322

DOCKETED 10/1/84
USNRC

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION '84 OCT -9 A10:46

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	
TEXAS UTILITIES GENERATING COMPANY, et al.	Docket Nos. 50-445-OL and 50-446-OL
(Comanche Peak Steam Electric Station Station, Units 1 and 2)	

CASE'S ANSWER TO APPLICANTS' 9/13/84 SUPPLEMENT TO
MOTION FOR AUTHORIZATION PURSUANT TO 10 C.F.R. 50.57(c)

Applicants' filed a Motion for Authorization to Issue a License to Load Fuel and Conduct Certain Precritical Testing on 8/7/84 (received by CASE on 8/8/84). CASE responded to Applicants' Motion on 8/18/84 (opposing it), and the NRC Staff responded on 8/22/84. On August 24, 1984, the Licensing Board issued its MEMORANDUM (Request for Evidence Relevant to Fuel Loading).

On September 13, 1984, Applicants filed their Supplement to Motion for Authorization Pursuant to 10 C.F.R. 50.57(c) (received by CASE on 9/14/84). In an off-the-record discussion initiated by CASE with Judge Bloch and Applicants' Counsel Nicholas Reynolds /1/ on 9/24/84, CASE sought and was granted (with no opposition from Applicants) leave to file our response to Applicants' Supplement after we had completed and filed our Answer to Applicants' pleading on A500 Steel (which the Board is treating as a Motion for Summary Disposition); it was agreed that CASE could file its response today, 10/1/84. (The NRC Staff counsel, Mr. Treby, had been advised earlier

/1/ It should be noted that the Board Chairman offered Applicants' counsel the option of the Board Chairman's not engaging in this off-the-record conference calls; however, Applicants' counsel stated that he had no objection.

DS03

in the day that CASE intended to contact the Board regarding this matter, but Staff counsel was unavailable at the time of the conference call.)

On Thursday, 9/27/84, CASE was contacted by Staff's counsel Mr. Scinto, with Mr. Mizuno also on the line, regarding the Staff's desire to postpone their response to Applicants' pleading until October 12, 1984. As CASE advised at that time, we have no objection to this postponement and will be back in touch should it appear that CASE also needs additional time to respond. CASE does, in fact, plan to supplement our response as soon as we have in hand additional information which we are currently preparing, and will ask at that time that the Board also consider this additional information. However, we wanted to go ahead and get the information contained in this pleading into the hands of the Board for their consideration.

Applicants' Response to the Board's Order Is Inadequate

In the Board's 8/24/84 Order, it stated, in part:

> ". . . the section [10 CFR 50.57(c), covering a license for low power testing] requires us to make the findings listed in 50.57(a) with respect to the contested activity sought to be authorized." (Emphasis in the original.)
>
> "The contested activities involve at least the following plant systems: (a) boron addition and monitoring equipment, (b) neutron monitoring equipment sufficient to detect significant increases in Keff above 0.95, (c) fuel handling equipment, and (d) reactor protection systems. Each of the components of these systems is relevant, including mechanical, electrical and instrumentation systems." (Emphasis added.)
>
> "Because of the broad quality control contention pending in this proceeding, we must have evidence concerning the adequacy of quality control for the contested systems. In particular, we require evidence concerning the current status of QA/QC oversight of these systems, including evidence that documentation is adequate to assure that unsatisfactory or non-conforming conditions have been corrected and evidence concerning whether or not there are allegations known to the applicants or Staff about the intimidation of QA/QC personnel who were working on these systems." (Emphases added.)

"We also require evidence: (1) that appropriate QA/QC procedures have been completed for all phases of the activities for which a license is sought, (2) concerning the maximum Keff to be permitted during pre-critical testing and the Keff that analysis suggests may be achieved during pre-critical testing if all control rods were inadvertently removed while the boron concentration was 2000 ppm, and (3) that non-borated water will never be injected into the core, substantially diluting the boron below 2000 ppm." (Emphasis added.)

". . . ORDERED:

"That Texas Utilities Electric Co., et al. shall supply the evidence requested in this order to facilitate further consideration of its Motion for Authorization to Issue a License to Load Fuel and Conduct Certain Precritical Testing."

CASE submits that Applicants' response to the Board's Order is inadequate, as discussed below.

Applicants' Witness Antonio Vega states in his 9/13/84 Affidavit (Attachment 1 to Applicants' Supplement, Affidavit of Antonio Vega Concerning Board Questions Regarding QA/QC Oversight), at page 2:

"In response to the Board's request, an evaluation of all plant systems was conducted to determine the systems that fell into the category specified by the Board, as noted above. Ten systems/equipment groupings were identified. These systems are listed in Attachment B. With regard to these systems, a thorough review was conducted to determine if all required inspections had been conducted and verified, as applicable. This review reflected that QC inspections have been performed and documented on the necessary mechanical, electrical and instrumentation components of these systems. These inspections include in-process inspections, final inspections, as-built verification inspections, and Authorized Nuclear Inspector (ANI) inspections, as applicable. Continuing reinspections will be made as appropriate to preserve the integrity of completed inspections." (Emphases added.)

To begin with, it appears that rather than making a genuine attempt to ascertain which other systems might need to be looked at in addition to those specifically mentioned by the Board, Applicants have instead used the Board's specified systems to limit their review.

Further, CASE challenges the statements made by Applicants' witness. It is obvious that Mr. Vega could not have personally thoroughly reviewed

such documentation himself (and he does not claim that he did so), especially in light of the fact that he was testifying (and it is reasonable to assume, preparing to testify) in the intimidation portion of these operating license proceedings during part of the just less than three weeks' time between the filing of the Board's Order (August 24, 1984) and the filing of Mr. Vega's Affidavit (September 13, 1984). It is therefore reasonable to assume that Mr. Vega is relying heavily upon someone else's actual review of the systems in question.

Further, based on the personal experiences of CASE's representatives and our witnesses, as well as on common sense and logic, regarding the amount of time necessary to thoroughly review such documentation, CASE submits that there is simply no way that Mr. Vega could have had others make the necessary evaluation to identify plant systems, then thoroughly review the necessary documentation (much less for him to have done so himself even to the extent that he would be able to state with certainty that his statements are accurate) in the less than three weeks' time. (Indeed, based upon recent past performance in regard to Applicants' ability to find, much less produce, documentation in the intimidation portion of the proceedings and in regard to Motions for Summary Disposition, one must seriously question Mr. Vega's statements.) Since no documentation was attached to support Applicants' assertions in this regard, CASE proposes a test of Applicants' assertions, as will be discussed later herein.

Mr. Vega further states (Affidavit at page 3):

> "In addition, an extensive testing program on these systems has been implemented and will be completed prior to fuel load, including, as applicable, hydrostatic tests on pressure retaining systems, prerequisite testing on components to assure proper component

> functional operability, and preoperational testing to assure proper operation as a system. Preoperational testing provides assurance that the systems in question will operate as designed by requiring demonstration testing of the capability of the systems to meet safety-related performance requirements. A summary of the preoperational testing for each of the ten systems in question is set forth in Attachment C."

It should be noted that Applicants' witness has not made even a minimal attempt to respond to the specific concerns raised by CASE regarding the neutron detectors and neutron detector wells at pages 19-23 in CASE's 8/18/84 Partial Answer (in Opposition to Applicants' Motion for Authorization to Issue a License to Load Fuel and Conduct Certain Precritical Testing and Motion for Additional Time to Respond). CASE submits that, absent documented proof to the contrary by Applicants, the Board should accept the documented information provided by CASE as being the current status of the items in question. Further, at least a portion of the matters discussed by CASE (item (1) on page 19) appears to have been confirmed by the NRC's Technical Review Team (TRT)'s request for additional information, attached to the 9/18/84 letter from Darrell G. Eisenhut, Director, Division of Licensing, NRR, to TUGCO's M. D. Spence, at page 7, item II.a. (*See* Attachment A hereto.) /1/

The representations in Mr. Vega's Affidavit (page 3) regarding the "extensive testing program" give the impression that (1) the testing will be completed prior to fuel load, and (2) all is well with the testing program. However, it appears that the TRT Report calls into question certain statements made by Mr. Vega regarding the testing program. *See* Attachment A hereto, pages 11-14. The information in the TRT Report takes on added

---

/1/ An inquiry from the Board to Mr. Ippolito's Technical Review Team regarding the amount of time necessary to *thoroughly* review documentation would in itself undoubtedly be very enlightening.

significance in light of the events leading up to and culminating in the Licensing Board's 10/1/84 Memorandum (Concerns About Start-Up Quality Assurance). /2/ We call the Board's attention especially to item III.b. Containment Integrated Leak Rate Testing (CILRT) at pages 13-14 of the TRT Report, which states, in part:

> "Apparently after repairing leaks found during the first two attempts, the third attempt at a CILRT was successful. It was successfully completed after three electrical penetrations were isolated because the leakage through them could not be stopped. Though the leaks were subsequently repaired and individually tested with satisfactory results, NRC approval was not obtained to perform the CILRT with these penetrations isolated. In addition, leak rate calculations were performed using ANSI-ANS 56.8, which is neither endorsed by the NRC nor in accordance with FSAR commitments." (Emphases added.)
>
> "Accordingly, TUEC shall identify to NRC any other differences in the conduct of the CILRT as a result of using ANSI/ANS 56.8 rather than ANSI N45.4-1972. Additionally, TUEC shall identify to NRC all other deviations from FSAR commitments."

Although CASE applauds the fact that the TRT identified this problem, a reading of the transcript and attachments of the TRT's 9/18/84 meeting with Applicants is not quite so encouraging. However, in any event, the Licensing Board must go beyond the additional information which the TRT is requiring Applicants to provide. The Board must also be concerned with how and why and at whose instigation this and other testing problems have been allowed to arise at this late date, as well as the extent of the QA/QC breakdown indicated by these events.

We also call the Board's attention to the attached 7/18/84 NRC Region IV Inspection and Enforcement (I&E) Report 84-21 advising of two Notices of Violation regarding preoperational testing and procedures, and the attached

---

/2/ We also invite the Board's attention once again to CASE's 10/13/83 (1) Motion to Add a New Contention, (2) Motion for Discovery, and (3) Offer of Proof, regarding hot functional and other preoperational and acceptance tests.

8/14/84 follow-up Report 84-21 (Attachments B and C hereto, respectively). Although there are several portions of these reports which are pertinent to the issue at hand (and we urge that the Board read them in their entirety), it should be especially noted that on page 8 of the Appendix attached to the 8/14/84 letter (Attachment C hereto), it is stated, in part:

> "Prior to issuance of this inspection report, the Notice of Violation was transmitted to the licensee as Severity Level IV Violation 445/8421-01. This is the <u>second violation issued in recent weeks pertaining to lack of procedure compliance</u>. The previous violation was identified as 445/8418-01 and contains three examples of failure to follow procedures. The licensee was made aware by the resident inspectors of the <u>importance of decisive permanent corrective action by senior management to prevent future procedure violations as the pace of testing and operations increases at CPSES</u>." (Emphases added.)

Again, although CASE is gratified that the NRC inspectors identified these problems, the Licensing Board must go beyond the future corrective actions being requested by the inspectors. The Board must also be concerned with <u>how</u> and <u>why</u> and <u>at whose instigation</u> this and other similar problems have been allowed to arise at this late date, as well as the extent of the QA/QC breakdown indicated by these events.

They also cast doubt on the statements made in Mr. Vega's Affidavit at page 3 regarding the "methods of documentation" which "assure positive control and tracking of such conditions to preclude inadvertent use of defective materials, components or systems," and about their timely resolution.

In addition, the Board should be aware that, contrary to the implications in Mr. Vega's affidavit regarding the status and adequacy of the testing program, the NRC Staff has stated that they will allow the Applicants to defer the following preoperational tests (<u>see</u> Attachment D hereto, 8/17/84 letter from B. J. Youngblood, Chief, Licensing Branch No. 1,

Division of Licensing, NRC, Washington, to M. D. Spence, TUGCO, under Subject: Acceptance of Preoperational Test Deferrals for Comanche Peak Steam Electric Station, Unit 1):

1. Containment Cooling Systems
2. Safety Injection System Check Valve Leakage
3. Turbine Drive Auxiliary Feedwater Pump Steam Supply Line Check Valve and Drain Pot Level Control Valve
4. Reactor Coolant Pump Seal Performance
5. Thermal Expansion Testing
6. Control Room Ventilation System

We're sure the Board will be as relieved as CASE was to find out that, according to the Staff's proposed findings for inclusion in a future SER supplement (see Attachment D hereto, Enclosure page 3, second paragraph):

> "The deferral of the thermal expansion retest is acceptable because it is consistent with approved industry practice on other plant test programs. . . " (Emphasis added.)

The Board should also be aware that the NRC Staff has recently approved an exemption from GDC 4 to the installation of jet impingement shields in Unit 1 (see Attachment E hereto, 8/28/84 letter from B. J. Youngblood, Chief, Licensing Branch No. 1, Division of Licensing, to M. D. Spence, TUGCO, under subject of: Applicants' Request for Exemption from a Portion of General Design Criterion 4 of Appendix A to 10 CFR Part 50 Regarding the Need to Analyze Large Primary Loop Pipe Ruptures as the Structure D gn Basis for Comanche Peak Steam Electric Station (Units 1 and 2), and attachment thereto; see also Attachment F hereto, pages 35058-35061 of the 9/5/84 FEDERAL REGISTER, especially page 35060, right-hand column, second paragraph of item (6)).

CASE also calls the Board's attention to the problems with the concrete which have been identified in the following:

1. Technical Review Team (TRT) Report, Attachment A hereto, pages 7 and 8, item II.b. Falsification of Concrete Compression Strength Test Results.
2. CASE's 9/10/84 Answer to Applicants' Motion for Summary Disposition Regarding Richmond Inserts, especially answer 8, pages 13-22, of Affidavit of CASE Witness Mark Walsh, and Attachment D thereto -- regarding compressive strength of concrete at Comanche Peak.
3. 8/24/83 Deposition of Arvill "J. R." Dillingham, Jr., pages 65-70, sent to Board and parties attached to 3/7/84 OI Case No. 4-84-006, re: Alleged Intimidation of QC Personnel -- regarding Unit 1 stainless steel liner hollow places.
4. NRC Staff's Request for Additional Information Concerning the Handling of Heavy Loads at Comanche Peak (Units 1 and 2) (<u>see</u> Attachment G hereto, letter from B. J. Youngblood, Chief, Licensing Branch No. 1, Division of Licensing, NRC, Washington, to M. D. Spence, TUGCO).
5. There are also many other documents in the record regarding defective concrete, etc., which we have not yet <u>thoroughly</u> analyzed (as we did recently regarding the NCR's on the compressive strength of the concrete).

It should also be noted that neither Applicants nor NRC Staff have yet responded to the Board's directive to address the substantive portions of

CASE's requests for information on drug-related terminations of QC inspectors, etc. It is unknown, therefore, what the effect may be on safety-related systems pertinent to Applicants' Motion to Load Fuel.

In addition to the preceding and the items discussed in our 8/18/84 Partial Answer, CASE is currently working on additional pleadings which also should be considered in regard to Applicants' Motion to load fuel. We hope to have them in the hands of the Board and parties within the next week.

An additional area of concern is the the adequacy and safety of the fuel pool, refueling cavity, transfer canal, stainless steel liner plates, etc., for both Units 1 and 2 and the transfer canal. (<u>See</u> CASE's 9/27/84 Evidence of A Quality Control Breakdown, attachments thereto, and referenced documents.) In addition, the NRC Technical Review Team has expressed concern regarding alleged unauthorized cutting of rebar in the Fuel Handling Building (<u>see</u> Attachment A hereto, page 11, item II.e.).

As discussed herein, there are far too many unanswered (as well as unfavorably answered) questions for the Board to allow Applicants to load fuel at this point in time or even in the near future, since it will obviously take some time for these matters to be resolved.

With regard to the fuel pool liners, transfer canal, etc., <u>CASE moves</u> that the Board order Applicants to provide CASE with complete documentation regarding the fuel pool, refueling cavity, transfer canal, stainless steel liner plates, Unit 1, etc. (similar to the documents provided recently in

the intimidation portion of these proceedings for only Unit 2). As discussed on page 4 of this pleading, CASE proposes that this discovery serve the additional purpose of testing the representations made by Applicants' Witness Vega regarding the thorough review of documentation in response to the Board's 8/24/84 Order. This can be done quite simply by having CASE's witnesses and representatives keep up with the amount of time necessary to thoroughly review the documents and provide the Board and parties with an analysis of the results.

CASE further moves that the Board accept the following documents into the record (see detailed list at end of this pleading): Attachments A, B, C, and E hereto.

Respectfully submitted,

Juanita Ellis

(Mrs.) Juanita Ellis, President
CASE (Citizens Association for Sound Energy)
1426 S. Polk
Dallas, Texas 75224
214/946-9446

Attachments:

Attachment A	9/18/84 letter from Darrell G. Eisenhut, Director, Division of Licensing, NRR, to TUGCO's M. D. spence, to which is attached the NRC's Technical Review Team (TRT) Report -- see page 5 of this pleading
Attachment B	NRC Inspection and Enforcement (I&E) Report 84-21 under cover letter of 7/18/84 -- see page 6 of this pleading
Attachment C	NRC Inspection and Enforcement (I&E) follow-up Report 84-21 under cover letter of 8/14/84 -- see page 6 of this pleading

Attachment D	8/17/84 letter from B. J. Youngblood, Chief, Licensing Branch No. 1, Division of Licensing, NRC, Washington, to M. D. Spence, TUGCO, under Subject: Acceptance of Preoperational Test Deferrals for Comanche Peak Steam Electric Station, Unit 1 -- see pages 7 and 8 of this pleading
Attachment E	8/28/84 letter from B. J. Youngblood, Chief, Licensing Branch No. 1, Division of Licensing, to M. D. Spence, TUGCO, under subject of: Applicants' Request for Exemption from a Portion of General Design Criterion 4 of Appendix A to 10 CFR Part 50 Regarding the Need to Analyze Large Primary Loop Pipe Ruptures as the Structure Design Basis for Comanche Peak Steam Electric Station (Units 1 and 2), and attachment thereto -- see page 8 of this pleading
Attachment F	Pages 35058-35061 of 9/5/84 FEDERAL REGISTER, granting exemption discussed in Attachment E preceding -- see page 8 of this pleading
Attachment G	9/21/84 letter from B. J. Youngblood, Chief, Licensing Branch No. 1, Division of Licensing, NRC, Washington, to M. D. Spence, TUGCO -- see page 9 of this pleading

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

ATTACHMENT A

SEP 13 1984

Dockets: 50-445
50-446

Texas Utilities Electric Company
Attn: M. D. Spence, President, TUGCO
Skyway Tower
400 North Olive Street
Lock Box 81
Dallas, Texas 75201

Dear Mr. Spence:

SUBJECT: COMANCHE PEAK REVIEW

On July 9, 1984, the staff began an intensive onsite effort designed to complete a portion of the reviews necessary for the staff to reach its decision regarding the licensing of Comanche Peak Unit 1. The onsite effort covered a number of areas, including allegations of improper construction practices at the facility.

The NRC assembled a Technical Review Team (TRT) responsible for evaluating most of the technical issues at Comanche Peak, including allegations. The TRT has recently identified a number of items that have potential safety implications for which we require additional information. These items are listed in the enclosure to this letter. Further background information regarding these issues will be published in a Supplement to a Safety Evaluation Report (SSER), which will document the overall TRT's assessment of the significance of the issues examined.

The items in the enclosure to this letter, which are in the general areas of electrical/instrumentation, civil/structural and test programs, cover only a portion of the TRT's effort. The TRT evaluation of items in the areas of mechanical, QA/QC, and coatings, and its consideration of the programmatic implications of these findings, are still is progress. A summary of these issues will be provided to you at a later date.

You are requested to submit additional information to the NRC, in writing, including a program and schedule for completing a detailed and thorough assessment of the issues identified. This program plan and its implementation will be evaluated by the staff before NRC considers the issuance of an operating license for Comanche Peak, Unit 1. The program plan should address the root cause of each problem identified and its generic implications on safety-related systems, programs, or areas. The collective significance of these deficiencies should also be addressed. Your program plan should also include the proposed TUGCO action to assure that such problems will be precluded from occurring in the future.

This request is submitted to you in keeping with the NRC practice of promptly notifying applicants of outstanding information/evaluation needs that could potentially affect the safe operation of their plant. Further requests for additional information of this nature will be made, if necessary, as the activities of the TRT progress.

Sincerely,

Darrell G. Eisenhut, Director
Division of Licensing, NRR

Enclosure:
As stated

cc w/enclosure
See next page

COMANCHE PEAK

Mr. M. D. Spence
President
Texas Utilities Generating Company
400 N. Olive St., L.B. 81
Dallas, Texas 75201

cc: Nicholas S. Reynolds, Esq.
Bishop, Liberman, Cook,
Purcell & Reynolds
1200 Seventeenth Street, N. W.
Washington, D. C. 20036

Robert A. Wooldridge, Esq.
Worsham, Forsythe, Sampels &
Wooldridge
2001 Bryan Tower, Suite 2500
Dallas, Texas 75201

Mr. Homer C. Schmidt
Manager - Nuclear Services
Texas Utilities Generating Company
Skyway Tower
400 North Olive Street
L. B. 81
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

Renea Hicks, Esq.
Assistant Attorney General
Environmental Protection Division
P. O. Box 12548, Capitol Station
Austin, Texas 78711

Mrs. Juanita Ellis, President
Citizens Association for Sound
Energy
1426 South Polk
Dallas, Texas 75224

Ms. Nancy H. Williams
CYGNA
101 California Street
San Francisco, California 94111

Mr. James E. Cummins
Resident Inspector/Comanche Peak
Nuclear Power Station
c/o U. S. Nuclear Regulatory
Commission
P. O. Box 38
Glen Rose, Texas 76043

Mr. John T. Collins
U. S. NRC, Region IV
611 Ryan Plaza Drive
Suite 1000
Arlington, Texas 76011

Mr. Lanny Alan Sinkin
114 W. 7th, Suite 220
Austin, Texas 78701

B. R. Clements
Vice President Nuclear
Texas Utilities Generating Company
Skyway Tower
400 North Olive Street
L. B. 81
Dallas, Texas 75201

William A. Burchette, Esq.
1200 New Hampshire Avenue, N. W.
Suite 420
Washington, D. C. 20036

Ms. Billie Pirner Garde
Citizens Clinic Director
Government Accountability Project
1901 Que Street, N. W.
Washington, D. C. 20009

David R. Pigott, Esq.
Orrick, Herrington & Sutcliffe
600 Montgomery Street
San Francisco, California 94111

Anthony Z. Roisman, Esq.
Trial Lawyers for Public Justice
2000 P. Street, N. W.
Suite 611
Washington, D. C. 20036

ENCLOSURE 1

REQUEST FOR ADDITIONAL INFORMATION

I. Electrical/Instrumentation Area

a. Electrical Cable Terminations

The Technical Review Team (TRT) inspected random samples of safety-related terminations, butt splices inside panels, and vendor-installed terminal lugs in General Electric (GE) motor control centers, and reviewed documentation relative to the installations.

1. The TRT found a lack of awareness on the part of quality control (QC) electrical inspectors to document in the inspection reports when the installation of the "nuclear heat-shrinkable cable insulation sleeves" was required to be witnessed.

   Accordingly, TUEC shall clarify procedural requirements and provide additional inspector training with respect to the areas in which nuclear heat-shrinkable sleeves are required on splices and assure that such sleeves are installed where required.

2. The TRT found inspection reports that did not indicate that the required witnessing of splice installation was done. Examples are as follows:

IR ET-1-0005393	IR ET-1-0005396
IR ET-1-0005394	IR ET-1-0006776
IR ET-1-0005395	IR ET-1-0014790

   Accordingly, TUEC will assure that all QC inspections requiring witnessing for butt splices have been performed and properly documented; and verify that all butt splices are properly identified on the appropriate drawings and are physically identified within the appropriate panels.

3. The TRT found a lack of splice qualification requirements and provisions in the installation procedures to verify the operability of those circuits for which splices were being used.

   Accordingly, TUEC shall develop adequate installation/inspection procedures to assure that the wiring splicing materials are qualified for the appropriate service conditions, and that splices are not located adjacent to each other.

4. Selected cable terminations were found that did not agree with their locations on drawings. Examples are as follows:

Panel CP1-ECPRCB-14, Cable E0139880
Panel CP1-ECPRTC-16, Cable E0110040
Panel CP1-ECPRTC-16, Cable E0118262
Panel CP1-ECPRTC-27, Cable EG104796
Panel CPX-ECPRCV-01, Cable EG021856
Panel CP1-ECPRCB-02, Cable NK139853 (nonsafety)

Accordingly, TUEC shall reinspect all safety-related and associated terminations in the control room panels and in the termination cabinets in the cable spreading room to verify that their locations are accurately depicted on drawings. Should the results of this reinspection reveal an unacceptable level of nonconformance to drawings, the scope of this reinspection effort shall be expanded to include all safety-related and associated terminations at CPSES.

5. The TRT found cases where nonconformance reports (NCRs) concerning vendor-installed terminal lugs in GE motor control centers had been improperly closed. Examples are NCR Nos. E-84-01066 through NCR E-84-01076, inclusive.

   Accordingly, TUEC shall reevaluate and redisposition all NCRs related to vendor-installed terminal lugs in GE motor control centers.

b. Electrical Equipment Separation

The TRT reviewed the separation criteria between separate cables, trays and conduits in the main control room and cable spreading room in Unit 1, and the compatibility of the electrical erection specifications with regulatory requirements. The TRT reviewed documentation and inspected random samples of separation between safety-related cables, trays and conduits and between them and nonsafety-related cables, trays and conduits.

1. In numerous cases, safety-related cables within flexible conduits inside main control room panels did not meet minimum separation requirements. Examples are as follows:

   Panel CP1-EC-PRCB-02
   Panel CP1-EC-PRCB-07
   Panel CP1-EC-PRCP-06
   Panel CP1-EC-PRCB-08
   Panel CP1-EC-PRCB-09

   Accordingly, TUEC shall reinspect all panels at CPSES, in addition to those in the main control room for Unit 1, that contain redundant safety-related cables within conduits, or safety and non-safety related cables within conduits, and either correct each violation of the separation criteria, or

demonstrate by analysis the acceptability of the conduit as a barrier for each case where the minimum separation is not met.

2. In several cases, separate safety and nonsafety-related cables and safety and nonsafety-related cables within flexible conduits inside main control room panels did not meet minimum separation requirements (Table 1 identifies examples of these cases). No evidence was found that justified the lack of separation.

   Accordingly, TUEC shall reinspect all panels at CPSES, in addition to those in the main control room of Unit 1, and either correct each violation of the separation criteria concerning separate cables and cables within flexible conduits, or demonstrate by analysis the adequacy of the flexible conduit as a barrier.

3. The TRT found that the existing TUEC analysis substantiating the adequacy of the criteria for separation between conduits and cable trays had not been reviewed by the NRC staff.

   Accordingly, TUEC shall submit the analysis that substantiates the acceptability of the criteria stated in the electrical erection specifications governing the separation between independent conduits and cable trays.

4. The TRT found two minor violations of the separation criteria inside panels CP1-EC-PRCB-09 and CP1-EC-PRCB-03 concerning a barrier that had been removed and redundant field wiring not meeting minimum separation. The devices involved with the barrier were FI-2456A, PI-2453A, PI-2475A, and IT2450, associated with Train A; and FI-2457A, PI-2454A, PI-2476A, and IT-2451, associated with Train B. The field wiring was associated with devices HS-5423 of Train B and HS-5574, nonsafety-related.

   Accordingly, TUEC shall correct two minor violations of the separation criteria inside panels CP1-EC-PRCB-09 and CP1-EC-PRCP-03 concerning a barrier that had been removed and redundant field wiring not meeting minimum separation.

Table 1

Examples of Cases of Safety or Nonsafety-Related Cables

In Contact With Other Safety-Related Cables Within Conduits in Control Room

Panels

1. Control Panel CP1-EC-PRCB-02 - Containment Spray System

Cable No.	Train	Related Instrument
EG139373	B (green)	Undetermined
E0139010	A (orange)	Undetermined

2. Control Panel CP1-EC-PRCB-07 - Reactor Control System

Cable No.	Train	Related Instrument
EG139383	B (green)	Reactor manual trip switch
E0139311	A (orange)	Undetermined

3. Control Panel CP1-EC-PRCP-06 - Chemical & Volume Control System

Cable No.	Train	Related Instrument
EG139335	B (green)	LCV-112C
E0139301	A (orange)	Undetermined

4. Control Panel CP1-EC-PRCB-09 - Auxiliary Feedwater Control System

Cable No.	Train	Related Instrument
E0139753	A (orange)	FK-2453A
E0139754	A (orange)	FK-2453B
E0139756	B (green)	FK-2454A
EG139288	B (green)	FK-2454B

c. Electrical Conduit Supports

The TRT examined the nonsafety-related conduit support installation in selected seismic Category I areas of the plant. The support installation for non-safety related conduits less than or equal to 2 inches was inconsistent with seismic requirements and no evidence could be found that substantiated the adequacy of the installation for nonsafety-related conduit of any size. According to Regulatory Guide 1.29 and FSAR Section 3.7B.2.8, the seismic Category II and nonseismic items should be designed in such a way that their failure would not adversely affect the function of safety-related components or cause injury to plant personnel.

Accordingly, TUEC shall propose a program that assures the adequacy of the seismic support system installation for nonsafety-related conduit in all seismic Category I areas of the plant as follows:

1. Provide the results of seismic analysis which demonstrate that all nonsafety-related conduits and their support systems, satisfy the provisions of Regulatory Guide 1.29 and FSAR Section 3.7B.2.8.

2. Verify that nonsafety-related conduits less than or equal to 2 inches in diameter, not installed in accordance with the requirements of Regulatory Guide 1.29, satisfy applicable design requirements.

d. Electrical QC Inspector Training/Qualifications

The TRT examined electrical QC inspector training and certification files, and requirements for personnel testing, on-the-job training, and recertification. The TRT also interviewed selected electrical QA/QC personnel.

1. The TRT found a lack of supportive documentation regarding personnel qualifications in the training and certification files, as required by procedures and regulatory requirements. Also, the TRT found a lack of documentation for assuring that the requirements for electrical QC inspector recertification were being met. Specific examples are:

   ° One case of no documentation of a high schoc diploma or General Equivalency Diploma.

- One case of no documentation to waive the remaining 2 months of the required 1 year experience.

- One case where a QC technician had not passed the required color vision examination administered by a professional eye specialist. A makeup test using colored pencils was administered by a QC supervisor, was passed, and then a waiver was given.

- Two cases where the experience requirements to become a Level 1 technician were only marginally met.

- One case of no documentation in the training and certification files substantiating that the person met the experience requirements.

Accordingly, TUEC shall review all the electrical QC inspector training, qualification, certification and recertification files against the project requirements and provide the information in such a form that each requirement is clearly shown to have been met by each inspector. If an inspector is found to not meet the training, qualification, certification, or recertification requirements, TUEC shall then review the records to determine the adequacy of inspections made by the unqualified individuals and provide a statement on the impact of the deficiencies noted on the safety of the project.

2. The TRT found a lack of guidelines and procedural requirements for the testing and certifying of electrical QC inspectors. Specifically, it was found that:

   - No time limit or additional training requirements existed between a failed test and retest.

   - No controls existed to assure that the same test would not be given if an individual previously failed that test.

   - No consistency existed in test scoring.

   - No guidelines or procedures were available to control the disqualification of questions from the test.

   - No program was available for establishing new tests (except when procedures changed). The same tests had been utilized for the last 2 years.

   Accordingly, TUEC shall develop a testing program for electrical QC inspectors which provides adequate administrative guidelines, procedural requirements and test flexibility to assure that suitable proficiency is achieved and maintained.

The deficiencies identified with the electrical QC inspections have generic implications to other construction disciplines. The implications of these findings will be further assessed as part of the overall programmatic review of QC inspector training and qualification and the results of this review will be reported under the QA/QC category on "Training and Qualification."

II. Civil/Structural Area

a. Unable to Justify Reinforcing Steel Omitted in the Reactor Cavity

The TRT investigated a documented occurrence in which reinforcing steel was omitted from a Unit 1 reactor cavity concrete placement between the 812-foot and 819-foot ½-inch elevations. This reinforcement was installed and inspected according to drawing 2323-S1-0572, Revision 2. However, after the concrete was placed, Revision 3 to the drawing was issued showing a substantial increase in reinforcing steel over that which was installed. Gibbs & Hill Engineering was informed of the omission by Brown & Root Nonconformance Report CP-77-6. Gibbs & Hill Engineering replied that the omission in no way impaired the structural integrity of the structure. Nevertheless, the additional reinforcing steel was added as a precaution against cracking which might occur in the vicinity of the neutron detector slots should a loss of coolant accident (LOCA) occur. A portion of the omitted reinforcing steel was also placed in the next concrete lift above the 819-foot ½-inch level. This was done to partially compensate for the reinforcing steel omitted in the previous concrete lift and to minimize the overall area potentially subject to cracking.

The TRT requested documentation indicating that an analysis was performed supporting the Gibbs & Hill conclusion. The TRT was subsequently informed that an analysis had not been performed. Therefore, the TRT cannot determine the safety significance of this issue until an analysis is performed verifying the adequacy of the reinforcing steel as installed.

Accordingly, TUEC shall provide an analysis of the as-built condition of the Unit 1 reactor cavity that verifies the adequacy of the reinforcing steel between the 812-foot and 819-foot ½-inch elevations. The analysis shall consider all required load combinations.

b. Falsification of Concrete Compression Strength Test Results

The TRT investigated allegations that concrete strength tests were falsified. The TRT reviewed an NRC Region IV investigation (IE Report No. 50-445/79-09; 50-446/79-09) of this matter that included

interviews with fifteen individuals. Of these, only the alleger and one other individual stated they thought that falsification occurred, but they did not know when or by whom. The TRT also reviewed slump and air entrainment test results of concrete placed during the period the alleger was employed (January 1976 to February 1977) and did not find any apparent variation in the uniformity of the parameters for concrete placed during this period. Although the uniformity of the concrete placed appears to minimize the likelihood that low concrete strengths were obtained, other allegations were raised concerning the falsification of records associated with slump and air content tests. The Region IV staff addressed these allegations by assuming that concrete strength test results were adequate. Furthermore, a number of other allegations dealing with concrete placement problems (such as deficient aggregate grading and concrete in the mixer too long) were also resolved by assuming that concrete strength test results were adequate. The TRT agrees with Region IV that, while the preponderance of evidence suggests that falsification of results did not take place, the matter cannot be resolved completely on the basis of concrete strength test results, especially if there is any doubt about whether they may have been falsified. Due to the importance of the concrete strength test results, the TRT believes that additional action by TUEC is necessary to provide confirmatory evidence that the reported concrete strength test results are indeed representative of the strength of the concrete installed in the Category I concrete structures.

Accordingly, TUEC shall determine areas where safety-related concrete was placed between January 1976 and February 1977, and provide a program to assure acceptable concrete strength. The program shall include tests such as the use of random Schmidt hammer tests on the concrete in areas where safety is critical. The program shall include a comparison of the results with the results of tests performed on concrete of the same design strength in areas where the strength of the concrete is not questioned, to determine if any significant variance in strength occurs. TUEC shall submit the program for performing these tests to the NRC for review and approval prior to performing the tests.

c. Maintenance of Air Gap Between Concrete Structures

The TRT investigated the requirements to maintain an air gap between concrete structures. Based on the review of available inspection reports and related documents, on field observations, and on discussions with TUEC engineers, the TRT cannot determine whether an adequate air gap has been provided between concrete structures. Field investigations by B&R QC inspectors indicated unsatisfactory conditions due to the presence of debris in the air

gap, such as wood wedges, rocks, clumps of concrete and rotofoam. The disposition of the NCR relating to this matter states that the "field investigation reveals that most of the material has been removed." However, the TRT cannot determine from this report (NCR C-83-01067) the extent and location of the debris remaining between the structures.

Based on discussions with TUEC engineers, it is the TRT's understanding that field investigations were made but that no permanent records were maintained. In addition, it is not apparent that the permanent installation of elastic joint filler material ("rotofoam") between the Safeguards Building and the Reactor Building, and below grade for the other concrete structures, is consistent with the seismic analysis assumptions and dynamic models used to analyze the buildings, as these analyses are delineated in the Final Safety Analysis Report (FSAR). The TRT, therefore, concludes that TUEC has not adequately demonstrated compliance with FSAR Sections 3.4.1.1.1, 3.8.4.5.1, and 3.7.B.2.8, which require separation of Seismic Category I buildings to prevent seismic interaction during an earthquake.

Accordingly, TUEC shall:

1. Perform an inspection of the as-built condition to confirm that adequate separation for all seismic category I structures has been provided.

2. Provide the results of analyses which demonstrate that the presence of rotofoam and other debris between all concrete structures (as determined by inspections of the as-built conditions) does not result in any significant increase in seismic response or alter the dynamic response characteristics of the Category I structures, components and piping when compared with the results of the original analyses.

d. <u>Seismic Design of Control Room Ceiling Elements</u>

The TRT investigated the seismic design of the ceiling elements installed in the control room. The following matrix designates those ceiling elements present in the control room and their seismic category designation:

1. Heating, Ventilating and Air Conditioning	- Seismic Category I
2. Safety-Related Conduits	- Seismic Category I
3. Nonsafety-Related Conduits	- Seismic Category II
4. Lighting Fixtures	- Seismic Category II
5. Sloping Suspended Drywall Ceiling	- Non-Seismic
6. Acoustical Suspended Ceiling	- Non-Seismic
7. Lowered Suspended Ceiling	- Non-Seismic

According to Regulatory Guide 1.29 and FSAR Section 3.7B.2.8, the seismic Category II and nonseismic items should be designed in such a way that their failure would not adversely affect the functions of safety-related components or cause injury to operators.

For the nonseismic items (other than the sloping suspended drywall ceiling), and for nonsafety-related conduits whose diameter is 2 inches or less, the TRT could find no evidence that the possible effects of a failure of these items had been considered. In addition, the TRT determined that calculations for seismic Category II components (e.g., lighting fixtures) and the calculations for the sloping suspended drywall ceiling did not adequately reflect the rotational interaction with the nonseismic items, nor were the fundamental frequencies of the supported masses determined to assess the influence of the seismic response spectrum at the control room ceiling elevation would have on the seismic response of the ceiling elements.

Accordingly, TUEC shall provide:

1. The results of seismic analysis which demonstrate that the nonseismic items in the control room (other than the sloping suspended drywall ceiling) satisfy the provisions of Regulatory Guide 1.29 and FSAR Section 3.7B.2.8.

2. An evaluation of seismic design adequacy of support systems for the lighting fixtures (seismic Category II) and the suspended drywall ceiling (nonseismic item with modification) which accounts for pertinent floor response characteristics of the systems.

3. Verification that those items in the control room ceiling not installed in accordance with the requirements of Regulatory Guide 1.29 satisfy applicable design requirements.

4. The results of an analysis that justify the adequacy of the nonsafety-related conduit support system in the control room for conduit whose diameter is 2 inches or less.

5. The results of an analysis which demonstrate that the foregoing problems are not applicable to other Category II and nonseismic structures, systems and components elsewhere in the plant.

e. Unauthorized Cutting of Rebar in the Fuel Handling Building

The TRT investigated an alleged instance of unauthorized cutting of rebar associated with the installation of the trolley process aisle rails in the Fuel Handling Building. The claim is that during installation of 22 metal plates in January 1983, a core drill was used to drill about 10 holes approximately 9 inches deep. The TRT reviewed the reinforcement drawings for the Fuel Handling Building and determined that there were three layers of reinforcing steel in the top reinforcement layer of the slab. This reinforcement layer consisted of a No. 18 bar running in the east-west direction in the first and third layers, and a No. 11 bar running in the north-south direction on the second layer. The review also revealed that the layout of the reinforcement and the trolley rails was such that the east-west reinforcement would interfere with the drilling of holes along only one rail location. However, if 9-inch holes were drilled, both the first and third layers of No. 18 reinforcement would be cut. Design Change Authorization No. 7041 was written for authorization to cut the uppermost No. 18 bar at only one rail location, but did not reference authorization to cut the lower No. 18 bar. DCA-7041 also stated that the expansion bolts and base plates may be moved in the east-west direction to avoid interference with reinforcement running in the north-south direction. The information, described in DCA-7041, was substantiated by Gibbs & Hill calculations. If the ten holes were actually drilled 9 inches deep, then the allegation that the reinforcement was cut without proper authorization would be valid.

Accordingly, TUEC shall provide:

1. Information to demonstrate that only the No. 18 reinforcing steel in the first layer was cut, or

2. Design calculations to demonstrate that structural integrity is maintained if the No. 18 reinforcing steel on both the first and third layers was cut.

III. Test Programs Area

a. Hot Functional Testing (HFT)

The TRT reviewed a sample of the completed data packages for HFT preoperational test procedures, pertinent startup administrative procedures, NRC inspection reports, and the preoperational test index and its schedule. The TRT also inspected test deficiency reports

(TDRs) that were generated as a result of test deficiencies found prior to and during HFT.

1. Chapter 14 of the FSAR and Regulatory Guide 1.68 provide requirements for the conduct of preoperational testing. In reviewing test data packages, the TRT found that certain test objectives were not met. It appears that the Joint Test Group approved incomplete data packages for at least three preoperational hot functinal tests. These were:

Test Procedure	Deficiency
1CP-PT-02-12, "Bus Voltage and Load Survey"	Because acceptable voltages could not be achieved with the specified transformer taps, they were changed. A subsequent engineering evaluation required returning to the original taps, but no retest was performed.
1CP-PT-34-05, "Steam Generator Narrow Range Level Verification"	Level detectors 1-LT-517, 518 and 529 were replaced with temporary equipment of a design that was different from that which was to be eventually installed
1CP-PT-55-05 "Pressurizer Level Control"	Level detector 1-LT-461 appeared to be out of calibration during the test and was replaced after the test. The retest approved by the JTG was a cold calibration rather than a test consistent with the original test objective, which was to obtain satisfactory data under hot conditions.

Accordingly, TUEC shall review all complete preoperational test data packages to ensure there are no other instances where test objectives were not met, or prerequisite conditions were not satisfied. The three items identified by the TRT shall be included, along with appropriate justification, in the test deferral packages presented to the NRC.

2. The TRT noted during a review of HFT completed test data that the JTG did not approve the data until after cooldown from the test. The tests are not considered complete until this approval is obtained. In order to complete the proposed post-fueling, deferred preoperational HFT, the JTG, or a similarly qualified group, must approve the data prior to proceeding to initial criticality. The TRT did not find any document providing assurance that TUEC is committed to do this.

   Accordingly, TUEC shall commit to having a JTG, or similarly qualified group, review and approve all post-fueling preoperational test results prior to declaring the system operable in accordance with the technical specifications.

3. The TRT pointed out that in order to conduct preoperational tests at the necessary temperatures and pressures after fuel load, certain limiting conditions of the proposed technical specifications cannot be met, e.g., all snubbers will not be operable since some will not have been tested.

   Accordingly, TUEC shall evaluate the required plant conditions for the deferred preoperational tests against limiting conditions in the proposed technical specifications and obtain NRC approval where deviations from the technical specifications are necessary.

4. Data for the thermal expansion tests (which have not yet been approved by the JTG) did not provide for traceability between the calibration of the measuring instruments and the monitored locations, as required by Startup Administrative Procedure-7. The information was separately available in a personal log held by Engineering.

   Accordingly, TUEC shall incorporate the information necessary to provide traceability between thermal expansion test monitoring locations and measuring instruments. TUEC shall also establish administrative controls to assure appropriate test and measuring equipment traceability during future testing.

b. <u>Containment Intergrated Leak Rate Testing (CILRT)</u>

The TRT reviewed the data package for the CILRT performed on Unit 1, and discussed the conduct of the test with TUEC and NRC personnel who participated in or witnessed it.

Apparently after repairing leaks found during the first two attempts, the third attempt at a CILRT was successful. It was successfully completed after three electrical penetrations were isolated because the leakage through them could not be stopped. Though the leaks were subsequently repaired and individually tested with satisfactory results, NRC approval was not obtained to perform the CILRT with these penetrations isolated. In addition, leak rate calculations were performed using ANSI/ANS 56.8, which is neither endorsed by the NRC nor in accordance with FSAR commitments.

Accordingly, TUEC shall identify to NRC any other differences in the conduct of the CILRT as a result of using ANSI/ANS 56.8 rather than ANSI N45.4-1972. Additionally, TUEC shall identify to NRC all other deviations from FSAR commitments.

c. Prerequisite Testing

The TRT reviewed FSAR commitments, startup administrative procedures, prerequisite test records, craft personnel qualification records, and discussed them with startup and craft management personnel. The TRT also observed test support craft personnel at work and interviewed some of them to gain familiarity with their attitudes and capabilities.

The review of test records revealed that craft personnel were signing to verify initial conditions for tests in violation of startup Administrative Procedure-21, entitled: "Conduct of Testing" (CP-SAP-21). This procedure requires this function to be performed by System Test Engineers (STE). Startup management had issued a memorandum improperly authorizing craft personnel to perform these verifications on selected tests.

Accordingly, TUEC shall rescind the startup memorandum (STM-83084), which was issued in conflict with CP-SAP-21, and ensure that no other memoranda were issued which are in conflict with approved procedures.

d. Preoperational Testing

The TRT assessed the preoperational test program by reviewing administrative procedures, interviewing startup personnel, and examining test records, schedules, system assignments, subsystem definition packages, and the master data base.

Problems found with test data are addressed in section III.a of this enclosure. The TRT also found that STEs were not being provided with current design information on a routine, controlled basis, and had to update their own material when they considered it appropriate.

Accordingly, TUEC shall establish measures to provide greater assurance that STEs and other responsible personnel are provided with current controlled design documents and change notices.

J. Ellis
Rcd. 8/29/84

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

July 18, 1984

In Reply Refer To:
Docket: 50-445/84-21

ATTACHMENT B

Texas Utilities Electric Company
Attn: M. D. Spence, President, TUGCO
Skyway Tower
400 North Olive Street
Lock Box 81
Dallas, Texas 75201

Gentlemen:

This refers to the inspection conducted by Mr. W. F. Smith of this office during the period June 14-16, 1984, of activities authorized by NRC Construction Permit CPPR-126 for the Comanche Peak Facility, Unit 1, and to the discussion of our findings with Mr. J. T. Merritt and other members of your staff.

Areas examined during the inspection included preoperational test witnessing and test procedure review. Within these areas, the inspection consisted of selective examination of procedures and representative records, interviews with personnel, and observations of testing in progress by the inspector.

During this inspection, it was found that certain of your activities were in violation of NRC requirements. Consequently, you are required to respond to these violations in writing, in accordance with the provisions of Section 2.201 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations. Your response should be based on the specifics contained in the Notice of Violation enclosed with this letter.

Details of this inspection will be included in a report to be issued in the near future and identified as NRC Inspection Report 50-445/84-21.

This office calls your attention to the fact that similar violations were addressed during the exit meeting of June 1, 1984, which will be documented in NRC Inspection Report 50-445/84-18. This indicates that actions taken thus far do not appear to have been effective with regard to procedure compliance.

The response directed by this letter and the accompanying Notice is not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, Pl 96-511.

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

Sincerely,

Richard L Bangart

Richard L. Bangart, Director
Division of Radiation Safety and
Safeguards

Enclosure:
Appendix - Notice of Violation

cc w/enclosure:

Texas Utilities Electric Company
ATTN: H. C. Schmidt, Manager
Nuclear Services
Skyway Tower
400 North Olive Street
Lock Box 81
Dallas, Texas 75201

Texas Utilities Electric Company
ATTN: B. R. Clements, Vice President, Nuclear
Skyway Tower
400 North Olive Street
Lock Box 81
Dallas, Texas 75201

APPENDIX

NOTICE OF VIOLATION

Texas Utilities Electric Company
Comanche Peak Steam Electric Station

Docket: 50-445/84-21
Construction Permit: CPPR-126

Based on the results of an NRC inspection conducted during the period of June 14-16, 1984, and in accordance with the NRC Enforcement Policy (10 CFR Part 2, Appendix C), 49 FR-8583, dated March 8, 1984, the following violations were identified:

1. Criterion V of Appendix B to 10 CFR 50 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 . . . ."

   Contrary to the above, on June 16, 1984, an operator proceeded to partially open Station Service Water Chlorination Valve XSW-042 in violation of Step 5.4.1.6 of System Operating Procedure SOP-501A (Rev. 0), "Station Service Water System," which requires XSW-036 to be opened. The operation was aborted and the valve restored to the shut position only after the NRC inspector pointed out the procedure violation. Subsequently, it was determined that the procedure was in error, thus was changed accordingly and the operation resumed by opening Valve XSW-042.

   This is a Severity Level IV Violation. (Supplement II-D) (445/8421-01)

2. Criterion XI of Appendix B to 10 CFR 50 states in part, ". . . the test program shall include, as appropriate, proof test prior to installation, preoperational tests, and operational tests during nuclear power plant or fuel reprocessing plant operation of structures, systems, and components. Test procedures shall include provisions for assuring that all prerequisites for the given test have been met, ..."

   a. Contrary to the above, during the performance of the Diesel Generator Control Circuit Functional and Start Test, 1CP-PT-29-02 RT-1, the NRC inspector noted that there was no prerequisite in the test procedure to provide for station service air so that Step 7.1.6.7 can be performed to operate the barring device, which requires service air to function. This became apparent to the NRC inspector when he noticed the service air piping was not connected to the barring device. In lieu of service air, the STE utilized temporary air from a portable air compressor, which is not addressed by the procedure.

b. Contrary to the above, the station service water flow balancing test procedure, ICP-PT-04-01, had no prerequisite requirement to ensure the flow gages used during Step 7.8 (Flow Adjustment) were properly filled and vented. Failure to fill and vent these detectors just prior to flow adjustment can cause erroneous flow gage indications. This can place the flow data in question. As a result, during conduct of Step 7.8 of the test, the service water flow gage for containment spray was pegged high with no flow. It was evident that the gage was malfunctioning due to air binding or other mechanical problem.

This is a Severity Level IV Violation. (Supplement II-E) (445/8421-02)

Pursuant to the provisions of 10 CFR 2.201, Texas Utilities Electric Company is hereby required to submit to this office within 30 days of the date of this Notice, a written statement or explanation in reply, including: (1) the corrective steps which have been taken and the results achieved; (2) corrective steps which will be taken to avoid further violations; and (3) the date when full compliance will be achieved. Consideration may be given to extending your response time for good cause shown.

Dated: July 18, 1984

J. Ellis

Rec'd. 8/29/84

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

August 14, 1984

In Reply Refer To:
Docket: 50-445/84-21

ATTACHMENT C

Texas Utilities Electric Company
ATTN: M. D. Spence, President, TUGCO
Skyway Tower
400 North Olive Street
Lock Box 81
Dallas, Texas 75201

Gentlemen:

This refers to the inspection conducted by Messrs. D. L. Kelley and W. F. Smith of this office during the period June 1-30, 1984, of activities authorized by NRC Construction Permit CPPR-126 for the Comanche Peak Facility, Unit 1, and to the discussion of our findings with Messrs. B. R. Clements and J. C. Kuykendall and other members of your staff at the conclusion of the inspection.

Areas examined during the inspection included: (1) plant procedures inspection; (2) preoperational test witnessing; and (3) plant tours.

Within these areas, the inspection consisted of selective examination of procedures and representative records, interviews with personnel, and observations by the inspectors. These findings are documented in the enclosed inspection report.

During this inspection, it was found that certain of your activities were in violation of NRC requirements. The two violations reported in paragraph 3 of the enclosed inspection report were forwarded to you by our letter and Notice of Violation, dated July 18, 1984; therefore, this letter does not require further written response.

In accordance with 10 CFR 2.790(a), a copy of this letter and the enclosure will be placed in the NRC Public Document Room unless you notify this office, by telephone, within 10 days of the date of this letter, and submit written application to withhold information contained therein within 30 days of the date of this letter. Such application must be consistent with the requirements of 2.790(b)(1).

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

Sincerely,

Richard L Bangart

Richard L. Bangart, Director
RIV Comanche Peak Task Force

Enclosure:
Appendix - NRC Inspection Report
50-445/84-21

cc w/enclosure:

Texas Utilities Electric Company
ATTN: H. C. Schmidt, Manager
Nuclear Services
Skyway Tower
400 North Olive Street
Lock Box 81
Dallas, Texas 75201

Texas Utilities Electric Company
ATTN: B. R. Clements, Vice President, Nuclear
Skyway Tower
400 North Olive Street
Lock Box 81
Dallas, Texas 75201

APPENDIX

U. S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-445/84-21 Construction Permit: CPPR-126

Docket: 50-445 Category: A2

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

Facility Name: Comanche Peak Steam Electric Station (CPSES), Unit 1

Inspection At: Glen Rose, Texas

Inspection Conducted: June 1-30,1984

Inspectors: Dennis L. Kelley 8/8/84
D. L. Kelley, Senior Resident Reactor Inspector (SRRI) (paragraphs 1, 3, 4, and 5) Date

W. F. Smith 8/8/84
W. F. Smith, Resident Reactor Inspector (RRI) (paragraphs 1, 2, 3, and 5) Date

Approved: D M Hunnicutt 8/9/84
D. M. Hunnicutt, Team Leader, Task Force Date

Inspection Summary

Inspection Conducted: June 1-30, 1984 (Report 50-445/84-21)

Areas Inspected: Routine, announced inspection of (1) plant procedures inspection; (2) preoperational test witnessing; and (3) plant tours.

The inspection involved 180 inspector-hours onsite by two NRC inspectors.

Results: Within the areas inspected, two violations were identified and were transmitted under separate cover to the licensee on July 18, 1984, as Severity Level IV Violations, 445/8421-01, Supplement II-D, and 445/8421-02, Supplement II-E.

DETAILS

1. Persons Contacted

Licensee Personnel

*B. R. Clements, Vice President, Nuclear Operations
*J. C. Kuykendall, Manager, Nuclear Operations
*R. A. Jones, Manager, Plant Operations
J. T. Merritt, Assistant Project General Manager
*J. H. Roberts, Construction Startup Turnover Surveillance Supervisor
*T. P. Miller, Lead Startup Engineer
*R. B. Seidel, Operations Superintendent
*H. A. Lancaster, Startup Quality Assurance Specialist
*J. C. Smith, Quality Assurance
*T. L. Gosdin, Support Services Superintendent
*D. E. Deviney, Operations Quality Assurance Supervisor
*S. M. Franks, Startup Special Projects
R. R. Wistrand, Administrative Superintendent
J. Moorefield, Office Services Coordinator CPSES
D. C. Hisey, System Test Engineer
J. A. Van Gulik, System Test Engineer
K. B. Becker, System Test Engineer
K. E. Hemmila, System Test Engineer
S. E. Harvey, Assistant Shift Supervisor
R. L. Fortenberry, Shift Supervisor
M. S. Harris, System Test Engineer
M. Smith, Shift Supervisor
R. Beck, System Test Engineer

Others

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

*Denotes those present during the exit interview.

2. Plant Procedures Inspection

The objective of this inspection was to determine that the scope of the plant procedures system is adequate to control safety-related operations within applicable regulatory requirements and to determine the adequacy of management controls in implementing and maintaining a viable procedure system.

The first segment of this inspection module was accomplished during the period March 1 through April 30, 1984. The results of the inspection are

described in NRC Inspection Report 50-445/84-15, dated July 3, 1984. The plant procedures inspection was completed June 30, 1984. Detailed operating, emergency, and maintenance procedure inspections will be conducted separately and reported in subsequent inspection reports.

The following procedures were reviewed during this inspection period:

STA-605 (Rev.3)	"Clearance and Safety Tagging"
STA-707 (Rev.1)	"Safety Evaluations"
STA-606 (Rev.3)	"Maintenance Action Requests"
STA-608 (Rev.5)	"Control of Measuring and Test Equipment"
STA-616 (Rev.0)	"Control Room and Observation Area Access"
SOP-609A (Rev.0)	"Diesel Generator System"
SOP-501A (Rev.0)	"Station Service Water System"
ODA-202 (Rev.2)	"Preparation of System Operating Procedures"
ODA-301 (Rev.3)	"Operating Logs"

The RRI verified that responsibilities have been assigned to assure that site procedures such as those listed above are reviewed, updated, approved, and that 10 CFR 50.59 considerations are included in the review. In addition, the NRC inspector verified that when special orders are used, administrative controls have been established that provide a mechanism for their review, issuance, distribution, and limitations for use.

The RRI interviewed a reactor shift supervisor to determine whether or not he understood the systems established for controlling temporary changes to procedures. Several pertinent questions were asked, and all of the answers provided by the shift supervisor were correct. As the NRC inspector witnessed the daily progress of preoperational testing, he noted that the shift supervisors as a group were sufficiently aware of the established systems.

The NRC inspector reviewed the above listed procedures to ensure that:

- The review, approval, and updating had been done in accordance with station administrative requirements.

- The issuance and superseding of the procedures were done in accordance with the established controls.

- The procedures were formatted properly.
- The procedures were free of typographical errors, conflicts, or editorial errors.
- The procedures were adequate for the intended purpose and scope.
- The working copy at key locations such as the control room is the latest revision.

Upon completion of the above review, the NRC inspector did not identify any significant deficiencies; however, the following comments were offered to the licensee for consideration to reduce the possibility of problems in the future:

STA-605

Section 4.1.9 does not require independent verification of danger tags for nonsafety-related systems that do not affect the safe operation of the plant. Discussion with shift supervisors revealed a tendency on their part to be conservative in actual practice and require such verification checks on most, if not all, tagouts. The NRC inspector stated to representatives of the licensee that some plants require independent verification checks in those situations where operating system pressure, temperature, electrical, or radiological conditions could result in equipment damage or injury to personnel. Usually the pressure is defined as greater than 150 pounds per square inch gage and/or temperatures greater than 200 degrees Fahrenheit. The licensee agreed to consider this matter for the next revision of STA-605.

Section 4.1.9 states that the hanging of danger tags "should not normally" be done simultaneously with the independent verification check. The NRC inspector recommended that the statement be changed to "shall not." The NRC inspector was concerned that the power of suggestion in watching a tag being hung on a component could lead the verifier into believing the component was correct instead of the verifier independently determining the component was correct. This can defeat the concept of independent verification.

The NRC inspector noted that Attachment 1 of STA-605, "Clearance Report," did not have a column for the independent verification of tag removal and restoration of each component in accordance with Section 4.2.2.3. This action is not documented except by a single signature. Such a column would be a good tool to help the verifier check off each item, and would provide better assurance to the shift supervisor that none were inadvertently overlooked.

The above comments were discussed with the licensee's representative, who indicated that the comments are under consideration and that some of them are already incorporated into a major rewrite of STA-605 that is currently underway. In particular, the attachment, such as the clearance report, has been improved significantly to incorporate such features as verification columns discussed above.

STA-707

Section 4.1.5 does not clearly implement the requirement of 10 CFR 50.59(b) to publish a periodic (at least annual) report of changes made as permitted by 10 CFR 50.59(a). During discussion between the RRI and the licensee's representative, two major points relative to this report were brought out by the RRI.

IE Circular 80-18 dated August 22, 1980, clarifies the NRC requirements for the report. In short, the Circular points out that, for all cases requiring a written safety evaluation, the safety evaluation must set forth the bases and criteria used to determine that the proposed change does not involve an "unreviewed safety question." Though the annual report can be brief, a simple statement of conclusion in itself is not sufficient. The regulation requires a summary of the safety evaluation. Changes made under the authority of 10 CFR 50.59(a) are reportable and should appear in the annual report, if a change in the facility or procedure generates a need for revision of any of the text or drawings in the current Safety Analysis Report (SAR).

In addition, tests or experiments not described in the current SAR shall also be reported if they are to be added to the SAR. Section 4.1.5 of STA-707 should more clearly address the requirements of 10 CFR 50.59.

ODA-202

Section 4.2.6.1 of ODA-202, Revision 2, requires the "Instructions" section of system operating procedures to be subdivided into specific evolutions of operation. Examples are, "Startup," "Normal Operations," "Shutdown," and "Draining." Because of the differences between systems it is not practical to use the subsections specified by ODA-202. Consequently, some standard operating procedures (SOPs)do not follow the formatting requirement, such as SOP-503, "Surface Water Pretreatment System," SOP-607A, "118 VAC Distribution System and Inverters," SOP-706, "Digital Radiation Monitoring System," and SOP-710, "Incore Instrumentation System." The NRC inspector discussed this with the

licensee, who stated that there is a major rewrite of SOPs in progress, and formatting problems such as this are being corrected. Since these procedures are scheduled for NRC review after publication, the NRC inspector indicated that this area would be reinspected at a later date.

There were no other concerns, deficiencies, or violations noted during the procedures inspection.

3. Preoperational Test Witnessing

a. 1CP-PT-37-01, RT-1, "Auxiliary Feedwater System (Motor Driven Pumps)"

The purpose of this retest was to verify those items which remained open items to 1CP-PT-37-01, Rev. 0, and to retest certain items which required retest due to rework. The items to be tested and reason for retest were:

(1) Auxiliary feedwater valves control logic due to major rework of control boards, analog racks, relay racks, and cable spread room wiring.

(2) Motor-driven auxiliary feedwater pumps 1-1 and 1-2 control logic due to major rework of control boards, analog racks, relay racks, and cable spread room wiring.

(3) Motor-driven auxiliary feedwater pumps 1-1 and 1-2 hydraulic performance due to redesign of test line, mini-flow orifice, and a 1CP-PT-37-01, Rev. 0, open item.

(4) Auxiliary feedwater system response time because system response time was not determined during performance of 1CP-PT-37-01, Rev. 0.

(5) Auxiliary feedwater pumps 1-1 and 1-2 endurance test due to redesign of test line orifice due to unsatisfactory operation of original test line orifice.

The NRC inspector noted that during the section of the test to verify item (1) above, the timing of the feedwater valves was not within the range specified in the test. An error in the calibration of the timing logic was found to be the problem. The logic was recalibrated and the test section was reperformed and the results were satisfactory. During the 48-hour performance run of pump 1-1, a high temperature condition developed in the pump outboard bearing. This resulted in having to stop the pump and check for bearing misalignment or other problems. There were no apparent problems.

Several attempts were made to re-initiate the 48-hour run. These were unsuccessful until it was determined that there was too much oil in the outboard bearing. The level was adjusted and the 48-hour performance run of pump 1-1 was successfully completed.

b. <u>1CP-PT-49-01, Rev. 3, "Letdown, Charging and Seal Water System"</u>

The purpose of the test was:

(1) To verify proper operation of control and interlock functions for various valves and pumps in the Chemical and Volume Control System (CVCS) (see Section 2.0 for components tested).

(2) To verify response of various CVCS valves and pumps to Solid State Protection System (SSPS) signals such as safety injection, including response times of valves.

(3) To verify hydraulic performance of the positive displacement charging pump.

(4) To verify proper operation of the volume control tank diversion valves.

The NRC inspector observed portions of the last phases of this test. A review of the completed portion of the test was also performed, including a review of the test log entries, test procedure deviations and test deficiency reports, if any. The witnessing of this test was in addition to the preoperational tests preselected by the NRC inspectors for observation.

c. <u>1CP-PT-57-02, RT-1, "Centrifugal Charging Pump Test"</u>

The purpose of this test was to verify proper operation of control and interlock functions for various valves in the CVCS which are related to the centrifugal charging pump high head injection flowpaths. The retest was required as a result of electrical rework for train separation criteria and walkdown deficiencies.

The NRC inspector witnessed portions of the test performance from the control room and hot shutdown panel. There were no problems encountered with the test.

The NRC inspector, however, noted that when a transfer switch is operated to transfer control of a device from the control room to the hot shutdown panel the device being transferred will go to the position dictated by the control switch on the hot shutdown panel. This will result in valves changing position unless the hot shutdown panel

valve control switches are matched to the actual valve positions prior to operating the transfer switches. This concern was discussed with the licensee, who indicated that procedural or hardware changes are under consideration. The NRC inspector will followup on this during subsequent inspections.

d. <u>1CP-PT-04-01, RT. 1, "Station Service Water (SSW)"</u>

The purpose of this test was to verify the operating characteristics and to demonstrate the capability of each train of the SSW to supply adequate flow to each of the components served.

As independent inspection effort, the NRC inspectors witnessed the performance of this preoperational test over a period of several days. There were no major problems associated with obtaining satisfactory test results. The system performed as expected. However, the NRC inspector observed problems which resulted in two violations:

(1) During the flow balancing of the SSW system in accordance with 1CP-PT-04-01, it was necessary to place the SSW Chlorination System in operation in accordance with System Operating Procedure SOP-501A, "Station Service Water System." Step 5.4.1.6 of SOP-501A directs the operator to open SSW Chlorination Valve XSW-036. The operator, in the presence of the System Test Engineer (STE), noticed that what appeared to be the correct valve was labeled "XSW-042." Instead of halting the test to determine whether the valve label or the procedure was in error, the operator proceeded to open the valve. When the NRC inspector brought his attention to the procedure violation, the operation was aborted and the valve restored to the shut position. Subsequently, it was determined the procedure was in error; thus, it was changed accordingly and the operation resumed.

Prior to issuance of this inspection report, the Notice of Violation was transmitted to the licensee as Severity Level IV Violation 445/8421-01. This is the second violation issued in recent weeks pertaining to lack of procedure compliance. The previous violation was identified as 445/8418-01 and contains three examples of failure to follow procedures. The licensee was made aware by the resident inspectors of the importance of decisive permanent corrective action by senior management to prevent future procedure violations as the pace of testing and operations increases at CPSES.

(2) During the flow balancing of the SSW system, when the procedure required the STE to record the flow of service water to containment spray cooling, the installed gage was pegged high with or without flow. It became evident that the gage was malfunctioning due to air binding. There was no prerequisite in the 1CP-PT-04-01 to provide for filling and venting of the installed instruments used for this test, just prior to the test. Without such a prerequisite, the data is subject to question. because air in the instrument lines will cause erroneous readings. This is contrary to Criterion XI of Appendix B to 10 CFR 50, and Notice of Violation was transmitted to the licensee prior to this inspection report in which the violation was identified as a Level IV Violation 445/8421-02.

e. 1CP-PT-29-02, RT-1, "Diesel Generator (DG) Control Circuit Functional and Start Test"

The purpose of this test was to functionally demonstrate electrical and pneumatic control circuit operability in the manual mode of operation for Train A diesel generator.

The NRC inspector witnessed parts of this test to verify that the testing was conducted in accordance with approved procedures, that the observations recorded by the STE were consistent with the observations of the NRC inspector, that test results were adequately documented, and that the procedure is adequate to accomplish the intended purpose.

The test was conducted in a professional efficient manner. There were no problems observed by the NRC inspector with regard to the above attributes; however, as the NRC inspector observed the interlock testing associated with the DG barring device (the "Barring Device" is an air-operated jacking mechanism installed on the DG for the purpose of rotating the crankshaft during maintenance), he noticed that service air was not connected. Instead, a temporary air hose was connected from a portable diesel air compressor outside. 1CP-PT-29-02 did not have a prerequisite requiring service air (or a temporary source of air) to conduct the test. This left the STE to his own devices to perform the test and as such is contrary to Criterion XI of Appendix B to 10 CFR 50.

f. In addition to the above tests that were completed during this reporting period, these three additional tests were started but are still in progress. These tests are:

1CP-PT-48-02, "Containment Spray System

1CP-PT-02-02, "118 VAC RPS Inverters"

1CP-PT-34-01, Rev. 1, "Main Steam Isolation Valves"

No violations or deviations were found during witnessing of the above three operational tests.

4. Plant Status

The following is a status of TUEC (TUGCO) manning levels for operations and plant testing activities as of June 30, 1984.

a. Operations Manning Status

Authorized Personnel Level (including maintenance, operations, administration, quality assurance, and engineering) - 553

Number Presently on Board - 482

b. Plant Testing Status

The present status of the NRC preoperational testing phase inspection program is approximately 60 percent complete.

The licensee preoperational testing program is as follows:

Test Completion Status

Preoperation Tests-136

Acceptance Tests-64

5. Exit Interview

An exit interview was conducted July 6, 1984, with licensee representatives (identified in paragraph 1). During this interview, the SRRI and RRI reviewed the scope and discussed the inspection findings. The licensee acknowledged the findings.

Rc2. 8/23/84

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

AUG 17 1984

Docket No.: 50-445

ATTACHMENT D

Mr. M. D. Spence
President
Texas Utilities Generating Company
400 N. Olive Street
L. B. 81
Dallas, Texas 75201

Dear Mr. Spence:

Subject: Acceptance of Preoperational Test Deferrals for Comanche Peak Steam Electric Station, Unit 1

The staff has completed its review of the following preoperational tests requested by letters dated May 29, June 5, June 8 and June 15, 1984 from B. R. Clements:

1. Containment Cooling Systems
2. Safety Injection System Check Valve Leakage
3. Turbine Drive Auxiliary Feedwater Pump Steam Supply Line Check Valve and Drain Pot Level Control Valve
4. Reactor Coolant Pump Seal Performance
5. Thermal Expansion Testing
6. Control Room Ventilation System

Enclosed are the staff's evaluations which are the proposed findings for inclusion in a future SER supplement. These proposed findings indicate that the requested deferrals are acceptable. Therefore, the Unit 1 Operating License will contain license conditions consistent with your commitments on conducting the tests prior to initial criticality.

Sincerely,

B. J. Youngblood

B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

Enclosure: As stated

cc: See next page

COMANCHE PEAK AUG 17 1984

Mr. M. D. Spence
President
Texas Utilities Generating Company
400 N. Olive St., L.B. 81
Dallas, Texas 75201

cc: Nicholas S. Reynolds, Esq.
Bishop, Liberman, Cook,
Purcell & Reynolds
1200 Seventeenth Street, N. W.
Washington, D. C. 20036

Robert A. Wooldridge, Esq.
Worsham, Forsythe, Sampels &
Wooldridge
2001 Bryan Tower, Suite 2500
Dallas, Texas 75201

Mr. Homer C. Schmidt
Manager - Nuclear Services
Texas Utilities Generating Company
Skyway Tower
400 North Olive Street
L. B. 81
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

Renea Hicks, Esq.
Assistant Attorney General
Environmental Protection Division
P. O. Box 12548, Capitol Station
Austin, Texas 78711

Mrs. Juanita Ellis, President
Citizens Association for Sound
Energy
1426 South Polk
Dallas, Texas 75224

Ms. Nancy H. Williams
CYGNA
101 California Street
San Francisco, California 94111

Mr. James E. Cummins
Resident Inspector/Comanche Peak
Nuclear Power Station
c/o U. S. Nuclear Regulatory
Commission
P. O. Box 38
Glen Rose, Texas 76043

Mr. John T. Collins
U. S. NRC, Region IV
611 Ryan Plaza Drive
Suite 1000
Arlington, Texas 76011

Mr. Lanny Alan Sinkin
114 W. 7th, Suite 220
Austin, Texas 78701

B. R. Clements
Vice President Nuclear
Texas Utilities Generating Company
Skyway Tower
400 North Olive Street
L. B. 81
Dallas, Texas 75201

William A. Burchette, Esq.
1200 New Hampshire Avenue, N. W.
Suite 420
Washington, D. C. 20036

Ms. Billie Pirner Garde
Citizens Clinic Director
Government Accountability Project
1901 Que Street, N. W.
Washington, D. C. 20009

David R. Pigott, Esq.
Orrick, Herrington & Sutcliffe
600 Montgomery Street
San Francisco, California 94111

Anthony Z. Roisman, Esq.
Trial Lawyers for Public Justice
2000 P. Street, N. W.
Suite 611
Washington, D. C. 20036

SUPPLEMENTAL SAFETY EVALUATION REPORT
DEFERRAL OF CERTAIN PREOPERATIONAL TESTS
COMANCHE PEAK UNIT 1

Texas Utilities Generating Company in letters from B. R. Clements to H. R. Denton, NRC, dated May 29, June 5, June 8 and June 15, 1984, requested approval to defer six preoperational tests until after fuel loading. The testing would be completed prior to initial criticality with the exception of a portion of the thermal expansion test. This test requires heatup and return to cold shutdown conditions for completion and is scheduled at the completion of the 30 percent power plateau.

A. <u>1CP-PT-45-06, Containment Cooling Systems</u>

The applicant has requested that this test be repeated after fuel loading. Testing of the containment cooling systems were performed during the normal preoperational test program; however, test deficiencies were identified requiring system modifications which could not be retested prior to the scheduled fuel loading.

The repeat of this test after fuel loading is acceptable because only limited portions of the system require retesting, no technical specification exceptions are required and, for operation to continue, the system must still meet technical specifications temperature limits in critical areas.

B. <u>1CP-PT-57-09, Check Valve and Hot Functional Safety Injection</u>

The applicant has requested that this test be repeated after fuel loading. During the initial test, a number of check valves leaked in excess of their acceptance criteria. These valves have been repaired or replaced. The repeat testing of these valves would be performed as required by the technical specifications surveillance tests for check valves. It is acceptable to defer repeating portions of this test until after fuel loading, but before criticality, because (1) it is consistent with the technical specifications which control normal operation and define check valve operability and (2) presents no safety problem because retesting is completed prior to criticality.

C. 1CP-PT-37-03, Turbine Driven Auxiliary Feedwater Pump Steam Supply Line Check Valve and Drain Pot Level Control Valve

In performing this preoperational test, a faulty level switch and corroded and bent disks in the steam supply line check valve were discovered. The applicant has made the necessary repairs and requests approval to complete the test after fuel loading (which will be the next scheduled heatup). This request is acceptable because there will have been no power operation prior to the retest and repeat of the necessary portions of this test will be completed prior to criticality.

D. 1CP-PT-55-09, Reactor Coolant Pump (RCP) Test and 1CP-PT-49-02, Seal Water and Letdown Flow Performance

Several test deficiencies relating to the RCP seals were identified during the performance of these preoperational tests. Modifications to correct the deficiencies have been completed. The applicant proposed to incorporate the portions of the tests to be repeated into the appropriate startup test procedures to be performed after fuel loading, but prior to criticality. This schedule for retesting of the RCP seals is acceptable because it would be consistent with normal operating maintenance and test procedures and prior to initial criticality these systems are not required for plant safety.

E. 1CP-PT-55-11, Thermal Expansion Preoperational Test

During the performance of the thermal expansion test, a number of test deficiencies were noted pertaining to snubbers, springs and supports. These deficiencies were of three categories:

(1) installed items did not meet acceptance criteria;

(2) installed items removed due to interferences, and;

(3) items not installed for the test.

The applicant will have corrected these deficiencies and proposes that the test be repeated after fuel loading when the next plant heatup is completed for initial criticality. Final cold setting of retest items would be accomplished at the shutdown scheduled at the end of the 30% power plateau.

The deferral of the thermal expansion retest is acceptable because it is consistent with approved industry practice on other plant test programs. Furthermore, compliance with Technical Specifications relating to piping supports will be required for plant operation to proceed.

F. Control Room Ventilation System

During performance of the Control Room Ventilation System preoperational test, it was determined that the system provided more than adequate air supply to the control room area for Unit 1, but less than design air flow was supplied to Unit 2 control room area. The applicant is proceeding with modifications to the ventilation system to correct the design deficiency. The applicant plans to start retesting the modified system, but anticipates not being able to complete the testing prior to scheduled Unit 1 fuel loading. The applicant, therefore, requests deferral of completion of the test until after fuel loading.

Based on the condition that this deferral is a retest of a system which was already determined to be acceptable for the Unit 1 control area, we find the deferral of the retesting of the Control Room Ventilation System until completion of the initial fuel loading of Unit 1 (and before initial criticality) to be acceptable.

In summary, the deferral of these six preoperational tests represent retesting of modifications made to correct identified system deficiencies in the respective systems. Retesting these systems after initial fuel loading, but prior to initial criticality, will pose no safety problem, will be controlled by the plant Technical Specifications and are consistent with other plant test programs. On this basis, the requested deferrals are approved.

Rcd. 9/1/84

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

ATTACHMENT E

Docket Nos. 50-445
and 50-446

AUG 28 1984

Mr. M. D. Spence
President
Texas Utilities Generating Company
400 N. Olive St., L. B. 81
Dallas, Texas 75201

Dear Mr. Spence:

Subject: Request for Exemption from a Portion of General Design Criterion 4 of Appendix A to 10 CFR Part 50 Regarding the Need to Analyze Large Primary Loop Pipe Ruptures as the Structure Design Basis for Comanche Peak Steam Electric Station (Units 1 and 2)

By letter dated October 31, 1983, Texas Utilities Generating Company (TUGCO) provided Westinghouse Report MT-SME-3135 (proprietary) as the technical basis in support of its request for exemption from a portion of the requirements of General Design Criterion (GDC) 4 of Appendix A to 10 CFR Part 50. The Westinghouse report addressed the "leak-before-break" concept as an alternative to providing protective devices against the dynamic effects of postulated ruptures in the primary coolant loops. My letter to R. J. Gary dated March 2, 1984, requested responses to questions and comments raised by the staff based on its review of Westinghouse Report MT-SME-3135 and its generic review of Westinghouse Generic Report WCAP-10456 (proprietary), which provided an analysis of the fracture toughness of piping materials under thermal aging conditions.

Your letter to H. R. Denton dated April 23, 1984, submitted a revision to Westinghouse Report MT-SME-3135, identified as Westinghouse Report WCAP-10527 (proprietary), which responded to the questions and comments furnished by my letter dated March 2, 1984. In a separate letter to H. R. Denton (also dated April 23, 1984), you provided a value-impact analysis, associated with your exemption request.

Since the Westinghouse Report WCAP-10527 provided supporting analyses encompassing other structures in both Units 1 and 2, and seemed to be in conflict with the scope of the exemption requested in an earlier TUGCO letter dated February 17, 1984, TUGCO was requested to clarify this apparent inconsistency. H. C. Schmidt's letter to me, dated June 7, 1984, provided clarification stating that the exemption requested from the GDC 4 requirements was limited to the installation of jet impingement shields associated with postulated pipe breaks in eight (8) locations per loop in Unit 1, as specified in Section 4.0 of the value-impact analysis submitted by TUGCO letter to H. Denton dated April 23, 1984.

On the basis of the staff's evaluation of these submittals, the Commission has granted your exemption request for Comanche Peak, Unit 1, which is enclosed. The exemption pertains only to the installation of jet impingement shields as reflected in Mr. H. C. Schmidt's letter to me dated June 7, 1984. The enclosed exemption is being forwarded to the Office of the Federal Register for publication. We are also processing your request for amendment of the construction permit for Unit 1 to reflect this exemption.

Sincerely,

B J Youngblood

B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

Enclosure: As stated

cc: See next page

COMANCHE PEAK

Mr. M. D. Spence
President
Texas Utilities Generating Company
400 N. Olive St., L.B. 81
Dallas, Texas 75201

cc: Nicholas S. Reynolds, Esq.
Bishop, Liberman, Cook,
Purcell & Reynolds
1200 Seventeenth Street, N. W.
Washington, D. C. 20036

Robert A. Wooldridge, Esq.
Worsham, Forsythe, Sampels &
Wooldridge
2001 Bryan Tower, Suite 2500
Dallas, Texas 75201

Mr. Homer C. Schmidt
Manager - Nuclear Services
Texas Utilities Generating Company
Skyway Tower
400 North Olive Street
L. B. 81
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

Renea Hicks, Esq.
Assistant Attorney General
Environmental Protection Division
P. O. Box 12548, Capitol Station
Austin, Texas 78711

Mrs. Juanita Ellis, President
Citizens Association for Sound
Energy
1426 South Polk
Dallas, Texas 75224

Ms. Nancy H. Williams
CYGNA
101 California Street
San Francisco, California 94111

Mr. James E. Cummins
Resident Inspector/Comanche Peak
Nuclear Power Station
c/o U. S. Nuclear Regulatory
Commission
P. O. Box 38
Glen Rose, Texas 76043

Mr. John T. Collins
U. S. NRC, Region IV
611 Ryan Plaza Drive
Suite 1000
Arlington, Texas 76011

Mr. Lanny Alan Sinkin
114 W. 7th, Suite 220
Austin, Texas 78701

B. R. Clements
Vice President Nuclear
Texas Utilities Generating Company
Skyway Tower
400 North Olive Street
L. B. 81
Dallas, Texas 75201

William A. Burchette, Esq.
1200 New Hampshire Avenue, N. W.
Suite 420
Washington, D. C. 20036

Ms. Billie Pirner Garde
Citizens Clinic Director
Government Accountability Project
1901 Que Street, N. W.
Washington, D. C. 20009

David R. Pigott, Esq.
Orrick, Herrington & Sutcliffe
600 Montgomery Street
San Francisco, California 94111

Anthony Z. Roisman, Esq.
Trial Lawyers for Public Justice
2000 P. Street, N. W.
Suite 611
Washington, D. C. 20036

UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of TEXAS UTILITIES GENERATING COMPANY (Comanche Peak Steam Electric Station, Units 1 and 2)	))))))	Docket Nos. 50-445 and 50-446

EXEMPTION

I.

On July 20, 1973, the Texas Utilities Generating Company (the applicant) tendered an application for licenses to construct Comanche Peak Steam Electric Station, Units 1 and 2 (Comanche Peak or the facility) with the Atomic Energy Commission (currently the Nuclear Regulatory Commission or the Commission). Following a public hearing before the Atomic Safety and Licensing Board, the Commission issued Construction Permit Nos. CPPR-126 and CPPR-127 permitting the construction of Units 1 and 2, respectively, on December 19, 1974. Each Unit of the facility is a pressurized water reactor, combining a Westinghouse Electric Company nuclear steam supply system, located at the applicant's site in Somervell/Hood Counties, Texas, approximately 40 miles southwest of Fort Worth, Texas.

On February 27, 1978, the applicant tendered an application for Operating Licenses for each Unit of the facility, currently in the licensing review process, with Unit 1 licensing to occur in the near term.

II.

The Construction Permits issued for constructing the facility provide, in pertinent part, that the facility Units are subject to all rules, regulations and Orders of the Commission. This includes General Design Criterion (GDC) 4 of Appendix A to 10 CFR 50. GDC 4 requires that structures, systems and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with the normal operation, maintenance, testing and postulated accidents, including loss-of-coolant accidents. These structures, systems and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, discharging fluids that may result from equipment failures, and from events and conditions outside the nuclear power unit.

By a submittal dated October 31, 1983, the applicant requested an exemption from a portion of the requirements of GDC 4 to: (1) eliminate the need to postulate circumferential and longitudinal pipe breaks in the Reactor Coolant System (RCS) primary loop (hot leg, cold leg and cross-over leg piping); (2) eliminate the need to install pipe whip restraints and jet impingement shields associated with previously postulated breaks in the RCS primary loops and; (3) to eliminate the need to consider dynamic effects and loading conditions associated with previously postulated pipe breaks in the RCS primary loop, including jet impingement loads, cavity pressure loads, blowdown loads in the RCS and attached piping, and subcompartment pressure loads. In support of this exemption request, the applicant's submittal enclosed Westinghouse Report MT-SME-3135 (Reference 1) containing the technical basis for their request.

Based on its review of the applicant's submittal, the NRC staff requested additional information and provided comments on the reports (References 1 and 9) which were transmitted to the applicant in the form of questions by NRC letter dated March 2, 1984, (Reference 2).

By a submittal dated April 23, 1984, the applicant responded to the staff's questions (Reference 2) and provided a revision to the Reference 1 report identified as Westinghouse Report WCAP-10527 (Reference 3). In a separate submittal, also dated April 23, 1984, the applicant provided a value-impact analysis which, together with the technical information contained in the Reference 3 report, provided a comprehensive justification for requesting a partial exemption from the requirements of GDC 4.

From the deterministic fracture mechanics analysis contained in the technical information furnished, the applicant stated that the postulated double-ended guillotine breaks (DEGB) of the primary loop coolant piping will not occur in Comanche Peak Units 1 and 2 and, therefore, need not be considered as a design basis for installing protective structures, such as pipe whip restraints and jet impingement shields, to guard against the dynamic effects associated with such postulated breaks.

By letter dated June 7, 1984 (Reference 10), the applicant clarified the scope of its request for exemption from GDC 4 requirements. Since the Westinghouse Report WCAP-10527 provided analyses encompassing other structures in both Comanche Peak Units 1 and 2, and seemed to be in conflict with the scope of the exemption requested in an earlier letter dated February 17, 1984 (Reference 11), the applicant stated in the Reference 10 letter that, although the analyses contained in the Report WCAP-10527 encompassed relief from the need to install pipe break protective devices in both Units 1 and 2, the exemption being requested pertained solely to the installation of jet impingement shields associated with such breaks in eight (8) locations per loop in Comanche Peak Unit 1, as specified in Section 4.0 of the value-impact analysis submitted by the applicant's letter dated April 23, 1984.

III.

The Commission's regulations require that applicants provide protective measures against the dynamic effects of postulated pipe breaks in high energy fluid system piping. Protective measures include physical isolation from postulated pipe rupture locations if feasible or the installation of pipe whip restraints, jet impingement shields or compartments. In 1975, concerns arose as to the asymmetric loads on pressurized water reactor (PWR) vessels and their internals which could result from these large postulated breaks at discrete locations in the main primary coolant loop piping. This led to the establishment of Unresolved Safety Issue (USI) A-2, "Asymmetric Blowdown Loads on PWR Primary Systems."

The NRC staff, after several review meetings with the Advisory Committee on Reactor Safeguards (ACRS) and a meeting with the NRC Committee to Review Generic Requirements (CRGR), concluded that for certain facilities an exemption from the regulations would be acceptable as an alternative for resolution of USI A-2 for sixteen facilities owned by eleven licensees in the Westinghouse Owner's Group (one of these facilities, Fort Calhoun has a Combustion Engineering nuclear steam supply system). This NRC staff position was stated in Generic Letter 84-04, published on February 1, 1984 (Reference 4). The generic letter states that the affected licensees must justify an exemption to GDC 4 on a plant-specific basis. Other PWR applicants or licensees may request similar exemptions from the requirements of GDC 4 provided that they submit an acceptable technical basis for eliminating the need to postulate pipe breaks.

The acceptance of an exemption was made possible by the development of advanced fracture mechanics technology. These advanced fracture mechanics techniques deal with relatively small flaws in piping components (either postulated or real) and examine their behavior under various pipe loads. The objective is to demonstrate by deterministic analyses that the detection of small flaws by either inservice inspection or leakage monitoring systems is assured long before the flaws can grow to critical or unstable sizes which could lead to large break areas such as the DEGB or its equivalent. The concept underlying such analyses is referred to as "leak-before-break" (LBB). There is no implication that piping failures cannot occur, but rather that improved knowledge of the failure modes of piping systems and the application of appropriate remedial measures, if indicated, can reduce the probability of catastrophic failure to insignificant values.

Advanced fracture mechanics technology was applied in topical reports (References 5, 6 and 7) submitted to the staff by Westinghouse on behalf of the licensees belonging to the USI A-2 Owners Group, Although the topical reports were intended to resolve the issue of asymmetric blowdown loads that resulted from a limited number of discrete break locations, the technology advanced in these topical reports demonstrated that the probability of breaks occurring in the primary coolant system main loop piping is sufficiently low such that these breaks need not be considered as a design basis for requiring installation of pipe whip restraints or jet impingement shields. The staff's Topical Report Evaluation is attached as Enclosure 1 to Reference 4.

Probabilistic fracture mechanics studies conducted by the Lawrence Livermore National Laboratories (LLNL) on both Westinghouse and Combustion Engineering nuclear steam supply system main loop piping (Reference 8) confirm that both the probability of leakage (e.g., undetected flaw growth through the pipe wall by fatigue) and the probability of a DEGB are very low. The results given in Reference 8 are that the best-estimate leak probabilities for Westinghouse nuclear steam supply system main loop piping range from 1.2×10^{-8} to 1.5×10^{-7} per plant year and the best-estimate DEGB probabilities range from 1×10^{-12} to 7×10^{-12} per plant year. Similarly, the best-estimate leak probabilities for Combustion Engineering nuclear steam supply system main loop piping range from 1×10^{-8} per plant year to 3×10^{-8} per plant year, and the best-estimate DEGB probabilities range from 5×10^{-14} to 5×10^{-13} per plant year. These results do not affect core melt probabilities in any significant way.

During the past few years it has also become apparent that the requirement for installation of larg[illegible] ive pipe whip restraints and jet impingement shields is not necessarily the most cost effective way to achieve the desired level of safety, as indicated in Enclosure 2, Regulatory Analysis, to Reference 4. Even for new plants, these devices tend to restrict access for future inservice inspection of piping; or if they are removed and reinstalled for inspection, there is a potential risk of damaging the piping and other safety-related components in this process. If installed in operating plants, high occupational radiation exposure (ORE) would be incurred while public risk reduction would be very low. Removal and reinstallation for inservice inspection also entail significant ORE over the life of a plant.

IV.

The primary coolant system of Comanche Peak Units 1 and 2, described in Reference 3, has four main loops each comprising a 33.9 inch diameter hot leg, a 36.2 inch diameter crossover leg and 32.14 inch diameter cold leg piping. The material in the primary loop piping is cast stainless steel (SA 351 CF8A). In its review of Reference 3, the staff evaluated the Westinghouse analyses with regard to:

- the location of maximum stresses in the piping, associated with the combined loads from normal operation and the SSE;
- potential cracking mechanisms;
- size of through-wall cracks that would leak a detectable amount under normal loads and pressure;
- stability of a "leakage-size crack" under normal plus SSE loads and the expected margin in terms of load;

- margin based on crack size; and
- the fracture toughness properties of thermally-aged cast stainless steel piping and weld material.

The NRC staff's criteria for evaluation of the above parameters are delineated in its Topical Report Evaluation, Enclosure 1 to Reference 4, Section 4.1, "NRC Evaluation Criteria", and are as follows:

(1) The loading conditions should include the static forces and moments (pressure, deadweight and thermal expansion) due to normal operation, and the forces and moments associated with the safe shutdown earthquake (SSE). These forces and moments should be located where the highest stresses, coincident with the poorest material properties, are induced for base materials, weldments and safe-ends.

(2) For the piping run/systems under evaluation, all pertinent information which demonstrates that degradation or failure of the piping resulting from stress corrosion cracking, fatigue or water hammer is not likely, should be provided. Relevant operating history should be cited, which includes system operational procedures; system or component modification; water chemistry parameters, limits and controls; resistance of material to various forms of stress corrosion, and performance under cyclic loadings.

(3) A through-wall crack should be postulated at the highest stressed locations determined from (1) above. The size of the crack should be large enough so that the leakage is assured of detection with adequate margin using the minimum installed leak detection capability when the pipe is subjected to normal operational loads.

(4) It should be demonstrated that the postulated leakage crack is stable under normal plus SSE loads for long periods of time; that is, crack growth, if any, is minimal during an earthquake. The margin, in terms of applied loads, should be determined by a crack stability analysis, i.e., that the leakage-size crack will not experience unstable crack growth even if larger loads (larger than design loads) are applied. This analysis should demonstrate that crack growth is stable and the final crack size is limited, such that a double-ended pipe break will not occur.

(5) The crack size should be determined by comparing leakage-size crack to critical-size cracks. Under normal plus SSE loads, it should be demonstrated that there is adequate margin between the leakage-size crack and the critical-size crack to account for the uncertainties inherent in the analyses, and leakage detection capability. A limit-load analysis may suffice for this purpose, however, an elastic-plastic fracture mechanics (tearing instability) analysis is preferrable.

(6) The materials data provided should include types of materials and materials specifications used for base metal, weldments and safe-ends, the materials properties including the J-R curve used in the analyses, and long-term effects such as thermal aging and other limitations to valid data (e.g. J maximum, ximum crack growth).

V.

Based on its evaluation of the analysis contained in Westinghouse Report WCAP-10527 (Reference 3), the staff finds that the applicant has presented an acceptable technical justification, addressing the above criteria, for not installing protective devices to deal with the dynamic effects of large pipe ruptures in the main loop primary coolant system piping of Comanche Peak, Units 1 and 2. This finding is predicated on the fact that each of the parameters evaluated for Comanche Peak is enveloped by the generic analysis performed by Westinghouse in Reference (5), and accepted by the staff in Enclosure 1 to Reference 4. Specifically:

(1) The loads associated with the highest stressed location in the main loop primary system piping are considerably lower than the bounding loads used by Westinghouse in Reference 5, or those established by the staff as limits (e.g. a moment of 42,000 in-kips in Enclosure 1 to Reference 4).

(2) For Westinghouse plants, there is no history of cracking failure in reactor primary coolant system loop piping. The Westinghouse reactor coolant system primary loop has an operating history which demonstrates its inherent stability. This includes a low susceptibility to cracking failure from the effects of corrosion (e.g. intergrannular stress corrosion cracking), water hammer, or fatigue (low and high cycle). This operating history totals over 400 reactor-years, including five plants each having 15 years of operation and 15 other plants with over 10 years of operation.

(3) The results of the leak rate calculations performed for Comanche Peak, using an initial through-wall crack are identical to those of Enclosure 1 to Reference (4). The Comanche Peak plant has an RCS pressure boundary leak detection system which is consistent with the guidelines of Regulatory Guide 1.45, and it can detect leakage of one (1) gpm in one hour. The calculated leak rate through the postulated flaw is large relative to the sensitivity of the Comanche Peak plant leak detection system.

(4) The expected margin in terms of load for the leakage-size crack under normal plus SSE loads is within the bounds calculated by the staff in Section 4.2.3 of Enclosure (1) to Reference 4. In addition, the staff found a significant margin in terms of loads larger than normal plus SSE loads.

(5) The margin between the leakage-size crack and the critical-size crack was calculated. Again, the results demonstrated that a significant margin exists and is within the bounds of Section 4.2.3 of Enclosure 1 to Reference 4.

(6) As an integral part of its review, the staff's evaluation of the material properties data of Reference 9 is enclosed as Appendix 1 to this Exemption. In Reference 9, data for ten (10) plants, including the Comanche Peak Units, are presented, and lower bound or "worst case" materials properties were identified and used in the analysis performed in the Reference 3 report by Westinghouse. The staff's upper bound of 3000 in-lb/in^2 on the applied J (refer to Appendix 1, page 6) was not exceeded; the applied J for Comanche Peak in Reference 3 was substantially less than 3000 in-lb/in^2.

In view of the analytical results presented in the Westinghouse Report for Comanche Peak (Reference 3) and the staff's evaluation findings related above, the staff concludes that the probability or likelihood of large pipe breaks occuring in the primary coolant system loop of Comanche Peak Units 1 and 2 is sufficiently low such that such pipe breaks need not be considered as a design basis for requiring protective devices. However, the pipe whip restraints have already been installed in Unit 1, and the applicant has limited the scope of its exemption request to the installation of jet impingement shields in Unit 1 only. The requested exemption from GDC 4 is limited to exemption from the need to install jet impingement shields at specified locations in Unit 1.

The staff also reviewed the value-impact analysis provided by the applicant for not providing protective structures against postulated reactor coolant system loop pipe breaks to assure as low as reasonably achievable (ALARA) exposure to plant personnel. Consideration was given to design features for reducing doses to personnel who must operate, service and maintain the Comanche Peak instrumentation, controls, equipment, etc. Normally, facilities and equipment are designed to <u>save</u> person-rems; however, the Comanche Peak value-impact analysis shows that the addition of protective devices for RCS pipe breaks will <u>cost</u> about 2 person-rems annually due to the slowing down of normally anticipated work, and increasing the scope of routine maintenance in radiation areas that would be involved. The analysis provides a reasonable estimate for this additional radiological cost. In view of the very low probability of pipe breaks at the specified locations covered by this exemption, the reduction of occupational exposure resulting from this exemption outweighs the potential accident exposure reduction that might result from installation of the jet impingement barriers.

VI.

In view of the staff's evaluation findings, conclusions, and recommendations above, the Commission has determined that, pursuant to 10 CFR 50.12(a), this Exemption is authorized by law and will not endanger life or property or the common defense [illegible] is otherwise in the public interest. The Commission he[illegible] ited exemption from GDC 4 o[illegible] nsee not to install

jet impingement shields associated with postulated pipe breaks of the eight (8) locations per loop in the Comanche Peak Unit 1 primary coolant system, as specified in Section 4.0 of the value-impact analysis submit by the applicant's letter dated April 23, 1984. This Exemption does not pertain to the installation of pipe whip restraints, already installed in Unit 1, or to the installation of pipe whip restraints and jet impingement shields in Comanche Peak Unit 2. The portion of the request concerning Unit 2 will be dealt with in a separate NRC action.

The Commission has determined that the issuance of the exemption will have no significant environmental impact on the environment (49 FR 33945).

FOR THE NUCLEAR REGULATORY COMMISSION

Frank Miraglia

Frank Miraglia, Deputy Director
Division of Licensing
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 28th day of August 1984

REFERENCES

(1) Westinghouse Report MT-SME-3135, "Technical Bases for Eliminating Large Primary Loop Pipe Ruptures as the Structural Design Basis for Comanche Peak Units 1 and 2," October 1983, Westinghouse Class 2 proprietary.

(2) Letter to R. J. Gary of Texas Utilities Generating Company, "Request for Additional Information Concerning Leak-Before-Break Analysis for Comanche Peak Steam Electric Station (Units 1 and 2)," dated March 2, 1984.

(3) Westinghouse Report WCAP-10527, "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Comanche Peak Units 1 and 2," April 1984, Westinghouse Class 2 proprietary.

(4) NRC Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Breaks in PWR Primary Main Loops," February 1, 1984.

(5) Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Throughwall Crack, WCAP-9558, Rev. 2, May 1981, Westinghouse Class 2 proprietary.

(6) Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation, WCAP-9787, May 1981, Westinghouse Class 2 proprietary.

(7) Westinghouse Response to Questions and Comments Raised by Members of ACRS Subcommittee on Metal Components During the Westinghouse Presentation on September 25, 1981, Letter Report NS-EPR-2519, E. P. Rahe to Darrell G. Eisenhut, November 10, 1981, Westinghouse Class 2 proprietary.

(8) Lawrence Livermore National Laboratory Rpeort, UCRL-86249, "Failure Probability of PWR Reactor Coolant Loop Piping," by T. Lo, H. H. Woo, G. S. Holman and C. K. Chou, February 1984 (Preprint of a paper intended for publication).

(9) Westinghouse Report WCAP-10456, "The Effects of Thermal Aging on the Structural Integrity of Cast Stainless Steel Piping for Westinghouse Nuclear Steam Supply Systems," November 1983, Westinghouse Class 2 proprietary.

(10) Texas Utilities Generating Company letter TXX-4197, "Request for Partial Exemption" (H. C. Schmidt to B. J. Youngblood) dated June 7, 1984.

(11) Texas Utilities Generating Company letter TXX-4118, "Request for Partial Exemption," (R. J. Gary to B. J. Youngblood) dated February 17, 1984.

Notes: See next page

REFERENCES

NOTE. Non-proprietary versions of References 1, 3, 5, 6, 7 and 9 are available in the NRC Public Document Room as follows:

(1) MT-SME-3136
(3) WCAP 10528
(5) WCAP 9570
(6) WCAP 9788
(7) Non-proprietary version attached to the Letter Report
(9) WCAP 10457

APPENDIX 1

Evaluation of Westinghouse Report
WCAP 10456, "The Effects of Thermal Aging on the Structural Integrity of Cast Stainless Steel Piping for Westinghouse Nuclear Steam Supply Systems"

INTRODUCTION

The primary coolant piping in some Westinghouse Nuclear Steam Supply Systems (NSSS) contain cast stainless steel base metal and weld metal. The base metal and weld metal are fabricated to produce a duplex structure of delta (δ) ferrite in an austenitic matrix. The duplex structure produces a material that has a higher yield strength, improved weldability and greater resistance to intergranular stress corrosion cracking than a single phase austenitic material. However, as early as 1965 (Ref.1), it was recognized that long time thermal aging at primary loop water temperatures (550°F-650°F) could significantly affect the Charpy impact toughness of the duplex structured alloys. Since the Charpy impact test is a measure of a material's resistance to fracture, a loss in Charpy impact toughness could result in reduced structural stability in the piping system.

The purpose of Report WCAP 10456 is to evaluate whether cast stainless steel base metal and weld metal containing postulated cracks will be sensitive to unstable fracture during the 40 year life of a nuclear power plant. In order to determine whether a piping system will behave

in such a fashion, the pipe materials' mechanical properties, design criteria and method of predicting failure must be established. In this evaluation, we assess the mechanical properties of thermally aged cast stainless steel pipe materials, which are reported in Report WCAP 10456.

DISCUSSION

1. Weld Metal

Report WCAP 10456 refers to test results reported in a paper by Slama, et.al. (Ref. 2) to conclude that the weld metal in primary loop piping would not be overly sensitive to aging and that the aged cast pipe base metal material would be structurally limiting. In the Slama report eight (8) welds were evaluated. The tensile properties were only slightly affected by aging. The Charpy U-notch impact energy in the most highly sensitive weld decreased from 7daJ/cm^2 (40 ft-lbs) to near 4daJ/cm^2 (24 ft-lbs) after aging for 10,000 hours at 400°C (752°F). This change was not considered significant. The relatively small effect of aging on the weld, as compared to cast pipe material was reported to be caused by a difference in microstructure and lower levels of ferrite in the weld than in the cast pipe material.

2. Cast Stainless Steel Pipe Base Metal

Report WCAP 10456 contains mechanical property test results from a number of heats of aged cast stainless steel material and a metallurgical study, which was performed by Westinghouse, to support a statistically based model for predicting the effect of thermal aging on the Charpy impact test properties of cast stainless steel. As a result of these tests and the proposed model, Westinghouse concludes that the fracture toughness test results from one heat of material tested represents end-of-life conditions for the ten (10) plants surveyed. The ten (10) plants surveyed are identified as Plants A through J.

a. Mechanical Property Test Results Reported in WCAP 10456

Mechanical property test results on aged and unaged cast stainless steel materials, as reported in papers by Landerman and Bamford (Ref. 3), Bamford, Landerman and Diaz (Ref. 4), Slama et al. (Ref. 2), were discussed in Report 10456. In addition, Westinghouse performed confirmatory Charpy V notch and J-integral tests on aged cast stainless steel material, which was tested and evaluated by Slama et al.

The results of these tests indicate that:

(1) The fatigue crack growth rates of aged or unaged material in air and pressurized water reactor environments were equivalent.

(2) Tensile properties were essentially unaffected except for a slight increase in tensile strength and a decrease in ductility.

(3) J-integral test results indicate that the J_{1C} and tearing modulus, T, are affected by aging.

b. Mechanism Study in WCAP 10456

The tests and literature survey conducted by Westinghouse indicate that the proposed mechanism of aging occurs in the range of operating temperatures for pressurized water reactors and the data from accelerated aging studies can be used to predict the behavior at operating temperatures.

c. Cast Stainless Steel Pipe Test

The materials data discussed in the previous section of this evaluation were obtained from small specimens. As a consequence, the J-R results are limited to relatively short crack extensions. To investigate the behavior of cast stainless steel in actual piping geometry, Westinghouse performed two experiments, one of which was with thermally aged cast stainless steel and the other test was identical except that the steel was not thermally aged.

Each pipe tested contained a throughwall circumferential crack to the extent specified in WCAP 10456. The pipe sections were closed at the ends, pressurized to nominal PWR operating pressure and then bending loads were applied.

The results of the tests were very similar, in that both pipes displayed extensive ductility, and stable crack extension. There was no observed unstable crack extension or fast fracture.

The results of the Westinghouse pipe experiments indicate that cast stainless steel, both aged and unaged, can withstand crack extensions well beyond the range of the J-R results with small specimens. However, if crack extension is predicted in an actual application of thermally aged cast stainless steel in a piping system, we believe that it is prudent to limit the applied J to 3000 in-lbs/in^2 or less unless further studies and/or experiments demonstrate that higher values are tolerable. Loss of initial toughness due to thermal aging of cast stainless steels at normal nuclear facility operating temperatures occurs slowly over the course of many years; therefore, continuing study of the aging phenomenon may lead to a relaxation of this position. Conversely, in the unlikely event that the total loss of toughness and the rate of toughness are greater than those projected in this evaluation, the staff will take appropriate action to limit the values to that which can be justified by experimental data. Because the aging is a slow process, the staff believes there would be sufficient time for the staff to recognize the problem and to rectify the situation. However, the staff believes this situation is highly unlikely because the staff has accepted only the lower bounds of data that were gathered among ten plants encompassing the range of materials in use

d. Effects of Thermal Aging on Westinghouse Supplied Centrifugally Cast Reactor Coolant Piping Reported in WCAP 10456

The reactor coolant cast stainless steel piping materials in the plants identified in WCAP 10456 as A through J, were produced to the specification SA-351, Class CF8A as outlined in ASME Code Section II, Part A and also to Westinghouse Equipment Specification G-678864, as revised. For these materials, Westinghouse has calculated the predicted end-of-life Charpy U-notch properties, based on their proposed model. The two (2) standard deviation end-of-life lower limit value for all the plants surveyed was greater than the Charpy V notch properties of the aged reference materials, which Westinghouse indicates represents end-of-life properties for all the plants. As a result, Westinghouse concluded that the amount of embrittlement in the aged reference material exceed the amount projected at end-of-life for all cast stainless steel pipe materials in Plants A through J.

Conclusions

Based on our review of the information and data contained in Westinghouse Report WCAP 10456, we conclude that:

1. Weld metal that is used in cast stainless steel piping system is initially less fracture resistant than the cast stainless steel base metal. However, the weld metal is less susceptible to thermal aging than the cast stainless steel base metal. Hence, at end-of-life the cast stainless steel base metal is anticipated to be the least fracture resistant material.

2. The Westinghouse proposed model may be used to predict the relative amount of embrittlement on a heat of cast stainless steel material. The two standard deviation lower confidence limit for this model will provide a useful engineering estimate of the predicted end-of-life Charpy impact properties for cast stainless steel base metal.

3. Since there is considerable scatter in J-integral test data for the heats of material tested, lower bound values for J_{1c} and T should be used as engineering estimates for the fracture resistance of the aged reference material. We believe these values should also provide a lower bound for the fracture resistance of aged and unaged weld metal. If crack extension is predicted in an actual application of cast stainless steel in a piping system, we conclude that the applied J should be limited to 3000 in-lbs/in^2 or less unless further studies and tests demonstrate that higher values are tolerable. The Westinghouse pipe tests demonstrate that this may be possible.

4. Since the predicted end-of-life Charpy impact values for the materials in Plants A through J are greater than the value measured for the aged reference material, the lower bound fracture properties for aged reference material may be used to determine the fracture resistance for the cast stainless steel material in Plants A through J.

REFERENCES

(1) F. H. Beck, E. A. Schoefer, J. W. Flowers, M. E. Fontana, "New Cast High Strength Alloy Grades by Structural Control," ASTM STP 369 (1965)

(2) G. Slama, P. Petrequin, S. H. Masson, T. R. Mager, "Effect of Aging on Mechanical Properties of Austenitic Stainless Steel Casting and Welds," presented at SMIRT 7 Post Conference Seminar 6 - Assuring Structural Integrity of Steel Reactor Pressure Boundary Components, August 29/30, 1983, Monterey, Ca.

(3) E. I. Landerman and W. H. Bamford, "Fracture Toughness and Fatigue Characteristics of Centrifugally Cast Type 316 Stainless Steel After Simulated Thermal Service Conditions. Presented at the Winter Annual Meeting of the ASME, San Francisco, Ca., December 1978 (MPC-8 ASME)

(4) W. H. Bamford, E. I. Landerman and E. Diaz, "Thermal Aging of Cast Stainless Steel and Its Impact on Piping Integrity." Presented at ASME Pressure Vessel and Piping Conference, Portland, Oregon, June 1983. To be published in ASME Trans.

ATTACHMENT F

Alternative Use of Resources

This action does not involve the use of resources not previously considered in connection with the Final Environmental Statement Relating to Operation of Millstone Unit 2.

Agencies and Persons Consulted

The NRC staff reviewed the licensee's request and did not consult other agencies or persons.

Finding of No Significant Impact

The Commission has determined not to prepare an environmental statement for the proposed relief.

Based upon the foregoing environmental assessment, we conclude that the proposed action will not have a significant effect on the quality of the human environment.

For further details with respect to this action, see the application for relief dated May 4, 1984, which is available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C. and at the Waterford Public Library, Waterford, Connecticut.

Dated at Bethesda, Maryland this 28th day of August 1984.

For the Nuclear Regulatory Commission.

Gus C. Lainas,

Division of Licensing Office of Nuclear Reactor Regulation.

[FR Doc. 84-23454 Filed 9-4-84; 8:45 am]

BILLING CODE 7590-01-M

[Docket No. 50-275]

Pacific Gas and Electric Co. (Diablo Canyon Nuclear Power Plant, Unit 1); Issuance of a Director's Decision Under 10 CFR 2.206

Notice is hereby given that the Director, Office of Nuclear Reactor Regulation, has issued a decision concerning Petitions dated February 2, March 1, March 23, April 12, May 3, June 21, June 22, July 11, July 18, and July 23, 1984 filed by the Government Accountability Project on behalf of the San Luis Obispo Mothers for Peace. The Petitioner requested that the Commission defer all licensing decisions on the Diablo Canyon Nuclear Power Plant, Unit 1 until a number of specified actions were taken including, *inter alia*, a comprehensive third-party reinspection of all safety-related equipment, an independent management audit and a full investigation of questions of harassment. The Petitioner alleged numerous violations of Commission requirements as the basis for its request. The petitions were referred to the Director, Office of Nuclear Reactor Regulation for treatment pursuant to 10 CFR 2.206 of the Commission's regulations and a final Director's Decision has been issued by the Director denying the Petitioner's request. The reasons for this denial are explained in the "Director's Decision under 10 CFR 2.206" (DD-84-20), which is available for inspection in the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C. 20555 and at the Local Public Document Room at the Robert E. Kennedy Library, California Polytechnic State University, San Luis Obispo, California 93407.

A copy of the decision will be filed with the Secretary for Commission review in accordance with 10 CFR 2.206(c). As provided in 10 CFR 2.206(c), the decision will become the final action of the Commission 25 days after issuance, unless the Commission, on its own motion, takes review of the decision within that time.

Dated at Bethesda, Maryland, this 20th day of August 1984.

For the Nuclear Regulatory Commission.

Harold R. Denton,

Director, Office of Nuclear Reactor Regulation.

[FR Doc. 84-23453 Filed 9-4-84; 8:45 am]

BILLING CODE 7590-01-M

[Docket Nos. 50-445 and 50-446]

Texas Utilities Generating Co. (Comanche Peak Steam Electric Station, Units 1 and 2); Exemption

I

On July 20, 1973, the Texas Utilities Generating Company (the applicant) tendered an application for licenses to construct Comanche Peak Steam Electric Station, Units 1 and 2 (Comanche Peak or the facility) with the Atomic Energy Commission (currently the Nuclear Regulatory Commission or the Commission). Following a public hearing before the Atomic Safety and Licensing Board, the Commission issued Construction Permit Nos. CPPR-126 and CPPR-127 permitting the construction of Units 1 and 2, respectively, on December 19, 1974. Each Unit of the facility is a pressurized water reactor, combining a Westinghouse Electric Company nuclear steam supply system, located at the applicant's site in Somervell/Hood Counties, Texas, approximately 40 miles southwest of Fort Worth, Texas.

On February 27, 1978, the applicant tendered an application for Operating Licenses for each Unit of the facility, currently in the licensing review process, with Unit 1 licensing to occur in the near term.

II

The Construction Permits issued for constructing the facility provide, in pertinent part, that the facility Units are subject to all rules, regulations and Orders of the Commission. This includes General Design Criterion (GDC) 4 of Appendix A to 10 CFR 50. GDC 4 requires that structures, systems and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with the normal operation, maintenance, testing and postulated accidents, including loss-of-coolant accidents. These structures, systems and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, discharging fluids that may result from equipment failures, and from events and conditions outside the nuclear power unit.

By a submittal dated October 31, 1983, the applicant requested an exemption from a portion of the requirements of GDC 4 to: (1) Eliminate the need to postulate circumferential and longitudinal pipe breaks in the Reactor Coolant System (RCS) primary loop (hot leg, cold leg and cross-over leg piping); (2) eliminate the need to install pipe whip restraints and jet impingement shields associated with previously postulated breaks in the RCS primary loops and; (3) to eliminate the need to consider dynamic effects and loading conditions associated with previously postulated pipe breaks in the RCS primary loop, including jet impringement loads, cavity pressure loads, blowdown loads in the RCS and attached piping, and subcompartment pressure loads. In support of this exemption request, the applicant's submittal enclosed Westinghouse Report MT-SME-3135 (Reference 1) containing the technical basis for their request.

Based on its review of the applicant's submittal, the NRC staff requested additional information and provided comments on the reports (References 1 and 9) which were transmitted to the applicant in the form of questions by NRC letter dated March 2, 1984, (Reference 2).

By a submittal dated April 23, 1984, the applicant responded to the staff's questions (Reference 2) and provided a revision to the Reference 1 report identified as Westinghouse Report WCAP-10527 (Reference 3). In a separate submittal, also dated April 23, 1984, the applicant provided a value-impact analysis which, together with the technical information contained in the

Reference 3 report, provided a comprehensive justification for requesting a partial exemption from the requirements of GDC 4.

From the deterministic fracture mechanics analysis contained in the technical information furnished, the applicant stated that the postulated double-ended guillotine breaks (DEGB) of the primary loop coolant piping will not occur in Comanche Peak Units 1 and 2 and, therefore, need not be considered as a design basis for installing protective structures, such as pipe whip restraints and jet impingement shields, to guard against the dynamic effects associated with such postulated breaks.

By letter dated June 7, 1984 (Reference 10), the applicant clarified the scope of its request for exemption from GDC 4 requirements. Since the Westinghouse Report WCAP-10527 provided analyses encompassing other structures in both Comanche Peak Units 1 and 2, and seemed to be in conflict with the scope of the exemption requested in an earlier letter dated February 17, 1984 (Reference 11), the applicant stated in the Reference 10 letter that, although the analyses contained in the Report WCAP-10527 encompassed relief from the need to install pipe break protective devices in both Units 1 and 2, the exemption being requested pertained solely to the installation of jet impingement shields associated with such breaks in eight (8) locations per loop in Comanche Peak Unit 1, as specified in Section 4.0 of the value-impact analysis submitted by the applicant's letter dated April 23, 1984.

III

The Commission's regulations require that applicants provide protective measures against the dynamic effects of postulated pipe breaks in high energy fluid system piping. Protective measures include physical isolation from postulated pipe rupture locations if feasible or the installation of pipe whip restraints, jet impingement shields or compartments. In 1975, concerns arose as to the asymmetric loads on pressurized water reactor (PWR) vessels and their internals which could result from these large postulated breaks at discrete locations in the main primary coolant loop piping. This led to the establishment of Unresolved Safety Issue (USI) A-2, "Asymmetric Blowdown Loads on PWR Primary Systems."

The NRC staff, after several review meetings with the Advisory Committee on Reactor Safeguards (ACRS) and a meeting with the NRC Committee to Review Generic Requirements (CRGR), concluded that for certain facilities an exemption from the regulations would be acceptable as an alternative for resolution of USI A-2 for sixteen facilities owned by eleven licensees in the Westinghouse Owner's Group (one of these facilities, Fort Calhoun has a Combustion Engineering nuclear steam supply system). This NRC staff position was stated in Generic Letter 84-04, published on February 1, 1984 (Reference 4). The generic letter states that the affected licensees must justify an exemption to GDC 4 on a plant-specific basis. Other PWR applicants or licensees may request similar exemptions from the requirements of GDC 4 provided that they submit an acceptable technical basis for eliminating the need to postulate pipe breaks.

The acceptance of an exemption was made possible by the development of advanced fracture mechanics technology. These advanced fracture mechanics techniques deal with relatively small flaws in piping components (either postulated or real) and examine their behavior under various pipe loads. The objective is to demonstrate by deterministic analyses that the detection of small flaws by either inservice inspection or leakage monitoring systems is assured long before the flaws can grow to critical or unstable sizes which could lead to large break areas such as the DEGB or its equivalent. The concept underlying such analyses is referred to as "leak-before-break" (LBB). There is no implication that piping failures cannot occur, but rather that improved knowledge of the failure modes of piping systems and the application of appropriate remedial measures, if indicated, can reduce the probability of catastrophic failure to insignificant values.

Advanced fracture mechanics technology was applied in topical reports (References 5, 6 and 7) submitted to the staff by Westinghouse on behalf of the licensees belonging to the USI A-2 Owners Group. Although the topical reports were intended to resolve the issue of asymmetric blowdown loads that resulted from a limited number of discrete break locations, the technology advanced in these topical reports demonstrated that the probability of breaks occurring in the primary coolant system main loop piping is sufficiently low such that these breaks need not be considered as a design basis for requiring installation of pipe whip restraints or jet impingement shields. The staff's Topical Report Evaluation is attached as Enclosure 1 to Reference 4.

Probabilistic fracture mechanics studies conducted by the Lawrence Livermore National Laboratories (LLNL) on both Westinghouse and Combustion Engineering nuclear steam supply system main loop piping (Reference 8) confirm that both the probability of leakage (e.g., undetected flaw growth through the pipe wall by fatigue) and the probability of a DEGB are very low. The results given in Reference 8 are that the best-estimate leak probabilities for Westinghouse nuclear steam supply system main loop piping range from 1.2 x 10^{-5} to 1.5 x 10^{-7} per plant year and the best-estimate DEGB probabilities range from 1 x 10^{-12} to 7 x 10^{-12} per plant year. Similarly, the best-estimate leak probabilities for Combustion Engineering nuclear steam supply system main loop piping range from 1 x 10^{-8} per plant year to 3 x 10^{-4} per plant year, and the best-estimate DEGB probabilities range from 5 x 10^{-14} to 5 x 10^{-13} per plant year. These results do not affect core melt probabilities in any significant way.

During the past few years it has also become apparent that the requirement for installation of large, massive pipe whip restraints and jet impingement shields is not necessarily the most cost effective way to achieve the desired level of safety, as indicated in Enclosure 2, Regulatory Analysis, to Reference 4. Even for new plants, these devices tend to restrict access for future inservice inspection of piping; or if they are removed and reinstalled for inspection, there is a potential risk of damaging the piping and other safety-related components in this process. If installed in operating plants, high occupational radiation exposure (ORE) would be incurred while public risk reduction would be very low. Removal and reinstallation for inservice inspection also entail significant ORE over the life of a plant.

IV

The primary coolant system of Comanche Peak Units 1 and 2, described in Reference 3, has four main loops each comprising a 33.9 inch diameter hot leg, a 36.2 inch diameter crossover leg and 32.14 inch diameter cold leg piping. The material in the primary loop piping is cast stainless steel (SA 351 CF8A). In its review of Reference 3, the staff evaluated the Westinghouse analyses with regard to:

—The location of maximum stresses in the piping, associated with the combined loads from normal operation and the SSE;
—Potential cracking mechanisms;
—Size of through-wall cracks that would leak a detectable amount under normal loads and pressure;

—Stability of a "leakage-size crack" under normal plus SSE loads and the expected margin in terms of load;
—Margin based on crack size; and
—The fracture toughness properties of thermally-aged cast stainless steel piping and weld material.

The NRC staff's criteria for evaluation of the above parameters are delineated in its Topical Report Evaluation, Enclosure 1 to Reference 4, Section 4.1, "NRC Evaluation Criteria", and are as follows:

(1) The loading conditions should include the static forces and moments (pressure, deadweight and thermal expansion) due to normal operation, and the forces and moments associated with the safe shutdown earthquake (SSE). These forces and moments should be located where the highest stresses, coincident with the poorest material properties, are induced for base materials, weldments and safe-ends.

(2) For the piping run/systems under evaluation, all pertinent information which demonstrates that degradation or failure of the piping resulting from stress corrosion cracking, fatigue or water hammer is not likely, should be provided. Relevant operating history should be cited, which includes system operational procedures; system or component modification; water chemistry parameters, limits and controls; resistance of material to various forms of stress corrosion, and performance under cyclic loadings.

(3) A through-wall crack should be postulated at the highest stressed locations determined from (1) above. The size of the crack should be large enough so that the leakage is assured of detection with adequate margin using the minimum installed leak detection capability when the pipe is subjected to normal operational loads.

(4) It should be demonstrated that the postulated leakage crack is stable under normal plus SSE loads for long periods of time; that is, crack growth, if any, is minimal during an earthquake. The margin, in terms of applied loads, should be determined by a crack stability analysis, i.e., that the leakage-size crack will not experience unstable crack growth even if larger loads (larger than design loads) are applied. This analysis should demonstrate that crack growth is stable and the final crack size is limited, such that a double-ended pipe break will not occur.

(5) The crack size should be determined by comparing leakage-size cracks to critical-size cracks. Under normal plus SSE loads, it should be demonstrated that there is adequate margin between the leakage-size crack and the critical-size crack to account for the uncertainties inherent in the analyses, and leakage detection capability. A limit-load analysis may suffice for this prupose, however, an elastic-plastic fracture mechanics (tearing instability) analysis is preferable.

(6) The materials data provided should include types of materials and materials specificatons used for base metal, weldments and safe-ends, the materials properties including the J–R curve used in the analyses, and long-term effects such as thermal aging and other limitations to valid data (e.g. J maximum, maximum crack growth).

V

Based on its evaluation of the analysis contained in Westinghouse Report WCAP-10527 (Reference 3), the staff finds that the applicant has presented an acceptable technical justification, addressing the above criteria, for not installing protective devices to deal with the dynamic effects of large pipe ruptures in the main loop primary coolant system piping of Comanche Peak, Units 1 and 2. This finding is predicated on the fact that each of the parameters evaluated for Comanche Peak is *enveloped* by the generic analysis performed by Westinghouse in Reference (5), and accepted by the staff in Enclosure 1 to Reference 4. Specifically:

(1) The loads associated with the highest location in the main loop primary system piping are considerably lower than the bounding loads used by Westinghouse in Reference 5, or those established by the staff as limits (e.g., a moment of 42,000 in-kips in Enclosure 1 to Reference 4).

(2) For Westinghouse plants, there is no history of cracking failure in reactor primary coolant system loop piping. The Westinghouse reactor coolant system primary loop has an operating history which demonstrates its inherent stability. This includes a low susceptibility to cracking failure from the effects of corrosion (e.g. intergranular stress corrosion cracking), water hammer, or fatigue (low and high cycle). This operating history totals over 400 reactor-years, including five plants each having 15 years of operation and 15 other plants with over 10 years of operation.

(3) The results of the leak rate calculations performed for Comanche Peak, using an initial through-wall crack are identical to those of Enclosure 1 to Reference (4), the Comanche Peak plant has an RCS pressure boundary leak detection system which is consistent with the guidelines of Regulatory Guide 1.45 and it can detect leakage of one (1) gpm in one hour. The calculated leak rate through the postulated flaw is large relative to the sensitivity of the Comanche Peak plant leak detection system.

(4) The expected margin in terms of load for the leakage-size crack under normal plus SSE loads is within the bounds calculated by the staff in Section 4.2.3 of Enclosure (1) to Reference 4. In addition, the staff found a significant margin in terms of loads larger than normal plus SSE loads.

(5) The margin between the leakage-size crack and the critical-size crack was calculated. Again, the results demonstrated that a significant margin exists and is within the bounds of Section 4.2.3 of Enclosure 1 to Reference 4.

(6) As an integral part of its review, the staff's evaluation of the material properties data of Reference 9 is enclosed as Appendix 1 to this Exemption. In Reference 9, data for ten (10) plants, including the Comanche Peak Units, are presented, and lower bound or "worst case" materials properties were identified and used in the analysis performed in the Reference 3 report by Westinghouse. The staff's upper bound of 3000 in-lb/in^2 on the applied J (refer to Appendix 1, page 6) was not exceeded; the applied J for Comanche Peak in Reference 3 was substantially less than 3000 in-lb/in^2.

In view of the analytical results presented in the Westinghouse Report for Comanche Peak (Reference 3) and the staff's evaluation findings related above, the staff concludes that the probability or likelihood of large pipe breaks occurring in the primary coolant system loop of Comanche Peak Units 1 and 2 is sufficiently low so such that such pipe breaks need not be considered as a design basis for requiring protective devices. However, the pipe whip restraints have already been installed in Unit 1, and the applicant has limited the scope of its exemption request to the installation of jet impingement shields in Unit 1 only. The requested exemption from GDC 4 is limited to exemption from the need to install jet impingement shields at specified locations in Unit 1.

The staff also reviewed the value-impact analysis provided by the applicant for not providing protective structures against postulated reactor coolant system loop pipe breaks to assure as low as reasonably achievable (ALARA) exposure to plant personnel. Consideration was given to design features for reducing doses to personnel who must operate, service and maintain the Comanche Peak instrumentation.

controls, equipment, etc. Normally, facilities and equipment are designed to *save* person-rems; however, the Comanche Peak value-impact analysis shows that the addition of protective devices for RCS pipe breaks will *cost* about 2 person-rems annually due to the slowing down of normally anticipated work, and increasing the scope of routine maintenance in radiation areas that would be involved. The analysis provides a reasonable estimate for this additional radiological cost. In view of the very low probability of pipe breaks at the specified locations covered by this exemption, the reduction of occupational exposure resulting from this exemption outweighs the potential accident exposure reduction that might result from installation of the jet impingement barriers.

VI

In view of the staff's evaluation findings, conclusions, and recommendation above, the Commission has determined that, pursuant to 10 CFR 50.12(a), this Exemption is authorized by law and will not endanger life or property or the common defense and security, and is otherwise in the public interest. The Commission hereby approves the requested limited exemption from GDC 4 of Appendix A to 10 CFR Part 50, to permit the licensee not to install jet impingement shields associated with postulated pipe breaks of the eight (8) locations per loop in the Comanche Peak Unit 1 primary coolant system, as specified in Section 4.0 of the value-impact analysis submit by the applicant's letter dated April 23, 1984. This Exemption does not pertain to the installation of pipe whip restraints, already installed in Unit 1, or to the installation of pipe whip restraints and jet impingement shields in Comanche Peak Unit 2. The portion of the equest Unit 2 will be dealt with in a separate NRC action.

The Commission has determined that the issuance of the exemption will have no significant environmental impact on the environment (49 FR 33945).

Dated at Bethesda, Maryland this 28th day of August 1984.

For the Nuclear Regulatory Commission.

Frank Miraglia,

Deputy Director, Division of Licensing, Office of Nuclear Reactor Regulation.

References

(1) Westinghouse Report MT-SME-3135, "Technical Bases for Eliminating Large Primary Loop Pipe Ruptures as the Structural Design Basis for Comanche Peak Unit 1 and 2," October 1983, Westinghouse Class 2 proprietary.

(2) Letter to R.J. Gary of Texas Utilities Generating Company, "Request for Additional Information Concerning Leak-Before-Break Analysis for Comanche Peak Steam Electric Station (Units 1 and 2)," dated March 2, 1984.

(3) Westinghouse Report WCAP-10527, "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Comanche Peak Units 1 and 2," April 1984, Westinghouse Class 2 proprietary.

(4) NRC Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Breaks in PWR Primary Main Loops," February 1, 1984.

(5) Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Throughwall Crack, WCAP-9558, Rev. 2, May 1981, Westinghouse Class 2 proprietary.

(6) Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation, WCAP-9787, May 1981, Westinghouse Class 2 proprietary.

(7) Westinghouse Response to Questions and Comments Raised by Members of ACRS Subcommittee on Metal Component During the Westinghouse Presentation on September 25, 1981, Letter Report NS-EPR-2519, E.P. Rahe to Darrell G. Eisenhut, November 10, 1981, Westinghouse Class 2 proprietary.

(8) Lawrence Livermore National Laboratory Report, UCRL-86249, "Failure Probability of PWR Reactor Coolant Loop Piping," by T. Lo, H.H. Woo, G. S. Holman and C.K. Chou, February 1984 (Preprint of a paper intended for publication).

(9) Westinghouse Report WCAP-10456, "The Effects of Thermal Aging on the Structural Integrity of Cast Stainless Steel Piping for Westinghouse Nuclear Steam Supply Systems," November 1983, Westinghouse Class 2 proprietary.

(10) Texas Utilities Generating Company letter TXX-4197, "Request for Partial Exemption" (H.C. Schmidt to B.J. Youngblood) dated June 7, 1984.

(11) Texas Utilities Generating Company letter TXX-4118, "Request for Partial Exemption," (R.J. Gary to B.J. Youngblood) dated February 17, 1984.

References

Note.—Non-Proprietary versions of References 1, 3, 5, 6, 7 and 9 are available in the NRC Public Document Room as follows: (1) MT-SME-3136, (3) WCAP 10528, (5) WCAP 9570, (6) WCAP 9788, (7) Non-proprietary version attached to the Letter Report, (9) WCAP 10457.

[FR Doc. 84-23451 Filed 9-4-84; 8:45 am]

BILLING CODE 7590-01-M

[Docket No. 50-266]

Wisconsin Electric Power Co.; Environmental Assessment and Finding of No Significant Impact

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of relief from the requirements of Section XI of the ASME Boiler and Pressure Vessel Code as specified by the provisions of 10 CFR 50.55a(b) to Wisconsin Electric Power Company (the licensee), for the Point Beach Nuclear Plant Unit No. 1, located in the Town of Two Creeks, Manitowoc County, Wisconsin.

Environmental Assessment

Identification of Proposed Action

The action would provide relief from the requirement to perform surface examinations of the safety injection reducer-to-safe end welds as required by Section XI of the ASME Boiler and Pressure Vessel Code which has been incorporated by reference in the requirements of 51 CFR 50.55a relating to Inservice Inspection of Safety Related Components. Volumetric examinations of these welds would be performed every 10 years as required.

The Need for the Proposed Action

The proposed relief is required because surface examinations of these welds are not possible due to the inaccessibility of the weld surfaces. The welds are located between the reactor vessel and the biological shield wall.

Environmental Impacts of the Proposed Action

The proposed relief is allowed by the provisions of 10 CFR 50.55a(g)(6)(i) where the tests or examinations required by the code are determined impractical to perform. As the surfaces of the welds in question are inaccessible, a surface examination has been determined by the licensee and evaluated by the Commission as impractical to perform. The staff has determined that the required volumetric inspection of the welds once every 10 years will provide adequate assurance of the structural integrity of the welds. Identical relief to that requested for Unit 1 was provided for Point Beach Unit 2 by the Commission's Safety Evaluation and letter of March 29, 1984.

Consequently, as the Commission has determined that the welds will retain adequate structural integrity utilizing the licensee's proposed alternate examination (volumetric examination once every 10 years), the probability of weld failure has not been increased significantly and the consequences of post-weld failure radiological releases will not be greater than previously determined nor does the requested relief otherwise affect radiological plant effluents. Therefore, the Commission has determined that there are no significant radiological environmental impacts associated with the requested relief.

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

Rcd. 9/29/84

SEP 2 1 1984

ATTACHMENT G

Docket Nos.: 50-445
and 50-446

Mr. M. D. Spence
President
Texas Utilities Generating Company
400 N. Olive Street
Lock Box 81
Dallas, Texas 75201

Dear Mr. Spence:

Subject: Request for Additional Information Concerning the Handling of Heavy Loads at the Comanche Peak Steam Electric Stations (Units 1 and 2)

As a part of its continuing review of the Comanche Peak FSAR, the staff has requested that the following additional information be provided regarding the handling of heavy loads:

1. Describe the means provided to assure the integrity of concrete structures, lifting eyes, and any other heavy loads so that they will not fall apart while being handled during refueling should the lifting eye fail or the plug impact other structures.

2. Alternatively, describe the consequences of failure of concrete structures or other heavy loads during handling. Your response should contain an evaluation to confirm that unacceptable fuel damage or damage to safety related equipment will not occur.

Your staff was advised of this forthcoming request by telephone. This letter serves to formally document the staff's request.

Please advise Mr. John Stefano of my staff when we may expect to receive your response upon receipt of this letter.

Sincerely,

B. J. Youngblood

B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

cc: See next page

COMANCHE PEAK

Mr. M. D. Spence
President
Texas Utilities Generating Company
400 N. Olive St., L.B. 81
Dallas, Texas 75201

cc: Nicholas S. Reynolds, Esq.
Bishop, Liberman, Cook,
Purcell & Reynolds
1200 Seventeenth Street, N. W.
Washington, D. C. 20036

Robert A. Wooldridge, Esq.
Worsham, Forsythe, Sampels &
Wooldridge
2001 Bryan Tower, Suite 2500
Dallas, Texas 75201

Mr. Homer C. Schmidt
Manager - Nuclear Services
Texas Utilities Generating Company
Skyway Tower
400 North Olive Street
L. B. 81
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

Renea Hicks, Esq.
Assistant Attorney General
Environmental Protection Division
P. O. Box 12548, Capitol Station
Austin, Texas 78711

Mrs. Juanita Ellis, President
Citizens Association for Sound
Energy
1426 South Polk
Dallas, Texas 75224

Ms. Nancy H. Williams
CYGNA
101 California Street
San Francisco, California 94111

Mr. James E. Cummins
Resident Inspector/Comanche Peak
Nuclear Power Station
c/o U. S. Nuclear Regulatory
Commission
P. O. Box 38
Glen Rose, Texas 76043

Mr. John T. Collins
U. S. NRC, Region IV
611 Ryan Plaza Drive
Suite 1000
Arlington, Texas 76011

Mr. Lanny Alan Sinkin
114 W. 7th, Suite 220
Austin, Texas 78701

B. R. Clements
Vice President Nuclear
Texas Utilities Generating Company
Skyway Tower
400 North Olive Street
L. B. 81
Dallas, Texas 75201

William A. Burchette, Esq.
1200 New Hampshire Avenue, N. W.
Suite 420
Washington, D. C. 20036

Ms. Billie Pirner Garde
Citizens Clinic Director
Government Accountability Project
1901 Que Street, N. W.
Washington, D. C. 20009

David R. Pigott, Esq.
Orrick, Herrington & Sutcliffe
600 Montgomery Street
San Francisco, California 94111

Anthony Z. Roisman, Esq.
Trial Lawyers for Public Justice
2000 P. Street, N. W.
Suite 611
Washington, D. C. 20036

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	}{	
	}{	
TEXAS UTILITIES ELECTRIC COMPANY, *et al.*	}{	Docket Nos. 50-445-1 and 50-446-1
(Comanche Peak Steam Electric Station, Units 1 and 2)	}{	

CERTIFICATE OF SERVICE

By my signature below, I hereby certify that true and correct copies of CASE's Answer to Applicants' 9/13/84 Supplement to Motion for Authorization Pursuant to 10 C.F.R. 50.57(c); and CASE's Answer to Applicants' Reply to CASE's Answer to Applicants' Motion for Summary Disposition Regarding Consideration of Friction Forces

have been sent to the names listed below this ~~1st~~ 2nd day of October, 1984, by: Express Mail where indicated by * and First Class Mail elsewhere. Hand-delivered at operating License Hearings where indicated by **

** Administrative Judge Peter B. Bloch
U. S. Nuclear Regulatory Commission
4350 East/West Highway, 4th Floor
Bethesda, Maryland 20814

** Ms. Ellen Ginsberg, Law Clerk
U. S. Nuclear Regulatory Commission
4350 East/West Highway, 4th Floor
Bethesda, Maryland 20814

* Dr. Kenneth A. McCollom, Dean
Division of Engineering, Architecture and Technology
Oklahoma State University
Stillwater, Oklahoma 74074

** Dr. Walter H. Jordan
881 W. Outer Drive
Oak Ridge, Tennessee 37830

** Nicholas S. Reynolds, Esq. (both hand-del. & mailed)
Bishop, Liberman, Cook, Purcell & Reynolds
1200 - 17th St., N. W.
Washington, D.C. 20036

** Geary S. Mizuno, Esq.
Office of Executive Legal Director
U. S. Nuclear Regulatory Commission
Maryland National Bank Bldg. - Room 10105
7735 Old Georgetown Road
Bethesda, Maryland 20814

Chairman, Atomic Safety and Licensing Board Panel
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Chairman
Atomic Safety and Licensing Appeal
Board Panel
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Renea Hicks, Esq.
Assistant Attorney General
Environmental Protection Division
Supreme Court Building
Austin, Texas 78711

John Collins
Regional Administrator, Region IV
U. S. Nuclear Regulatory Commission
611 Ryan Plaza Dr., Suite 1000
Arlington, Texas 76011

Lanny A. Sinkin
114 W. 7th, Suite 220
Austin, Texas 78701

Dr. David H. Boltz
2012 S. Polk
Dallas, Texas 75224

Michael D. Spence, President
Texas Utilities Generating Company
Skyway Tower
400 North Olive St., L.B. 81
Dallas, Texas 75201

Docketing and Service Section
(3 copies)
Office of the Secretary
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Juanita Ellis

(Mrs.) Juanita Ellis, President
CASE (Citizens Association for Sound Energy)
1426 S. Polk
Dallas, Texas 75224
214/946-9446