December 13, 1979, Applicants informed the NRC RRI of this matter and that a consultant would be hired to perform a sonic (microseismic) investigation of the walls involved.

In early January 1980, services of a consultant were retained for the purpose of microseismically evaluating the inaccessible portions of the subject placement. A series of measurements of the placement were taken over a period of four days from both sides of the placement. In addition, Engineering personnel conducted physical investigations to ascertain better the nature of certain anomalies identified by the microseismic investigation. Subsequently, further microseismic tests were then conducted based on data derived from the physical investigation.

Upon evaluation by Engineering personnel, based on all data available, it was concluded that the inaccessible portions of the placement, excluding the honeycombed portions already identified, met or exceeded design requirements or contained no hidden internal defects which would be detrimental to the safety or utility of the structure.

A subsequent examination by the NRC RRI of the repair work on the honeycombing was conducted in March, 1980. At

that time the RRI found that work was being accomplished in accordance with detailed instructions generated at the site and the recommendations set forth in "applicable portions of the U.S. Bureau of Reclamation 'Concrete Manual', a recognized authoritative publication on concrete work." 5/ The NRC Staff conducted extensive reviews in April and May, 1980 and concluded that no items of noncompliance or deviations existed. 6/ Further, TUGCO/TUSI Quality Assurance personnel verified that all repair work was conducted in accordance with appropriate specifications and procedures.

B. Mislocated Reactor Vessel
Support Structure

On February 20, 1979, Applicants reported to the NRC RRI that an error had been discovered in the design of the Unit 2 reactor vessel support structure. 7/ Applicants

5/ I&E Inspection Report 80-08, at p. 6, transmitted by cover letter of April 18, 1980, from W.C. Seidle (NRC) to R.J. Gary (TUGCO). Attachment 5.

6/ I&E Report 80-11, at p. , transmitted by cover letter of June 17, 1980, from W.C. Seidle (NRC) to R.J. Gary (TUGCO). Attachment 6.

7/ See I&E Report 79-03, transmitted by cover letter dated March 14, 1979 from W.C. Seidle (NRC) to R.J. Gary (TUGCO). Attachment 7.

reported that the reactor vessel support shoes, the ventilation duct work, and the surrounding reinforcing steel had been rotated forty-five degrees from correct positions through a design error.

The Unit 2 reactor vessel support misorientation resulted from a combination of several factors, including inadequate communication of evolving design criteria to designers and a breakdown in the vendor drawing review interface between Gibbs & Hill and Westinghouse. The evolving design included changes to the reactor vessel supports to reflect the revised design of the Unit 2 reactor vessel to be identical to the Unit 1 vessel, with associated changes to the primary coolant systems. As a result, the mounting pads for the Unit 2 vessel were misoriented by about 45 degrees.

As reported to the NRC, Applicants chose to place additional reinforcing steel in the reactor vessel support structure under each new location of the vessel pads to correct this situation. This corrective action required coring holes in existing concrete to embed new tiebolts and reinforcing steel. During the performance of this corrective action, the NRC RRI identified as an unresolved item the replacement of reinforcing steel which had been severed by the drilling of holes into existing concrete. 8/

8/ I&E Report 79-07, at pp. 3-4, transmitted by cover letter dated March 23, 1979, from W.C. Seidle (NRC) to R.J. Gary (TUGCO). Attachment 8.

This item was later resolved upon examination of the reinforcing steel being added to replace the reinforcement cut during drilling. 9/

On March 27, 1979, Applicants and NRC representatives held a meeting to discuss the repair procedures for relocating the vessel support pads. The meeting focused on differences between the original design of the reactor vessel support pedestal and the repaired pedestal. At that meeting, Applicants presented their position that the repair has no safety impact and that the repaired pedestal will not be structurally different from the Unit 1 design. The NRC concluded that "no unresolved safety concerns associated with the repair design for the Unit 2 pedestal were identified at the meeting." 10/

C. Placement of Concrete on Unit 1 Dome

As described in I&E Report 79-11, 11/ on March 30, 1979, the NRC Region IV office received a telephone call from an individual who identified himself as a former CPSES

9/ I&E Report 79-13, at p. 2, transmitted by cover letter dated June 18, 1979, from W.C. Seidle (NRC) to R.J. Gary (TUGCO). Attachment 9.

10/ NRC Meeting Summary, May 15, 1979, "Summary of March 27, 1979 Meeting on Repair of Reactor Vessel Support Pedestal for Comanche Peak, Unit 2." Attachment 10.

11/ I&E Report 79-11, at pp. 3-5, 6-10, transmitted by cover letter dated May 14, 1979, from W.C. Seidle (NRC) to R.J. Gary (TUGCO). Attachment 4.

employee. The individual alleged that during a concrete pour on the Unit 1 dome in January, 1979, a rain washed away part of newly poured concrete, and that the affected area was repaired with grout. The NRC RRI subsequently contacted the individual and learned that the concrete used for repair was known to contain gravel.

Subsequent investigation by the NRC RRI indicated that the incident apparently occurred on January 18, 1979. It was determined that on that day a concrete pour on the Unit 1 dome was begun under good weather conditions. The weather subsequently deteriorated to the point that heavy rain stopped work at about 7:30 p.m. The pour area was covered and the incoming shift was instructed to clean the area so the pour could resume the next day. The RRI subsequently identified time sheets of individuals which indicated they had been "placing concrete" during the later shift. On April 17, 1979, two senior Brown & Root construction management personnel asked the RRI whether they could check with their personnel regarding the allegations. They were informed that since the RRI had completed his on-site investigation they could do so. The next day, the Applicants reported to the RRI that they had identified the craft General Foreman who was involved. As a result of interviews with this individual it was determined that concrete placement had occurred that evening to replace concrete which had washed away in the rain. The General Foreman had himself

mixed the concrete (one-half yard) in accordance with design mix data for the dome concrete, although no QA personnel were present. Consequently, a Notice of Violation was issued, citing a breakdown in the QA program on the evening of January 18, 1979. 12/

The NRC Office of Inspection and Enforcement formally closed the Infraction cited in I&E Report 79-11 regarding the failure to implement the Quality Assurance program in I&E Report 79-24/23. 13/ As stated in that Report, Applicants notified the Region IV Office of Inspection & Enforcement that reviews by consultants and Gibbs & Hill had been completed and the in-place concrete was found satisfactory. In addition, Applicants informed the Region IV Office of Inspection and Enforcement that construction supervisory personnel have been informed that should a similar situation occur, no additional concrete shall be batched or placed without prior notification to senior construction management.

12/ See May 14, 1979 letter from W.C. Seidle to R.J. Gary transmitting I&E Report 79-11, Enclosure 1. Attachment 4.

13/ I&E Report 79-24/23, at p. 2, transmitted by cover letter dated November 27, 1979 from W.C. Seidle (NRC) to R.J. Gary (TUGCO). Attachment 11. The structural integrity of the concrete used in this pour was cited as an unresolved item in I&E Report 79-24/23. This item will be closed out following the Structural Integrity Tests to be performed pursuant to 10 C.F.R. Part 50, Appendix J, which test the leak-tight integrity of the entire primary reactor containments of both Units 1 and 2, as described in SER § 2.8.1 at p. 2-18.

D. Inspection of Coatings

In I&E Report 81-15, 14/ the NRC raised certain matters concerning documentation of coating applications for miscellaneous steel, cable tray supports or pipe supports inside Units 1 and 2 Containment Buildings. Also, the records reviewed by the RRI for the Unit 2 Containment Steel Liner revealed incomplete checklists without recorded visual inspections and Dry Film Thickness readings. In this regard, the seal coat had allegedly been applied over surfaces to which the records review pertained.

The discrepancies cited in I&E Report 81-15 have been identified as nonconforming conditions in accordance with established QA procedures. In response, reinspections of the subject coatings using both scratch and adhesion tests to evaluate the condition of the applied coatings are being conducted. A complete review of existing records and reinspection of affected areas is being conducted. Any discrepancies are being identified and corrected in accordance with approved procedures.

To prevent recurrence of this matter, Application (Construction) Procedures were revised and reissued to clearly indicate pot life at all temperatures within the

14/ I&E Report 81-15. Attachment 3.

applicable range for application of the coating. In addition, Inspection (Quality) Procedures/Instructions were revised to clarify applicable requirements and were reissued. Also, an identification system providing traceability of inspection documentation from blasting through installation and final coating for miscellaneous steel and supports was established.

Formal close-out of this item will occur upon verification by the NRC Office of Inspection and Enforcement of satisfactory completion of the actions described above.

III. CONCLUSION

As demonstrated above, all matters raised in NRC I&E Reports cited by Intervenor CASE as pertinent to Contention 5 have been resolved to the satisfaction of the NRC Staff, or Applicants have taken corrective action which is subject to verification by the NRC Office of Inspection and Enforcement prior to formal close-out of the item by the NRC.

Susan L. Spencer

Susan L. Spencer

Subscribed and sworn to before me this 7th day
of May 1982.

Glenda Benson

Notary Public

GLEND A BENSON, Notary Public
In and for Dallas County, Texas
My Commission Expires 2-17-85

SUSAN L. SPENCERSTATEMENT OF EDUCATIONAL
AND PROFESSIONAL QUALIFICATIONS

POSITION: Quality Assurance Auditor

FORMAL EDUCATION: Texas Tech University, 1974-75
Texas A&M University, 1976-78
BBA-Management

OTHER: Certified in auditing nuclear quality assurance programs by Stat-A-Matrix Institute - 1979.

Eighty hours of formal training in Radiographic Testing - Westinghouse Electric Corporation - 1979.

Forty hours of training in Ultrasonic Examination - Rockwell International - 1979.

EXPERIENCE:

1979 to Present Texas Utilities Generating Company, Dallas, Texas, Quality Assurance Auditor. Responsible for coordinating responses to Nuclear Regulatory Commission Inspection and Enforcement Reports. Perform audits as necessary to support QA Division activities.

PROFESSIONAL: American Society for Quality Control, Member



UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 1000
ARLINGTON, TEXAS 76012

October 21, 1980

RECEIVED

OCT 22 1980

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Received

OCT 22 1980

R. J. GARY

In Reply Refer To:

RIV

Docket No. 50-445/Rpt. 80-20
50-446/Rpt. 80-20

Texas Utilities Generating Company
ATTN: Mr. R. J. Gary, Executive Vice
President and General Manager
2001 Bryan Tower
Dallas, Texas 75201

Gentlemen:

This refers to the inspection conducted by our Resident Reactor Inspector, Mr. R. G. Taylor, during September 1980, of activities authorized by NRC Construction Permits No. CPPR-126 and 127 for the Comanche Peak facility, Units No. 1 and 2, and to the discussion of our findings with Mr. R. G. Tolson and other members of your staff at the conclusion of the inspection.

Areas examined during the inspection and our findings are discussed in the enclosed inspection report. Within these areas, the inspection consisted of selective examination of procedures and representative records, interviews with personnel, and observations by the inspector.

During the inspection, it was found that certain activities under your license appear to be in noncompliance with Appendix B to 10 CFR 50 of the NRC Regulation, "Quality Assurance Criteria for Nuclear Power Plants." The Notice of Violation for the item of noncompliance reported in the enclosed inspection report was forwarded to you by our letter dated September 24, 1980; therefore, this letter does not require further response regarding this matter.

During the inspection, it was found that certain of your activities appeared to deviate from commitments in the FSAR. This item and references to the specific commitments are identified in the enclosed Notice of Deviation. In your reply, please include your comments concerning this item, a description of any steps that have been or will be taken to correct it, a description of any steps that have been or will be taken to prevent recurrence, and the date all corrective actions or preventive measures were or will be completed.

Two new unresolved items are identified in paragraphs 4 and 7 of the enclosed report.

~~80-121-0621~~

Texas Utilities Generating Company

2

October 21, 1980

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and the enclosed inspection report will be placed in the NRC's Public Document Room. If the report contains any information that you believe to be proprietary, it is necessary that you submit a written application to this office, within 20 days of the date of this letter, requesting that such information be withheld from public disclosure. The application must include a full statement of the reasons why it is claimed that the information is proprietary. The application should be prepared so that any proprietary information identified is contained in an enclosure to the application, since the application without the enclosure will also be placed in the Public Document Room. If we do not hear from you in this regard within the specified period, the report will be placed in the Public Document Room.

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,



W. C. Seidle, Chief
Reactor Construction and
Engineering Support Branch

Enclosures:

1. Appendix B, Notice of Deviation
2. IE Inspection Report No. 50-445/80-20
50-446/80-20

cc: w/enclosures
Texas Utilities Generating Company
ATTN: Mr. H. C. Schmidt, Project Manager
2001 Bryan Tower
Dallas, Texas 75201

yc: D. W. Chapman ✓
D. G. Hampton

50-445/80-20
50-446/80-20

Appendix B

NOTICE OF DEVIATION

Based on the results of the NRC inspection conducted during September 1980, it appears that certain of your activities deviate from commitments made in your Final Safety Analysis Report (FSAR) as indicated below:

Incorrect Design of Pressurizer Spray Control Valve Piping

FSAR, Section 5.1, Figure 5.1-1, Note 2 states, "Spray pipe sloped to provide water seal between pressurizer and spray valves. Valves to be a minimum of 10 ft. below pressurizer top."

Contrary to the above:

The Resident Reactor Inspector observed that the partially installed pressurizer spray control valves were above the top of the pressurizer. Subsequent review of the engineered design as displayed on isometric drawings RC-1-RB-016, 017 and 018 established that valves 1-PVC-455B and C are placed 1 foot 4 inches and 2 feet 11 inches, respectively above the top of the pressurizer rather than at least 10 feet below.

This is a deviation.

~~84-112-14624~~

U. S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION IV

Report No. 50-445/80-20; 50-446/80-20

Docket No. 50-445; 50-446

Category A2

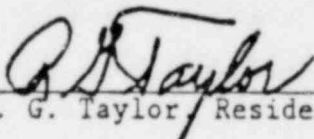
Licensee: Texas Utilities Generating Company
2001 Bryan Tower
Dallas, Texas 75201

Facility Name: Comanche Peak, Units 1 and 2

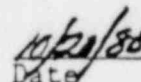
Inspection at: Comanche Peak Steam Electric Station, Glen Rose, Texas

Inspection conducted: September 1980


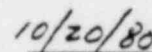
Inspector:



R. G. Taylor, Resident Reactor Inspector, Projects Section

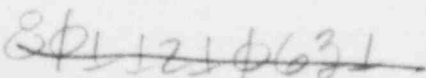

Date

Approved:


W. A. Crossman, Chief, Projects Section
DateInspection Summary:Inspection during September 1980 (Report No. 50-445/80-20; 50-446/80-20)

Areas Inspected: Routine, announced inspection by the Resident Reactor Inspector (RRI), with limited support by two Regional IE inspectors, including general site tours; piping, electrical and instrumentation, installation activities; protection of installed and uninstalled equipment and pipe supports and restraints. The inspection involved ninety-two inspector hours by the RRI and four hours by two Regional IE inspectors.

Results: No items of noncompliance were identified in five of the areas. One item of noncompliance and a deviation (infraction - unsuitable weld surface as required by magnetic particle test procedures - paragraph 6; deviation - incorrect design of pressurizer spray system control valve piping - paragraph 5) were identified in two other areas.



DETAILS1. Persons ContactedPrincipal Licensee Employees

- *R. B. Clements, TUGCO, Vice-President, Nuclear Operations
- *D. N. Chapman, TUGCO, Quality Assurance Manager
- *R. G. Tolson, TUGCO, Site Quality Assurance Supervisor

Other Persons

- J. V. Hawkins, Brown & Root, Project Quality Assurance Manager
- D. C. Frankum, Brown & Root, Construction Project Manager

The RRI also interviewed other licensee and Brown & Root employees during the inspection period including both craft labor and QA/QC personnel.

*Denotes those persons with whom the RRI held on-site management meetings during the inspection period.

2. Site Tours

The RRI toured the safety-related plant areas several times weekly during the inspection period to observe the general progress of construction and the practices involved. Four of the tours were accomplished during portions of the second shift where the main activity continues to be installation of electrical cables and the application of protective coatings.

No items of noncompliance or deviations were identified.

3. Protection of Major Installed Equipment

The RRI observed that the Reactor Vessel Internals (core support structure) were partially installed in the Unit 1 vessel as part of the final alignment machining operation. The internals and the vessel were well covered and protected from any construction debris. The Unit 2 Reactor Vessel continued to be protected during periods when personnel had to enter the vessel for coolant pipe welding and weld inspection.

The Unit 2 internals remain in their storage enclosures near a facility warehouse. The RRI observed that randomly selected electric prime movers for pumps have their space heaters energized as have the motors associated with motor operated valves. The safety-related switchgear and motor control centers either have their space heaters energized or have large light bulbs installed to keep them dry. The control room panels and boards were maintained in an air-conditioned environment during the period to preclude heat damage to the electronic components.

No items of noncompliance or deviations were identified.

4. Storage of Uninstalled Components

The RRI toured the warehouses and outdoor laydown areas during the inspection period. All components observed, including a substantial number of prefabricated pipe spools and cable reels, appeared to be well cared for in accordance with normal industry practice and the recommendations of the various suppliers of major components, with one exception. The RRI noted in touring an outdoor storage area what appeared to be spent fuel storage racks identified by an attached sign as "NON-Q" (not within the scope of the licensee's quality assurance program). Also the lay-down positions of some of the rack modules were such that water might collect in some areas in the event of rain.

The RRI reviewed the files of the on-site vendor coordination and found that the racks had been furnished by Westinghouse as Seismic I components as indicated by the FSAR Section 17.2 and that there was a Westinghouse quality release document on file. The RRI initiated a discussion with licensee QA personnel and was informed that they were aware of the problem and that a Nonconformance Report had been initiated but not yet finalized.

This situation will be considered to be an unresolved item pending completion of the Nonconformance Report and whatever actions are taken to rectify the matter.

5. Safety-Related Piping Installation and Welding

The RRI made several general observations of the handling practices for piping components during the inspection period both in the on-site fabrication shop and within the main plant buildings. The practices were found to be consistent with the construction practices outlined in Construction Procedure 35-1195-CPM 6.9 which in turn are consistent with the Project Specification MS-100 and good industry practice.

The RRI observed, during a portion of one of the tours, that the installation and welding of a portion of the pressurizer spray piping had evolved to a point where the final layout configuration of this subsystem of the reactor coolant system was evident. The RRI obtained the pertinent isometric drawings for the subsystem and verified that the piping was installed essentially as shown. The design layout, however, was found not to match the commitments of Section 5.1 of the FSAR in that Note 2 of Figure 5.1-1 had not been achieved. This note requires that pressurizer spray control valves be located a minimum of ten feet below the top of the pressurizer while the isometric drawings indicate that the two valves are actually located two feet eleven inches and one foot four inches, respectively above the top of the pressurizer.

This is considered to be a deviation to the FSAR commitments.

The RRI observed the following welds being made during the inspection period:

<u>Weld No.</u>	<u>Isometric</u>	<u>Filler Ht.</u>	<u>Welder(s)</u>	<u>Procedure</u>	<u>Process</u>
FW-4	AF-2-SB-003	87401	BPK	11021	GTAW
W-7-12	SW-1-SB-003	762550	BWP	88025	GTAW
FW-5	RC-2-520-001	434788	ARN-BRS-AFP	99028	GTAW (Machine)
FW-1	RH-1-RB-001	746100	AEN	88021	GTAW

The RRI verified that the weld filler metals being utilized in the above welds were certified by the suppliers via Certified Material Test Reports as meeting the applicable requirements of ASME Section II. The welders and weld procedures noted were verified to have been properly qualified in accordance with ASME Section IX.

The RRI also examined the radiographs of the following welds for compliance to the requirements of ASME Section III for weld quality and Section V for the quality of the radiographs themselves.

<u>Weld No.</u>	<u>Isometric</u>	<u>Line No.</u>
W-16	RC-2-RB-073	4-RC-2-091-2501R1
FW-16A	CS-1-RB-028	2-CS-1-112-1501R1
W-4	CS-2-RB-042	3-CS-2-235-2501R1
FW-14	SI-1-RB-017	6-SI-1-046-2501R1
FW-7A	CS-1-RB-028	2-CS-1-112-2501R1
FW-7	SI-1-RB-037	10-SI-1-179-2501R1
W-13	SI-2-RB-042	2-SI-2-036-2501R1
FW-11A	RC-1-RB-006	6-RC-1-070-2501R1
W-19	FW-2-SB-029	6-FW-2-097-1303

Except as noted above, no items of noncompliance or deviations were identified.

6. Piping System Supports and Restraints

The RRI initiated an inspection of a group of some one hundred piping system moment restraints that had been fabricated by Chicago Bridge & Iron Co. under Brown & Root (acting as a purchasing agent for the licensee) subcontract 35-1195-0585 and shipped to the site complete, except for sandblasting and painting.

The inspection was initiated as a result of an investigation of allegations received by Region IV as documented in Inspection Report 50-445/80-22; 50-446/80-20. The allegation essentially stated that some of the weld surfaces of the fabrications were such that a meaningful magnetic particle inspection within the requirements of ASME Section III, subsection NF could not have been done as certified to by the subcontractor.

A secondary consideration contained within the allegation, although not specifically stated, was that the Authorized Nuclear Inspector assigned to the facility was derelict in that the fabrication had been stamped and certified as having met the Code requirements. The RRI, with the assistance of two NRC inspectors specialized in weld inspections and nondestructive examination techniques, performed a magnetic particle and visual examination of selected portions of the welds of four of the fabrications using the Chicago Bridge & Iron procedures. Several of the selected areas examined revealed indications that could only be resolved by surface removal of the welds, thus substantiating the specific allegation. The secondary allegation referred to above was effectively refuted when the RRI found that the magnetic particle inspection was required by the Architect/Engineer as a supplemental addition beyond the requirements of the ASME Code. Subsection NA of the Code specifically excludes the Authorized Nuclear Inspector from taking responsibility for requirements imposed by a designer that are in excess of the Code requirements. The Code requirement was for a visual examination only of these components and each of the welds examined by the NRC personnel appeared to satisfy the Code stated acceptance criteria. The licensee was informed of the NRC findings informally on September 19, 1980, and formally by a Region IV letter dated September 24, 1980.

The licensee placed a QC hold on all of the components and has initiated a reinspection program involving an identified one hundred five units received under the order.

The RRI also became aware of another substantial group of large fabrications used as pipe whip restraints that were fabricated by Chicago Bridge & Iron under another subcontract and has initiated an inspection of these that is not complete.

7. Electrical Installation Activities

The RRI made a number of observations of electrical cable pulling operations during the period. The RRI observed the activities of each of the six cable pulling crews one or more times during the period in order to ascertain whether they were working within the parameters of the site installation procedures and good practices. The RRI also observed the activities of the QA/QC personnel assigned to monitor the pulling crews. The RRI found both groups (craft and QA/QC) to work consistently within their respective procedures, EEI-7 and QE-QP-11.3-26. The RRI also inspected randomly selected cable tray segments in Unit 1 Safeguards area for freedom from cable damaging burrs and excessive debris. The selected segments were found to be in a satisfactory condition for cable pulling.

The RRI examined the wiring within the 118 VAC Distribution Panels shown on FSAR Figure 8.3-15 and identified as IPC1 thru IPC4. It was found that from two to four nonsafety cables entered each panel

along with several safety-related cables. Within the panel, the safety and nonsafety cables were tightly tied together which appears to be contrary to commitments of the FSAR to maintain a six inch space between safety and nonsafety cables within panels. Reference to the above FSAR Figure revealed that safety and nonsafety cables were shown and that the nonsafety cable was to be landed on circuit breakers assembled among the safety-related breakers which would make the six inch spacing nearly impossible regardless of cable routing techniques. The same Figure also shows the nonsafety cabling being powered off a safety bus with no other means than a circuit breaker of an over-current type to serve as isolation. This situation appears to violate the recommendations of Regulatory Guide 1.75. It appears that the nonsafety cables should be designated as associated safety-related cables and given segregated routing through the cable raceway system.

Since the issue is clearly shown on the referenced FSAR Figure and since the electrical/instrument wiring scheme is still under active review by NRR, this matter will be considered to be an unresolved item rather than a deviation subject to either a FSAR revision or specific approval by NRR of the as-built design.

8. Instrument Installation Activities

The RRI observed during the inspection period that various instruments are being placed on their supports throughout the Unit 1 and Common Plant areas. The instruments observed were noted to be covered with wooden boxes wired in place for the purposes of physical protection. Where the instrument impulse tubing has not been connected to the instrument, the ports have been capped to protect from the entrance of debris, all considered to be good industry practices.

The RRI continued to examine the engineer's drawing in the electrical and instrument areas to establish whether all safety-related devices and circuits have been identified and to also develop the IE inspection plan. The RRI noted the Engineer's Instrument Tabulation List does not appear to address those safety-related devices associated with the heating and ventilation controls as being within the QA/QC scope even though they are connected with safety-related cabling to safety-related control circuits. The RRI discussed this matter with cognizant engineering personnel and was informed that the instrument list was under active review and would be corrected in a timely manner.

No items of noncompliance or deviations were identified.

9. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance, or deviations. Unresolved items disclosed during the inspection are discussed in paragraphs 4 and 7 and will, in future inspection reports, be referred to as:

- a. 50-445/80-20; 50-446/80-20 Spent Fuel Storage Racks
- b. 50-445/80-20; 50-446/80-20 Design of the AC Instrument Distribution Panels

10. Management Interviews

The RRI met with one or more of the persons identified in paragraph 1 on September 2, 3, 5, 12, 15, 19, and 24, 1980, to discuss inspection findings and the licensee's actions and positions.



UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 1000
ARLINGTON, TEXAS 76011

ATTACHMENT 3

November 6, 1981

In Reply Refer To:

RIV

Dockets: 50-445/Rpt. 81-15

50-446/Rpt. 81-15

Texas Utilities Generating Company
ATTN: Mr. R. J. Gary, Executive Vice
President and General Manager
2001 Bryan Tower
Dallas, Texas 75201

RECEIVED

NOV 9 1981

TUGBO QA
DALLAS

Gentlemen:

This refers to the inspection conducted by Mr. C. E. Johnson of our staff during the periods October 13-16 and 19-23, 1981, of activities authorized by NRC Construction Permits CPPR-126 and CPPR-127 for the Comanche Peak facility, Units 1 and 2, and to the discussion of our findings with Mr. R. G. Tolson and other members of your staff at the conclusion of the inspection.

Areas examined during the inspection and our findings are discussed in the enclosed inspection report. Within these areas, the inspection consisted of selective examination of procedures and representative records, interviews with personnel, and observations by the inspector.

During the inspection, it was found that certain activities under your license appear to be in violation of Appendix B to 10 CFR Part 50 of the NRC Regulation, "Quality Assurance Criteria for Nuclear Power Plants." The Notice of Violation for the violation reported in the enclosed inspection report was forwarded to you by our letter, dated October 23, 1981; therefore, this letter does not require further response regarding this matter.

One new unresolved item is discussed in paragraph 3.a of the enclosed report.

In accordance with 10 CFR 2.790 of the Commission's regulations, a copy of this letter and the enclosed inspection report will be placed in the NRC's Public Document Room. If this report contains any information that you believe to be exempt from disclosure under 10 CFR 9.5(a)(4), it is necessary that you (a) notify this office by telephone within 10 days from the date of this letter of your intention to file a request for withholding; and (b) submit within 25 days from the date of this letter a written application to this office to withhold such information. If your receipt of this letter has been delayed such that less than seven days are available for your review, please notify this office promptly so that a new due date may be established.

Received

NOV 9 1981

R. J. GARY

~~3111244747~~

Texas Utilities Generating
Company

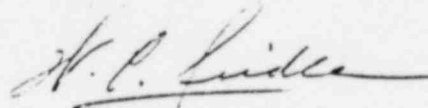
-2-

November 6, 1981

Consistent with Section 2.790(b)(1), any such application must be accompanied by an affidavit executed by the owner of the information which identifies the document or part sought to be withheld, and which contains a full statement of the reasons on the basis which it is claimed that the information should be withheld from public disclosure. This section further requires the statement to address with specificity the considerations listed in 10 CFR 2.790(b)(4). The information sought to be withheld shall be incorporated as far as possible into a separate part of the affidavit. If we do not hear from you in this regard within the specified periods noted above, the report will be placed in the Public Document Room.

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,



W. C. Seidle, Chief
Engineering Inspection Branch

Enclosure:
Appendix - IE Inspection Report 50-445/81-15
50-446/81-15

cc: w/enclosure
Texas Utilities Generating Company
ATTN: Mr. H. C. Schmidt, Project Manager
2001 Bryan Tower
Dallas, Texas 75201

cc: *BR Clements*
Ad Chapman
R Locke

APPENDIXU.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION IV

Report: 50-445/81-15; 50-446/81-15

Dockets: 50-445; 50-446

Category A2


Licensee: Texas Utilities Generating Company
2001 Bryan Tower
Dallas, Texas 75201

Facility Name: Comanche Peak, Units 1 and 2


Inspection at: Comanche Peak Site, Glen Rose, Texas

Inspection Conducted: October 13-16 and 19-23, 1981

Inspector:


C. E. Johnson, Reactor Inspector, Engineering and
Materials Section11/5/81
DateOther
Accompanying
Personnel:R. E. Hall, Acting Chief, Engineering and Materials Section
(October 23, 1981)

Approved:


R. E. Hall, Acting Chief, Engineering and
Materials Section11/5/81
DateInspection SummaryInspection Conducted October 13-16 and 19-23, 1981 (Report No. 50-445/81-15;
50-446/81-15)Areas Inspected: Routine, unannounced inspection of construction activities including a site tour and observation of work and review of records for safety-related pipe supports, protective coatings, and restraint systems in Units 1 and 2. The inspection involved sixty-five inspector-hours by one NRC inspector.Results: In the three areas inspected, no violations or deviations were identified in two areas. One violation was identified in the area of protective coatings (violation - failure to follow procedures for the inspection of coatings - paragraph 3.b).

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DETAILS1. Persons ContactedPrincipal Licensee Personnel

- *D. N. Chapman, Texas Utilities Generating Company (TUGCO), Quality Assurance Manager
- *R. G. Tolson, TUGCO, Site Quality Assurance Supervisor
- B. C. Scott, TUGCO, Quality Engineering

Other Personnel

- *J. R. Merritt, Texas Utilities Services, Inc., (TUSI), Engineering and Construction Manager
- J. V. Hawkins, Brown & Root (B&R), Project Quality Assurance Manager
- H. Williams, B&R, QC Superintendent, Non-ASME
- J. Ragan, B&R, Site QC Manager

The NRC inspector also contacted other licensee and Brown & Root employees during the inspection period including both craft labor and QA/QC personnel.

*Denotes presence at the exit interview conducted October 23, 1981.

2. Site Tour

The NRC inspector toured the site several times during the inspection to observe construction activities in progress and to inspect housekeeping. The areas toured included the Units 1 and 2 Reactor Containment Buildings, Safeguards Building, Auxiliary Building, paint shop, and paint warehouse.

There were no violations or deviations identified.

3. Protective Coatinga. Concrete Coatings Application

The NRC inspector reviewed six records of protective coating application on concrete in Reactor Containment Building No. 2. The NRC inspector identified some procedural inconsistencies including those listed below:

- (1) no inspector initials or signature
- (2) incorrect pot life specified or indicated
- (3) dates missing
- (4) induction time signed N/A when there should have been a minimum time listed

In four of the six records observed, the finish coat had been applied; however, checklists documenting the preceding applications had neither been completed nor finally accepted. The majority of the inspection documentation had not had final acceptance.

The NRC inspector informed the licensee representative of the discrepancies found. The licensee representative then informed the NRC inspector that they were aware of the problems. After a brief discussion, the licensee representative showed the NRC inspector an audit report performed during September 14-18, 1981, by TUGCO QA pertaining to this matter. The audit report was titled, "Protective Coatings," QA Audit File: TCP-24. This audit identified the same deficiencies as did the NRC inspector. The scope of the audit included receiving, storage and handling, certifications, measuring and testing equipment, application, and inspection documentation regarding protective coatings on concrete. This audit requested a response to the deficiencies and concerns identified by November 6, 1981.

This matter will be listed as an unresolved item until a satisfactory reply to the QA Audit, TCP-24 has been documented in a timely manner and reviewed by an NRC inspector on a subsequent inspection.

b. Steel Coatings Application

The NRC inspector also reviewed records on protective coating application on steel. During the review of these records, the NRC inspector noticed that the most current records for miscellaneous steel and supports dated from late September 1979 and back. The NRC inspector reviewed one package which contained several records of miscellaneous embedments and door frames in Reactor Building No. 2. This package contained documents that were not properly filled out or were incomplete. For example, dry film thickness (DFT) of primer after application was not recorded and initialed by Quality Control inspectors on the checklist prior to application of the seal coat as required. The NRC inspector also noticed incorrect pot life requirements on the protective coating material identification and mixing checklist. The licensee representative was informed of the deficiencies noted by the NRC inspector.

The NRC inspector inquired about records of inspection conducted since September 1979. The licensee representative informed the NRC inspector that from late September 1979, there had been no checklists maintained as required by the procedures for miscellaneous steel, cable tray supports, and pipe supports noted in paragraph 3.f below. The licensee representative presented a log book that Quality Control was maintaining on cable tray supports. The licensee representative also informed the NRC inspector that they have maintained a log book on miscellaneous steel, pipe hangers, and supports. After reviewing the log book, the NRC inspector determined that this did not appear to satisfy

the records requirements of the procedures. The NRC inspector also reviewed protective coatings records of Unit 2 Reactor Building steel liner plate, from September 1980 to October 1980. In reviewing these records, the NRC inspector noticed that documentation on the "Steel Substrate Primer Application Checklist" had not been completely filled out prior to seal coat applications as required by procedure.

This appears to be in violation of procedural requirements and 10 CFR 50, Appendix B, Criterion V.

c. Paint Shop

The NRC inspector toured the paint shop and reviewed coating records in process by a Quality Control inspector. All records reviewed at the paint shop appear to conform to Procedures QI-QP-11.4-1, QI-QP-11.4-2, and QI-QP-11.4-3 as listed in paragraph 3.f. The NRC inspector discussed procedural requirements with the Quality Control inspector in charge of the paint shop. The Quality Control inspector demonstrated an apparent level of knowledge sufficient to inspect protective coating on steel.

The NRC inspector also discussed procedural requirements and qualifications for applicators with a paint foreman. The paint foreman also appeared to have the level of knowledge required for protective coating applications.

During the tour, the NRC inspector observed some sandblasting taking place; however, there was no coating application going on at the time on safety-related items.

There were no violations or deviations identified.

d. Paint Warehouse Storage

The NRC inspector toured the "Q" paint warehouse and randomly inspected for opened containers and expired dates for shelf life of containers. There are two warehouses, one a non "Q" and the other a "Q" warehouse. The NRC inspector verified the readings of the temperature recorder which varied between 70 degrees and 75 degrees. The temperature was well within the specification. The temperature recorder calibration date was current. The NRC inspector reviewed records of the temperature charts and the surveillance checklist for storage and control of coatings. The temperature charts covered from January 1, 1981, to August 31, 1981, and the temperature was maintained within the specified range. The checklists were filled out and signed. The only discrepancy noted on the checklist was in the block "shelf life expired", which had been checked in the "yes" blocks. After inquiring into this matter, it was found to be in error since the shelf life had not been exceeded. This was an isolated case which was immediately corrected.

There were no violations or deviations identified.

e. Qualification Records (Coatings)

The NRC inspector reviewed qualification records for six Quality Control inspectors and ten painters. The Quality Control inspectors and painters selected for review were indicated on the records of six safety-related items which had been painted and inspected. The NRC inspector traced the qualifications of each Quality Control inspector and painter on the records during that time period to verify each individual's qualifications. All Quality Control inspectors and painters were qualified during the period of applications.

There were no violations or deviations identified.

f. Review of Coating Procedures

The NRC inspector reviewed the coating procedures listed below. All procedures reviewed appeared to include sufficient instructions and appropriate quantitative and qualitative acceptance criteria for determining that important activities would have been satisfactorily accomplished.

Procedures reviewed:

Steel

CP-QP-11.4, Revision 4, "Inspection of Protective Coatings"

QI-QP-11.4-1, Revision 3, "Inspection of Steel Substrate Surface Preparation"

QI-QP-11.4-2, Revision 4, "Inspection of Steel Substrate Protective Coating Mixing Operations"

QI-QP-11.4-3, Revision 5, "Inspection of Steel Substrate Prime Coat Applications"

QI-QP-11.4-5, Revision 4, "Inspection of Steel Substrate Finish Coat Application"

QI-QP-11.4-17, Revision 1, "Storage and Handling of Protective Coatings"

Concrete

QI-QP-11.4-10, Revision 1, "Inspection of Concrete Substrate Surface Preparation"

QI-QP-11.4-12, Revision 1, "Inspection of Concrete Substrate Surface Application"

QI-QP-11.4-14, Revision 1, "Inspection of Concrete Substrate
Finish Coat Application"

There were no violations or deviations identified.

4. Safety-Related Pipe Support and Restraints

a. Moment Limiting Components

The NRC inspector toured Unit 1 Reactor Containment Building and randomly selected three moment restraints listed below for inspection:

- (1) S-1-0538-06 - Safety Injection System (incomplete)
- (2) S-1-0538-02 - Safety Injection System (complete)
- (3) S-1-0538-08 - Chemical and Volume Control System (complete)

The NRC inspector visually inspected the completed moment restraints in accordance with referenced design drawings. Welds, attachments, and location all appeared to be in compliance with design drawings.

The NRC inspector reviewed the traveler package contents on work being performed in Unit 1 Reactor Containment Building, on the Safety Injection vertical line restraints in each of the four steam generator compartments. All documentation appeared to be in the traveler package as required. All changes and modifications had been approved.

The moment restraints reviewed were all fabricated by an off-site vendor. The constructor erected, aligned, and welded as required by the design drawings.

The records of the two completed restraints indicated that verification of elevation, location, plumb, and levelness had been verified by Quality Control. All welds were in accordance with design drawings and verified by Quality Control. Completion of all operations were verified, signed, and initialed by Quality Control. The NRC inspector verified welder qualifications from the completed travelers and also for work in process in the field. All welders were found to be qualified by procedure for the work being performed.

b. Pipe Supports

The NRC inspector reviewed the records of five pipe supports. The records indicated the type and classification of the supports. The records confirmed that the specifications and installation procedures had been met. The required scope of QA/QC inspections was met. Weld identification and location corresponded to respective weld card, drawing, and work order. Welders were qualified for the welding procedures used.

Records reviewed were for the pipe supports listed below:

CT-1-029-018-C82S - (Containment Spray System)

CT-1-014-421-C62R - (Containment Spray System)

CT-1-031-007-C92R - (Containment Spray System)

CS-1-074-047-S42R - (Chemical and Volume Control System)

CS-1-074-041-S42R - (Chemical and Volume Control System)

There were no violations or deviations identified.

5. Unresolved Items -----

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, violations, or deviations. An unresolved item disclosed during this inspection is discussed in paragraph 3.a.

6. Exit Interview

The NRC inspector met with the licensee representatives (denoted in paragraph 1) and R. G. Taylor, NRC Resident Reactor Inspector, on October 23, 1981, and summarized the purpose, scope, and findings of the inspection.



UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 1000
ARLINGTON, TEXAS 76011

ATTACHMENT 4

RECEIVED

MAY 16 1979

TUSI
NUCLEAR DIV.

May 14, 1979

In Reply Refer To:

RIV

Docket No. 50-445/Rpt. 79-11

50-446/Rpt. 79-11

Texas Utilities Generating Company
ATTN: Mr. R. J. Gary, Executive Vice
President and General Manager
2001 Bryan Tower
Dallas, Texas 75201

Gentlemen:

This refers to the investigation conducted by Mr. R. G. Taylor and other members of our staff on April 2-3 and April 13-23, 1979, of activities authorized by NRC Construction Permits No. CPPR-126 and 127 for the Comanche Peak facility, Units No. 1 and 2, concerning allegations by a former Comanche Peak employee.

The investigation and our findings are discussed in the enclosed investigation report.

During the investigation, it was found that certain activities under your license appear to be in noncompliance with Appendix B to 10 CFR 50 of the NRC Regulations, "Quality Assurance Criteria for Nuclear Power Plants." The item of noncompliance and references to the pertinent requirements are identified in the enclosed Notice of Violation.

This notice is sent to you pursuant to the provisions of Section 2.201 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations. Section 2.201 requires you to submit to this office, within 30 days of your receipt of this notice, a written statement or explanation in reply including: (1) corrective steps which have been taken by you, and the results achieved; (2) corrective steps which will be taken to avoid further noncompliance; and (3) the date when full compliance will be achieved.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and the enclosed investigation report will be placed in the NRC's Public Document Room. If the report contains any information that you believe to be

79-473-242

Texas Utilities Generating
Company

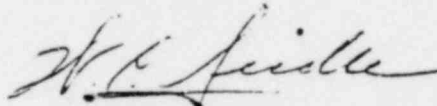
-2-

May 14, 1979.

proprietary, it is necessary that you submit a written application to this office, within 20 days of the date of this letter, requesting that such information be withheld from public disclosure. The application must include a full statement of the reasons why it is claimed that the information is proprietary. The application should be prepared so that any proprietary information identified is contained in an enclosure to the application, since the application without the enclosure will also be placed in the Public Document Room. If we do not hear from you in this regard within the specified period, the report will be placed in the Public Document Room.

Should you have any questions concerning this investigation, we will be pleased to discuss them with you.

Sincerely,



W. C. Seidle, Chief
Reactor Construction and
Engineering Support Branch

Enclosures:

1. Appendix A, Notice of Violation
2. IE Investigation Report No. 50-445/79-11
50-446/79-11

cc: w/enclosures
Texas Utilities Generating Company
ATTN: Mr. H. C. Schmidt, Project Manager
2001 Bryan Tower
Dallas, Texas 75201

50-445/79-11

50-446/79-11

Appendix A

NOTICE OF VIOLATION

Based on the results of the NRC investigation conducted during the periods April 2-3 and April 13-23, 1979, it appears that certain of your activities were not conducted in full compliance with the conditions of your NRC Construction Permit No. CPPR-126 as indicated below:

Failure to Implement the Quality Assurance Program For Civil Construction

10 CFR 50, Appendix B, Criterion II requires that a quality assurance program be established and implemented for the construction of the structures important to safety of the nuclear plant. The Texas Utilities Generating Company Comanche Peak Steam Electric Station Quality Assurance Plan affirms the intention to fulfill this requirement. The CPSES "Civil Inspection Manual" provides a body of inspection and testing procedures required to implement the Quality Assurance Plan.

Contrary to the above:

On January 18, 1979, personnel of the civil construction labor force placed an undetermined amount of concrete of an unknown quality on the dome of the Unit 1 containment without the knowledge of your Quality Assurance organization and without benefit of required inspections and testing of the concrete.

This is an infraction.

7907300250

U. S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION IV

Report No. 50-445/79-11; 50-446/79-11

Docket No. 50-445; 50-446

Category A2

Licensee: Texas Utilities Generating Company
2001 Bryan Tower
Dallas, Texas 75201

Facility Name: Comanche Peak, Units 1 & 2

Investigation at: Comanche Peak Steam Electric Station, Glen Rose, Texas

Investigation Conducted: April 2-3 and April 13-23, 1979

Inspectors:

W. A. Crossman
for R. G. Taylor, Resident Reactor Inspector, Projects
Section5/10/79
Date*R. E. Hall*
for D. P. Tomlinson, Reactor Inspector, Engineering
Support Section (April 13, 1979, Interview)5/10/79
Date*R. E. Hall*
for A. B. Beach, Reactor Inspector, Engineering
Support Section (April 23, 1979, Interview)5/10/79
Date

Approved:

W. A. Crossman
W. A. Crossman, Chief, Projects Section5/10/79
Date*R. E. Hall*
R. E. Hall, Chief, Engineering Support Section5/10/79
Date~~7907300256~~

Investigation Summary:

Investigation on April 2-3 and April 13-23, 1979 (Report 50-445/79-11;
50-446/79-11)

Areas Investigated: Special investigation of allegations received indicating that concrete had been placed on the Unit 1 dome during a rainstorm in January 1979, without QC or documentation; that pipe with sandblasted-off markings was being used in Unit 1; that steam system pipe was damaged by a handling accident and covered up; and that welders were not being properly qualified. The investigation involved thirty-six inspector-hours by the Resident Reactor Inspector and three inspector-hours by two Region IV based inspectors.

Results: The allegation relative to the concrete placement was confirmed (noncompliance - failure to implement the QA program - infraction). No items of noncompliance or deviations were identified relative to the balance of the allegations.

INTRODUCTION

Comanche Peak Steam Electric Station (CPSES), Units 1 and 2, are under construction in Somerville County, Texas, near the town of Glen Rose, Texas. Texas Utilities Generating Company is the Construction Permit holder with Brown and Root, Inc., as the constructor and Gibbs and Hill, Inc., as the Architect/Engineer.

REASON FOR INVESTIGATION

The Region IV Reactor Construction and Engineering Support Branch office received a telephone call from a former CPSES employee who reported several allegations indicating a potential breakdown in the CPSES Quality Assurance program.

1

SUMMARY OF FACTS

On March 30, 1979, the Region IV Reactor Construction and Engineering Support Branch received a telephone call from a party who identified himself as a former CPSES employee. The call was taken by an on-duty Reactor Inspector in the Projects Section who in turn provided the information to the assigned Resident Reactor Inspector at CPSES on April 2, 1979. The allegations, as received on March 30, 1979, were:

1. During a concrete pour on the Unit 1 containment dome in January 1979, a rain occurred which washed away part of the concrete. The affected area was repaired by the use of grout. Workers involved were requested to "keep it quiet." Two workers, who are still at the site, have knowledge of this occurrence.
2. The identity of a lot of "Q" and "non-Q" pipe (6" or less) being used for Unit 1 has been lost due to obliteration of heat numbers by sandblasting and loss of identifying tags. Workers are guessing as to the proper identification of the pipe.
3. A steam pipe intended for the Unit 1 turbine fell off of a truck and struck a railroad track. It was taken back to a storage area and hidden.
4. Third class helpers are being qualified in less than three months and are being used for safety related welding on Unit 1.

On April 13, 1979, the Resident Reactor Inspector assigned to CPSES and accompanied by another Region IV inspector interviewed the alleged in an effort to obtain additional information on the allegations. The additional information is summarized as follows:

1. The concrete used for the repair was not grout as originally indicated but was known to contain gravel. The concrete came from the batch plant where it was mixed on the ground and carried in a bucket to a tower crane at the Unit 1 Containment Building and hoisted to the dome area. The work was accomplished sometime during the middle of the second shift, possibly around 10:00 to 10:30 p.m. (January 1979, no day specified).
2. The pipe in question was not prefabricated pipe but rather bulk pipe joints. Sometimes, the pipe is sandblasted on the outside (rate of occurrence not identified) which removes all of the heat marking used for traceability.
3. The steam pipe was being moved during the second shift from the "Dodd's Spur" storage area to the plant area when it was dropped off the truck. A couple of the large "cherry-picker" type cranes were dispatched to the incident to pick up the pipe and place it back on the truck. The crew with the truck decided instead to put the pipe back into the storage area and leave it there for another shift to pick up and perhaps be blamed for damaging the pipe. The alleged did not know if the pipe had actually suffered any damage. He was aware the pipe in question was "non-Q" but expressed a concern that if the craft could get away with a cover-up on "non-Q," they probably are also doing it on the "Q" pipe and other equipment.
4. The alleged indicated he was concerned with what must be incompetent welders working on "Q" welds, since they could not have very much experience and still only be considered third class labor.

CONCLUSIONS

Research of various records and interviews with both craft labor and Brown & Root QC personnel produced the following conclusions:

1. The allegation relative to the concrete placement on the dome of Unit 1 is essentially correct and is evidence of a breakdown in the licensee's Quality Assurance program. The incident will be considered an item of noncompliance.

2. The allegation relating to the loss of pipe traceability markings could not be confirmed. The Resident Reactor Inspector's finding was that on occasion the sandblasting, with attendant loss of readily visible markings, probably does occur through human error, but that there are other means which will re-establish the identity of the pipe without guessing on the part of the craft labor force.
3. The piping in the "Dodd's Spur" storage area is for the turbine portion of the plant and is not safety related from a nuclear standpoint and is therefore not within the jurisdiction of the NRC inspection program. The more generalized concern of cover-up of improper handling practices is not consistent with the observations of the Resident Reactor Inspector and other NRC inspectors made during the course of routine inspections. The allegation cannot be verified or refuted at this time, but should subsequent observations verify that the alleged situation is occurring, appropriate action will be taken.
4. Welders are qualified in accordance with the provisions of the ASME Boiler and Pressure Vessel Code, Section IX, "Welding and Brazing Qualifications," as required by NRC regulations and the licensee's commitments as contained in the Safety Analysis Report submitted to obtain a Construction Permit. The labor classification, and therefore the pay, of the welders is not an element of the ASME Code welder qualification program, only the ability of the person being tested to weld on a specified weld coupon.

DETAILS1. Persons ContactedNon-Licensee or Contractor Persons

The alleged is a former employee of Brown & Root (the site general contractor). The person identified himself as a former equipment operator and foreman of equipment operators.

Principal Licensee Employees

Construction Manager, Texas Utilities Generating Co.
Supervisor of Product Assurance, Texas Utilities Generating Co./
Gibbs & Hill

Brown & Root, Inc.

Project General Manager
Construction Project Manager
General Foreman, Building Department
Superintendent, Building Department
Quality Control Inspector, Civil

2. Preliminary Investigation - April 2-3, 1979

- a. Allegation 1: The Resident Reactor Inspector (RRI) initiated a preliminary investigation of the allegation as soon as received. The RRI was aware that a number of concrete placements had been necessary to complete the dome area of Unit 1 and that a substantial portion of these placements occurred in January 1979. Schedule completion data indicated that five of the total of thirteen dome placements occurred in January 1979. Rainfall data for January was then obtained from the licensee's meteorology unit which indicated rain had fallen on January 15, 1979 (with the rainfall totalizer reset to zero) and again in the period between January 15 and 22, 1979, when the totalizer was again zeroed. The data suggested that placement 101-8805-013, the final placement on the dome, was the most likely candidate since 2.72 inches of rain had occurred about the placement date of January 18, 1979. The RRI then examined the QC inspection records for the placement which stated, "Pour stopped at 8:00 p.m. 1/18/79 due to inclement weather. Pour was topped out all but to a 30' radius which was cleaned up and finished 1/19/79."

The RRI then interviewed the QC inspector of record for the placement and was informed that the placement had started under good weather conditions on January 18, but that the

weather subsequently developed into a light mist and drizzle which did not interfere with the placement. By late evening, the weather deteriorated further and became a full rainstorm with thunder and lightning. By 7:30 p.m. or so it was decided that the placement would have to be stopped for reasons of personnel safety. The placement area was covered to keep the rain off the fresh concrete and the second shift was instructed to water blast and clean up the area so the placement could be resumed the following day.

- b. Allegations 2, 3 & 4: No attempt was made to perform a preliminary investigation of these allegations since the information was too vague.

3. Licensee/Contractor Report of Allegations

During the course of the above preliminary investigation, personnel of the licensee's management and QA organizations approached the RRI and stated that they too had received an allegation relative to the dome placement. It was stated that licensee management had received a telephone call on or about March 19, 1979, on the subject and that licensee management had visited the alleged at his home on March 20, 1979, to ascertain the facts of the allegation. The alleged then was invited to visit the site and discuss the allegation, which the alleged is reported to have done on March 26, 1979. On the basis of these interviews, the licensee's Product Assurance personnel undertook an investigation which concluded that the allegation had no merit.

4. Interview with Allegor by NRC Personnel

The Region IV office made several attempts to establish contact with the alleged during the period following March 30, 1979, when the allegation was received, through April 12, 1979, when the interview date and location were established. The RRI and another NRC inspector met with the alleged and a friend on April 13, 1979.

The alleged provided the following information about himself:

- a. He had been employed by Brown & Root at CPSES for 2-1/2 to 3 years and had quit in mid-March because he was dissatisfied with how the night shift equipment operators were being dispatched and supervised.
- b. He had been an equipment operator, primarily on cherry-pickers, and also a foreman for equipment operators at an earlier time.

- c. He stated that he had made the allegations to licensee management and Brown & Root management earlier but had not been at all satisfied with the answers he had received to his allegations.

The allegor provided the following additional information relative to each of the allegations:

Allegation 1: The incident occurred well after the time that the placement had been stopped. He could not be sure of the time but thought it was probably 10:00 to 10:30 p.m. when some equipment was dispatched to the concrete batch plant to bring down a bucket of concrete to Unit 1 and thought it strange. The concrete was taken to the dome by a tower crane. He was sure that the concrete was not batched by the batch plant and certainly was not delivered by the usual concrete mix truck.

Allegation 2: The allegor made it clear that he was not referring to completed pipe spools but rather to bulk pipe. The cherry-picker operators routinely move the pipe from one location to another on the site and that the pipe involved was bulk pipe or joints. He stated that the pipe was sometimes sandblasted in such way as to obliterate the heat number markings or tags and that he was pretty sure that there was a lot of unidentified pipe in the safety systems in Unit 1. This sandblasting sometimes happened to various steel forms used to make supports.

Allegation 3: The allegor described being dispatched with his equipment out to "Dodd's Spur" to pick up a length of pipe that had fallen off a truck after being loaded. The pipe had fallen on the spur railroad track. The RRI was not familiar with the term "Dodd's Spur." The allegor stated it was the area where the turbine components are stored. When he (the allegor) arrived at the site of the incident, he was told not to reload the pipe on the truck but to take it back into the storage area and put it down. The pipe crew indicated to him that they hoped that a day shift crew would come for the pipe and would probably be blamed for any damage that might have occurred to pipe when it fell. He stated that he did not know if the pipe had been damaged. He stated that he knew it was "non-Q" pipe but thought the NRC should be aware that such things were going on at the site.

5. Final Investigation - April 16-23, 1979

- a. Allegation 1. The RRI obtained the craft labor time sheets for both shifts for January 18 and 19, 1979. Review of the time sheets for the day shift on January 18 indicated that a portion of that shift worked on placement 101-8805-013. The records indicated that the day shift was terminated at approximately

8:30 p.m. relative to the placement as were the personnel at the concrete batch plant. The batch plant has no second shift operators. The RRI found that a large number of people, well in excess of fifty, had then worked on the placement during a substantial portion of the second shift. One crew of twelve people was shown by the time sheets to have been placing concrete, a notation not consistent with the fact that the batch plant was closed during the shift. The RRI then utilized the time sheets to develop a list of persons to be interviewed in connection with the incident with special concentration on the persons listed on the time sheet indicating "placing concrete 101-8805-013." The B&R personnel office records indicated that eight of the ten names included in this specific crew had been terminated at various times since January 18; the records did not suggest that any action was being taken to get rid of possible confirmatory personnel.

Late on April 17, 1979, two of the senior B&R construction management personnel very informally asked the RRI how the investigation of the allegations was coming along. The RRI responded that the on-site phase appeared to be complete and that NRC personnel would undertake the effort to locate and interview selected personnel immediately since it appeared that the allegation might be well founded. They asked the RRI if they could check with their people down to the General Foreman level as to the incident the night of January 18. The RRI indicated that such an inquiry on their part would probably not interfere with any future investigative action by the NRC.

On April 18, 1979, the licensee's Product Assurance Supervisor informed the RRI that he had information which indicated that the incident had occurred and that the craft General Foreman was the person responsible.

On April 23, 1979, the RRI, accompanied by another NRC Inspector, interviewed the General Foreman and his immediate supervisor, the night shift B&R Building Department Superintendent. These men related that on the night of January 18 the weather seemed to worsen and got to the point where the rain was so heavy that the people could hardly see. The freshly placed concrete developed into a problem when the plastic cover could not take the rainfall water load. Some of concrete began to sag back down the dome slope and one small area actually washed out and fell to the ground below. These men related that they and their entire crew of up to about one hundred-fifty worked on into the night trying to save a very bad situation. The sagged concrete was worked back into position and the crew protected it in any way they could to allow it to take a set.

The General Foreman went to the batch plant, got it open and operated the plant himself to make enough material to patch the washed out area. He stated that he found the design mix data used for the concrete on the dome and calculated the necessary weight of ingredients to prepare a half a cubic yard of concrete. The required data was put into the control system for the back-up dry batch plant, dropped into a skiff, and carried over to the quarter yard concrete mixer at the site test laboratory. It was mixed in two batches and placed into a skiff and carried to the dome where most of the half yard was used as a patch in the washed out area.

Both the General Foreman and his Superintendent were aware that there were no Quality Control personnel around to observe any of these actions since they had all gone home when the weather got really bad. Both men related to the RRI a picture of almost panic proportions in which the presence or absence of Quality Control simply did not matter; they were going to save a concrete placement from what they considered a disastrous situation, regardless. They indicated that while the night shift Assistant Construction Project Manager was generally aware of the situation on the dome that night, he probably was unaware of the fact that Quality Control personnel were not there or of the batching of the concrete under the conditions indicated.

In response to a question from the General Foreman as to "what happens now" the RRI stated that the NRC had no choice but to issue a Notice of Violation to the licensee since it had become very clear that the licensee's Quality Assurance program had broken down for the entire evening of January 18, 1979, and that a substantial amount of concrete on the dome was of an unknown quality.

- b. Allegation 2. The RRI visited the paint shop sandblasting area during the course of the final investigation to ascertain if this allegation could reasonably happen. The RRI interviewed a foreman of painters who is also in charge of the sandblasting activity and was told that three main categories of piping material routinely are sandblasted. These are:

- (1) Completed carbon steel spool pieces which are blasted on the outside prior to painting. The identity of these pieces is on an attached stainless steel band on which the identifying is encoded by stamping. Should the band come off, the spool piece identity can be re-established by the pipe fabrication shop since each spool is unique and is fully described by isometric drawings.

- (2) Carbon steel cut lengths, but otherwise in an unfabricated condition, are sent to sandblasting to have the inside cleaned prior to further fabrication. The outside, which usually carries the heat marking in paint is supposed to be untouched.
- (3) Bulk carbon steel pipe materials used for making equipment stands and supports is blasted and painted prior to fabrication. The material is used for such items as instrument supports.

The RRI found a number of examples of each of the above categories as well as steel shapes in the sandblast area. During the tour of the area, the RRI did not find any material that could not be identified except that in category three. The RRI interviewed one of the sandblasting personnel and came to the conclusion that the person might make an occasional mistake on category 2 material since he seemed confused when asked what he was going to do with a number of pieces ready for him to work on. It appeared that he might well blast the outside of a pipe when he should blast the inside.

Subsequent discussions with the paint shop foreman and with a Brown and Root Quality Control inspector in the pipe fabrication shop revealed that all cut, but unfabricated material, is transferred to the paint shop by memo which details the size, schedule and length of the cut section and the pipe spool isometric drawing involved. Should the outside of the pipe be inadvertently blasted, the piece can be reidentified relatively easy by measuring its size, schedule and length. The isometric drawing used to make the cut length is annotated with the pipe heat number prior to the cutting operation and verified by QC. It appeared most unlikely to the RRI that two otherwise identical pieces but with different heat numbers would be inadvertently blasted within the same time period.

The RRI concluded that the allegers remark that "workers are guessing on the identity of pipe" might be true, but that there was an adequate cross-check system built into the quality assurance program to preclude untraceable pipe from being installed in the safety related systems.

All of the steel shapes used in safety related supports for pipe and cable tray that have been examined by the RRI and other NRC inspectors have been sufficiently marked to establish their origin. These materials are also subject to a system of quality control verifications at various stages of fabrication sufficient to make it very unlikely that any improperly identified or unidentified material is used and installed.

- c. Allegation 3: Based on the interview with the alleged, no further action was taken to investigate the specifics of the allegation since the pipe in question was clearly not safety related and therefore not within the jurisdiction of the NRC inspection program. The more general concern that the pipe handling incident was a possible indicator of the general attitude of the craft personnel, particularly the riggers and pipefitters, appeared to be unfounded. The RRI has observed during many plant tours over the past nine months (since August 1978) that the material handling activities of the craft personnel have been accomplished under well controlled conditions in so far as they relate to safety related equipment and materials. An allegation of possible cover-up of improper actions by the craft personnel in behalf of other craft personnel is almost impossible to either confirm or completely refute.
- d. Allegation 4: No further investigation was made into the charge that third class welders are being used to perform safety related piping system welds on the basis that the welders are all qualified under a program prescribed by the ASME Boiler and Pressure Vessel Code Section IX, "Welding and Brazing Qualification." The application of the Section IX program has been reviewed a number of times by the RRI and other NRC inspectors since it was implemented at CPSES. The implementation has been found to be consistent with the requirements. These requirements, however, do not address themselves to the experience or inexperience of the person seeking qualification as a welder, but rather to whether he can accomplish a weld in one or more of the Code prescribed positions that will pass the test criteria imposed by the Code. The terminology "third class," as it applies to the labor force, relates primarily to the pay category in which a person is hired and previous experience is a factor in this determination.



UNITED STATES
NUCLEAR REGULATORY COMMISSION

REGION IV
611 RYAN PLAZA DRIVE, SUITE 1000
ARLINGTON, TEXAS 76012

ATTACHMENT 5

April 18, 1980

In Reply Refer To:

RIV

Docket No. 50-445/Rpt. 80-08
50-446/Rpt. 80-08

Texas Utilities Generating Company
ATTN: Mr. R. J. Gary, Executive Vice
President and General Manager
2001 Bryan Tower
Dallas, Texas 75201

Gentlemen:

This refers to the inspection conducted by our Resident Inspector, Mr. R. G. Taylor, during March 1980, of activities authorized by NRC Construction Permits No. CPPR-126 and 127 for the Comanche Peak facility, Units No. 1 and 2, and to the discussion of our findings with Mr. R. G. Tolson and other members of your staff at the conclusion of the inspection.

Areas examined during the inspection and our findings are discussed in the enclosed inspection report. Within these areas, the inspection consisted of selective examination of procedures and representative records, interviews with personnel, and observations by the inspector.

During the inspection, it was found that certain activities under your license appear to be in noncompliance with Part 50, Title 10, of the Code of Federal Regulations. One apparent item of noncompliance, which is discussed in paragraph 5 of the enclosed inspection report, was forwarded to you by our letter dated April 2, 1980. A second apparent item of noncompliance and references to the pertinent requirements is identified in the enclosed notice of Violation and in paragraph 6 of the enclosed inspection report.

This notice is sent to you pursuant to the provisions of Section 2.201 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations. Section 2.201 requires you to submit to this office, within 30 days of your receipt of this notice, a written statement of explanation in reply including: (1) corrective steps which will be taken by you, and the results achieved; (2) corrective steps which will be taken to avoid further noncompliance; and (3) the date when full compliance will be achieved.

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Texas Utilities Generating
Company

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April 18, 1980


We have also examined the actions you have taken with regard to previously identified inspection findings. The status of these items is identified in paragraph 2 of the enclosed report.

Two new unresolved items are identified in paragraph 6 of the enclosed report.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and the enclosed inspection report will be placed in the NRC's Public Document Room. If the report contains any information that you believe to be proprietary, it is necessary that you submit a written application to this office, within 20 days of the date of this letter, requesting that such information be withheld from public disclosure. The application must include a full statement of the reasons why it is claimed that the information is proprietary. The application should be prepared so that any proprietary information identified is contained in an enclosure to the application, since the application without the enclosure will also be placed in the Public Document Room. If we do not hear from you in this regard within the specified period, the report will be placed in the Public Document Room.

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,


for W. C. Seidle, Chief

Reactor Construction and
Engineering Support Branch

Enclosures:

1. Appendix A, Notice of Violation
2. IE Inspection Report No. 50-445/80-08
50/446/80-08

cc: w/enclosures
Texas Utilities Generating Company
ATTN: Mr. H. C. Schmidt, Project Manager
2001 Bryan Tower
Dallas, Texas 75201

Docket No. 50-445/Rpt. 80-08
50-446/Rpt. 80-08

Appendix A

NOTICE OF VIOLATION

Based on the results of an NRC inspection conducted during March 1980, it appears that certain of your activities were not conducted in full compliance with the conditions of your NRC Construction Permits No. CPPR-126 and 127 as indicated below:

Failure to Follow Procedures for Reporting and Repair of Damaged Electrical Cable

10 CFR 50, Appendix B, Criterion V states in part, "Activities affecting quality shall be prescribed by documented instructions, procedures or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings."

Brown and Root Procedure 35-1195-EEI-13, "Repair of Electrical Cable Jacket," requires that damaged cable shall be reported to Electrical Engineering by the craft or by QC via a Field Deficiency Report and requires that damaged jackets on control type cable be repaired using Okonite No. 35 Jacketing Tape in accordance with Okonite Drawing No. D-5715 when the damage involves penetration through the jacket.

Contrary to the above:

The Resident Reactor Inspector (RRI) learned of a Safety Train A control cable that had been damaged during pulling of the cable through the buried conduit system from the primary power plant building to the Service Water Intake Building. The cable was pulled back on or about March 28, 1980, and it was verified by the RRI that the damage had been sustained, the jacket was penetrated, and the repair was made with standard vinyl electrical tape rather than Okonite No. 35 and not in the method required by Okonite Drawing No. D-5715. The original damage and initial repair were not reported via a Field Deficiency Report.

This is an infraction.

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U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION IV

Report No. 50-445/80-08; 50-446/80-08

Docket No. 50-445; 50-446

Category A2

Licensee: Texas Utilities Generating Company
2001 Bryan Tower
Dallas, Texas 75201

Facility Name: Comanche Peak, Units 1 and 2

Inspection at: Comanche Peak Steam Electric Station, Glen Rose, Texas

Inspection conducted: March 1980

Inspector: *R. G. Taylor*
for R. G. Taylor, Resident Reactor Inspector
Projects Section4/16/80
DateApproved: *W. A. Crossman*
W. A. Crossman, Chief, Projects Section4/16/80
DateInspection Summary:Inspection During March 1980 (Report 50-445/80-08; 50-446/80-08)Areas Inspected: Routine, announced inspection by the Resident Reactor Inspector (RRI) including follow up to previous inspection findings; general site tours; safety related piping installations; concrete placement activities; electrical installation activities; and protection of major components. The inspection involved ninety-seven inspector-hours by one NRC inspector.Results: Of the six areas inspected, no items of noncompliance were identified in four areas; two items of noncompliance were identified in two areas (infraction - failure to follow procedures for repair of damaged cable - paragraph 6; deficiency - failure to report a significant construction deficiency - paragraph 5).8006120231

DETAILS SECTION1. Persons ContactedPrincipal Licensee Employees

- *J. B. George, TUSI, Project General Manager
- *J. T. Merritt, TUSI, Construction and Engineering Manager
- *D. N. Chapman, TUGCO, Quality Assurance Manager
- *R. G. Tolson, TUGCO, Site Quality Assurance Supervisor
- *J. C. Kuykendall, TUGCO, Plant Superintendent

The RRI also interviewed other licensee and Brown & Root employees during the inspection period including several randomly selected craft and Quality Control personnel that were interviewed in a semi-formal, private atmosphere.

*Denotes those persons with whom the RRI held on-site management meetings during the inspection period.

2. Action on Previous Inspection Findings

(Closed) Unresolved Item (50-445/79-07; 50-446/79-07): ITT-Grinnell Piping Assemblies Containing Bends. The licensee informed the RRI that a complete records search had been accomplished at ITT-Grinnell which identified all pipe spool components that had been hot bent in the fabrication process and had provided the site with documentation to support that each had been solution annealed after the hot bending process. The RRI selected at random a total of eight spools, three installed and five in storage, that by either of the wall thickness or the size had probably been hot bent. The necessary documentation was filed with the spool fabrication records attesting to the items having been solution annealed by various subcontractors to ITT-Grinnell.

The RRI had no further questions on this matter.

(Closed) Unresolved Item (50-445/79-08; 50-446/79-08): Pressurizer Safety Valve Spring Material. The licensee informed the RRI that the Crosby Valve drawing clearly indicated the material composition of the primary valve closure spring and that Westinghouse had approved the drawing and certified it to the owner.

The RRI reviewed drawing DC-C-56964, Revision C and the Westinghouse letter WPT-877 and had no further questions on this matter.

(Closed) Unresolved Item (50-445/79-16; 50-446/79-16): Design of Service Water System Cooling for Diesel Generators. The licensee notified RIV, by a letter dated October 5, 1979, that a potential design problem with valves and controls at the Service Water System and Auxiliary Feedwater System interface had been analyzed and found not to be a problem. The letter further indicated that data to support this contention was on file in the offices of the Architect/Engineer (A&E), Gibbs & Hill, Inc.

The RRI referred the matter to the Vendor Inspection Branch for review during a routine inspection of that A/E. This review was accomplished by VIB personnel who informed the RRI of their concurrence with the A/E's analysis. For further information see Inspection Report 99900524/80-01.

(Closed) Unresolved Item (50-445/80-01; 50-446/80-01): Engineer's Review of Test Reports. The licensee obtained a clarification statement from his A/E relative to the meaning of the stamp appearing on various test report documents indicating "Approved For Arrangement Only." The A/E has stated that the review is for complete compliance to related contract requirements placed on the vendor and that the test report indicates that the equipment satisfied contractual requirements.

The RRI had no further questions on this matter.

(Closed) Unresolved Item (50-445/80-03; 50-446/80-03): Materials For Service Water Valve Discs. The licensee informed RIV, by letter dated March 25, 1980, that they have reviewed the vendor reported lack of ASME Code required heat treatment of aluminum-bronze discs after welding has been performed and are reasonably sure that no safety problem exists. However, the licensee is returning the valves to the vendor for the Code required treatment prior to fuel loading or at some other time that will not interfere with early phases of pre-operational testing. This is a necessary step to prevent abrogation of the ASME certification for the valves.

The RRI had no further questions on this matter but will follow it as an element of routine inspection.

(Closed) Infraction (50-445/80-03; 50-446/80-03): Failure to Follow Procedures for Cable Pulling. The licensee notified RIV, by letter dated March 5, 1980, that the specific cables in question had been pulled back and visually examined and electrically tested with no damage being evident as a result of the failure to follow proper procedures. The licensee also committed to having foreman level supervision in attendance during all cable pulling activities in the future and to use cable lubrication when prescribed by approved site procedures. The RRI has verified through several inspections that each of the corrective actions has been implemented. The RRI is also satisfied, based upon document reviews and interviews with appropriate personnel, that the cables did not suffer damage.

The RRI had no further questions on this matter.

(Closed) Unresolved Item (50-445/79-23; 50-446/79-22): Component Installation Activities. This unresolved item was written to express a possibility that neither Operational Travelers nor Engineering Instructions would be utilized for the setting and alignment of mechanical equipment and, therefore, that the work might be improperly accomplished and/or not inspected to a proper set of instructions. The RRI has no evidence gained during various inspections over the past twenty months that would indicate that this possibility is also a probability. The evidence, as based on these inspections, is that the

necessary instructions for setting and aligning equipment have been issued, have been followed, and have been verified by the site QA/QC organization.

The RRI has no further questions on this matter.

3. Site Tours

The RRI toured the safety-related plant areas several times weekly during the inspection period to observe the progress of construction and the general practices involved. Three of the tours were conducted during portions of the construction second shift, with a primary emphasis on electrical cable pulling activities.

No items of noncompliance or deviations were identified during the tours.

4. Safety-Related Piping Installations and Welding

The RRI made several observations of the general handling and installation practices of safety-related piping components including spool pieces and valves with a primary concern for those fabricated from stainless steel. These observations included operations being carried on within the plant primary buildings, the pipe fabrication shop and finally, how the finished, uninstalled components are stored. The RRI observed the work of one relatively new welder that had not been previously observed. The welder observed was BFZ accomplishing the post-fitup tack welds on FW-2 of isometric CT-2-RB-005 in line 4-CT-2-097-301R2. The weld rod being used as obtained from the rod flag-tag and the weld filler metal log was from Sandvik heat 463638. The weld procedure being employed was 88021. The RRI subsequently verified that the welder, weld material, weld procedure and the adjacent components being welded all were consistent with the requirements of the ASME Code at that point in time. The RRI also observed the activities of the QA/QC person present and found him diligent and apparently knowledgeable of requirements.

The RRI also observed the activities of a QC person performing a liquid penetrant examination of two socket weld joints of a single spool in the containment spray system being finalized in the fabrication shop. The liquid penetrant examination was being carefully accomplished in a manner consistent with Brown & Root Procedure CP-NDEP-300, ASME Code requirements and good practice.

In addition to the above, the RRI also examined the following weld radiographs which were found to be consistent with the requirements of ASME, Section III as to weld quality and ASME, Section V for radiograph quality:

<u>Weld</u>	<u>Iscmetric</u>	<u>Line Identification</u>
W-11	BRP-SI-1-RB-27	1.5-SI-1-057-2501R1
W-4	BRP-SI-2-RB-48	1.5-SI-2-028-2501R1
W-6	BRP-SI-2-RB-008	3-SI-2-003-2501R1
W-4	BRP-SI-2-RB-008	3-SI-2-339-2501R1
W-3	BRP-SI-2-RB-008	3-SI-2-339-2501R1
W-5	BRP-RC-2-RB-22	1.5-RC-2-079-2501R1
W-12	BRP-CS-1-RB-029	2-CS-1-112-2501R1
W-7	BRP-RC-1-RB-016	2-RC-1-132-2501R1
W-24	BRP-RC-1-RB-032	1-RC-1-159-2501R1
W-6	BRP-RC-1-RB-016	2-RC-1-132-2501R1
W-24	BRP-RC-1-RB-033	1-RC-1-159-2501R1
W-15	BRP-SI-1-RB-022	1.5-RC-1-079-2501R1
W-8	BRP-SI-2-RB-008	1.5-SI-2-027-2501R1
FW-2	BRP-SI-1-RB-014	2-SI-1-059-2501R1
FW-8	BRP-CS-2-RB-021	2-CS-2-112-2501R1
FW-7	BRP-CS-2-RB-021	2-CS-2-112-2501R1
FW-9	BRP-CS-2-RB-021	2-CS-2-112-2501R1
W-20	BRP-SI-2-RB-060	6-SI-2-092-2501R1
FW-17	BRP-SI-1-RB-038	10-RC-1-092-2501R1
W-10	BRP-RC-1-RB-008	3-RC-1-052-2501R1
W-18	BRP-RC-1-RB-008	3-RC-1-052-2501R1
FW-9	BRP-FW-1-SB-019	18-FW-1-26-2002-2
FW-12	BRP-FW-1-SB-017	18-FW-1-34-2003-2

During an interview with a pipefitter welder, the RRI was informed that sometimes the pipefitter foreman is able to convince the Quality Control personnel to accept an out-of tolerance fitup at a weld joint. The welder indicated that this seemed to occur mainly when the joint was in a pipe size where the inside of the joint could be ground after welding. The RRI obtained enough information from the welder to pinpoint one weld where this had occurred, but by the time the RRI could get to the joint to make an examination, access to the inside of the pipe had become blocked by additional installations. The only recourse under this circumstance was then to rely on examination of the final weld radiographs and an ultrasonic measurement of the pipe wall thickness in the near vicinity of the weld and through the weld. The radiographs indicate a Code acceptable weld which does indicate evidence that some amount of internal grinding was accomplished prior to the radiograph having been taken. The ultrasonic wall thickness measurements show that adequate wall thickness was maintained even though the weld area was ground on both the interior and exterior surfaces. The RRI also interviewed the QC inspector who performed the fitup inspection. The inspector related that the joint in question was marginal in fitup, but that a consumable ring had been used and that the fitup could not be more than a very few thousandths of an inch over specification. The RRI inquired as to what the inspector and the welders considered to be the requirements and was informed that 3/32nds of an inch was the maximum offset allowable. The Code, however, allows as much as 3/16ths inch offset in the wall thickness involved. The Code further allows the fairing (grinding) of the interior of such a joint to provide a smooth transition across the weld. It appears that the tighter tolerance used by Brown & Root primarily comes from the verbiage of the "General Piping Procedure" CPM-6.9 and further that the welders much prefer the better fitup since it is easier for them to achieve a satisfactory weld.

The RRI has concluded that the weld joint in question is satisfactory; i.e., it meets all applicable requirements of the ASME Code and that the information received from the welder was largely based on a misunderstanding of the requirements.

No items of noncompliance or deviations were identified.

5. Concrete Placement Activities

The RRI examined the area preparation and the special formwork being installed preparatory to repairing the "honeycomb" condition in the interior walls of the Unit 2 Containment Building as discussed in Inspection Report 50-446/80-01. The work is being accomplished in accordance with a site generated set of detailed instructions while the RRI's basis for inspection is both these instructions and the applicable portions of the U.S. Bureau of Reclamation "Concrete Manual", a recognized authoritative publication on concrete work. The work to date appears to be progressing in accordance with the site instruction and the recommendations of the referenced publication.

The RRI did note that a period of nearly three months had passed between the time that the RRI was first notified of the "honeycomb" situation and the initiation of significant repair efforts. Based on the work performed by a consultant to the licensee (see Inspection Report No. 50-446/80-01); the observed trips to the site by representatives of the A/E; and the time span before repairs were started; it appeared that an extensive engineering review had occurred either for the purpose of determining the method of repair or to develop a basis for possibly not needing to make the repair at all for other than cosmetic reasons; i.e., that the structural soundness of the walls was not affected sufficiently to have a safety impact. 10 CFR 50.55(e) in sub-paragraph (1) (iii) indicates that a deficiency is reportable if an extensive engineering evaluation is required to simply determine the safety significance of such a deficiency as appears to be the situation. Since the licensee did not file an interim or final written report to the RIV office within thirty days following immediate notification of the incident to the RRI on December 13, 1979, the licensee was found to be in noncompliance with paragraph (3) of 10 CFR 50.55(e). A formal Notice of Violation was sent to the licensee on April 2, 1980.

²During an interview with a craft person, the person related a concern that some of the concrete in the ceiling over a corridor in the facility Common Fuel Building was not what it is supposed to be. He stated that sometime over a year ago, he was drilling holes in the concrete to insert "Hilti" bolts from which he was going to suspend a pipe hanger. He said that the dust which came from the drilling was nearly coal black rather than nearly white which is usually encountered. He had asked his foreman to check with someone on the apparent problem. He was subsequently told to go ahead and drill the holes; that nothing was wrong, but was given no explanation. The RRI determined from the person's description of the location of the incident and by reference to the civil/structural design drawings that the concrete in the particular ceiling was designated to be "heavy weight" concrete to provide added personnel shielding from radiation since the ceiling is also the support floor slab for the refueling canal which connects the spent fuel pools to the containments and the corridor below is a primary passageway. The job specifications require that a magnetite ore be used as both the fine and coarse aggregate in the concrete to achieve the much higher density. With the aid of a licensee employee, the RRI located a block of heavy weight concrete that had been originally cast as a test weight for hoisting equipment and drilled a hole in the block with a standard "Hilti" bolt carbide drill. The resulting drilling dust was nearly coal black just as described by the person. Although it has not yet been done, the RRI intends to locate the person expressing the concern and inform him that his concern is needless while thanking him for relating his concerns to the RRI.

Except as noted above, no items of noncompliance or deviations were identified of a technical safety nature in this area of the inspection.

6. Electrical System Installation Activities

The RRI made an on-going series of observations of the labor and QC activities as they relate to electrical cable installation and termination. The RRI observed a number of various sizes of control and low voltage power cables being installed in both of the primary safety trains. The RRI found that the labor force is carefully handling the cables and is lubricating them thoroughly when pulling through conduits already containing cables. Random checks in the cable tray system indicates that the trays are properly installed and adequately clean and further that there is presently no evidence of intermixing of either of the trains with each other or of either with nonsafety cable. No attempt was made to trace any given cable through its routing system since this is more efficiently and effectively done by other means than visual.

The RRI randomly selected two wire lug crimping tools observed in use and examined them for apparent wear or evidence of careless use. The tools appeared to be in good condition. The tools (CT-1224 and CT-1323) were also used as a vehicle for examining the crimping tool control system. The tools are checked each three months by the site calibration facility using vendor recommended procedures and certified go-no-go gauges in accordance with procedures IEI-98 and IEI-103, respectively. The RRI examined, on a random basis, the actual terminations in various cabinets and observed that the lugs were correctly crimped onto the wires and would not pull loose with application of reasonable force.

The RRI selected a cable type observed being installed as a vehicle for examining the qualification of the cable as required by the commitments contained in the FSAR, Section 8. The cable selected was W-847 from reel W-847-2 and is a 12 conductor unit made up of number 12 AWG individual wires and including their insulation with filler plus a jacket. The cable was supplied by Rockbestos in accordance with Project Specification ES-13B.1. The Project Specification and the referenced FSAR section both require that the cable be qualified in accordance with IEEE-383-74, "IEEE Standard For Type Test of Class IE Electric Cables, Field Splices and Connections for Nuclear Generating Plants." The RRI obtained documentation indicating that Rockbestos had performed the stipulated type and production tests required by the specification with the exception that there was no clear evidence that the three separate type tests of cables, as required by paragraph 2.5.4.3 of IEEE-383, had been accomplished nor was there evidence in the report that individual conductor tests had been performed as required by paragraph 2.5.6. The report did contain an attachment indicating that such tests had been performed and were successful.

This matter will be considered an unresolved item pending receipt of specific test data to substantiate the vendor statement.

During an interview, the RRI was informed that a safety-related control cable had been damaged while pulling it into the buried conduit system running from the main plant buildings to the Service Water Intake Building. The interviewee related that early in January 1980, a 5 or 7 conductor, orange (Safety Train A) control cable had slipped between a pulley wheel and the pulley frame during an interruption in pull and that when the pull was resumed, the jacket of the cable had been cut open. The pulley had been installed in the first manhole outside of the main buildings to aid the electricians in making the nearly ninety degree direction change through the manhole. The person further related that the electrical crew foreman in charge had instructed his people to tape up the damaged area with standard electrical tape (Scotch 33) and continue the pull which placed the damaged area somewhere in the buried conduit that is nearly five hundred feet long. Neither the foreman or any of his crew had apparently seen fit to report the incident nor was QC apparently aware of it. The RRI discussed the matter with the site electrical engineering personnel who indicated that the standard electrical tape used in the repair was probably not adequate as a jacket repair considering the location and the time that cable would have performed a safety function; i.e., forty years. The engineering personnel determined that only some eight 5 and/or 7 conductor, orange cables were likely to be involved since the balance of the cable in the particular conduit was either much larger or smaller and the person interviewed had been very specific in his relation of the event. The RRI verified that each of the identified cables had been pulled within the time frame of the related event by review of the electrical cable pull cards. The licensee elected to determinate the cables and draw them back to the manhole. The RRI examined the single cable found to have the damage and found that a cut of about one inch long had occurred that penetrated the cable jacket, but also found that no damage had occurred to the individual wire insulation within the cable.

Based on the RRI's knowledge of the characteristics of the wire insulation material, cross-linked polyethylene (XLPE), it is very doubtful that the functionality of the cable would have even been impaired and, therefore, the cut jacket has no direct impact on safety. The implications of the incident do, however, have a potential impact on safety in that it is indicative of a breakdown in the Construction Quality Assurance Program as evidenced by the fact that an electrician foreman took it upon himself to determine the need for the type of repair that was to be made to a damaged cable rather than reporting the matter through proper channels and allowing engineering to make the decision.

The incident is a violation of the intent of Appendix B, 10 CFR 50, Criterion V in that cable was not repaired in accordance with the applicable procedure.

For the record, the RRI would note that at the time of the incident, specific instructions had been issued addressing the area of cable damage or repair after the damage had occurred. The procedure provided for reporting damage to engineering and also provided for the use of a self-vulcanizing rubber tape to make jacket repairs rather than using Scotch 33.

In regard to the aforementioned cable repair procedure (EEI-13), the RRI's review, along with discussions with appropriate Quality Engineering personnel indicate a lack of clarity in its requirements. The procedure currently allows the replacement of wire insulation material in a multiconductor cable with the self-vulcanizing tape. There is currently no evidence available which would show that the tape has the same or better flame retardance characteristics as the factory applied XLPE insulation. The verbiage utilized in the procedure also essentially requires discussion with the procedure writers in order to achieve an understanding of what was intended by the writers.

This matter will be considered to be unresolved pending clarification.

7. Protection of Major Safety-Related Components

The RRI verified that the reactor vessels in both units are adequately protected to prevent likely damage and/or contamination. The Unit 1 reactor vessel head is well covered and protected in its lay-down area. The Unit 1 reactor vessel core-support components (internals) remain in their enclosed lay-down areas within the refueling pool area.

No items of noncompliance or deviations were identified.

8. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance or deviations. Two such items are discussed in paragraph 6 and will be referenced in the future as:

- a. Clarification of Rockbestos Electrical Cable Qualification
- b. Clarification of Electrical Cable Repair Procedures

9. Management Interviews

The RRI met with one or more of the persons identified in paragraph 1 on March 3, 4, 18, 19 and 29, 1980, to discuss various inspection findings and to discuss licensee actions and positions.



UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 1000
ARLINGTON, TEXAS 76012

ATTACHMENT 6

June 17, 1980

In Reply Refer To:

RIV

Docket No. 50-445/Rpt. 80-11

50-446/Rpt. 80-11

Texas Utilities Generating Company
ATTN: Mr. R. J. Gary, Executive Vice
President and General Manager
2001 Bryan Tower
Dallas, Texas 75201

RECEIVED

JUN 20 1980

TUGCO QA
DALLAS

Gentlemen:

This refers to the inspection conducted by our Resident Reactor Inspector, Mr. R. G. Taylor, during April and May 1980 of activities authorized by NRC Construction Permits No. CPPR-126 and 127 for the Comanche Peak Facility, Units No. 1 and 2, and to the discussion of our findings with Mr. R. G. Tolson and other members of your staff at the conclusion of the inspection.

Areas examined during the inspection and our findings are discussed in the enclosed inspection report. Within these areas, the inspection consisted of selective examination of procedures and representative records, interviews with personnel, and observations by the inspector.

During the inspection, it was found that certain activities under your license appear to be in noncompliance with Part 50, Title 10, of the Code of Federal Regulations. An apparent item of noncompliance, which is discussed in paragraph 7 of the enclosed inspection report, was forwarded to you by our letter dated April 9, 1980. You responded to our letter by your letter of May 5, 1980, and our Resident Reactor Inspector has verified implementation of your commitment as discussed in paragraph 2 of the enclosed report. No further response is required.

We have also examined the actions you have taken with regard to previously identified inspection findings. The status of these items is identified in paragraph 2 of the enclosed report.

~~8047180567~~

Texas Utilities Generating
Company

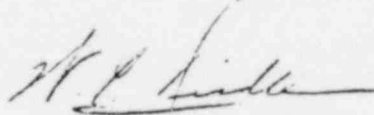
2

June 17, 1980

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and the enclosed inspection report will be placed in the NRC's Public Document Room. If the report contains any information that you believe to be proprietary, it is necessary that you submit a written application to this office, within 20 days of the date of this letter, requesting that such information be withheld from public disclosure. The application must include a full statement of the reasons why it is claimed that the information is proprietary. The application should be prepared so that any proprietary information identified is contained in an enclosure to the application, since the application without the enclosure will also be placed in the Public Document Room. If we do not hear from you in this regard within the specified period, the report will be placed in the Public Document Room.

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,



W. C. Seidle, Chief
Reactor Construction and
Engineering Support Branch

Enclosure:

IE Inspection Report No. 50-445/80-11
50-446/80-11

cc: w/enclosure
Texas Utilities Generating Company
ATTN: Mr. H. C. Schmidt, Project Manager
2001 Bryan Tower
Dallas, Texas 75201

yc: D. H. Chapman ✓
D. J. Hampton

U. S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION IV

Report No. 50-445/80-11; 50-446/80-11

Docket No. 50-445; 50-446

Category A2

Licensee: Texas Utilities Generating Company
2001 Bryan Tower
Dallas, Texas 75201

Facility Name: Comanche Peak, Units 1 and 2

Inspection at: Comanche Peak Steam Electric Station, Glen Rose, Texas

Inspection conducted: April and May 1980

Inspector: *W. A. Crossman*
for R. G. Taylor, Resident Reactor Inspector
Projects Section6/16/80
DateApproved: *W. A. Crossman*
W. A. Crossman, Chief, Projects Section6/16/80
DateInspection Summary:Inspection During April and May 1980 (Report 50-445/80-11; 50-446/80-11)Areas Inspected: Routine, announced inspection by the Resident Reactor Inspector (RRI) including follow up to previous inspection findings; general site tours; safety-related pipe and equipment installations; concrete repair activities; electrical installation activities; and protection of major components. The inspection involved one hundred fifty-five inspector-hours by one NRC inspector.Results: Of the seven areas inspected, no items of noncompliance were identified in six areas. One item of noncompliance was identified in one area (infraction - failure to follow piping installation procedures - paragraphs 2 and 7).8467184574

DETAILS1. Persons ContactedPrincipal Licensee Employees

- *J. B. George, TUSI, Project General Manager
- *J. T. Merritt, TUSI, Construction and Engineering Manager
- *D. N. Chapman, TUGCO, Quality Assurance Manager
- *R. G. Tolson, TUGCO, Site Quality Assurance Supervisor

The RRI also interviewed other licensee and Brown & Root employees during the inspection period including both craft labor and QA/QC personnel.

*Denotes those persons with whom the RRI held on-site management meetings during the inspection period.

2. Action on Previous Inspection Findings

(Closed) Infraction (50-445/79-18): Failure to Control Inspection Stamps. As noted in paragraph 2 of Inspection Report 50-445/79-27; 50-446/79-26, the use of numbered inspection stamps has been discontinued and the implementing procedure cancelled. The licensee's Site Surveillance Group interviewed all QC personnel to whom such stamps had been issued and who had failed to return them when the cancellation took place to ascertain the reason for the nonreturn and approximately when the stamp was lost or misplaced. Personnel no longer in QC by reason of termination were not interviewed. The dates of loss and/or termination were then used as the basis for an extensive QA records search to determine if the missing stamps had been improperly used. The records search failed to reveal any such improper use and the licensee concluded that the loss of the stamps was attributed to personnel carelessness rather than any overt act.

The RRI had no further questions on this matter.

(Closed) Unresolved Item (50-445/80-01; 50-446/80-01): Class 1 to Class 2 Transition Orifices. The licensee has issued Component Modification Cards 33001 and 33002 which require the installation of the required transition orifices in the manner originally called for in the design drawings at a location approximately six inches from the improperly sized orifices. The improperly sized orifices will be plugged and seal welded.

The RRI had no further questions on this matter but will follow the implementation of the above Component Modification Cards during routine inspections.

(Closed) Infraction (50-445/80-01; 50-446/80-01): Failure to Provide Instructions and Procedures Appropriate to the Circumstances. The licensee informed RIV, by letter dated February 19, 1980, that their analysis of

the as-built mounting of the battery chargers indicated that the mounting provided adequate strength to satisfy seismic requirements. The licensee also stated that engineering procedures were being revised to require an Architect/Engineer review of equipment mounting details in addition to that already required by the equipment vendor. The RRI has verified that the procedure has been revised and implemented.

The RRI had no further questions on this matter.

(Closed) Deficiency (50-445/80-08; 50-446/80-08): Failure to Report a Significant Construction Deficiency. The licensee informed RIV, by letter dated April 21, 1980, that a review of the reporting requirements of 10 CFR 50.55(e) had been accomplished and that a meeting in the RIV office, as documented by Inspection Report 50-445/80-12; 50-446/80-12, had rendered further clarification of the requirements. The licensee stated that necessary instructions had been given to appropriate personnel in the matter. The RRI has interviewed these personnel and is satisfied that they now understand and will implement the requirements fully. For further information relative to the "honeycomb" condition referred to in the original finding, see paragraph 4 of this report.

This item is considered closed.

(Closed) Infraction (50-445/80-11; 50-446/80-11): Failure to Follow Piping Installation Procedures. This infraction, which is discussed in paragraph 7 of this report, was forwarded by RIV letter dated April 9, 1980. The licensee reported to RIV, by letter dated May 5, 1980, that an analysis of the reported situation showed that no excessive strain had been placed on the pump nozzle involved. The RRI reviewed these calculations with the NSSS supplier and was satisfied that no damage had been incurred. The licensee also committed to additional inspection for like items which was accomplished and the results documented. Other situations were found of a like nature and fortunately no harm to equipment was involved. The licensee stated that piping installation procedures have been revised to make it clear to the craft labor force that piping connections to equipment are not to be made until the piping is supported properly with hangers rather than by simple cribbing.

The RRI observed, during tours of the facility, that the revised procedures had been implemented and had no further questions.

(Closed) Infraction (50-445/80-08; 50-446/80-08): Failure to Follow Procedures for Reporting and Repair of Damaged Electrical Cable. The licensee informed RIV, by letter dated May 14, 1980, that a new cable would be pulled through the buried bus duct to replace the damaged cable. In addition, new cables were also pulled to replace several other cables in the duct that were damaged in the search for the cable originally reported. The licensee also stated, in the referenced letter, that Management/Supervisory Seminars had been held to emphasize the need to follow all project procedures. The RRI reviewed documentation indicating that eighty-two persons, including

electrical department superintendents, general foremen, and foremen, attended one of two such seminars. Interviews with two electrical crew foremen indicate that they are aware of the procedural requirements.

The RRI had no further questions.

3. Site Tours

The RRI toured the safety-related plant areas several times weekly during the inspection period to observe the general progress of construction of the practices involved. Five of the tours were accomplished during portions of the second shift. Since the principal effort of the second shift is the installation of electrical cables, primary emphasis was placed on this activity.

No items of noncompliance or deviations were identified.

4. Concrete Repair Activities

The RRI observed substantial portions of the activities involved in the removal of the defective concrete in the "honeycomb" areas of the Unit 2 Reactor Containment Building internal walls as discussed in Inspection Reports 50-445/80-01; 50-446/80-01 and 50-445/80-08; 50-446/80-08.

The RRI examined a number of the cavities after removal of the "honeycomb," after application of concrete bonding agents, and again after the repair formwork was in place for the concrete pour back. In one area, the sleeve through the wall for the reactor coolant pipe had to be partially removed to gain access to the defective concrete. The RRI observed portions of the weld repair to the sleeve to re-establish its original configuration. The welding was accomplished in accordance with the engineer's instructions by qualified welders utilizing qualified weld procedures. As of the end of the inspection period, the entire repair effort was essentially complete and appeared to have been done in a sound manner in accordance with recognized concrete repair practices.

The licensee officially informed RIV of the above matter as required by 10 CFR 50.55(e) in a letter dated April 21, 1980. The report outlines the engineering evaluations performed, the safety impact had the defects gone unrecognized and/or unrepaired, and the repair methods to be utilized.

No items of noncompliance or deviations were identified.

5. Major Component Installation Activities

During the inspection period, the RRI observed the efforts involved in installing the last two steam generators and the first two Reactor Coolant pumps in the Unit 2 Reactor Containment Building. The RRI observed the

initial preparation of the steam generators for hoisting into the building, the actual hoisting and movement, and finally the setting and alignment of the units on their support columns. Each step was observed to be in accordance with Operation Travelers RI80-369-3400 and ME80-2005-5500 governing the work of the riggers and millwrights, respectively. The RRI also reviewed the steps indicated by the two Operation Travelers with the NSSS supplier representatives on site and verified that the steps utilized were in consonance with the supplier's written recommendations. The RRI reviewed data developed by the site field engineers (surveyors) which showed that the generators are well within the established vertical requirements of the vendor and that each of the four support columns are carrying approximately equal load. In regard to the Reactor Coolant pump installation, the RRI observed the work involved in setting the pumps on their columns and establishing the pumps into an essentially level position.

The RRI also observed the preliminary installation of two of the Reactor Coolant pipe legs through the sleeves leading to the Reactor Pressure Vessel. These pipe sections were carefully handled and placed into position in accordance with good practice.

No items of noncompliance or deviations were identified.

6. Reactor Coolant Pressure Boundary Piping Installation

The RRI made limited observations of piping component handling in the Reactor Coolant Pressure Boundary area during the period. The RRI observed two welds in process as follows:

Weld Number:	FW-3A	FW-20
Isometric:	RC-1-RB-026	SI-1-RB-037
Line Identification:	14-RC-1-135-2501R1	10-RC-1-021-2501R1
Welder Identification:	AWT and BMK	BAG
Weld Procedure:	99025 (Machine GTAW)	88025 (Manual GTAW)
Filler Metal Identification:	463870	762550

Subsequent to the observation of welding, the RRI verified that the welders, weld procedures and weld filler metals were each properly qualified in accordance with the ASME Code, Section III or IX as appropriate. In addition, the RRI also examined the radiographs taken of the welder qualification test coupons for welders BAG, BLU, AXC, BPA and AED. These

radiographs, which are an examination alternative of ASME, Section IX (the other alternative is prescribed bend tests), indicated a sensitivity technically acceptable per Section V of the Code. The RRI discussed the radiographs with the supervising radiographer who stated that the fuzziness of the radiographs was caused by energy scatter from the source (Iridium 192). Since the radiographs met all technical requirements of the Code, he felt there was no problem. The RRI agreed that the Code had been technically satisfied, but at a marginal or minimum level and the radiographs could be substantially improved by a better technique. The RRI will pursue this matter during future inspections. The above discussed radiographs indicated that each welder had accomplished a weld or welds that satisfied the Code requirements and were, thus, fully qualified to perform production welding.

The RRI also examined radiographs of the following reactor coolant boundary (Class 1) welds:

<u>Weld Identification</u>	<u>Isometric</u>	<u>Line Number</u>
W-6	SI-2-RB-042	2-SI-2-086-2501R1
FW-12	SI-1-RB-21	3-SI-1-033-2501R1
W-14	SI-2-RB-042	2-SI-2-086-2501R1
W-14	SI-1-RB-020	1.5-SI-1-020-2501R1
W-12	SI-2-RB-042	2-SI-2-086-2501R1
W-6	CS-1-RB-031B	2-CS-1-105-2501R1
FW-1-1	RC-1-RB-15	3-RC-1-111-2501R1
FW-10-2	RC-1-RB-15	3-RC-1-111-2501R1
FW-38-1	RC-1-RB-15	3-RC-1-146-2501R1
W-10	CS-1-RB-031B	1.5-CS-1-249-2501R1
W-8	CS-1-RB-031B	1.5-CS-1-105-2501R1
W-9	CS-1-RB-031B	1.5-CS-1-105-2501R1
W-7	CS-1-RB-031B	1.5-CS-1-105-2501R1
W-2	CS-2-RB-074	2-CS-2-112-2501R1
W-5	CS-1-RB-031B	2-CS-1-105-2501R1

W-3	CS-1-RB-028	2-CS-1-112-2501R1
W-18	RC-1-RB-15	3-RC-1-111-2501R1
FW-42	RC-1-RB-15	3-RC-1-146-2501R1
FW-6	RC-1-RB-08	3-RC-1-052-2501R1
W-6	RC-1-RB-06	6-RC-1-70-2501R1
FW-3	RC-1-RB-017	4-RC-1-075-2501R1
FW-38-2	RC-1-RB-05	3-RC-1-146-2501R1
FW-2	RC-1-RB-017	4-RC-1-075-2501R1
W-5	SI-2-RB-042	2-RC-2-086-2501R1
W-35	SI-1-RB-015	2-SI-1-086-2501R1
FW-11	SI-1-RB-021	3-SI-1-033-2501R1
FW-1	RC-1-RB-06	12-RC-1-069-2501R1
FW-5A	RH-1-RB-02	12-RH-1-022-2501R1

No items of noncompliance or deviations were identified.

7. Other Safety-Related Piping Installation Activities

The RRI observed welder AHI during a period when the welder was working on joint FW-5 as identified on isometric CT-1-RB-17 in line 10-CT-1-027-301R2. The welder was working to Weld Procedure 88021 using filler metal Heat Number 463638. The qualification of the procedure and this heat of filler metal have been verified several times during previous inspection. Review of the welder qualification records for AHI indicate that he has been properly qualified in accordance with ASME, Section IX.

The RRI also examined the licensee actions in regard to implementation of his commitment to radiograph and repair those field welds in the Safety Class 3 Component Cooling Water and Auxiliary Feedwater Systems that do not require radiographs under the Code. (For more information regarding this commitment, see Inspection Reports 50-445/79-12 and 50-445/79-17.) The personnel managing the program indicated that approximately 56% of the 1842 welds involved have, to date, been radiographed and that about 37% of those requiring repair have been repaired. The RRI randomly selected the following radiographs for review:

<u>Weld Identification</u>	<u>Isometric</u>	<u>Line Number</u>
FW-10	AF-1-SB-23	4-AF-1-102-152-3
FW-25	AF-1-YD-05	3-AF-1-86-152-3
FW-13	CC-1-RB-042	3-CC-1-232-152-3
FW-10A	CC-1-RB-58A	3-CC-1-234-152-3
FW-30	CC-1-RB-58A	3-CC-1-234-152-3
FW-1	CC-2-AB-045	3-CC-2-118-152-3
FW-22-R1	AF-1-SB-10	6-AF-1-33-152-3
FW-28-R1	AF-1-SB-15	4-AF-1-102-152-3
FW-3-R1	AF-1-SB-72	3-AF-1-72-152-3
FW-24-R1	CC-1-RB-041	3-CC-1-232-152-3

The RRI made numerous observations of the general pipe and component handling operations in both Units 1 and 2 during the inspection period and found that good practices were being followed as outlined in the General Piping Procedure CPM-6.9. In one instance however, the RRI observed a situation that was of concern in that possible major safety component damage might have occurred which could easily have gone undetected. The RRI found that a pipe assembly, consisting of several feet of six inch diameter pipe, was being entirely suspended by attachment to the suction nozzle flange of the Unit 2 Train A Safety Injection pump TCX-SIAPSI-01. Further investigation developed that the pipe assembly would place a torque load on the nozzle of between 1500 and 2000 foot-pounds. The RRI found that CPM 6.9 did not provide instructions on this matter to the labor force, although the project Mechanical Erection Specification (MS-100) specifically prohibited such practices. The RRI notified the licensee of the situation which was in turn followed up with a Notice of Violation dated April 9, 1980.

The licensee responded to the initial notification by having the other installed pumps and valves in Unit 2 checked for like situations. A very limited number of other comparable situations were identified during this inspection.

The RRI identified situation and others identified by the licensee were detailed on Nonconformance Reports which were submitted to the component

vendor, Westinghouse, for analysis of possible damage to the components. The analysis indicated that no damage was likely to have occurred due to the static loading on the nozzles, although had the pipe been of a heavier schedule or longer in length, such damage would have occurred. The Westinghouse analysis was reviewed by the RRI who had no question of its accuracy.

The licensee's investigation of the circumstances surrounding the incident indicated that the pipefitters had the pipe supported by temporary wooden blocks or jack stands when they left the work area. These workers were subsequently reassigned to other work and did not return to the area. In the meantime, it appears that a group of painters were assigned to paint the floors in the area and removed the temporary shoring under the piping leaving it suspended from the nozzles.

The labor force was notified that this practice must cease and the licensee also revised CPM 6.9 to provide specific instructions in the matter. All of these actions were consummated during the period covered by this report, and as noted in paragraph 2, this item of noncompliance is considered to have been satisfactorily closed.

Except as noted above, no items of noncompliance or deviation were identified.

8. Electrical Installation Activities

The RRI made a number of observations of electrical cable installations during the inspection period. The primary inspection effort was directed toward observing the activities of the various cable pulling crews and toward this end at least five crews were checked. During most of the period there were seven active crews working safety-related cable. Each of the crews observed appeared to be knowledgeable of the prescribed methods of pulling cable and of the limitations imposed by site procedures and good practice. The RRI also examined most of the Main Control Room cabinets and the termination cabinets in the Cable Spread Room of Unit 1 relative to the quality of the workmanship displayed in termination of the cables. No instances were found in which the termination was less than satisfactory as evidenced by the application of the correct size wire lug that was properly crimped and tightly installed on the terminal boards. The RRI also examined a number of terminations for correct connection on the terminals as indicated on the electrical design drawings with no errors being detected. This effort was primarily directed toward the main 6.9 KV switchgear in Safety Train A.

No items of noncompliance or deviations were identified.

9. Protection of Major Safety-Related Equipment

During the course of general plant tours, the RRI noted that the major plant components continue to be well cared for as evidenced by space

heaters being energized and where appropriate, because of on-going work, the equipment is adequately covered. The Unit 1 and 2 Reactor Vessels were noted to be well protected even though extensive civil construction work was in progress in the immediate vicinity. The Unit 1 Reactor Vessel internals were noted to be in their enclosures and apparently adequately protected as was the Unit 1 Vessel head with the installed Control Rod Drive Mechanisms.

No items of noncompliance or deviations were identified.

10. Management Interviews

The RRI met with one or more of the persons identified in paragraph 1 on April 2, 3, 9, 10, 15, 18, and May 13 and 29, 1980, to discuss inspection findings and to discuss licensee actions and positions.



UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 1000
ARLINGTON, TEXAS 76011

ATTACHMENT 7

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MAR 15 1979

COMANCHE PEAK STEAM
ELECTRIC STATION

March 14, 1979

In Reply Refer To:

RIV

Docket No. 50-445/Rpt. 79-03
50-446/Rpt. 79-03

Texas Utilities Generating Company
ATTN: Mr. R. J. Gary, Executive Vice
President and General Manager
2001 Bryan Tower
Dallas, Texas 75201

Gentlemen:

This refers to the inspection activities performed by our Resident Inspector, Mr. R. G. Taylor, during the period February 2-28, 1979, of activities authorized by NRC Construction Permit Nos. CPPR-126 and 127 for the Comanche Peak facility, Units No. 1 and 2, and to the discussion of our findings with Mr. R. G. Tolson and other members of your staff during the course of the inspection.

Areas examined during the inspection and our findings are discussed in the enclosed inspection report. Within these areas, the inspection consisted of selective examination of procedures and representative records, interviews with personnel, and observations by the inspector.

During the inspection, it was found that certain activities under your license appear to be in noncompliance with Appendix B to 10 CFR 50 of the NRC Regulations, "Quality Assurance Criteria for Nuclear Power Plants."

The Notice of Violation for the item of noncompliance reported in paragraph 9 of the enclosed inspection report was forwarded to you by our letter, dated February 20, 1979; therefore, this letter does not require further response regarding this matter.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and the enclosed inspection report will be placed in the NRC's Public Document Room. If the report contains any information that you believe to be proprietary, it is necessary that you submit a written application to

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Texas Utilities Generating
Company

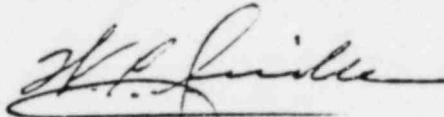
-2-

March 14, 1979

this office, within 20 days of the date of this letter, requesting that such information be withheld from public disclosure. The application must include a full statement of the reasons why it is claimed that the information is proprietary. The application should be prepared so that any proprietary information identified is contained in an enclosure to the application, since the application without the enclosure will also be placed in the Public Document Room. If we do not hear from you in this regard within the specified period, the report will be placed in the Public Document Room.

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,



W. C. Seidle, Chief
Reactor Construction and
Engineering Support Branch

Enclosure:

IE Inspection Report No. 50-445/79-03
50-446/79-03

cc: w/enclosure
Texas Utilities Generating Company
ATTN: Mr. H. C. Schmidt, Project Manager
2001 Bryan Tower
Dallas, Texas 75201

U. S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION IV

Report No. 50-445/79-03; 50-446/79-03

Docket No. 50-445; 50-446

Category A2

Licensee: Texas Utilities Generating Company
2001 Bryan Tower
Dallas, Texas 75201

Facility Name: Comanche Peak, Units 1 & 2

Inspection at: Comanche Peak Steam Electric Station, Glen Rose, Texas

Inspection conducted: February 2-28, 1979, and meeting at
TUGCO Corporate Office on February 2, 1979

Inspector:

for *W. A. Crossman*
R. G. Taylor, Resident Reactor Inspector
Projects Section3/13/79
DateOther Accompanying
Personnel at Corporate
Meeting onFebruary 2, 1979: W. C. Seidle, Chief, Reactor Construction and
Engineering Support Branch

R. E. Hall, Chief, Engineering Support Section

Approved:

W. A. Crossman
W. A. Crossman, Chief, Projects Section3/13/79
DateInspection Summary:

Inspection on February 2-28, 1979 (Report No. 50-445/79-03; 50-446/79-03)

Areas Inspected: Routine inspection by the IE Resident Reactor Inspector (RRI) of safety related construction activities including installation and welding of reactor coolant and other piping systems; structural building activities; construction fire protection; and follow up on licensee identified items. The inspection involved seventy-five inspector-hours by one NRC inspector.Results: One item of noncompliance (infraction - failure to follow concrete placement procedures - paragraph 9) was identified in one of the ten construction areas inspected.

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DETAILS1. Persons ContactedPrincipal Licensee Employees

- *J. B. George, TUSI, Project General Manager
- *R. G. Tolson, TUGCO, Site QA Supervisor
- *J. V. Hawkins, TUGCO/G&H Product Assurance Supervisor
- *D. N. Chapman, TUGCO Quality Assurance Manager
- C. C. Buffkin, TUGCO/G&H QA Welding Specialist
- *D. E. Deviney, TUGCO QA Technician
- *R. J. Gary, TUGCO, Executive Vice President & General Manager
- *L. F. Fikar, TUSI, Executive Vice President & General Manager

Other Personnel

- *H. O. Kirkland, Project General Manager, Brown & Root (B&R)
- *U. D. Douglas, Construction Project Manager, B&R
- *J. H. Magner, Construction Project Engineer, B&R

The Resident Reactor Inspector (RRI) also interviewed a number of other site construction and quality control personnel during the course of the inspection.

*denotes those individuals attending the management interviews.

2. Licensee Construction Deficiency Reports (10 CFR 50.55(e))

The licensee has determined that two of three potentially reportable construction deficiencies discussed in IE Inspection Report 50-445/79-01; 50-446/79-01 are not formally reportable as 50.55(e) items. The status of the third item has not yet been determined.

- a. The licensee notified RIV by letter, dated February 6, 1979, that the problem of leakage through a designed expansion joint would not meet the criteria of 50.55(e) and was, therefore, not considered reportable. The RRI reviewed the referenced supporting documentation which essentially indicated that while leakage would occur, the worst case level of contamination in the leakage would be below the limits imposed by 10 CFR 20 Appendix B, Table 1. The RRI discussed the licensee's analysis with knowledgeable NRC personnel who concurred. This item is considered closed.

- b. The licensee notified RIV by letter, dated February 20, 1979, that the matter of the reversed reactor coolant loop elbows was not reportable. The RRI reviewed the referenced documentation which concludes that the reversal is readily corrected by addition of weld metal at the connecting joints and that the initial report was premature since appropriate subsequent QA and construction actions had not yet taken place. The evaluation by the licensee concluded that the item was not reportable within the context of 50.55(e). The RRI had no further questions and the item is considered closed.

3. Mislocated Reactor Vessel Support Structure

The licensee reported to the RRI on February 20, 1979, that a major error had been detected in the design of the Unit 2 reactor vessel support structure. It had been determined that the reactor vessel support shoes, their ventilation duct work, and the surrounding reinforcing steel had been rotated forty-five degrees from the correct positions through a design error. As a result of the rotation, the reactor vessel would not match the vessel support feet nor would the piping system to the other reactor loop components fit. The licensee concluded that since the vessel could not be installed and connected this precluded the deficiency from being reported as a 50.55(e) item, but was necessarily of concern because significant design and construction changes would be required to accommodate the vessel and its connecting piping. As of the date of this report, it has been determined that the existing structure can be modified to reorient the vessel correctly to the balance of the system. The exact details of the modification, however, are not known at this time. The RRI will follow the modification effort in detail until it has been concluded.

4. Plant Tours

The RRI toured one or more plant areas several times weekly during the reporting period to observe the progress of construction of safety related structures and the installation of equipment. Two of the tours were conducted during second shift hours of work. The RRI observed that housekeeping practices have improved during the period, apparently through the combined efforts of the construction and quality control forces. The RRI has been informally advised that a general housekeeping program is now operative, although undocumented. The work force and work effort continues as previously indicated.^{1/}

No items of noncompliance or deviations were identified.

5. Construction Fire Protection

The RRI observed, during various plant tours, that installed equipment is adequately protected from incidental damage from probable heat sources such as area welding operations. Protection is primarily by covering of equipment with plywood and fire retardant plastic and by availability of properly charged portable fire extinguishers. Interviews with randomly selected craft personnel indicated an awareness of where the fire extinguishers were available.

The RRI has no further questions regarding fire protection at this stage of construction.

6. Reactor Vessel and Internal Protection

The RRI observed that the Unit 1 vessel internals remained stored in site warehouse facilities and continue to be protected from probable contaminants. The RRI observed that the prefabricated steel reactor vessel cover in Unit 1 was removed on or about February 27, and replaced with a wood beam and plastic cover. All vessel nozzles are now partially or completely welded to piping. Access control and personnel logging for work on or in the vessel continues in effect.

The RRI reviewed plans and procedures related to receipt, off-loading and temporary storage of the Unit 2 reactor vessel scheduled to arrive early in March 1979. Records of tests and maintenance indicate that the handling equipment proposed has met all requirements and that planned storage is consistent with the supplier's (Westinghouse) recommendations.

No items of noncompliance or deviations were identified.

7. Reactor Coolant System Installation and Welding

The RRI observed the welding of Control Rod Drive Mechanisms (CRDM) to the reactor vessel head CRDM nozzles during this period. The total construction effort involved seventy-eight assembly and welding operations with associated nondestructive and hydrostatic testing over a period of approximately seven days. The welds involved are nonstrength seal welds referred to as canopy welds and falling under the provisions of ASME, Section III, Article NB-4360. The welding observed was accomplished using an automatic gas tungsten machine with an especially designed welding head powered by an "Astro-Arc" control and supply unit. The consumable insert, the only weld filler

metal involved, was tacked in place prior to CRDM assembly by welder AFK while the machine was operated by welder ABT during the time of observation. The machine welding was being performed in accordance with Weld Procedure Specification (WPS) 99029, Revision 0 as evidenced by the control unit settings and the strip chart recorder attached. Review of WPS 99029 indicated that the procedure and three welders, including AFK and ABT, had been qualified as required by the above referenced article of ASME, Section III. Each of the welds involved was authorized and documented by individual Weld Data Cards. The RRI reviewed seven of the cards and found them to have been completed consistent with project procedures.

The RRI also observed a portion of the welding operations on FW-23, which joins the cold leg reactor pipe to the coolant pump in loop 3. The welders observed were BBI, AXB and AZF using a Dimetrics automatic TIG machine in accordance with WPS 99028, Revision 1. The weld filler metal was of Heat No. 434788 as evidenced by Weld Material Requisitions A175373 and A175374 and vendor identification on the wire supply reel. The machine control panel settings were verified as being consistent with WPS 99028 requirements for the observed weld pass; i.e., number seven. The RRI reviewed the qualification documents relative to the above identified persons and found them to be in accordance with ASME, Section IX requirements.

The RRI reviewed acceptance level radiographs for Reactor Coolant Loop welds FW-10, FW-26 and FW-31 during the report period. These welds join piping to the loop 2 steam generator hot leg; the loop 4 steam generator hot leg, and loop 4 pump to cold leg, respectively. The radiographs indicated acceptable weld quality in accordance with ASME, Section III and displayed sensitivity as required by ASME, Section V.

No items of noncompliance or deviations were identified.

8. Other Safety Related Installation and Welding

The RRI observed several piping installation and welding operations during the period with emphasis on stainless steel components. It was observed that in all instances the pipe components, including valves, were handled in accordance with project procedures to prevent carbon steel contamination via the use of nonmetallic slings, use of marked "for stainless only" grinders, brushes and files.

Two welds were selected for in-depth observation and document review. The RRI observed welder ARP working on repairing FW-5 as shown on isometric BRP-CT-1-SB-09-0 in Containment Spray line 16"-CT-1-SB-014-301R2. The welder was utilizing WPS 88023, Revision 4, a manual

TIG process. Filler metal was ER308, Heat No. 463730. Review of documentation indicated that repair involved removal of five zones of root lack of fusion totaling approximately 3.3 inches in about 50 inches of original weld.

The RRI observed welder AGP working on joint FW-2B identified on isometric BRP-RH-1-SB-02-0 in line 12-RH-1-003-601R2, a Residual Heat Removal system line. This weld was the second full replacement weld for an original weld. WPS 88021, Revision 0 was being utilized with filler metal Heat No. 463730.

The RRI subsequently verified that welders ARP and AGP were both qualified to the welding processes involved in accordance with ASME, Section IX. Documentation for the off-site fabricated spool pieces being joined by the above welds indicated that the items met Project Specification MS-43A and ASME, Section III requirements.

The RRI also reviewed radiographs of the following safety system welds:

<u>Weld</u>	<u>Isometric</u>	<u>Line</u>
FW-12A	BRP-SI-1-SB-14	4-SI-1-300-1501R2
FW-13A	BRP-SI-1-SB-14-5	4-SI-1-045-1501R2
FW-5	BRP-CT-1-SB-05	16-CT-004-151R2

The RRI observed the conduct of liquid penetrant examination of FW-3, isometric BRP-CS-1-SB-01-0 in line 8-CS-1-063-151R2. All of the identified nondestructive examination efforts indicated acceptable weld joints with examination techniques consistent with ASME Code requirements.

No items of noncompliance or deviations were identified.

9. Concrete Construction activities

The RRI selected Placement No. 201-5805-019 in the Unit 2 containment as typical of concrete construction activities for the period. The concrete batch plant was again batching out ten cubic yard loads of Design Mix 133 concrete from the automatic stationary mixer utilizing calibrated scales. The site test laboratory personnel were observed at both concrete pumping stations taking required field tests of slump, air content and temperature at the correct frequency and utilizing project proceduralized techniques. The RRI verified that the placement formwork was tight and had adequate clearance from the reinforcing steel

to provide specified concrete cover; and that the lower preceding placement was clean and damp. It was observed that the placement crews were consolidating the freshly deposited concrete in a thorough but not excessive manner.

After a change from one deposit chute to another, the RRI observed that the falling concrete from the new chute was aimed such that a portion of the stream impinged on a cross piece (shear tie) reinforcing steel element which shredded the stream and caused the coarse aggregate to segregate from the stream and fly around the placement area. The RRI pointed out the situation to the Brown & Root QC inspector who ordered the crew to stop and add additional length to the chute (elephant trunk) to get the concrete down through the rebar without direct impingement. The RRI verified that the other placement crew had also added the elephant-trunk extenders after a short shutdown. Since neither the Brown & Root crew nor the attending QC inspector had taken any immediate corrective action without being prompted by the RRI, it appeared that an undocumented nonconformance to project specifications and industry standards had occurred and might have been allowed to continue.

ACI-301-72, paragraph 8.3 as invoked by Project Specification 2323-SS-9, "Concrete," prohibits any placement procedure which will cause segregation. Project QC procedure QI-QP-11.0-3 also references SS-9 as criteria with an inspection checkpoint for segregation. The RRI considered this to be in noncompliance with 10 CFR 50, Appendix B and so informed the licensee.

10. Management Meetings

The RRI met with licensee site representatives on February 14, 15, 22 and 23, 1979, to discuss inspection findings and the licensee's plans for modification to the Unit 2 containment for the problem identified in paragraph 3. The RRI also accompanied the RIV Reactor Construction Branch Chief and Projects and Engineering Support Section Chiefs during a meeting on February 2, 1979, at the licensee's corporate headquarters to discuss safety related piping system welding problems. The RIV representatives indicated that welding reject rates with the accompanying repair cycles had created concern since a repaired weld is frequently not as satisfactory as an original weld due to metallurgical effects in the heat affected zone adjacent to the weld. The licensee stated that the site construction management had developed a program endorsed by licensee management that would improve the situation as follows:

- a. Improve the welder training and qualification program beyond that required by the ASME Code such that qualification testing would be more representative of field production conditions.

- b. Tighten the administrative production controls such that immediate identification of inadequately trained welders could be achieved; these to be retrained as necessary.
- c. Provide highly qualified welding technicians to the field to work with the welders in a training and advisory capacity.

The licensee management acknowledged their concern for the reject rate and expressed hope that the above program would substantially reduce the rate in the near future. The licensee representatives emphasized that welds accepted had met all requirements of ASME Code and FSAR commitments; a condition acknowledged by the RIV representatives.



UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 1000
ARLINGTON, TEXAS 76011

ATTACHMENT 8

March 23, 1979

In Reply Refer To:

RIV

Docket No. 50-445/Rpt. 79-07
50-446/Rpt. 79-07

RECEIVED

MAR 27 1979

TUGCO QA
DALLAS

Texas Utilities Generating Company
ATTN: Mr. R. J. Gary, Executive Vice
President and General Manager
2001 Bryan Tower
Dallas, Texas 75201

Gentlemen:

This refers to the inspection conducted by Mr. A. B. Beach and other members of our staff during the period March 8, 12 and 13, 1979, of activities authorized by NRC Construction Permits No. CPPR-126 and 127 for the Comanche Peak facility, Units No. 1 and 2, and to the discussion of our findings with Mr. R. G. Tolson and other members of your staff at the conclusion of the inspection.

Areas examined during the inspection and our findings are discussed in the enclosed inspection report. Within these areas, the inspection consisted of selective examination of procedures and representative records, interviews with personnel, and observations by the inspectors.

Within the scope of the inspection, no items of noncompliance were identified.

Two new unresolved items are identified in paragraphs 3 and 4 of the enclosed report.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and the enclosed inspection report will be placed in the NRC's Public Document Room. If the report contains any information that you believe to be proprietary, it is necessary that you submit a written application to this office, within 20 days of the date of this letter, requesting that such information be withheld from public disclosure. The application must include a full statement of the reasons why it is claimed that the information is proprietary. The application should be prepared so that

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Texas Utilities Generating
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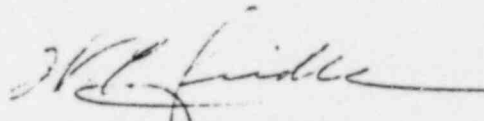
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March 23, 1979

any proprietary information identified is contained in an enclosure to the application, since the application without the enclosure will also be placed in the Public Document Room. If we do not hear from you in this regard within the specified period, the report will be placed in the Public Document Room.

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,



W. C. Seidle, Chief
Reactor Construction and
Engineering Support Branch

Enclosure:
IE Inspection Report No. 50-445/79-07
50-446/79-07

cc: w/enclosure
Texas Utilities Generating Company
ATTN: Mr. H. C. Schmidt, Project Manager
2001 Bryan Tower
Dallas, Texas 75201

yc: D. N. Chapman ✓

U. S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION IV

Report No. 50-445/79-07; 50-446/79-07

Docket No. 50-445; 50-446

Category A2

Licensee: Texas Utilities Generating Company
2001 Bryan Tower
Dallas, Texas 75201

Facility Name: Comanche Peak, Units 1 & 2

Inspection at: Comanche Peak Site, Glen Rose, Texas

Inspection conducted: March 8, 12 and 13, 1979

Inspectors:

A. B. Beach
for A. B. Beach, Reactor Inspector, Engineering
Support Section (Paragraphs 1, 2, 3, 5 & 6)

3/23/79
Date

A. B. Rosenberg
A. B. Rosenberg, Reactor Inspector, Engineering
Support Section (Paragraphs 1, 2, 3, 5 & 6)

3/23/79
Date

I. Barnes
I. Barnes, Contractor Inspector, Vendor
Inspection Branch (Paragraph 4)

3/23/79
Date

J. E. Ellershaw
J. E. Ellershaw, Contractor Inspector, Vendor
Inspection Branch (Paragraph 4)

3/23/79
Date

Other
Accompanying
Personnel:

R. E. Hall, Chief, Engineering Support Section

Reviewed:

C. R. Otero
for C. R. Otero, Resident Reactor Inspector, Projects
Section

3/23/79
Date

7945150561

Approved: C. R. Oberly
for W. A. Crossman, Chief, Projects Section

3/23/79
Date

R. E. Hall
for R. E. Hall, Chief, Engineering Support Section

3/23/79
Date

Inspection Summary:

Inspection on March 8, 12 and 13, 1979 (Report No. 50-445/79-07; 50-446/79-07)

Areas Inspected: Routine, unannounced inspection of construction activities including review of ITT Grinnell documentation for austenitic stainless steel piping assemblies containing bends; observation of work; and the review of working drawings and procedures related to the corrective measures being performed as a result of the misorientation of the reactor vessel supports. The inspection involved forty-six inspector-hours on site by four NRC inspectors.

Results: No items of noncompliance were identified in the areas inspected.

DETAILS1. Persons ContactedPrincipal Licensee Employees

- *J. R. Ainsworth, TUGCO, QA Engineer
- *D. N. Chapman, TUGCO, QA Manager
- *D. E. Deviney, TUGCO, QA Technician
- R. E. Heim, G&H, Resident Engineer
- *H. O. Kirkland, TUSI/B&R, Project General Manager
- *R. G. Tolson, TUGCO, Site QA Supervisor

Other Personnel

- W. Hanshaw, Project Civil Engineer, Brown and Root (B&R)/TUSI
- R. Williams, Civil Engineer, TUSI
- G. Westlund, Civil Engineer, B&R

The IE inspectors also interviewed other licensee and contractor employees during the course of the inspection.

*denotes those attending the exit interview.

2. Site Tour

The IE inspectors toured the various areas of the site to observe construction activities in progress in both Unit 1 and Unit 2.

No items of noncompliance or deviations were identified.

3. Misorientation of the Reactor Vessel Support Structures - Unit 2

The IE inspectors observed activities relative to the relocation of the structural concrete support for the reactor vessel steel support plates in the Unit 2 Containment Building. The reactor vessel concrete support structure is misoriented approximately forty-five degrees. By design, the reinforcing steel, which supports the reactor vessel support plates, is of a greater design strength than the reinforcing steel which is beneath the unsupported reactor vessel nozzles. Therefore, there will be a loss of shear strength in those areas where the concrete base for the steel support plates are to be relocated. Hence, the design repairs have centered on the placement of additional shear reinforcement in the areas where the steel support plates are to be relocated.

This additional reinforcement will consist of 5-1/2" x 5-1/2" x 1" steel plates attached by B series Cadwelds to #11 steel reinforcing bars. These reinforcing bars, in turn, will be embedded by grout into 2-1/2 inch diameter holes which have been drilled into the existing concrete approximately 4 feet in depth. These reinforcement plates will be arranged in a matrix pattern in the areas beneath the location of each reactor vessel steel support plate.

The IE inspectors reviewed the documentation which precipitated the design changes, DCA 3872, Rev. 1. The drawings which reflected these design changes, drawing numbers 2323-S2-0572-3, 4 and 5, were also reviewed. Actual work being performed in the reactor vessel cavity at the time of the inspection was observed to be in compliance with these drawings.

An investigation of Procedure 35-1195-CCP-12, Rev. 2, "Concrete Patching, Finishing and Preparation of Construction Joints," revealed a discrepancy between the drawings and the procedure. The procedure required that the #11 steel reinforcing bars be embedded with grout into 3 inch diameter holes per Interim Change 2. The drawings and the actual hole diameters, which have been drilled into the existing concrete in the reactor vessel cavity, show that the #11 steel reinforcing bars are to be grouted into 2-1/2 inch diameter holds. This procedure is now being revised to reflect the 2-1/2 inch diameter size.

The justification for using 2-1/2 inch diameter holes versus 3 inch diameter holes was of some concern to the inspectors; however, investigation of the Gifford-Hill supreme grouting procedures and investigation of site pull tests for the Gifford-Hill supreme grout indicated that there is a sufficient margin between the break strength of the reinforcing bar and the required strength of the redesign of the concrete support structure to compensate for any possible loss of bonding due to the smaller sized 2-1/2 inch diameter hole.

The procedure for replacing reinforcing steel which has been severed due to the drilling of the holes into the existing concrete must be determined. The IE inspectors, in discussions with licensee and contractor personnel, discovered that there are two separate views as to how this reinforcement steel is to be replaced. One view is to replace steel equal to an amount that assumes all the steel in the existing concrete has been severed. The other view is to replace only that reinforcement steel which has been documented to have been severed by the drilling. This could be a problem, as the unavailability of documentation would make it difficult to determine exactly what existing reinforcing steel has been severed. The replacement steel will be placed in the elevations from 819 feet to 824 feet in the area directly below the reactor vessel nozzles. This item is considered to be unresolved.

4. ITT Grinnell Piping Assemblies Containing Bends

An inspection was performed on March 8, 1979, by two Vendor Inspection Branch (VIB) IE inspectors to determine the heat treatment condition of bends in austenitic stainless steel assemblies that had been received at the Comanche Peak Site from ITT Grinnell Industrial Piping, Incorporated (GCO). This inspection resulted from the discovery by a VIB IE inspector of austenitic stainless steel assemblies in active fabrication at GCO for the Comanche Peak Site, which contained hot bends that had not been processed to require a subsequent solution annealing heat treatment.

GCO documentation packages were reviewed for eighteen austenitic stainless steel assemblies that were observed in storage to contain bends. The IE inspectors established that there was no documentation at the site to indicate that a solution annealing heat treatment had been performed on bends in five of the assemblies (i.e., Piece Mark Nos. CT-1-RB-43-5, SF-1-SB-03-1, CT-2-RB-541, CT-1-RB-46-5 and SI-1-RB-33-2).

The compliance of GCO with the solution annealing heat treatment requirements of the applicable Design Specification, Gibbs & Hill Specification No. 2323-MS-43A, Revision 3, could not be established at the site for the five assemblies, in that the GCO Operations Record (which is the manufacturing process control document that defines whether a specific bend was made hot or cold) was not obtained from GCO as part of the vendor documentation requirements.

This matter is considered unresolved pending the establishment of those system assemblies containing bends, identification of GCO practices used for performing the bends and verification that a solution annealing heat treatment has been performed on all hot bends and those cold bends made at a radius of less than five pipe diameters.

5. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance, or deviations. One item relating to the misorientation of the reactor vessel support is identified in paragraph 3 and one item relating to stainless steel pipe bends which were not solution annealed is in paragraph 4.

6. Exit Meeting

The IE inspectors met with licensee representatives (denoted in paragraph 1) at the conclusion of the inspection on March 13, 1979. The IE inspectors summarized the scope and findings of the inspection.



UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 1000
ARLINGTON, TEXAS 76012

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ATTACHMENT 9
JUN 20 1979
TUSI
NUCLEAR DIV.

June 18, 1979

In Reply Refer To:

RIV

Docket No. 50-445/Rpt. 79-13

50-446/Rpt. 79-13

RECEIVED

JUN 21 1979

Texas Utilities Generating Company
ATTN: Mr. R. J. Gary, Executive Vice
President and General Manager
2001 Bryan Tower
Dallas, Texas 75201

TUGCO QA
DALLAS

Gentlemen:

This refers to the inspection activities performed by our Resident Inspector, Mr. R. G. Taylor, during the period May 14-31, 1979, of activities authorized by NRC Construction Permit Nos. CPPR-126 and 127 for the Comanche Peak facility, Units No. 1 and 2, and to the discussion of our findings with Mr. R. G. Tolson and other members of your staff during the inspection period.

Areas examined during the inspection and our findings are discussed in the enclosed inspection report. Within these areas, the inspection consisted of selective examination of procedures and representative records, interviews with personnel, and observations by the inspector.

Within the scope of the inspection, no items of noncompliance were identified.

Two new unresolved items are identified in paragraph 3 of the enclosed report.

We have also examined actions you have taken with regard to previously identified inspection findings. The status of these items is identified in paragraph 2 of the enclosed report.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and the enclosed inspection report will be placed in the NRC's Public Document Room. If the report contains any information that you believe to be proprietary, it is necessary that you submit a written application to this office, within 20 days of the date of this letter, requesting that such information be withheld from public disclosure. The application must include a full statement of the reasons why it is claimed that the information is proprietary. The application should be prepared so that any proprietary information identified is contained in an

7948470125

Texas Utilities Generating Company

-2-

June 18, 1979

enclosure to the application, since the application without the enclosure will also be placed in the Public Document Room. If we do not hear from you in this regard within the specified period, the report will be placed in the Public Document Room.

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,



W. C. Seidle, Chief
Reactor Construction and
Engineering Support Branch

Enclosure:

IE Inspection Report No. 50-445/79-13
50-446/79-13

cc: w/enclosure
Texas Utilities Generating Company
ATTN: Mr. H. C. Schmidt, Project Manager
2001 Bryan Tower
Dallas, Texas 75201

U. S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION IV

Report No. 50-445/79-13; 50-446/79-13

Docket No. 50-445; 50-446

Category A2

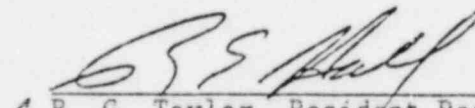
Licensee: Texas Utilities Generating Company
2001 Bryan Tower
Dallas, Texas 75201

Facility Name: Comanche Peak, Units 1 & 2

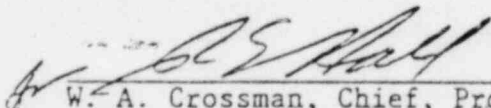
Inspection at: Comanche Peak Steam Electric Station, Glen Rose, Texas

Inspection conducted: May 14-31, 1979

Inspector:


R. G. Taylor, Resident Reactor Inspector, Projects Section6/15/79
Date

Approved:


W. A. Crossman, Chief, Projects Section6/15/79
DateInspection Summary:Inspection on May 14-31, 1979 (Report No. 50-445/79-13; 50-446/79-13)Areas Inspected: Routine inspection by the Resident Reactor Inspector (RRI) of safety related construction activities including installation and welding of reactor coolant and other piping systems, piping system hangers, containment penetrations; and follow up on various unresolved items. The inspection involved fifty-six inspector-hours by the Resident Reactor Inspector.Results: No items of noncompliance or deviations were identified.

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DETAILS1. Persons Contacted

*R. G. Tolson, TUGCO, Site QA Supervisor

*J. V. Hawkins, TUGCO/G&H Product Assurance Supervisor

*Denotes those who attended the management interviews.

The RRI also interviewed and had discussions with various persons employed by the constructor, Brown & Root, in quality control and construction capacities.

2. Licensee Action on Previous Findings

(Closed) Unresolved Item (50-446/79-07): Misorientation of Reactor Vessel Support Structures. The RRI observed, during the course of several inspections, that the reactor vessel supports have been placed in their correct positions and the necessary shear steel has been grouted in place. The reinforcing steel added to replace the reinforcement cut during the drilling for the shear steel has been placed as designed by the engineer. The work has progressed to a point where the normal work activities will progress toward placement of the reactor vessel in the near future.

3. Potential Significant Construction Deficiencies (10 CFR 50.55(e))

- a. On May 18, 1979, the licensee reported that it appeared that an unidentified quantity of pipe hangers designed by ITT-Grinnell may not meet seismic criteria. It appears that a method of computing seismic stresses may not have been appropriately conservative as compared to a newer computer program currently in use by Gibbs & Hill, the project A/E. This item will be considered an unresolved matter pending receipt of further information regarding the extent of the problem.
- b. On May 22, 1979, the licensee reported that the A/E had discovered that pipe wall thickness requirements established in the project piping specifications may not have been adequately conservative to meet ASME Code stress requirements. The licensee is evaluating the extent of the problem and corrective action requirements. Until such information is available, this matter shall be considered unresolved.

4. Safety Related Pipe Installation and Welding

The RRI observed a portion of the welding of field weld FW-5 as shown on isometric drawing BRP-SI-1-RB-037-0 in line 10-SI-103-2501R2. The line will run from safety injection accumulator TBX-SIATAT-01 through several valves and connect to the reactor coolant loop. The weld was being performed by an automatic GTAW machine in accordance with qualified Welding Procedure 99025-2. Qualified welding machine operators BAL and AIZ were controlling the process. Weld material was issued on Weld Material Requisition A019881 covering Sandvik heat 745599. The RRI reviewed Certified Material Test Reports for the weld material and examined the fabrication records for the two pipe spools being joined by weld FW-5. All documentation was found to be in accordance with the requirements of the project specifications and the ASME Code for Class 2 piping with the exception that there was no documentation to the effect that the -4 pipe spool had been solution annealed after hot bending. The licensee informed the RRI that the necessary documentation does exist at the vendor's facility (ITT-Grinnell) and will be forwarded to the site in the near future. The missing documentation of solution annealing was the subject of an unresolved item in Inspection Report No. 50-445/79-07 and will not be separately considered here. The RRI also reviewed the radiographic film for the single shop weld in pipe spool -3 as made by the vendor and the radiograph film for the examination of weld FW-5. Both sets of film displayed sensitivity meeting the requirements of ASME Section V and the welds met the acceptance criteria of ASME Section III for Class 2 welds.

No items of noncompliance or deviations were identified.

5. Reactor Coolant Pipe Welding

The RRI reviewed all of the radiographic examination films for reactor coolant loop welds FW-1, FW-2, FW-4, FW-6, FW-7, FW-9, FW-12 thru 14, FW-17, FW-18, FW-24, FW-25, and FW-27. All of these welds met the acceptance criteria of ASME Section III for Class 1 welds and the film displayed sensitivities required by ASME Section V. The only significant anomaly indicated by any of the radiographs was a linear indication outside of the weld zone in a reactor vessel "safe-end." This indication has been the subject of a potential significant construction deficiency as indicated in Inspection Report No. 50-445/79-01, paragraph 3.c. This indication, which appears to be entrapped slag, is on the borderline of being either acceptable or non-acceptable. It is understood that the license has made the decision to direct the vessel fabricator, Combustion Engineering, through the licensee's vendor, Westinghouse, to excavate the attachment, examine the indication and make appropriate repairs in the interest of safe and reliable operation of the reactor. The exact time frame for this work has not been fully determined. The RRI will follow this work closely as it occurs.

No items of noncompliance or deviations were identified.

6. Containment Mechanical Penetrations

The RRI selected containment mechanical penetration M1-4 as being typical of the larger penetrations. This particular unit is one of four main steam components. The component was fabricated by Gulf-Western Energy Products Group in accordance with Project Specification MS-74, "Mechanical Penetrations." The component, which is a single, relatively large forging, was fabricated from material meeting the requirements of SA-350, Grade 1F-2. According to the vendor documentation, as verified by the licensee, the forging was ultrasonically examined prior to being machined into its final shape and found satisfactory. The component was hydrostatically tested as required by the ASME Code for Class 2 components but was not Code stamped since no welding was involved in the fabrication. Under these conditions, the component is treated as material which is not required to be verified by an Authorized Nuclear Inspector.

No items of noncompliance or deviations were identified.

7. Review of Quality Assurance Procedures for Installation of Piping System Hangers

The RRI accomplished an extensive review of the licensee QA procedures for quality control inspection and verification of piping system supports and hangers. The FSAR in Section 3.2 commits the licensee to design, fabricate and install safety related piping systems in accordance with ASME Section III, including the Summer 1974 Addenda. Section III includes Sub-section NF, "Component Supports." With this document as a basis, the RRI reviewed Project Specification MS-46A, "Nuclear Safety Class Pipe Hangers and Supports." This document appeared to be consistent with the requirements of the Code document. The licensee's production and QC program is contained within CP-CPM-6.9, "General Piping and Inspection Procedure," and referenced supplementary procedures. The QC program depicted by CP-CPM-6.9 appeared to be adequate for the intended purpose; i.e., provide instructions for inspection of hanger installations and documentation thereof.

No items of noncompliance or deviations were identified.

8. Site Tours

The RRI toured one or more plant areas several times weekly during the reporting period to observe the progress of construction and the general practices involved, particularly in regard to facility housekeeping. Two of the tours were made during portions of the labor force second shift.

No items of noncompliance or deviations were identified.

9. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance, or deviations. Two such items are identified in paragraph 3 and will be referred to in future reports as:

50-445/79-13 - Pipe Hanger Seismic Design

50-445/79-13 - Pipe Wall Thickness Criteria

10. Management Interviews

The RRI met with licensee representatives (denoted in paragraph 1) on May 18, 1979, to discuss findings which had developed during the period. In addition, the RRI met informally nearly every day during the period with the licensee site QA supervisor to discuss items of immediate interest such as significant construction deficiencies. Included within these informal discussions has been the continuing progress of the program offered by the licensee to improve pipe welding activities (see Inspection Report No. 50-445/79-03). Licensee generated statistical data and other day-to-day data available to the RRI indicate a significant improvement in welding performance with a comparable significant reduction in the amount of repair work required to obtain acceptable welds. The licensee representative reiterated that all welds accepted by him will have fully complied with the ASME Code requirements and commitments made in the FSAR.



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAY 15 1979

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MAY 29 1979

TUSI
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DOCKET NOS. 50-445
AND 50-446

APPLICANT: TEXAS UTILITIES GENERATING COMPANY

FACILITY: COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 & 2

SUBJECT: SUMMARY OF MARCH 27, 1979 MEETING ON REPAIR OF REACTOR VESSEL SUPPORT
PEDESTAL FOR COMANCHE PEAK, UNIT 2

Summary

A meeting was held with representatives of the Texas Utilities Generating Company on March 27, 1979 in Bethesda, Maryland. The purpose of the meeting was to discuss the repair procedures for relocating the misoriented vessel support pads on Comanche Peak Steam Electric Station, Unit 2. The discussion was directed primarily at how the design of the repaired pedestal will differ from the original design of the reactor vessel support pedestal. The applicant described the repair design for the load carrying members as structurally equivalent and physically similar to those on the original design on Unit 1. A meeting attendance list is enclosed.

Background

The misorientation of the reactor vessel support structure at Comanche Peak, Unit 2 was described in our "Preliminary Notification of Event or Unusual Occurrence" PNO-79-028, issued by the Office of Inspection and Enforcement on February 22, 1979. The applicant reported that the reactor vessel support pedestal was being constructed such that its mounting pads would not mate with the support pads on the reactor vessel. The architect-engineer designed many features of the Unit 2 containment building, including the reactor vessel pedestal, as a mirror image of Unit 1. However, the nuclear steam system supplier did not design the primary coolant systems mirror image for this two unit station. The reactor vessel supplied for Unit 2 is a duplicate of the reactor vessel installed in Unit 1. As a result, the mounting pads on the Unit 2 pedestal are misoriented around the reactor vessel's vertical axis by about 45 degrees. The applicant elected to modify the top of the pedestal to install the mounting pads in the proper orientations. Construction had progressed to elevation 819 feet, several feet below the support pad, when the error was discovered.

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MAY 15 1979

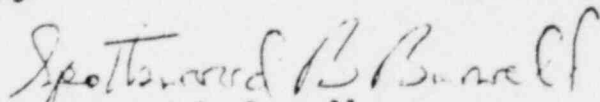
Meeting Details

The original design for the Unit 2 pedestal anticipated that the vessel would be supported on four pads which are approximately equally spaced around the periphery of the vessel. The revised design shifts the location of each pad 45 degrees about the vertical axis of the reactor vessel. The reactor vessel support pedestal is a reinforced concrete structure which by design includes a greater density of reinforcing steel beneath the four vessel support pads. Shifting the support pads 45 degrees therefore requires the installation of additional reinforcing steel under the new pad locations. The applicant was strengthening the pad locations by drilling 64 holes in the face of the pedestal under each new pad location and grouting in #11 rebar. The holes are spaced on about 10 inch centers and extend almost five feet into the pedestal. Each new pad position also will have about 12 vertical #9 rebar installed in a similar fashion. The applicant states that the steel installed beneath the new pad position is equivalent to the initial design for this service, and that the new pads are designed to support the same loads as the pads on Unit 1.

The applicant advised that the modification of the Unit 2 pedestal required that the ventilation ducts to the reactor vessel supports be extended to the new support locations. This did not cause any change in the ventilation systems performance, and was of no safety significance.

The applicant considers this repair of the Unit 2 reactor vessel support pedestal a field design change, having no safety impact. The applicant further does not consider the repaired pedestal to be structurally different from the Unit 1 design and therefore does not plan to make separate structural load analyses for each unit.

No unresolved safety concerns associated with the repair design for the Unit 2 pedestal were identified at the meeting.


Spottswood B. Burwell
Light Water Reactors Branch No. 2
Division of Project Management

Enclosure:
Attendance List

ccs w/enclosure:
See next page

MAY 15 1977

Texas Utilities Generating Company

ccs:

Nicholas S. Reynolds, Esq.
Debevoise & Liberman
1200 Seventeenth Street
Washington, D.C. 20036

Spencer C. Relyea, Esq.
Worsham, Forsythe & Sampels
2001 Bryan Tower
Dallas, Texas 75201

Mr. Homer C. Schmidt
Project Manager - Nuclear Plants
Texas Utilities Generating Company
2001 Bryan Tower
Dallas, Texas 75201

Mr. H. R. Rock
Gibbs and Hill, Inc.
393 Seventh Avenue
New York, New York 10001

Mr. A. T. Parker
Westinghouse Electric Corporation
P. O. Box 355
Pittsburgh, Pennsylvania 15230

Mr. R. J. Gary
Executive Vice President
and General Manager
Texas Utilities Generating Company
2001 Bryan Tower
Dallas, Texas 75201

MAY 15 1979

ATTENDANCE LIST

COMMANCHE PEAK STEAM ELECTRIC STATION

MARCH 27, 1979

NRC - STAFF

S. B. Burwell
F. Rinaldi
R. E. Shewmaker

TEXAS UTILITIES GENERATING COMPANY

H. C. Schmidt

GIBBS & HILL

H. R. Rock
E. G. Gibson
E. L. Bezkor
Cherim Zion



UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 1000
ARLINGTON, TEXAS 76012

ATTACHMENT 11

November 27, 1979

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NOV 28 1979

R. J. GARY

In Reply Refer To:

RIV

Docket No. 50-445/Rpt. 79-24
50-446/Rpt. 79-23

Texas Utilities Generating Company
ATTN: Mr. R. J. Gary, Executive Vice
President and General Manager
2001 Bryan Tower
Dallas, Texas 75201

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NOV 28

TUCCO 04
DALLAS

Gentlemen:

This refers to the inspection conducted by our Resident Inspector, Mr. R. G. Taylor, during the period of October 1979 of activities authorized by NRC Construction Permits No. CPPR-126 and 127 for the Comanche Peak facility, Units No. 1 and 2, and to the discussion of our findings with Mr. R. G. Tolson and other members of your staff during the inspection period.

Areas examined during the inspection and our findings are discussed in the enclosed inspection report. Within these areas, the inspection consisted of selective examination of procedures and representative records, interviews with personnel, and observations by the inspector.

Within the scope of the inspection, no items of noncompliance were identified.

We have also examined actions you have taken in regard to a previously identified finding. The status of this item is identified in paragraph 2 of the enclosed report.

Two new unresolved items are identified in paragraphs 2 and 7.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and the enclosed inspection report will be placed in the NRC's Public Document Room. If the report contains any information that you believe to be proprietary, it is necessary that you submit a written application to this office, within 20 days of the date of this letter, requesting that such information be withheld from public disclosure. The application must include a full statement of the reasons why it is claimed that the information is proprietary. The application should be prepared so that any proprietary information identified is contained in an enclosure to the application, since the application without the enclosure

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Texas Utilities Generating Company

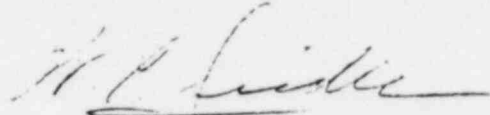
-2-

November 27, 1979

will also be placed in the Public Document Room. If we do not hear from you in this regard within the specified period, the report will be placed in the Public Document Room.

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,



W. C. Seidle, Chief
Reactor Construction and
Engineering Support Branch

Enclosure:

IE Inspection Report No. 50-445/79-24
40-446/79-23

cc: w/enclosure
Texas Utilities Generating Company
ATTN: Mr. H. C. Schmidt, Project Manager
2001 Bryan Tower
Dallas, Texas 75201

yc: D. N. Chapman ✓
L. J. Hampton

U. S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION IV

Report No. 50-445/79-24; 50-446/79-23

Docket No. 50-445; 50-446

Category A2

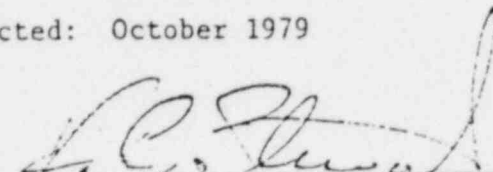
Licensee: Texas Utilities Generating Company
2001 Bryan Tower
Dallas, Texas 75201

Facility Name: Comanche Peak, Units 1 & 2


Inspection at: Comanche Peak Steam Electric Station, Glen Rose, Texas

Inspection conducted: October 1979

Inspector:


R. G. Taylor, Resident Reactor Inspector, Projects
Section11/23/79
Date

Approved:


W. A. Crossman, Chief, Projects Section11/23/79
DateInspection Summary:Inspection during October 1979 (Report No. 50-445/79-24; 50-446/79-23)Areas Inspected: Routine inspection by the Resident Reactor Inspector (RRI) of construction progress and practices; concrete construction activities; piping system installation and welding; storage and maintenance of equipment; construction fire protection; electrical cable installation; and followup on previous inspection findings. The inspection involved sixty-nine inspector-hours by one NRC inspector.Results: No items of noncompliance or deviations were identified.

2001 BRYAN TOWER

DETAILS1. Persons ContactedPrincipal Licensee Employees

- *R. G. Tolson, TUGCO, Site QA Supervisor
- *J. B. George, TUSI, Project General Manager
- *J. R. Merritt, TUSI, Construction and Engineering Manager
- *D. N. Chapman, TUGCO, Quality Assurance Manager

Brown and Root Employees

- *U. D. Douglas, Construction Project Manager
- J. P. Clarke, Project QA Manager
- J. V. Hawkins, QC Supervisor

The RRI also interviewed other licensee and Brown and Root employees during the inspection period.

*Denotes those persons with whom the RRI held on-site management meetings.

2. Action on Previous Inspection Findings

(Closed) Infraction (50-445/79-11): Failure to Implement the Quality Assurance Program for Civil Construction. The licensee notified RIV by a letter dated September 17, 1979, that all contemplated actions by his consultants and the Architect/Engineer have been completed and that the in-place concrete was found satisfactory. The substantiating data to support this contention were reviewed by the RRI and personnel of the RIV Engineering Support Section. The IE inspectors found that the stated positions were essentially based upon an examination of the in situ concrete by a sonic technique. This technique was developed, used and is interpreted by only one person in the industry and, as such, is not verifiable by any other party.

Pending some additional and verifiable assurance that the in situ concrete has the necessary qualities required by the design, this matter will be considered as an unresolved item.

(Closed) Unresolved Item (50-445/79-13): Potential Deficiency Regarding Design of Pipe Supports. The licensee notified RIV by letter dated September 8, 1979, that this matter had been investigated and deemed to be not reportable within the context of 50.55(e). Supporting data reviewed by the RRI and discussions with cognizant site personnel substantiated this determination.

The RRI had no further questions on this matter.

3. Site Tours

The RRI toured the safety-related plant areas several times during the inspection period to observe the progress of construction and the general practices involved. Three of these tours were conducted during portions of the construction labor second shift which continues to be relatively small and substantially devoted to electrical installation activities.

During several of these tours, the RRI observed a general deterioration in plant area housekeeping. This matter was brought to the attention of licensee management who responded immediately. The construction force was directed to cleanup and remove the accumulated construction debris which was promptly done.

No items of noncompliance or deviations were identified.

4. Concrete Construction Activities

The RRI observed a portion of the concrete placement activities for the Unit 2 dome. This placement, identified as 201-8805-013, was the final placement in the Unit 2 containment shell exclusive of the construction opening.

The RRI observed the preparation of the concrete at the batch plant and the condition of the cement and aggregate storage activities. The RRI also observed the transportation of the concrete to the placement area via trucks and two yard buckets including performance of required tests for slump, temperature and air content of the fresh material.

On October 24, 1979, at approximately 11:15 a.m., the RRI received a call on the plant area telephone system. The caller, who refused to identify himself, stated that he and several other persons, also unidentified, had overheard the Brown and Root QC inspector say, "I didn't inspect this placement, but since the trucks are here go ahead." The caller said that the pour was in progress inside Reactor Building 2. The RRI went immediately to the placement area, which was a portion of an interior wall, and discussed the accusation with the QC inspector of record. The QC inspector promptly and emphatically denied having made the statement and stated positively that he had inspected the placement area.

The RRI asked that all personnel associated with the activity be made available for an interview. A subsequent and longer interview with the QC inspector of record indicated that he had inspected the area on October 23, 1979, and was satisfied, except for cleanup, an element which was satisfactorily verified between 6:00 a.m. and 7:00 a.m. on October 24, 1979. The inspector did relate that he found one small area of the placement that had to be fully inspected just prior to initial delivery of concrete which was held up for a few minutes. This occurred, the

inspector said, because he had misconstrued the exact placement boundaries. The inspector indicated that he had discussed the localized lack of inspection with a craft general foreman in charge when he discovered his error and was immediately informed as to what the boundary really was. The QC inspector reiterated that the entire placement area had been properly inspected prior to initiation of concreting.

A second B&R QC inspector who had assisted the inspector of record between 6:00 a.m. and 7:00 a.m. could shed no light on the quality of the inspection on October 23, 1979, but stated that the placement area was clean and ready prior to placement. He also indicated that he was aware of the inspector of record's problem with the small uninspected area but had no reason to raise a question since he had observed that the area had finally been inspected.

The RRI subsequently interviewed some seventy-four persons of the labor force who might have possibly overheard the alleged conversation or might have some knowledge of the quality of the placement. With two exceptions, no one admitted to being a party to or overhearing the alleged conversation. One of two exceptions was the previously referenced general foreman who recalled the conversation with the inspector of record about the small uninspected area and the short ensuing delay, but could not recall the exact words used. The other exception was a carpenter crew foreman who said that he overheard a portion of a conversation between the general foreman and the inspector. The foreman stated that to the best of his recollection the inspector said, "I didn't inspect that, but I'll get on it," and indicated that the inspector was pointing to an area of the placement. The foreman was aware that the placement was held up shortly for QC to finish inspecting the area.

The various general foremen and foremen actively involved in the placement activity and cleanup process stated that they had observed and assisted the inspector of record on October 23 and October 24, 1979, and had no question as to his thoroughness. A few workers substantiated this review. Most of the workers indicated that they were not in a position to have had any specific knowledge relative to the quality of the inspections.

Based upon the results of the interviews and upon the lapsed time between when the conversation had to have taken place; ie., approximately 7:00 a.m. and the receipt of the phone call (11:15 a.m.), the RRI can only conclude that the call was a hoax. The purpose of the hoax could not be identified.

No items of noncompliance or deviation were identified.

5. Piping Systems Installation and Welding

The RRI observed the general handling and installation of Reactor Coolant Pressure Boundary and other safety-related piping system components during the inspection period. These activities were accomplished in accordance

with good industry practices. The RRI examined the following weld joint radiographs for conformance to the requirements of ASME Section III:

<u>Joint Number</u>	<u>Isometric Drawing</u>	<u>Line Number</u>
FW-19	BRP-RC-1-520-1	Reactor Main Loop
FW-14	BRP-RC-1-520-1	Reactor Main Loop
FW-20	BRP-RC-1-520-1	Reactor Main Loop
FW-21	BRP-RC-1-520-1	Reactor Main Loop
FW-22	BRP-RC-1-520-1	Reactor Main Loop
FW-29	BRP-RC-1-520-1	Reactor Main Loop
FW-14-1	BRP-S1-1-RB-21	3-S1-1-339-2501R1
FW-1	BRP-S1-1-RB-053	6-S1-1-329-2501R1
FW-2	BRP-RH-1-RB-002	12-RH-1-002-2501R1
W-4	BRP-RC-1-RB-05	6-RC-1-008-2501R1
W-2	BRP-RC-1-RB-028B	6-RC-1-096-2501R1
W-3	BRP-RC-1-RB-028B	6-RC-1-096-2501R1
W-4	BRP-RC-1-RB-028B	6-RC-1-096-2501R1
W-6	BRP-RC-1-RB-028B	6-RC-1-096-2501R1
W-7	BRP-RC-1-RB-028B	6-RC-1-096-2501R1
W-14	BRP-RC-1-RB-028B	6-RC-1-096-2501R1
W-16	BRP-RC-1-RB-028B	6-RC-1-096-2501R1
W-18	BRP-RC-1-RB-028B	6-RC-1-096-2501R1
W-20	RRP-RC-1-RB-028B	6-RC-1-096-2501R1
W-21	BRP-RC-1-RB-028B	6-RC-1-096-2501R1
W-2	BRP-RC-1-RB-028A	6-RC-1-108-2501R1
W-3	BRP-RC-1-RB-028A	6-RC-1-108-2501R1
W-5	BRP-RC-1-RB-028A	6-RC-1-108-2501R1

W-6	BRP-RC-1-RB-028A	6-RC-1-108-2501R1
W-7	BRP-RC-1-RB-028A	6-RC-1-108-2501R1
W-9	BRP-RC-1-RB-028A	6-RC-1-108-2501R1
W-10	BRP-RC-1-RB-028A	6-RC-1-108-2501R1
W-17	BRP-RC-1-RB-028A	6-RC-1-108-2501R1
W-10	BRP-S1-1-RB-017	6-S1-102-2501R1
W-9	BRP-S1-1-RB-017	6-S1-102-2501R1
W-12	BRP-S1-1-RB-017	6-S1-102-2501R1
FW-2	BRP-S1-1-RB-053	6-S1-1-330-2501R1
FW-6	BRP-CT-2-RB-09	16-CT-2-014-301R2

The six weld joints noted above as being in the Reactor Main Coolant Loop are the last of thirty-two field welded connections in the Unit 1 Main Loop piping.

No items of noncompliance or deviations were identified.

6. In-place Storage and Maintenance of Safety-Related Components

The RRI randomly selected several mechanical and electrical components during the period to observe the storage and maintenance practices being employed. Among these components were safety-related motor operated valves, main control boards, switchgear cabinets, heat exchangers, reactor pressure vessels in both units and the Unit one Reactor Vessel internals. Each of the components observed were protected by adequate covering and were being maintained in a manner commensurate with supplier instructions and/or good industry practice.

No items of noncompliance or deviations were identified.

7. Construction Fire Protection

The RRI verified that an adequate number of portable fire extinguishers displaying a properly charged condition were present in areas where welding and/or flame cutting operations were observed. The RRI observed on one occasion that welding operations of a structural nature were being carried on above a cable tray containing safety-related electrical cable that was unprotected from the weld spatter. Although none of spatter fell on the cable during the sustained period of observation, this was judged to be more of a fortunate accident than a deliberate action. The RRI ascertained that at present there is no coordinated inter-craft method of controlling such welding operations. The RRI discussed the

matter with licensee construction and Quality Assurance management, both indicated an awareness of the potential problem and stated that a control method would be developed.

This matter will be considered an unresolved item pending an opportunity to review and observe implementation of such controls.

8. Electrical Cable Installation

During this period, the RRI observed the installation of a three conductor, number 6 AWG safety train A cable. The cable which runs from motor control center IEB1 to the Channel static inverter was approximately 410 ft. in length going through various segments of cable tray and conduit runs. The RRI verified that the cable utilized was of the type specified and verified, on a selective basis, that the cable was being routed as shown on the engineer furnished cable pull card. The RRI observed a portion of the cable through conduit pulling operation for consistency with project procedures. The RRI interviewed and observed the activities of the QC inspector assigned to the activity. The QC inspector appeared to be knowledgeable of the requirements and diligent in his work effort.

As a result of a licensee management audit and review of the cable installation program, the licensee determined that it was desirable to stop all safety-related cable pulling activities to allow time for an in depth review of project specifications, construction procedures and quality control procedures along with a review of appropriate personnel qualifications. The review was initiated in the latter part of the period and will probably last two to four weeks according to the licensee provided information.

No items of noncompliance or deviations were identified.

9. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance or deviations. Two such items are discussed in this report. The applicable paragraph and item title reference are as follows:

Paragraph 2: Unit 1 Containment Dome Concrete

Paragraph 7: Protection of Installed Electrical Cable

10. Management Interviews

The RRI met with one or more of the persons identified in paragraph 1 on October 9, 11, 15, and 19, 1979, to discuss various inspection findings and to discuss licensee actions and positions.