NUREG-0797
Supplement No. 1

Safety Evaluation Report

related to the operation of Comanche Peak Steam Electric Station, Units 1 and 2

Docket Nos. 50-445 and 50-446

Texas Utilities Generating Company, et al.

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

October 1981

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ABSTRACT

Supplement No. 1 to the Safety Evaluation Report (SER) related to the operation of the Comanche Peak Steam Electric Station, Units 1 and 2, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Somervell County, Texas. Subject to favorable resolution of the items identified in this supplement, the staff concludes that the facility can be operated by the applicant without endangering the health and safety of the public. This document provides the NRC staff's evaluation of the outstanding and confirmatory issues that have been resolved, and addresses changes to the SER which have resulted from the receipt of additional information from the applicant.

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

APPENDICES

1 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 Introduction

The Nuclear Regulatory Commission's "Safety Evaluation Report Related to the Operation of Comanche Peak Steam Electric Station, Units 1 and 2," (NUREG-0797), hereinafter referred to as the SER, was issued in July 1981. The SER provided a summary and the results of the staff's radiological safety review of the application by the Texas Utilities Generating Company (applicant) for an operating license for Comanche Peak. The SER concluded that upon favorable resolution of outstanding matters described therein, the plant could be operated without endangering the health and safety of the public.

By Amendments 24 through 27 to the Final Safety Analysis Report (FSAR) and by letters identified in Appendix A to this supplement, the applicant has provided additional information regarding the outstanding issues in the SER.

This document is SER Supplement No. 1 (supplement or SSER). The purpose of this supplement is to provide the staff's evaluation of the outstanding and confirmatory issues that have been resolved and to address changes to the SER which have resulted from the receipt of additional information from the applicant.

In the SER, the staff categorized issues that were not resolved with the applicant under three categories: (1) Outstanding Issues, (2) Confirmatory Issues, and (3) License Conditons. These three categories are retained in Sections 1.7, 1.8, and 1.9 respectively of this supplement.

Each section of this supplement is numbered and titled the same as the corresponding section of the SER that has been affected by the additional evaluation. Except as noted, each section is supplementary to the corresponding section in the SER. Appendix A to this supplement is a continuation of the chronology of principal actions related to the staff's safety review of the application.

The SER noted that the applicant had requested amendments to the construction permits to alter the fractional share ownership among existing co-owners and to add a new co-owner, Tex-La Electric Cooperative of Texas, Inc. The NRC amended the construction permits to accept both of these requests on September 30, 1981.

The "Final Environmental Statement Related to the Operation of Comancne Peak Steam Electric Station, Units 1 and 2," NUREG-0775) was issued in September 1981.

The SER noted that the staff had requested that the applicant verify that Comanche Peak meets the applicable requirements of 10 CFR Parts 20, 50, and 100. The applicant has since submitted a response which the staff is reviewing. A status report on this review is included in Section 3.1 of this supplement.

In a letter dated July 17, 1981, the NRC advised the applicant that the SER had been issued, however, the staff had deferred scheduling the Comanche Peak review by the Advisory Committee on Reactor Safeguards (ACRS) until such time that sufficient information has been provided to resolve the large number of open issues. The ACRS's report to the Commission will be included in a future supplement to the SER.

Copies of the SSER are available for inspection at the NRC Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Somervell County Public Library, Glen Rose, Texas. Single copies may be purchased from the sources indicated on the inside front cover.

The NRC Project Manager assigned to the Operating License application for Comanche Peak is Mr. Spottswood B. Burwell. Mr. Burwell may be contacted by calling (301) 492-7563 or writing:

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U.S. Nuclear Regulatory Commission
Washington, DC 20555

This SSER is a product of the NRC staff. NRC staff members who were principal contributors to this report are identified in Appendix D.

In addition, a number of outside organizations assisted the staff in the review supporting the SER and this supplement. Participating members of these organizations are also listed in Appendix D. The supporting organizations are as follows:

Argonne National Laboratory
BioTechnology, Inc.
Brookhaven National Laboratory
Energy Technology Engineering Center
Gage-Babcock & Associates, Inc.
Idaho National Engineering Laboratory; EG&G, Inc.
Lawrence Livermore Laboratory
Naval Surface Weapons Center
Sandia National Laboratory

1.7 Summary of Outstanding Issues

As a result of the staff review of the safety aspects of the Comanche Peak application, a number of items remain outstanding at the time of issuance of this supplement. The partial or complete resolution of some of the outstanding issues identified in the SER is described in appropriate sections of this supplement. The outstanding issues remaining in the staff operating license review are listed below, with the appropriate section numbers in the SER or this supplement that describe the issue. The staff will complete its review of these items before the operating license is issued. The resolution of these items will be discussed in a future supplement to the SER.

(1) Pipe-break damage analysis for high- and moderate-energy pipes outside containment (SER Sections 3.6.2, 9.3.3, and 10.4.9)

(2) Seismic and dynamic qualification of mechanical and electrical equipment (SER Section 3.10 and SSER Section 3.10)

(3) Environmental qualification of safety-related electrical equipment (SER Section 3.11 and SSER Section 3.11)

(4) Preservice and inservice inspection program for compliance to 10 CFR 50.55a(g) (SER Sections 5.2.4.1 and 6.6.1 for Unit 1, SER Sections 5.2.4.2 and 6.6.2 for Unit 2)

(5) Transfer of the containment spray system from injection mode to recirculation mode (SER Section 6.5.2)

(6) Low-temperature overpressure protection system control design (SER Section 7.6.4)

(7) Consequences of control system failures from a harsh environment associated with a high-energy line break (SER Section 7.7.2)

(8) Fire protection program

 (a) Alternate remote shutdown system (SER Sections 7.4.2, 8.4.6 and 9.5.1.5, and SSER Section 9.5.1.8)

 (b) Conformance with Appendix R (SER Sections 9.5.1 and 9.5.1.7, and SSER Sections 9.5.1.7 and 9.5.1.8)

(9) TMI Action Plan (SER Section 22 and SSER Section 22)

I.C.1	Guidance for evaluation and development of procedures for transients and accidents (SER)
I.C.8	Pilot monitoring of selected emergency procedures for near-term operating license applicants (SER)
I.D.1	Control room design review (SER)
II.B.2	Plant shielding to provide access to vital areas and protect safety equipment for postaccident operation (SER)
II.F.1	Additional accident monitoring instrumentation, attachments 1, 2, and 3 (SER)
II.F.2	Instrumentation for detection of inadequate core cooling (SER)
III.A.1.2	Upgrading emergency support facilities (SER and SSER)
III.A.2	Improving licensee emergency preparedness, long term (SER and SSER)
III.D.1.1	Integrity of systems outside containment likely to contain radioactive material (SER)

1.8 Confirmatory Issues

At this point in the staff review, there are a few items which have essentially been resolved to the staff's satisfaction, but for which certain confirmatory information has not yet been provided by the applicant. In these instances, the applicant has committed to provide the confirmatory information. The staff is awaiting confirmation of the applicant's commitment to comply with these positions and/or receipt of the appropriate confirmatory information. These items, with reference to the applicable sections of the SER and SSER, are identified below.

(1) Missile barriers between redundant trains of safety chilled-water system pumps and chillers (SER Sections 3.5.1.1 and 9.4.6)

(2) Staff review of WECAN computer program incomplete (SER Section 3.9.1)

(3) Dynamic analysis for asymetric loads on the reactor coolant system (SER Section 3.9.2.3)

(4) Stress limits for Class 2 and 3 austenitic pipe bends and elbows (SER Section 3.9.3.1 and SSER Section 3.9.3.1)

(5) Periodic leak testing of pressure isolation valves (SSER Section 3.9.6)

(6) Staff review of PAD-3.3 computer program incomplete (SER Section 4.2.2 and 4.4.4)

(7) Fracture-toughness properties of Unit 2 reactor vessel materials (SER Sections 5.3.1.2, 5.3.1.3, 5.3.2, and 5.3.3 and SSER Sections 5.3.1.2, 5.3.1.3, 5.3.2, and 5.3.3)

(8) Containment ventilation system isolation signals blocked by reset (SER Section 7.3.2.1)

(9) Steam generator reference leg temperature compensation and low-low steam generator level setpoint per IE Bulletin 79-21 (SER Section 7.3.2.2)

(10) Confirmation of procedure review per IE Bulletin 79-27 (SER Section 7.4.5)

(11) Handling of heavy loads in conformance with the guidelines of NUREG-0612 (SER Section 9.1.4 and SSER Section 9.1.4)

(12) Verification that auxiliary building post-LOCA radiation levels will be low enough to permit manual operation of exhaust damper CPX-VADPOC-83 (SER Section 9.4.4)

(13) Documentation of applicant's commitments on fire protection (SSER Section 9.5)

(14) Protection against flooding of safety-related compartments from a failure in the circulating water expansion joint (SER Section 10.4.5)

(15) TMI Action Plan (SER Section 22 and SSER Section 22)

I.C.2 Shift and relief turnover procedures (SER and SSER)

I.C.5 Procedures for feedback of operating experience to plant staff (SER and SSER)

I.G.1 Special low-power testing and training (SER)

II.B.1 Reactor coolant system vents (SER)

II.D.1 Performance testing of BWR and PWR relief and safety valves (SER and SSER)

II.E.1.1 Recommendation GL-3: Verification by test of the capability of the turbine-driven AFW pump to operate for two hours without ac power (SER)

II.E.4.2 Containment isolation dependability (SER)

II.K.2.13 Thermal mechanical report--effect of high-pressure injection on vessel integrity for small-break LOCA accident with no auxiliary feedwater (SER and SSER)

II.K.2.17 Potential for voiding in the reactor coolant system during transients (SER and SSER)

II.K.3.1 Installation and testing of automatic PORV isolation system (SER and SSER)

II.K.3.2 Report on overall safety effect of PORV isolation system (SER and SSER)

II.K.3.5 Automatic trip of reactor coolant pumps during LOCA (SER)

II.K.3.11 Justification of use of certain PORVs (SER and SSER)

II.K.3.30 Revised small-break LOCA methods to show compliance with 10 CFR 50, Appendix K (SER)

II.K.3.31 Plant-specific calculations to show compliance with 10 CFR 50.46 (SER)

1.9 License Conditions

There are several issues for which a license condition may be desirable to ensure that staff requirements are met during plant operation. The license condition may be in the form of a condition in the body of the Operating Licenses, or a limiting condition for operation in the Technical Specifications appended to the licenses. These items, with appropriate references to subsections of the SER and SSER, are listed below.

(1) The applicant must control mineral exploration within the exclusion area (SER Section 2.1.2).

(2) The applicant must implement the secondary water monitoring and control program proposed in a letter dated August 19, 1981 (SSER Section 5.4.2.3).

(3) The applicant must provide control room position indication and alarm for manual valves ISI-047 and I-8717 (SSER Section 6.3.3.2).

(4) The applicant must conform to Regulatory Guide 1.97, Revision 2 (SER Section 7.5.2).

(5) The bore and keyways of the low-pressure turbine discs must be inspected for cracks during the first refueling outage (SER Section 10.2.2).

(6) If contractor personnel are used to provide experience in plant operations to operating shifts, these augmentation personnel must remain on shift until the plant attains a 100 percent power level (SSER Section 13.1.2.1 and SER Section 13.1.2.1).

(7) TMI Action Plan

II.B.3 Postaccident sampling capability (SER Section 22).

2 SITE CHARACTERISTICS

2.4 Hydrologic Engineering

2.4.3 Flood Potential

2.4.3.1 Probable Maximum Flood on Squaw Creek

In Section 2.4.3.1 of the SER, the staff concluded that the thickness of the rip-rap on the safe-shutdown impoundment dam is sufficient to accommodate the weight and size of stone necessary to resist wave action as long as the stone is hard and dense so that it can resist long exposure to weathering. The stone should be well graded so that the smaller sized fragments fill the voids between the larger stones. Further, the staff noted that the applicant had neither provided the gradation limits of the rip-rap nor the results of tests conducted to determine the resistance of the rip-rap to weathering.

Subsequent to this evaluation, the FSAR was amended and the staff was informed that the safe shutdown impoundment dam does not have a separate layer of rock (rip-rap) for slope protection as had been stated in the FSAR. Since the entire outer shell of the safe shutdown impoundment dam consists of graded rock, it is not necessary to provide a separate layer of rip-rap. The staff agrees that a separate layer of rock for slope protection is not necessary as long as the exposed rock is of sufficient size to withstand wave attack and erosion. An independent analysis by the staff showed that rock with a maximum diameter of about 18 inches, would be sufficient to withstand these forces.

Normal construction techniques used in constructing the safe shutdown impoundment dam resulted in the larger rock sizes accumulating on the edges of the slopes. Some rock fragments are as large as 30 inches in diameter. The exposed rock layer is reasonably well graded without any open voids. Since the rock is larger than required and is well graded, the staff concludes that the stability of the safe shutdown impoundment dam will not be affected by erosion or wind-wave activity.

The rock used in construction is specified to be unweathered limestone. Most of the rock meets this specification; however, during a site visit, the staff observed some weathering of claystone particles above the water line. The applicant in a letter dated July 2, 1981 committed to perform inspections of the safe shutdown impoundment dam in accordance with Regulatory Guide 1.127, Inspection of Water Control Structures Associated with Nuclear Power Plants, Rev 1, (3/78). The integrity of the rock shell will be inspected and documented annually and any deterioration will be repaired. The staff will incorporate this inspection in the Technical Specifications.

The staff has reviewed the latest information submitted by the applicant and has performed independent evaluations. The staff concludes that the safe shutdown impoundment meets the requirements of GDC-2 with respect to flooding. This outstanding issue is resolved.

2.4.8 Monitoring and Maintenance

In Section 2.4.8 of the SER, the staff required that the applicant commit to a program for monitoring sediment buildup in the service water intake (SSI) channel. By letter dated August 14, 1981, the applicant committed to inspect the SSI channel for sediment buildup at the frequencies stated in Regulatory Guide 1.127, "Inspection of Water-Control Structures Associated with Nuclear Power Plants, Revision 1." This inspection will consist of a determination of the elevation at the bottom of the SSI channel at three locations along its length. The same location will be used each time the inspection is performed. If it is determined that sediment has accumulated enough to increase the bottom elevation 1.5 feet, measures will be employed to remove the sediment from the SSI channel. The staff has reviewed this inspection program and concludes that it is adequate to detect and remove excessive sediment buildup at the SSI channel and is, therefore, acceptable. The staff will incorporate this inspection in the Technical Specifications. This outstanding issue is resolved.

2.4.9 Conclusions

The staff's concerns regarding the protection of the safe-shutdown impoundment dam against erosion and weathering, and buildup of sediment in the service water intake channel have been resolved as discussed above. The staff concludes that the requirements of GDC 2 with respect to a flood hazard have been met.

2.5 Geology and Seismology

2.5.4 Stability of Subsurface Materials and Foundations

2.5.4.3 Foundation Stability

The SER identified an outstanding issue that required the applicant to provide the design basis for the cyclic strength criteria specified in FSAR Table 2.5.4-11 for Class I backfill and bedding material.

In a letter dated August 24, 1981, the applicant stated that there is no liquefaction potential for Class I backfill and bedding materials at the Comanche Peak site. The applicant also informed us that the cyclic strength criteria, provided by Gibbs & Hill Specification 2323-SS-8, were used only for liquefaction analysis.

The staff has determined that the maximum shear stresses that could develop during the SSE (0.12g maximum horizontal ground surface acceleration) are smaller than the corresponding design cyclic shear stresses and the applicant has shown that the Class I backfill and bedding materials will not liquefy when subjected to the design cyclic shear stresses. Therefore, the staff concurs in the applicant's conclusion that these materials will not liquefy at the Comanche Peak site.

Based upon its evaluation of the additional information noted above, the staff finds the cyclic-strength design criteria acceptable as used in the liquefaction analyses and concurs in the applicant's findings that the Class I backfill and Class I bedding will remain stable during the SSE. This outstanding issue is resolved.

2.5.4.5 Conclusion

The staff has concluded that the site and plant foundations will be adequate to safely support the powerblock structures, the service water intake structure, and associated buried pipelines and conduits of the Comanche Peak plant.

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Conformance with General Design Criteria and NRC Regulations

One of the outstanding issues in the SER was the applicant's response to the staff's request for verification that Comanche Peak meets regulatory requirements in Title 10 of the Code of Federal Regulations (10 CFR) Parts 20, 50, and 100. By a letter dated August 24, 1981, the applicant provided a comparison of the application with the regulations. The staff's review of this submittal is not complete.

The staff will require that the Comanche Peak facility be maintained and operated in conformance with the NRC regulatory requirements except where it is specifically granted authorization to use alternative criteria. Since this matter is a legal requirement (as opposed to a safety requirement) this matter will not be listed separately in Section 1.7, 1.8 or 1.9 of this and future supplements.

3.5 Missile Protection

3.5.1 Internally Generated Missiles

3.5.1.3 Turbine Missiles

According to GDC 4, Appendix A to 10 CFR Part 50, nuclear power plant structures, systems, and components important to safety shall be appropriately protected against dynamic effects, including the effects of turbine missiles. Systems important to safety are defined to be those structures, systems, and components necessary to ensure:

(1) The integrity of the reactor coolant pressure boundary,

(2) The capability to shut down the reactor and maintain it in a cold shutdown condition, and

(3) The capability to prevent accidents that could result in potential offsite exposures that are a significant fraction of the guideline exposures of 10 CFR Part 100, "Reactor Site Criteria."

The turbine-generator placement and orientation of Comanche Peak Units 1 and 2 are favorable relative to the essential plant buildings; that is, there are no structures, systems, or components important to safety inside the low trajectory missile (LTM) strike zone (see Regulatory Guide 1.115). Protecting systems important to safety from LTMs by excluding them from the region in which LTM ejection is most likely is preferable to other methods of protection. On this basis, therefore, the staff concludes that systems important to safety at Comanche Peak SES are adequately protected from low trajectory turbine missiles.

With regard to high trajectory missiles (HTMs), the applicant has performed an analysis and calculated that for an estimated target area of 104,000 square feet, the HTM strike probability is 2.4×10^{-3} per turbine failure per unit. The target area was conservatively taken to be the total horizontal surface area of the reactor containment buildings - Units 1 and 2, the safeguards building - Units 1 and 2, the diesel generator buildings - Units 1 and 2, the auxiliary building, the fuel building, the electrical and control building, and the service water intake structure. The applicant conservatively assumes a roof thickness over all targets of 21 inches of reinforced concrete, and postulated a uniform missile velocity spectrum, for each turbine missile, over the range from the roof scabbing velocity to the maximum escape velocity of the missile from the turbine casing. The staff has reviewed the applicant's analysis and concludes that the probability of striking systems important to safety at Comanche Peak with high trajectory turbine missiles is sufficiently low that the risk rate for unacceptable damage due to HTMs is less than 10^{-7} per year per unit, which the staff considers an acceptable risk rate for the loss of an essential system from a single event.

In summary, the staff concludes that the total turbine missile risk from high and low trajectory missiles for the Comanche Peak SES Unit Nos. 1 and 2 design and layout is acceptably low so that the plant structures, systems, and components important to safety are adequately protected against potential turbine missile damage.

3.9 Mechanical Systems and Components

3.9.3 ASME Code Class 1, 2 and 3 Components, Component Supports, and Core Support Structures

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

Functional capability refers to the ability of the pipjng system to deliver rated flow when the piping system is stressed to the maximum allowable value at specified locations. In the SER, the staff stated that it had provided the applicant with simplified screening criteria which assure functional capability.

The applicant has proposed alternate criteria for the functional capability of ASME Code Class 2 and 3 austenitic pipe bends and elbows. The staff has reviewed these criteria and has determined that their use within the staff's stated limits of $\frac{D_o}{t}$ less than 50 (where D_o = pipe outside diameter, and t = wall thickness of the bend or elbow) will not result in collapse of these piping fittings, thus assuring functional capability. For any bends and elbows in which $\frac{D_o}{t}$ is greater than 50, the staff's original screening criteria remain applicable. The applicant proposes to demonstrate that these criteria have been satisfied by performing a reevaluation of a random sample of austenitic pipe bends and elbows in a number of Class 2 and 3 piping systems. It is anticipated that the results of the evaluation will be available to the staff by April 1, 1982. The staff considers this approach to be acceptable, and will report in a future supplement to the SER if the applicant indicates problems in meeting these criteria.

3.9.6 Inservice Testing of Pumps and Valves

The inservice testing program for pumps and valves is intended to demonstrate that they will maintain operational readiness at any time during the plant life. Tests and parameter measurements are performed to detect long term degradation in accordance with the rules of Section XI of the ASME Code and to verify that pumps and valves operate successfully when called upon.

The staff's review under Standard Review Plan Section 3.9.6 covers the applicant's program for preservice and inservice testing of pumps and valves, and emphasizes those areas of the test program for which the applicant requests relief from the requirements of Section XI of the ASME Code.

The staff has not completed its detailed review of the applicant's submittal. However, based on its preliminary review, the staff finds that it is impractical within the limitations of design, geometry, and accessibility for the applicant to meet certain of the American Society of Mechanical Engineers (ASME) Code requirements. Imposition of those requirements would, in the staff's view, result in hardships or unusual difficulties without a compensating increase in the level of quality or safety. The relief requested is authorized by law, and will not endanger life or property or the common defense and security, and is otherwise in the public interest. Therefore, pursuant to 10 CFR 50.55a(g)(2), (g)(4)(i), and (g)(6)(i), the staff recommends that the relief that the applicant has requested from the pump and valve testing requirements of the ASME Code be granted for that portion of the initial 120-month period during which the staff completes its review.

One area of concern during the staff's review was the periodic leak testing of pressure isolation valves. There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated reactor coolant system (RCS pressure). There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. To protect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure systems. The leaktight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure system thus causing an inter-system LOCA.

The applicant will be required to categorize its pressure isolation valves for the safety injection including the accumulator discharge check valves, residual heat removal, and boron injection systems as Category A or AC. These categorizations will meet the staff requirements. Pressure isolation valves are required to be Category A or AC and to meet the appropriate valve leak rate test requirements of IWV-3420 of Section XI of the ASME Code except as discussed below. The allowable leakage rate shall not exceed 1 gpm for each valve as will be stated in the Technical Specifications.

Limiting Conditions for Operation (LCO) will be added to the Technical Specifications which will require corrective action (shutdown or system isolation when the acceptance criteria is not met). Also, surveillance requirements, which state the acceptable testing frequency, will be provided in the Technical Specifications.

The staff will review the proposed Technical Specifications for compliace with the above periodic leak testing requirements. Conformance with the above requirements for pressure isolation valves between the reactor coolant system and low pressure systems will provide reasonable assurance that the design pressure of the low pressure systems will not be exceeded, and thus, reduce the probability of an occurrence of an inter-system LOCA.

3.10 Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment

On August 4, 1981, NRC staff members met with representatives from the applicant, NSSS vendor (Westinghouse) and the architect-engineer (Gibbs & Hill). The purpose of the meeting was to hold a working session to clarify and discuss the seismic qualification review procedure and requirements needed to complete the staff review. The staff bases for a Seismic Qualification Review Team (SQRT) audit, as recommended in the SER, were discussed and the conditions required prior to scheduling an audit were clarified.

The architect-engineer (AE) and NSSS vendor presented a brief summary of their equipment qualification programs and their procedures for implementation of the programs. Based on these presentations, the equipment qualification documentation for seismic Category I balance-of-plant equipment was approximately sixty percent complete. The NSSS documentation was nearly one-hundred percent complete for the mechanical equipment, and approximately seventy-five percent complete for the electrical equipment. The completion percentage of the equipment installed on-site was determined to be somewhat less than that.

General discussions and comments involving the qualification testing, methods of analysis, equipment test mounting verses as-built mounting, and the response spectra used in qualifying the equipment were made during the working session. The purpose and need for auditable links in the qualification documentation showing that the supplier/applicant had reviewed and approved the qualification test methods, analysis and results were pointed out by the staff. The AE representatives provided the staff with typical verification letters and discussed the sign off procedures they could readily implement for the audit qualification documentation. The staff agreed that the proposed AE methodology and implementation would provide an acceptable auditable link. It was also pointed out that the applicant has the overall responsibility (10 CFR Part 50, Appendix B, Section I) and that it was its task to assure an acceptable auditable link for the NSSS equipment.

Based on the information exchange obtained during the working session, no major disagreement or deficiency surfaced in the applicant/NSSS/AE seismic qualification program and implementation procedures. It was determined by the staff, and agreed to by the applicant, NSSS vendor, AE, that the sample equipment documentations reviewed during the working session was not sufficient for the staff to reach a conclusion on the acceptability of the overall seismic qualification program and implementation for Category I safety-related equipment. Thus, a confirmatory on-site audit should be scheduled.

Prior to scheduling the confirmatory audit, the equipment qualification documentation should be at least ninety percent complete and on-site. The same completion requirements apply to on-site installation of the equipment. Because the staff judgment on the acceptability of the applicant's seismic qualification of equipment will be weighted by the results of the confirmatory audit, the equipment audited must provide the staff with a high level of confidence in the applicant's overall program.

The staff will report on the results of its review of the seismic and dynamic qualification program in a future supplement to the SER.

3.11 Environmental Qualification for Safety-Related Electrical Equipment

In December 1979, the staff issued guidance for the environmental qualification of safety-related electrical equipment in NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment". By letters dated February 5 and February 21, 1981, the staff requested the applicant to review the environmental qualification documentation for each item of safety-related electrical equipment which could be exposed to a harsh environment so as to identify the degree to which the associated environmental qualification program complies with the staff's position as described in NUREG-0588. Further, where there are deviations, the applicant must commit to corrective action (requalification, replacement, relocation etc.) consistent with the requirements to establish qualification. If fuel loading occurs before the deadline, justification for operation until the corrective actions are completed must be provided.

The Commission Memorandum and Order, CLI-80-21, dated May 23, 1980, directed that by no later than June 30, 1982, all safety-related electrical equipment in operating reactors shall be in compliance with NUREG-0588 or the "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors". The applicable requirements for Comanche Peak are in Category I of NUREG-0588.

On July 15, 1981, the staff and the applicant met to discuss the applicant's environmental qualification program. During the course of the meeting the applicant described its overall environmental qualification program and the staff reviewed several qualification documentation packages representative of the information contained and to be contained in the applicant's environmental qualification central file. The staff is satisfied with the applicant's general review effort.

The staff will conduct an audit of the applicant's environmental qualification documentation after the applicant has completed its review. After the staff has completed the audit it will report on the environmental qualification of safety-related electrical equipment in a future supplement to the SER. The applicant indicated that it should be ready for an audit in early 1982.

4 REACTOR

4.2 Fuel Design

4.2.1 Description

The applicant amended the FSAR for Comanche Peak in January 1981, to allow the use of hafnium as a control rod absorber material. The hafnium would be a direct substitution for the silver-indium-cadmium (Ag-In-Cd) currently used in Westinghouse reactors and would be incorporated in the standard rod cluster control assembly (RCCA) rodlet design configuration with the same 304 stainless steel cladding as is used in the standard design.

Because of its higher melting temperature (2,222°C vs. 800°C), greater strength (>30,000 psi vs. ~7,000 psi at 0.2% yield strength) and greater corrosion resistance, hafnium appears to have significant material advantages relative to Ag-In-Cd as a control rod absorber, and the staff advised the applicant and Westinghouse that it was therefore receptive to the use of hafnium for that purpose. To date there has been some operating experience with hafnium control rods in a few commercial PWRs, although that experience was with cruciform shaped rods that were non-prototypic of the current design. Furthermore, Naval Reactor experience with hafnium control material is not totally relevant because the hafnium used in such applications is in a different design configuration and is unclad. While these experiences would tend to support the use of hafnium in control rods, the staff does not consider them conclusive. Therefore, the staff asked the applicant to propose and commit to performing a surveillance program that will include at least visual inspection of some control rods at each refueling. In addition, because hafnium was essentially a new material to be used in commercial PWR cores, the staff asked the applicant to consider the potential for adverse chemical reaction at the elevated temperatures that might be attained during a postulated (non-specific) accident.

In a recent submittal (FSAR Amendment 26), the applicant stated that visual inspection of selected D Bank control rods will be performed during the first, third, fifth, and ninth refueling outages, unless there are indications that warrant additional surveillance. The RCCA change fixture will be utilized as a point of visual inspection for evidence of control rod degradation such as deformation, bowing, or cladding defects. The staff believes that the frequency and spacing of the proposed examinations are appropriate because (1) fabrication defects and deficiencies would be expected to manifest observable effects fairly early in life (i.e., a kind of infant mortality effect), (2) observable effects are not expected and so need not be checked for at each refueling outage, and (3) advance indications of severely anomalous and unacceptable mechanical or nuclear performance should be detectable from the drop time verification tests and reactivity worth measurements that will be performed each cycle. The visual examinations are intended only to provide additional verification of adequate performance. The staff, therefore, concludes that the proposed visual surveillance program is acceptable.

With regard to the potential for adverse reaction under severe high temperature accident conditions, the applicant has pointed out that the lowest eutectic temperature in the iron-hafnium (Fe-Hf) system is 2322°F (1300°C). This eutectic temperature is above the temperature at which the control rod cladding would be likely to fail in a short time by stainless steel-water reaction or plastic creep deformation, and it is significantly higher than the 1454°F Ag-In-Cd melting temperature. At temperatures greater than 1454°F, a postulated cladding failure would result in the molten Ag-In-Cd leaking out, whereas the hafnium absorber would remain solid (at least up to the Hf-Fe eutectic temperature) and would begin to slowly react with the water to form an oxide with the slower release of energy and hydrogen than results from the zirconium-water reaction. On the basis of the above considerations, the staff concludes that there is reasonable assurance that the substitution of hafnium for Ag-In-Cd absorber in Comanche Peak control rods will not result in reduced safety margins or reliability and is, therefore, acceptable.

4.2.5 Conclusion

As discussed in Section 4.2.1 above, the staff concludes that the substitution of hafnium for Ag-In-Cd absorber in the control rod is acceptable.

4.3 Nuclear Design

4.3.2 Design Description

4.3.2.3 Control

General background information on the use of hafnium for control rods is contained in Section 4.2 of this supplement.

The staff has reviewed the information presented by the applicant in the FSAR supporting the Westinghouse ability to calculate the nuclear characteristics of hafnium for control purposes. The generally good agreement between Westinghouse calculations and critical experiment data provide confidence that the required control will be met by the hafnium design.

In addition the staff had its Technical Assistant consultants at Brookhaven National Laboratory[18] perform independent calculations of hafnium control characteristics in a LWR matrix. These calculations indicated hafnium will provide essentially the same control characteristics as Ag-In-Cd, the control material presently used in Westinghouse reactors. The staff therefore concludes that the hafnium control rod design proposed by the applicant is an acceptable alternative to the Ag-In-Cd design.

4.4 Thermal-Hydraulic Design

4.4.1 Thermal-Hydraulic Criteria and Design Basis

In the SER the staff concluded that the initial core design of the Comanche Peak units conformed to the requirements of GDC 10, 10 CFR Part 50 and Standard Review Plan Section 4.4. However, before the staff could conclude that the core design was acceptable for final design approval, it required that the applicant confirm that its thermal-hydraulic design methods appropriately bound future cycles. By letter dated October 2, 1981 the applicant provided its response. In its response, the applicant stated that the input parameter values used in the reference safety analysis were selected to bound the values expected in all future cycles. The applicant further stated that when all of the reload parameters for a given accident are bounded, the reference safety analysis is valid; however, if a parameter is not bounded, further evaluation is necessary. This further evaluation is to confirm that the margin of safety defined in the basis for any Technical Specification is not reduced.

Based on the information given above, the staff concludes that the applicant has adequately considered future cycle operation in the design safety analyses. The staff also concludes that the thermal-hydraulic design presented in Chapter 4.4 of the FSAR is acceptable.

4.7 References

18 Brookhaven National Laboratory Memorandum from Peter Cohut to J. S. Carew, "Hafnium Control Rod Worth Calculations," September 10, 1981.*

*Available for inspection in the NRC Public Document Room, 1717 H St., NW, Washington, DC 20555.

5 REACTOR COOLANT SYSTEM

5.2 Integrity of the Reactor Coolant Pressure Boundary

5.2.2 Overpressure Protection

5.2.2.2 Low-Temperature Operation

As discussed in the SER, the staff requested additional information with regard to the operability of the low-temperature overpressure protection system during a seismic event or during a vital dc bus power failure:

(1) After a seismic event the nonseismic air system may be assumed to be in its failed position, i.e., closed. Also, the charging line air-operated valves may be assumed to be in their failed position, i.e., open. It was hypothesized that these two selective failures would produce a pressurization event, that is, charging flow with no matching letdown flow. The staff requested that the applicant address the availability of overpressure protection during such a scenario and, in particular, discuss the requirements for an operating-basis earthquake (OBE) qualified system.

(2) The staff requested that the applicant address a scenario involving a vital dc bus failure with the reactor coolant system in a cooled down and depressurized condition in which the low-temperature overpressure protection system is required to be operable, and with charging and letdown established. It was hypothesized that a dc bus failure could cause normal letdown isolation and also render one of the two power operated relief valves (PORVs) incapable of opening when it is called upon to open by the low-temperature overpressure protection system circuitry. The above scenario would lead to an overpressurization transient with one operable PORV remaining. However, a single failure could then disable the remaining PORV.

The applicant responded to the staff's concern about availability of low-temperature overpressure protection following a seismic event in FSAR Amendment 22. The applicant's response indicated that the PORV bodies and the nitrogen accumulators attached to the valves are qualified to be operable and intact through a safe shutdown earthquake (SSE). The reference also indicated that the PORV operators will remain functional throughout a seismic event. The staff has reviewed this information and concludes that the two PORVs would be available to mitigate the subject overpressurization event. Accordingly, the staff concludes that this matter is resolved.

The applicant responded to the staff's concern about availability of low-temperature overpressure protection in the event of a failure of a vital dc bus in a letter dated August 14, 1981. In this response, the applicant stated that since the letdown valve will fail closed and the charging isolation valve will fail open due to loss of dc power to the solenoid valve which directs air away from the valve diaphram, an overpressurization transient will develop, i.e., charging flow without letdown. However, the applicant added, that between the charging pump and the normal charging isolation valve there are two normally throttled valves that receive their power from the process and control racks which are powered by the vital ac instrument buses. Therefore, these valves would be unaffected by a dc bus failure and would be expected to continue to function normally during the event. One of these valves is the charging flow control valve (FCV-121) which automatically regulates the charging flow to maintain a prescribed pressurizer level. Since this valve continues to function normally, as pressurizer level rises charging flow would be automatically reduced to that required for reactor coolant pump seal injection. The other valve is the charging flow back-pressure regulator (HCV-182) which is manually positioned to regulate flow to the reactor coolant pump seals. This valve would remain in its initial position but can be closed by the operator, thus terminating the transient.

Additionally, the applicant indicates that with reactor coolant system water temperature below 350°F, the residual heat removal system (RHRS) will normally be in operation or, at a minimum, will be lined up so that the RHRS suction relief valves are available for overpressure mitigation. During operations when the RHRS is not lined up with the reactor coolant system, a steam bubble of approximately 1350 ft^3 is maintained in the pressurizer. At a maximum charging rate of 120 gpm it would take in excess of 30 minutes to reach the Appendix G limit at 200°F, the coldest temperature at which RHRS is permitted to be isolated. Several alarms and indicators alert the operators to the ongoing transient (e.g., a dc bus failure alarm, an alarm before the pressure relief setpoint is reached and panel indication of system pressure).

Based on the above considerations, the staff concludes that the reactor coolant system is adequately protected against the overpressurization events discussed above and finds the design of the low-temperature overpressure protection system acceptable. The staff concludes that this matter is resolved.

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials

5.3.1.2 Compliance With Appendix H, 10 CFR Part 50

The materials surveillance program at Comanche Peak Units 1 and 2 will be used to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region that result from exposure to neutron irradiation and the thermal environment. Under the Comanche Peak surveillance programs, fracture toughness data will be obtained from material specimens that are representative of the limiting base, weld, and heat-affected zone materials in the beltline region. These data will permit the determination of the conditions under which the vessel can be operated with adequate margins of safety against fracture throughout its service life.

The fracture toughness properties of reactor vessel beltline materials must be monitored throughout the service life of the plant by a materials surveillance program that meets the requirements of the American Society for Testing and Materials (ASTM) Standard E 185-73, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels," and Appendix H of 10 CFR Part 50.

The staff has evaluated the applicant's information for degree of compliance with these requirements. Based on its evauation, the staff concludes that Comanche Peak Unit 1 has met the requirements of Appendix H and that additional confirmatory information for Comanche Peak Unit 2 will be required to demonstrate compliance with paragraph II.B of Appendix H.

Paragraph II.B of Appendix H requires that the surveillance program comply with ASTM E-185-73. ASTM E-185-73 requires the surveillance capsule materials be removed from beltline reactor vessel base metals and weld samples which represent the materials that may limit operation of the reactor vessel during its lifetime. The applicant has indicated in FSAR Section 5.3.1.6 that Unit 2 will comply with the surveillance program requirement of Appendix H, but has not submitted sufficient information for the staff to confirm that Unit 2 will comply with the requirements of paragraph II.B of Appendix H. Upon receipt of the Unit 2 reactor vessel beltline surveillance program, the staff will confirm that the applicant's program has complied with the requirements of paragraph II.B of Appendix H.

5.3.1.3 Conclusions

Based on its evaluation of compliance with Appendices G and H of 10 CFR Part 50, the staff concludes that for Unit 1 the applicant has met all the fracture toughness requirements of Appendix G and the surveillance program requirement of Appendix H. The evaluation indicates that for Unit 2 the applicant has met all the fracture toughness requirements of Appendix G, but must provide confirmatory information to demonstrate compliance with Paragraph II.B of Appendix H.

5.3.2 Pressure-Temperature Limits

Appendices G and H of 10 CFR Part 50 describe the conditions that require pressure-temperature limits for the reactor coolant pressure boundary (RCPB) and provide the general bases for these limits. These appendices specifically require that pressure-temperature limits must provide safety margins for the RCPB at least as great as the safety margins recommended in the ASME Code, Section III, Appendix G. Appendix G of 10 CFR Part 50 requires additional safety margins whenever the reactor is critical, except for low-level physics tests.

The applicant has submitted pressure-temperature limit curves for Unit 1 but has not submitted curves for Unit 2. The staff concludes that the pressure-temperature limit curves for Unit 1 are acceptable for 32 effective full power years. The applicant in FSAR Section 5.3.2.1 indicated that the pressure-temperature limit curves will comply with the requirements of Appendix G of 10 CFR Part 50. Upon receipt of the Unit 2 reactor vessel pressure-temperature limit curves, the staff will confirm that the applicant's curves comply with the requirements of Appendix G of 10 CFR Part 50.

5.3.3 Reactor Vessel Integrity

The staff has reviewed the FSAR sections related to the reactor vessel integrity for Comanche Peak Units 1 and 2. Although most areas are reviewed separately in accordance with other review plans, reactor vessel integrity is of such importance that a special summary review of all factors relating to reactor vessel integrity is warranted.

The staff has reviewed the information in each area to ensure that it is complete and that no inconsistencies exist that would reduce the certainty of vessel integrity. The areas reviewed are

(1) design (SER Section 5.3.1)

(2) construction materials (SER Section 5.3.1)

(3) fabrication methods (SER Section 5.3)

(4) operating conditions (SER Section 5.3.2)

The staff has reviewed the factors which contribute to the structual integrity of the Comanche Peak Units 1 and 2 reactor vessels and concludes that the applicant has complied with Appendices G and H of 10 CFR Part 50 except that confirmatory information for Unit 2 are required for the following items:

(1) The material in the surveillance capsule has not been identified per ASTM E-185-73 (Paragraph II.B, Appendix H).

(2) The applicant has not submitted pressure-temperature limit curves to ensure the safe operation of the reactor vessel during normal operation, maintenance, and testing.

The staff has reviewed all factors contributing to the structural integrity of the reactor vessel and concludes there are no special considerations that make it necessary to consider potential reactor vessel failure for Comanche Peak Unit 1 and Unit 2.

5.4 Component and Subsystem Design

5.4.1 Reactor Coolant Pumps

The SER stated that the applicant's inservice inspection program for pump flywheels complies with Regulatory Guide 1.14 except that the applicant's plan did not include a surface examination of all exposed flywheel surfaces at approximately 10-yr intervals. This examination is identified in Paragraph C.4.b of Regulatory Guide 1.14; it is required to avoid reducing the margin of assurance against flywheel missiles. The staff noted its intent to include a license condition imposing this requirement.

In FSAR Amendment 27 the applicant stated,

> "Inservice inspection of each RCP flywheel will be performed by TUGCO in conformance with Regulatory Guide 1.14, Revision 1, August 1975. Insevice inspection will include a surface inspection of each RCP flywheel as outlined in Paragraph C.4.b.2 of the above mentioned Regulatory Guide."

The staff finds that commitment acceptable and concludes the need for a license condition resolved. The staff will require that the Technical Specifications contain a condition that is consistent with the above statement.

5.4.2 Steam Generators

5.4.2.3 Secondary Water Chemistry

In the SER, the staff described its requirements for the secondary water chemistry monitoring and control program. At the staff's request, the applicant provided additional information by letters of July 10 and 14, and August 19, 1981. The applicant addressed the six program criteria set forth in the staff's position as discussed below. The program is based on the steam generator water chemistry program recommended by the NSSS vendor.

The proposed program monitors the critical parameters to inhibit steam generator corrosion and tube degradation. The limits and sampling schedule for these parameters have been established for steam generator blowdown and feedwater/condensate under power operation, startup, shutdown, and wet layup conditions. The control points for the critical parameters and the process sampling points have been identified. The analytical techniques for measuring the values of the critical parameters are indicated, and reference to the plant chemical procedures is given for the complete procedures. The authority ultimately responsible for interpretation of secondary-side water chemistry data is the Chemistry and Environmental Engineer. The Manager of Plant Operations is responsible for modifying plant operations for abnormal chemical conditions.

The staff has reviewed the information submitted by the applicant, and finds that the applicant's secondary water chemistry monitoring and control program:

(a) is capable of reducing the probability of abnormal leakage in the reactor coolant pressure boundary by inhibiting steam generator corrosion and tube degradation, and thus meets the requirements of GDC 14;

(b) adequately addresses all of the program criteria delineated in the NRC staff position on control and monitoring of secondary water;

(c) is based on the NSSS vendor's recommended steam generator water chemistry program;

(d) monitors the secondary coolant purity in accordance with Branch Technical Position MTEB 5-3, revision 1, and thus meets acceptance criterion 3 of Standard Review Plan Section 5.4.2.1, "Steam Generator Materials," revision 1;

(e) monitors the water quality of the secondary side water in the steam generators to detect potential condenser cooling water in-leakage to the condensate, and thus meets Position 2 of Branch Technical Position MTEB 5-3, revision 1; and

(f) describes the methods for control of secondary side water chemistry data and record management procedures and corrective actions for off-control point chemistry, and thus meets Position 3 of Branch Technical Position MTEB 5-3, revision 1.

On the basis of its evaluation, the staff concludes that the proposed secondary water chemistry monitoring and control program meets (1) the requirements of GDC 14 insofar as secondary water chemistry control assures primary boundary material integrity, (2) Acceptance Criterion 3 of Standard Review Plan Section 5.4.2.1, revision 1, (3) Positions 2 and 3 of Branch Technical Position MTEB 5-3, revision 1, and (4) the program criteria in the staff's position and, therefore, is acceptable. The staff will condition the operating license to require that the proposed secondary water chemistry monitoring and control program be carried out.

6 ENGINEERED SAFETY FEATURES

6.3 Emergency Core Cooling System

6.3.3 Design Evaluation for Single Failures

The uncertainty associated with refueling water storage tank (RWST) low level signals and their impact on switching the low-head portion of the emergency core cooling system (ECCS) from the injection to the recirculation mode led the staff to request that the applicant show that the operator has sufficient time to complete the manual actions (e.g., closing the RHRS suction valves from the RWST) before the pumps that are not isolated from the RWST are damaged.

In a letter dated September 15, 1981, the applicant demonstrated by calculations that the failure of the operator to manually isolate the RHRS suction from the RWST would not expose the RHR pumps to a cavitation effect which may damage the pumps. After consideration of the different temperatures and pressures and line resistances at the sump and RWST, the calculations demonstrated that an adequate water leg would exist in the piping line from the RWST to the RHR pump suction. The staff has reviewed these calculations, and concludes that the applicant's design for switching the ECCS from the injection to the recirculation mode of operation is adequate. Accordingly, the staff considers that this outstanding issue is resolved.

6.3.3.1 Qualification of the Emergency Core Cooling System

By a letter dated May 21, 1981, from Mr. T. Anderson (Westinghouse) to Mr. V. Stello (NRC-I&E), the NRC was advised that a malfunction in the control grade volume control tank (VCT) level control circuitry may lead to cavitating the centrifugal charging pump(s), thus potentially degrading the core cooling capability during normal operation.

In response to the above concern, the applicant has reviewed the design of the VCT level control system for the Comanche Peak plant and verified that no single failure can disable both level control channels, namely LP-112 and LP-185.

The applicant stated that if LP-112 fails high, a failure which leads to decreasing the level in the VCT, an alarm will sound when the VCT level reaches the low level setpoint. Upon the actuation of that alarm, the operator has in excess of 10 min. to perform any one or a combination of the following: (1) to restore letdown to the VCT, (2) to restore manual makeup from the reactor makeup water system to the VCT, (3) to transfer the charging pump suction from the VCT to the RWST, or (4) to simply stop the charging pump. Furthermore, the applicant stated that if LP-185 fails high, a high level alarm will sound allowing the operator a longer time than that available following the failure of the other channel, LP-112.

In addition, an indication light on the control board would show whether the letdown flow has been diverted away from the VCT. Based on the above considerations, the staff concludes that the Comanche Peak plant is adequately protected against the VCT level control failure outlined above.

6.3.3.2 Instrumentation and Control

The staff understands that there are two ECCS manual valves, the mispositioning of either which could prevent or degrade the ECCS function. One valve, ISI-47, is on the line from the refueling water storage tank (RWST) to the ECCS pump suction. If this valve is left closed, the RWST water becomes unavailable and the ECCS is rendered ineffective. The other valve, I-8717, is on the RHRS return line to the RWST. If this valve is left open the ECCS flow will be diverted away from the reactor vessel, back to the RWST, thus degrading the ECCS performance. The staff requires that prior to issuance of an operating license these valves (ISI-047, I-8717) shall be provided with control room position indication and with an alarm that indicates whether they are out of position. If the required control room position indication and alarm are not installed prior to issuance of the operating license, the staff will include a license condition in the operating licenses for this requirement.

6.3.3.3 Functional Design

The staff's concern relative to the adequacy of the proposed design for switching the ECCS from the injection to the recirculation mode has been resolved as described in Section 6.3.3 of this supplement.

6.3.4 Tests and Inspection

6.3.4.2 Preoperational Tests

The staff noted in the SER that the applicant proposed using a one-to-one scale model test to demonstrate the sump design for ECCS recirculation mode. The staff requested that the applicant provide for its review a sump test report describing the test model, the test results and the acceptance criteria, when the test is completed. The SER stated that the staff will review the model and test results when the test is completed. The applicant advised that the sump tests were conducted in August 1981, that the tests demonstrated the sump design was satisfactory and that it expects to send the NRC a test report in approximately one month. Based on the above discussion and on experience with previous sump test information, there is reasonable assurance that acceptable performance will be verified. Should the applicant's submittal of test results show otherwise, the staff will require further consideration of improvement to sump performance. Therefore, this item is resolved contingent on staff acceptance of the test results.

7 INSTRUMENTATION AND CONTROLS

7.7 Control Systems Not Required for Safety

7.7.2 Conclusions

The staff requested that the applicant identify any power sources, sensors, or sensor impulse lines which provide power or signals to two or more control systems and demonstrate that failures of these powers sources, sensors, or sensor impulse lines will not result in consequences outside the bounds of the Chapter 15 analyses or beyond the capability of operators or safety systems.

The applicant has submitted a response describing the results of a study to determine the effect on the plant of the following:

(1) Loss of power to all control systems powered by a single power supply

(2) Failure of each instrument sensor which provides a signal to two or more control systems

(3) Break of any sensor impulse line which is used for sensors providing signals to two or more control systems

The applicant has provided a summary of the events resulting from each postulated failure and identified the specific Chapter 15 analysis which delineates the bounding consequences of the failure.

The staff has reviewed the bases for the applicant's study and concludes there is reasonable assurance that the consequences of single failures within the control systems are bounded by analyses in Chapter 15 of the FSAR. The staff finds the results of the applicant's study acceptable and considers this outstanding issue resolved.

Unresolved Safety Issue A-47, "Safety Implications of Control Systems" will address control system designs and the need for any control system design modifications. The applicant will be required to address any new guidance which may result from the resolution of the unresolved safety issue.

8 ELECTRIC POWER SYSTEMS

8.1 General Considerations

A site visit was made in order to view the installation and arrangement of electrical equipment and cables for the purpose of verifying the proper implementation of the design criteria. During the staff's site visit, the following discrepancies were discovered. These items were discussed with the applicant and their proposed modifications are discussed below.

8.4 Other Electrical Features and Requirements for Safety

8.4.4 Physical Identification and Independence of Redundant Safety-Related Electrical Systems

Physical and electrical separation between Class 1E and non-Class 1E cables was not properly maintained inside the Class 1E 118 volt ac instrumentation power supply panelboards IPC1, IPC2, IPC3 and IPC4. Class 1E cables inside the panelboards were bundled with non-Class 1E cables. The non-Class 1E cables leaving these four panelboards were routed through common conduit, cable tray and junction box, thereby exposing all four Class 1E panels to failure from one single event. This does not conform to Regulatory Guide 1.75.

For each non-Class 1E circuit powered from IPC1, IPC2, IPC3 and IPC4, the applicant has agreed to install a non-Class 1E breaker or fuse in series with the panel Class 1E breaker. Class 1E and non-Class 1E wiring within the panels will be installed in different wire bundles and these bundles will be separated from each other.

The applicant described the shutdown transfer panel (STP) that is being added to improve the reliability of the remote shutdown system. The panel will be connected to the hot shutdown panel (HSP) and control room panels. After the STP is installed, no single event such as a fire in the HSP will prevent the operators from bringing the plant to a hot shutdown. Class 1E and non-Class 1E wires are terminated at the same switch in the shutdown transfer panel. This does not conform to Regulatory Guide 1.75.

The applicant has committed to install isolation cards or interposing relays, in order to bring this item into compliance with Regulatory Guide 1.75.

Based upon its review, the staff finds these proposed modifications to be in accordance with the guidelines of Regulatory Guide 1.75 and, therefore, acceptable.

9 AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.4 Fuel Handling System

In Section 9.1.4 of the SER, the staff stated that it will require the applicant to implement certain interim measures identified in Enclosure 2 to the December 22, 1980 generic letter prior to issuance of the Comanche Peak operating license. In a letter dated October 8, 1981, the applicant committed to implement the interim measures prior to fuel load of Unit 1. Implementation of the interim measures of NUREG-0612, as required by the December 22, 1980 letter, provides reasonable assurance of safe handling of heavy loads and the staff concludes that the heavy load handling system is acceptable for licensing until completion of the staff's evaluation of the applicant's complete response to NUREG-0612.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection Review

9.5.1.1 Fire Protection Systems Description and Evaluation

Sprinkler and Standpipe System

In the SER, the staff expressed a concern that the standpipe supply pipe diameter was less than that specified in Appendix A guidelines. The staff requested that the applicant perform a hydraulic calculation to verify that the standpipe system could supply two hose streams at the most remote plant area.

By a letter dated August 28, 1981, the applicant stated that a hydraulic calculation has been made which verifies that the standpipe system is adequate and meets the intent of Appendix A guidelines.

Based on this statement, the staff concludes that the standpipe system at Comanche Peak is adequate and meets the guidelines of Appendix A to BTP ASB 9.5-1, Section C.3.d, and is, therefore, acceptable.

Fire Detection System

In the SER, the staff expressed a concern that the power supply for the fire detection system was not connected to the plant emergency power supply. Also, the staff was concerned that the fire detection systems that actuate fire suppression systems were not Class A systems.

The applicant has verbally agreed to comply with the staff's guidelines concerning power supply and Class A detection systems. This commitment will be documented in FSAR Amendment 28.

Based on its review and subject to documentation of the above applicant's commitment, the staff concludes that the fire detection system design meets the guidelines of Appendix A to BTP ASB 9.5-1 and is, therefore, acceptable.

Fire Barrier, Fire Barrier Penetrations, and Fire Dampers

The staff was unable to verify the 3-hour fire rating of fire barriers and fire barrier penetration seals during the site visit.

The applicant has committed, at staff's request, by letter dated August 28, 1981, to provide fire barriers, fire dampers, and penetration seals that have been verified by a fire test design performed in accordance with the ASTM E-119 fire test method.

Based on the applicant's commitments, the staff concludes that the fire barriers, fire dampers, and fire penetration seals are adequate, meet Appendix A to BTP ASB 9.5-1 guidelines in Sections 4.a.1, and D.1.j, and are, therefore, acceptable.

Fire doors

Many door openings in fire rated barriers are provided with labeled fire doors. However, the staff could not determine if the labeled fire doors were the proper fire rating for the area. Also, several areas have doors which are not labeled fire doors. These doors are the water-tight/missile doors.

The applicant verbally agreed to provide fire doors with a fire rating commensurate with the fire rating of the wall assembly. For water-tight/missile doors, the applicant will ensure that these door are equivalent to labeled fire doors. These two commitments will be documented in FSAR Amendment 28.

Based on its review and subject to documentation of the above applicant's commitment, the staff concludes that the fire doors will be in conformance with the guidelines of Appendix A to BTP ASB 9.5-1, Section D.1.j, and are, therefore, acceptable.

Water Fire Extinguishers in the Control Room

In the SER, the staff expressed a concern that water fire extinguishers would not be installed in the control room.

By letter dated August 28, 1981, the applicant has committed to provide water fire extinguishers in the control room.

Based on the applicant's commitment the staff concludes that the installation of water fire extinguishers is adequate and meets the guidelines of Appendix A to BTP ASB 9.5-1, and is, therefore, acceptable.

Battery Room Air Flow Monitors

In the SER, the staff expressed a concern that the battery rooms would not have air flow monitors. The staff requested that the applicant commit to provide air flow monitors for the battery rooms and to describe these air flow monitors. During its site visit, the staff confirmed that the battery rooms had air flow monitors.

The applicant, by letter dated August 28, 1981, has committed to describe the battery room air flow monitors in a future amendment to the FSAR.

Based on the applicant's commitments and the staff's site visit, the staff concludes that the battery room air flow monitiors are adequate, meet the requirements of Appendix A to BTP ASB 9.5-1, and are, therefore, acceptable.

Other Plant Areas

In the SER, the staff stated that "other plants areas" would be evaluated during our site visit. The staff has evaluated other areas of the plant not specifically addressed in our SER and, also, the staff reviewed the plant's communication systems that will be used by the fire brigade.

The applicant has committed to install additional fire detectors, portable extinguishers and automatic sprinklers, prior to fuel loading. The applicant has, also, installed an adequate communication system for use by the fire brigade.

The staff finds that the fire protection for these areas, with the indicated modifications is in accordance with the guidelines of Appendix A to BTP ASB 9.5-1 and is, therefore, acceptable.

9.5.1.6 Administrative Controls, Fire Brigade, and Technical Specifications

In the SER, the staff stated that it did not have adequate information to complete its review regarding the Administrative Controls, Fire Brigade, and Technical Specifications.

The applicant, by letter dated August 28, 1981, has committed to implement the Technical Specifications in accordance with the NRC's Standard Technical Specifications.

The applicant has committed, by letter dated August 28, 1981, to implement the fire protection program contained in the staff's supplemental guidance, "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance," dated August 29, 1977 including (1) fire brigade training, (2) control of combustibles, (3) control of ignition sources, (4) fire fighting and (5) quality assurance.

The staff concludes that, with the applicant's commitments, the 5-man fire brigade, and the equipment and training for the brigade conforms to the recommendations of the National Fire Protection Association, to Appendix A to BTP ASB 9.5-1, to NRC supplemental staff guidelines and, to NRC Standard Technical Specifications, and, therefore, are acceptable.

9.5.1.7 Appendix R Statement

The applicant has not committed to meet the technical requirements of Appendix R to 10 CFR Part 50 or provide equivalent protection. Since the review of Comanche Peak was not specifically conducted to Appendix R requirements, we will require the applicant to meet the technical requirements of Appendix R to 10 CFR Part 50, or provide equivalent protection. This remains an outstanding issue.

9.5.1.8 Summary of Fire Protection Program Review

The remaining outstanding items related to the fire protection program are (1) the alternate safe shutdown system, and (2) a requirement that the fire protection program comply with all the technical requirements of Appendix R to 10 CFR Part 50 or provide equivalent protection. The staff will also confirm the documentation of the applicant's commitment discussed in Section 9.5 above.

9.5.2 Communication System

The staff has reviewed the communications system installed by the applicant for use by the fire brigade, and finds it to be adequate. The staff considers this matter resolved.

11 RADIOACTIVE WASTE MANAGEMENT

11.1 Summary Description

In the summary description of the SER the staff noted that it found the designed liquid and gaseous radioactive waste systems and associated process and effluent radiological monitoring and sampling systems acceptable, but that the staff;s evaluation of the increased storage capacity for the solid waste system would be provided in a supplement to the SER prior to the licensing of Unit 2. This outstanding issue has been resolved as described in Section 11.2.3.1. Therefore, the staff now finds the designed liquid, gaseous and solid radioactive waste systems and associated process and effluent radiological monitoring and sampling systems acceptable.

11.2 System Description and Evaluation

11.2.3 Solid Radioactive Waste Treatment System

The staff stated in the SER that "wet" solid wastes -- consisting of boron recycle systems, floor drain and waste evaporator bottoms, spent resin from deep bed and filter/demineralizers, reverse osmosis concentrations, and chemical drain tank effluents - are preconditioned to meet the feed conditions required by the ATCOR-132A cement solidification system in 50 ft^3 containers for offsite shipment. The staff also noted it had not completed its review of the ATCOR-132A system under the topical report program. In a letter dated September 3, 1981, to Mr. Martin Brownstein (ATCOR) from Mr. Robert L. Tedesco (NRC) the staff found the ATCOR topical report to be acceptable.

11.2.3.1 Conformance with Federal Regulations and Branch Technical Positions

The staff described the storage facilities for solid waste as including an area in the fuel building for approximately 35 50-ft^3 containers and 50 55-gal drums. The staff found that this space is adequate for operation of Unit 1, but questioned the adequacy for the design to accommodate the radwastes expected during normal operations, including anticipated operational occurrences, with both units operational. In FSAR Amendment 26, the applicant advised that storage capacity for up to 74 50-ft^3 containers is provided in the drum storage area and storage capacity of at least 144 55-gal drums is provided in Area 247 in the fuel building. Based on this additional information, the staff estimates that the facility provides at least 30 days storage capacity for packaged solid radwaste for each unit, and therefore finds the storage capacity adequate to meet the demands of Comanche Peak Unit 1 and 2 during normal operation. The staff concludes the solid radwaste system is acceptable.

11.4 Evaluation Findings

The evaluation findings presented in the SER noted the staff's concern that the storage area for the packaged wet waste might be inadequate for the facility with both units operational. This outstanding issue was resolved as described in Section 11.2.3.1 of this supplement.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure and Qualifications

13.1.1 Management and Technical Resources

13.1.1.1 General

An audit team composed of members of the Office of Nuclear Reactor Regulation and the Office of Inspection and Enforcement visited the applicant's corporate offices and the Comanche Peak site during the period of September 2-4, 1981. The purpose of the visit was to explore with the applicant the items of concern noted in the SER and to confirm the staff findings as stated in the SER. The results of that audit team visit are presented in the following sections.

13.1.1.2 Corporate Organization

The staff noted in the SER that the Texas Utilities Generating Company (TUGCO) was responsible for plant operation while Texas Utilities Services Inc. (TUSI) provided the technical resources. The staff had some concern that TUGCO might not be able to obtain technical support as needed to support plant operation, although the SER stated our belief that, in practical effect, the matrix organization involving the two companies is little different from the relationships that exist between different divisions in a single company. The staff confirmed this belief during the audit team visit.

The audit team was informed by Mr. M. D. Spence, President of TUGCO and TUSI, of his personal commitment to assure that members of TUGCO and TUSI operate as a team to attain a common corporate goal of having a safe, reliable plant. The audit team interviewed 18 individuals of both TUGCO and TUSI ranging from the level of Mr. Spence down through the corporate officers and the plant management to a shift supervisor and a shift technical advisor. Throughout these interviews, the audit team was impressed with the knowledge and enthusiasm and the obvious esprit that permeates these organizations. Each of the individuals interviewed, in addition to knowing his own job, seemed to have a good working knowledge of the responsibilities of others. The audit team found that personal rapport and easy work relationships seemed to exist throughout the organization, without regard to whether individuals were nominally employed by TUGCO or TUSI.

The audit team confirmed that those TUSI personnel supporting Comanche Peak are and will be dedicated to that support role. Further, the audit team confirmed that TUGCO has a significant in-house technical capability, even without considering the TUSI support. The plant staff will have about 39 engineers assigned for two unit operation. This technical capability will be backed up by an additional group of about 25 TUGCO engineers assigned to the corporate Operations Support Group, but who will be physically located in the nuclear operations support facility, immediately adjacent to the plant site. The TUSI Technical Support Group, consisting of about 40 engineers providing the in-depth technical support for plant operation, is scheduled to be located on site where they will be immediately available to support the plant as necessary.

The staff concludes that any concerns it may have had regarding the adequacy of technical support were unfounded. The applicant has provided or will provide adequate technical resources to support plant operation. Further, the staff concludes that the matrix organization involving TUGCO and TUSI is acceptable since the personnel involved, to all intents and purposes, are working together as though they are all members of the same company.

The staff also noted in the SER an apparent lack of commercial operating experience at the corporate level. The audit team explored this matter during the visit.

Our discussions with corporate officers indicated an awareness of the necessity for management commitment to nuclear affairs and confirmed that this commitment exists. All members of the corporate staffs whom the audit team interviewed seem dedicated to achieving a first class operation at Comanche Peak and they are aware of the effort and personal commitment that this will entail.

The audit team found that the senior corporate officers (Mr. Gary, Executive Vice-President and General Manager of TUGCO; Mr. Fikar, Executive Vice-President, TUSI, and Vice-President, TUGCO; Mr. Clements, Vice-President, Nuclear of TUGCO; and Mr. George, Vice-President and Project Manager of TUSI) hold weekly meetings each Monday to assure that each officer is aware of the plant status and that proper management oversight is taking place. In addition, the principals of each company involved in the Comanche Peak project, including major contractors, meet on a two-month cycle to assure that all are up-to-date on the project status.

The present design and construction efforts at Comanche Peak utilize personnel from a number of contractors. However, the effort is under the direct control of TUSI personnel. Each part of the design-construction force is under the control of TUSI. As explained by Mr. George, Vice-President and Project General Manager of TUSI, the intent is to retain selected individuals at the end of the design and construction phase so that the Technical Support Group will essentially be a mini-AE available for continuing support of plant operation.

Other than for the experience of Mr. Clements who was the commanding officer of a nuclear submarine for more than four years prior to joining TUGCO, the team found little reactor operating experience on the corporate staff. There is no commercial operation experience among the corporate officers. However, the audit team found that a total of 54 persons have now attended the Westinghouse training program at Zion. All but two of these people have been certified at the SRO or RO level and many of them, including the Manager of Nuclear Operations are on the corporate staff. The audit team also found that the corporate officers have considerable experience in fossil plant operations and that they are aware of the special problems of and the demands to be faced in nuclear plant operations. Further, as noted in the SER the corporate level support groups of both TUGCO and TUSI are comprised of well-qualified individuals who have had considerable experience in both fossil and nuclear power plants. A number of the corporate officers, including Mr. Gary, the Executive Vice-President and General Manager, TUGCO, serve on industry committees and groups where they are in frequent contact with other nuclear utility personnel and where they gain and maintain an awareness of special industry problems. As expressed by Mr. Gary, the top management of a utility must be involved with the nuclear plant and must be knowledgeable enough to know when the plant needs help. Mr. Gary now spends about 15 percent of his time with Comanche Peak and anticipates that this will increase to approximately 25-30 percent by the time Unit 1 goes into operation. Mr. Clements spends 100 percent of his time on Comanche Peak affairs.

Overall, the audit team found that the corporate officers are knowledgeable and dedicated to safe, reliable operation of Comanche Peak. The level of staffing at both the corporate and plant levels and the emphasis placed upon obtaining and training qualified individuals is indicative of this knowledge and commitment. While the corporate officers lack actual experience in commercial nuclear plant operation, they are experienced in the functions of management as regards supervision of plant operation. The staff concludes that the personal involvement and commitment of the corporate officers is such that little could be gained by requiring an augmentation of the corporate staff to include a person or persons with actual commercial nuclear experience. Accordingly, the staff does not proposed to require such augmentation.

13.1.2 Operating Organization

13.1.2.1 Plant Staff

The staff noted in the SER that it would require that, if contractor personnel are used to provide experience in plant operation to operating shifts, the NRC would require that these augmentation personnel remain on shift until the plant attains a 100% power level. During the audit team visit, the staff was informed that the applicant will retain such augmentation personnel on shift until the plant reaches a 100% power level. By letter dated September 19, 1981, the applicant informed the staff that one operator with at least one year of PWR operating experience will be available on each shift until the plant reaches the maximum licensed power level. The staff will include a license condition in the operating license for Unit 1.

13.3 Emergency Planning

The staff has reviewed the applicant's response dated August 24, 1981, to the list of deficiencies issued by the staff based on the review of the applicant's Emergency Plan, Revision 1. The staff has determined that a number of questions remain unresolved based on the applicant's responses and has scheduled a meeting with the applicant to clarify the scope and depth of information required from the applicant before the staff can make a favorable finding on the adequacy of the plan. The staff will complete its review of the applicant's Emergency Plan under TMI Action Plan Item III.A.2 (see Section 22).

13.4 Review and Audit

13.4.3 Independent Safety Engineering Group (ISEG)

In the SER, the staff noted that the applicant had not informed us as to the number and qualifications of the members of the ISEG. During the audit team visit, the team was informed that the ISEG will consist of five graduate engineers having at least three years of nuclear plant experience. By letter dated September 29, 1981, the applicant confirmed this commitment and stated that efforts are underway to hire these individuals so that they can become involved in the design, construction and pre-operational testing of the plant so as to maximize their nuclear plant experience. The staff finds this commitment to be acceptable.

13.5 Station Procedures

13.5.1 Administrative Procedures

See SER Section 13.5.

13.5.2 Operating and Maintenance Procedures

A review has been conducted of the applicant's plan for development and implementation of operating and maintenance procedures as described in the FSAR. The objective of the review was to determine the adequacy of the applicant's program for assuring that routine operating, off-normal, and emergency activities are performed in a safe manner. In determining the acceptability of the applicant's program, the following criteria were used: 10 CFR Part 50, §50.34 and Regulatory Guide 1.33.

The applicant has committed to a program in which all activities that affect safety-related structures, systems, and components are to be conducted in accordance with detailed, written, and approved procedures meeting the requirements of Regulatory Guide 1.33, Revision 2 dated February 1978, "Quality Assurance Program Requirements (Operation)," and ANSI 18.7-1976/ANS 3.2.

The applicant utilizes the following categories of procedures for those operations performed by licensed operators in the control room.

Intergrated Plant Operating Procedures

System Operating Procedures

Alarm Procedures

Abnormal Conditions Procedures

Emergency Operating Procedures

Radwaste Systems Procedures

Operating and maintenance procedures have also been prepared in the following areas:

Maintenance and Modification

Plant Radiation Protection

Emergency Preparedness

Chemical and Radiochemical Control

Instrument Calibration and Tests

Material Control

Security

Fuel Handling

Fire Protection

The staff concludes that the applicant's program for developing operating and maintenance procedures is acceptable and meets the relevant requirements of 10 CFR Part 34, and is consistent with the guidance provided in Regulatory Guide 1.33 and ANSI 18.7-1976/ANS 3.2. The applicant's programs for reanalysis of transients and accidents, and for development of upgraded emergency operating procedures, are contained in Chapter 22, Items I.C.1 and I.C.8.

13.6 Industrial Security

The applicant has submitted the "Comanche Peak Steam Electric Station (CPSES) Physical Security Plan," "Comanche Peak Steam Electric Station Security Training and Qualification Plan," and the "Comanche Peak Steam Electric Station Safeguards Contingency Plan" for the protection of Unit 1 from acts of radiological sabotage.

As a result of our evaluation, the staff has identified certain areas of the "CPSES Physical Security Plan" which require additional information and revision to satisfy the requirements of 10 CFR Part 73, Section 73.55.

The applicant has submitted detailed commitments to revise the physical security plan in response to staff comments. The security plan, when revised in accordance with the applicant's written commitments, will be considered to meet the requirements of 10 CFR Part 73, and therefore, to be acceptable.

The "CPSES Security Training and Qualification Plan" and the "CPSES Safeguards Contingency Plan" have been determined to meet the requirements of 10 CFR Part 73 and, therefore, are approved.

The identification of vital areas and measures used to control access to these areas, as described in the security plan, may be subject to amendment in the future based on a confirmatory evaluation of the plant to determine those areas where acts of sabotage might cause a release of radionuclides in sufficient quantities to result in dose rates equal to or exceeding 10 CFR Part 100 limits.

The applicant's plans are being withheld from public disclosure in accordance with the provisions of Section 2.790(d) of 10 CFR Part 2.

15 ACCIDENT ANALYSIS

15.3 Infrequent Transients and Postulated Accidents

15.3.8 Loss-of-Coolant Accident (LOCA)

In the LOCA calculations, the applicant assumed that the temperature of the water in the upper-head region is equal to the cold-leg temperature as a result of the increased bypass flow that cools the upper-head region. The increased bypass flow was considered in the LOCA analyses. In the SER, the staff noted that the applicant had committed to verify the adequacy of the upper-head-region temperature assumption for the plant LOCA calculations.

By a letter dated September 14, 1981, the applicant submitted a description of the tests and analyses performed to confirm that the temperature assumed for the upper-head region of the reactor vessel is appropriate for the LOCA analysis. The staff reviewed the submittal, and its supporting reference, and agrees that the temperature assumed for the upper-head region is acceptable for the LOCA calculations. The staff concludes this item is resolved.

15.3.9 Anticipated Transients Without Scram (ATWS)

In the SER the staff noted that the applicant is required to take two steps prior to operation, and before completing the plant modification determined to be necessary by the Commission. These two steps are as follows:

(1) Develop emergency procedures to train operators to recognize ATWS events, including consideration of scram indicators, rod position indicators, flux monitors, pressurizer level and pressure indicators, pressurizer relief valve and safety indicators, and any other alarms annunicated in the control room, with emphasis on alarms not processed through the electrical portion of the reactor scram system.

(2) Train operators to take actions in the event of an ATWS, including consideration of manually scramming the reactor by using the manual scram button, prompt actuation of the auxiliary feedwater system to ensure delivery to the full capacity of this system, and initiation of turbine trip. The operators should also be trained to initiate boration by actuating the high-pressure safety-injection system to bring the facility to a safe-shutdown condition.

A review of the applicant's ATWS procedure for conformance to these procedural requirements will be conducted in conjunction with the review performed for TMI Item I.C.1. A description of the review and a finding on the acceptability of the Comanche Peak Anticipated Transient Without Scram procedure will be included under TMI Item I.C.1 in a future supplement to the SER.

16 TECHNICAL SPECIFICATIONS

In the SER, the staff identified certain conditions which must be included in the Comanche Peak Technical Specifications in addition to Standard Technical Specifications. In this supplement, the staff has identified the need for additional plant-specific Technical Specifications. The complete list of plant-specific Technical Specifications is given below; they are discussed further in sections of the SER or this supplement as indicated.

<u>Technical Specifications Identified in the SER</u>

(1) Control of mineral exploration within the exclusion area (Section 2.1.2).

(2) The constant axial offset control band will be limited to $\pm 5\%$ ΔI for the first 3000 MWD/MTU burnup, and to +3 to -12% ΔI thereafter (Section 4.3.2.1).

(3) DNBR penalty for rod bow (Section 4.4.1).

(4) Surveillance for detection of crud buildup (Section 4.4.1).

(5) Prohibit operation in the N-1 mode (Section 4.4.4).

(6) Pressure-temperature limits for the reactor vessel (Section 5.3.3).

(7) Inspection of steam generator tubes (Section 5.4.2.2).

(8) Restrictions on isolation valves in the containment purge system and containment pressure relief system (Section 6.2.3 and Section 22, II.E.4.2).

(9) Containment leak testing requirements (Section 6.2.5).

(10) Testing of safety system response times (Section 7.2.4).

(11) Testing of auxiliary feedwater pumps (Section 7.3.1.6).

(12) Methodology for establishing safety system set points (Section 7.3.2.6)

(13) Trip setpoints for offsite power undervoltage protection (Section 8.2.4)

(14) List valves that require power lockout (Section 8.4.3).

(15) Sampling and analysis of demineralizer resins from condensate cleanup system (Section 11.2.1.2).

(16) Process control program for solid waste system (Section 11.2.3).

(17) Restrictions on boron dilution rates (Section 15.2.3.1).

(18) Limitations and surveillance of auxiliary feedwater system (Section 22, II.E.1.1, Recommendations GS-1 and GS-2).

Additional Technical Specifications Identified in this Supplement

(1) Monitoring of safe shutdown impoundment dam for erosion and weathering (Section 2.4.3.1).

(2) Monitoring for sediment buildup in the service water intake channel (Section 2.4.8).

(3) Leak testing of pressure isolation valves for reactor coolant pressure boundary (Section 3.9.6)

(4) Inservice inspection of reactor coolant pump flywheel (Section 5.4.1).

(5) Adherence to accepted secondary water chemistry monitoring and control program (Section 5.4.2.3).

(6) Fire protection program (Section 9.5.1.6).

17 QUALITY ASSURANCE

17.5 List of Systems, Structures and Components Under Control of the QA Program

Section 17.5 of the SER stated that the applicant had not responded to shortcomings identified by the staff in the list of structures, systems, and components under the control of the QA program. By FSAR Appendix 17A (revised) and response to staff Question 260.1, differences between the staff and the applicant regarding the list have been resolved to the staff's satisfaction. The list has been expanded to include safety-related items reflected in NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980. Therefore, the staff has no open issues concerning the quality assurance program for operations or as to the items to which the program applies. Accordingly, the staff concludes that the applicant's description of the QA program is in compliance with applicable NRC regulations.

20 FINANCIAL QUALIFICATIONS

20.1 Business of Applicant

The SER noted that the NRC had received a request, dated May 28, 1981, from the applicant for an amendment to the Construction Permits that would authorize Tex-La Electric Cooperative, Inc. (Tex-La) to become a joint owner of Comanche Peak. On September 30, 1981, the NRC issued amendments to the Construction Permits authorizing the addition of Tex-La as a co-owner in the Comanche Peak facility. The amendments to the Construction Permits also recognized that with the addition of Tex-La as a co-owner/co-applicant, the percentages of ownership authorized agree with those given in this section of the SER and used by the staff in its earlier evaluation.

20.5 Reasonable Assurance of Funds, Costs of Operation

The SER noted that the review of Tex-La's financial qualifications will be completed and a separate report issued in response to the applicant's request for amendments to the Construction Permit. The staff reviewed the financial qualifications of Tex-La in a "Safety Evaluation Supporting Amendment No. 4 to CPPR-126 and CPPR-127," issued September 30, 1981. That document states,

> "Since the date of the applicants' request to amend the Construction Permits, Tex-La's plan for financing its full ownership share has virtually been realized. On August 3, 1981, the U.S. Rural Electrification Administration issued a $180 million loan guarantee commitment notice to Tex-La, an amount substantially in excess of Tex-La's estimated total capital contribution to the subject facility ($135 million). Because this REA commitment for the loan guarantee is already in effect, Tex-La has satisfied NRC's financial qualifications requirements, as described below."

The evaluation of Tex-La's financial qualifications concluded:

> "In accordance with the provisions of 10 CFR 50.33(f) and Appendix C to 10 CFR Part 50, Tex-La Electric Cooperative of Texas, Inc. has demonstrated reasonable assurance that it can obtain the funds to purchase a 4-1/3 percent ownership interest in Comanche Peak. Accordingly, Tex-La is financially qualified under the provisions of the above regulations to purchase such an interest."

20.7 Conclusion

The conclusion stated in the SER, as it relates to Tex-La, was contingent upon a favorable conclusion by the NRC staff in its review of the applicant's request for amendments to the Construction Permits to include Tex-La as a co-owner of the facility. This favorable conclusion was issued on September 30, 1981, and Tex-La has been authorized to become a co-owner of the facility. With that action, the staff's review of the applicant's financial qualifications is complete and the staff reconfirms its finding that the applicant is financially qualified to operate and safely decommission the Comanche Peak facility.

22 TMI-2 REQUIREMENTS

22.2 Discussion of Requirements

The SER provided, for each of the TMI-2 Requirements, a statement on the staff's Position, a Clarification statement if applicable, and a Discussion and Conclusions on the resolution or status as it relates to Comanche Peak. This supplement will provide only the Discussion and Conclusions.

I.A.1.1 Shift Technical Advisor

During the audit team visit, the staff learned that the applicant now has three shift technical advisors assigned and plans to have six available by the time of plant startup. The incumbents are recent college graduates, all of whom hold bachelor of science degrees in nuclear engineering. Training courses for these individuals are underway. By the time of fuel loading, each of these individuals will have had some practical experience in addition to the special STA training. In addition, since the STAs report to the Operations Support Engineer, who also controls the Independent Safety Engineering Group (ISEG) (see I.B.1.2, below) the STAs will have the benefit of advice and assistance provided by the experienced engineers who will be members of the ISEG.

By a letter dated September 29, 1981, the applicant informed the staff that four STAs had been hired and that these individuals were to start their STA training in October 1981. Resumes of these four individuals, furnished with the September 29 letter, indicate that three hold bachelor of science degrees in nuclear engineering while the fourth individual has a bachelor of science degree in mechanical engineering. All are 1981 college graduates. In the September 29 letter, the applicant stated that all STAs will meet the education, experience and training requirements of NUREG-0737. The staff finds this commitment and the educational qualifications of the STA incumbents to be acceptable.

I.A.1.2 Shift Supervisor Administrative Duties

In connection with the audit team visit, the applicant furnished a copy of an office memorandum dated July 8, 1981 from Mr. B. R. Clements, Vice President, Nuclear to the senior plant staff and the shift supervisors, delineating the shift supervisor responsibilities. This memorandum assigns safe plant operation as the primary mangement responsibility of each shift supervisor on his shift. It tasks the shift supervisors with directing plant operations including the management of assigned personnel in accordance with applicable technical specifications, regulations, licenses and approved procedures. The memorandum further directs the shift supervisors to remain in the control room at all times during accident conditions to direct operating activities, but to maintain a broad perspective of operating conditions and avoid becoming totally involved in any single operation during emergency situations. The memorandum directs that in the event the shift supervisor is absent from the control room during normal operations, an assistant shift supervisor shall be designated to assume the shift supervisor's control room responsibilities.

The applicant also furnished and the audit team has reviewed a copy of Procedure No. ODA-102, Revision 3, from the Operations Department Administrative Manual. This procedure, entitled "Shift Complement Responsibilities and Authorities," was issued on January 27, 1981 and was immediately effective. The procedure delineates the responsibilities and authorities of shift members. The procedure reinforces the shift chain-of-command from the shift supervisor to the assistant shift supervisor to the control room operator.

By a letter dated September 29, 1981, the applicant informed the staff that a list of administrative duties of the Shift Supervisor will be provided to the Vice President, Nuclear, for review prior to fuel load. Administrative functions that detract from or are subordinate to the management responsibility for ensuring the safe operation of the plant will be delegated to other operations personnel with no control room operating duties. Auxiliary Operators and clerical personnel will assume these functions. The staff finds this commitment by the applicant to be acceptable and, accordingly, finds that the applicant has made adequate provisions to satisfy the requirements of Action Plan Item I.A.1.2.

I.A.1.3 Shift Manning

Procedure No. ODA-102, Revision 3, "Shift Complement Responsibilities and Authorities," describes the required shift manning for various modes of plant operation. It satisfactorily addresses the numbers of licensed and unlicensed operators to be present on each shift. However, it does not mention the shift technical advisor, the radiation protection technician, the chemistry technician, or the communicator, who are also required to be on each shift.

By a letter dated September 29, 1981, the applicant informed the staff that a Shift Technical Advisor (during Modes 1, 2, 3 & 4), a Communicator (member of the Shift Operating Organization), and two Radiation Protection Technicians (one qualified to initiate the post-accident sampling and analysis process) will be assigned to each shift. This commitment satisfies the staff's concerns regarding the adequacy of shift staffing and is acceptable.

Restrictions on use of overtime were incorporated in Procedure No. ODA-302, Revision 1, "Relief of Personnel and Overtime Restrictions." However, Revision 2 of ODA-302 has been retitled "Relief of Personnel" and the overtime restrictions are now incorporated in another plant procedure, STA-615. The September 29 letter furnished a copy of procedure STA-615. The staff has reviewed this procedure and has found it to be in accordance with the requirements outlined in NUREG-0737. Accordingly, the staff finds that the applicant has made adequate provision to restrict the use of overtime.

In summary the staff finds that the applicant has made acceptable plans for shift manning and for restrictions on the use of overtime. Accordingly, the staff finds that the applicant has acceptably met the requirements of Action Plan Item I.A.1.3.

I.B.1.2 Evaluation of Organization and Management Improvements of Near-Term Operating License Applicants

As noted in Section 13.4.3 of this supplement, the audit team was informed that the Independent Safety Engineering Group (ISEG) would be composed of five graduate engineers having at least three years of nuclear plant experience. This commitment was confirmed by the applicant's letter dated September 29, 1981, and accordingly the staff finds that the applicant has acceptably met the requirements of Action Plan Item I.B.1.2.

I.C.2 Shift Relief and Turnover Procedures

The audit team was furnished a copy of Procedure No. ODA-302, Revision 2, "Relief of Personnel," which describes the shift relief and turnover procedures to be used by the applicant. The procedure contains guidance and checksheets to be used for shift relief and turnover to assure that all oncoming operators are aware of the status of the plant or portions of the plant for which they are assuming responsibility. The relief checksheet for auxiliary operators is still under development. Guidance and a checksheet also is provided for the Operations Engineer to periodically monitor and evaluate the effectiveness and completeness of the turnover.

Except for the lack of a checksheet for the auxiliary operators, the staff finds the provisions for shift relief and turnover to be in accordance with the staff requirements as specified in Action Plan Item I.C.2, and, therefore, acceptable.

The applicant's letter of September 29, 1981, forwarded a copy of a memorandum entitled "Operator Guidelines" dated August 20, 1981, which the applicant states is a precursor to the checklist for auxiliary operators. The memorandum provides a comprehensive listing of the status of plant equipment that the auxiliary operators should check for. When fully developed and converted to a format for use in shift relief and turnover, it should be acceptable. However, final close-out of this matter must await the staff's further review of the developed check-list.

I.C.3 Shift Supervisor Responsibilities

As noted in Action Plan Item I.A.1.2, the staff has reviewed the memorandum from the Vice President, Nuclear, which emphasizes the primary management responsibility of the shift supervisor and which clearly defines his command duties. The staff has also reviewed Procedure No. ODA-102, "Shift Complement Responsibilities and Authorities." Based upon its review of these documents, the staff concludes that the provisions made by the applicant establishing the responsibilities of shift supervisors are in accordance with staff requirements and are, therefore, acceptable. Accordingly, this matter is resolved.

I.C.4 Control Room Access

The audit team was furnished a copy of Procedure No. ODA-306, Revision 2, "Control Room and Observation Area Access." This procedure, which was issued on February 2, 1981, establishes the authority, guidelines, and requirements for access to the control room and the observation area. The staff review indicates that the controls established governing access to the control room are in accordance with the staff requirements as specified in Action Plan item I.C.4 and are, therefore, acceptable. Further, the staff's review of Procedure No. ODA-102, Revision 3, "Shift Complement Responsibilities and Authorities" indicates that the applicant has established a clear line of authority and responsibility in the control room, in accordance with the requirements of Action Plan Item I.C.4. The staff thus concludes that the applicant has satisfied the requirements of Action Plan Item I.C.4 and considers this matter resolved.

I.C.5 Procedures for Feedback of Operating Experience to Plant Staff

The applicant furnished to the audit team a copy of draft procedure No. NOA-XXX, "Review of Significant Event Reports." This draft procedure establishes the manner by which the applicant will receive, evaluate and track information received from the Institute of Nuclear Power Operations (INPO) over the Significant Event Evaluation and Information Network (SEE-IN). The procedure appears to provide an adequate means for handling this information. By letter dated September 29, 1981, the applicant also furnished a copy of procedure ODA-107, "Reporting of Operational Incidents," which prescribes the handling of internally generated operating experience information. This procedure makes acceptable provisions for originating information within the Operations Department.

Procedure No. ODA-106, Revision 0, "Review of Documents," prescribes the methods to be used for disseminating information to the operations department personnel and documenting the review of this information. It includes provisions for the Operations Engineer to screen the available information to assure that department personnel are not burdened with extraneous and unimportant information. The procedure relies on having personnel read the available information to gain an awareness.

By letter dated September 29, 1981, the applicant informed the staff that the Operations Support Superintendent is responsible for dissemination of all pertinent operating experience information received from industry sources. Included is the INPO information received from the SSE-IN program which is disseminated per procedure NOA-XXX (to be numbered later). Other procedures will be developed to handle other sources of information such as NRC bulletins and orders, NRC information notices, and the many other sources of industry operating experience information. Procedures will also be developed which ensure that operating experiences reported onsite, including all licensee event reports, are reviewed by operations support section personnel (STAs) and routed to the appropriate departments including the training department.

The staff concludes that the applicant has made and is making generally acceptable provisions for the feedback of operating experience information to the plant staff. However, final close-out of this item must await receipt and review of the remaining procedures now under development.

I.C.6 Verify Correct Performance of Operating Activities

The applicant furnished to the audit team a copy of Procedure No. STA-605, Revision 2, "Clearance and Safety Tagging," which establishes a safe, reliable and uniform method of component, subsystem and system tagging. Incorporated in this procedure are requirements that a second, licensed operator must verify status of components that are removed from service or that are being returned to service. This applies to all safety-related systems, subsystems and components and to nonsafety-related systems or components whose function or loss of function affects safe plant operation.

Based upon the staff's review of this procedure, the staff concludes that the provisions for independent verification of operating activities are in accordance with the requirements of Action Plan Item I.C.6 and are, therefore, acceptable. The staff considers this matter to be closed.

II.D.1 Performance Testing of Boiling Water Reactor and Pressurized Water Reactor Relief and Safety Valves

The applicant has stated that it will participate in the EPRI/NSAC program to conduct performance testing of PWR relief and safety valves and associated piping and supports. The applicant has referenced the proposed EPRI program ("Program Plan for the Performance Verfication of PWR Safety/Relief Valves and Systems," dated December 13, 1979) for the performance testing of these valves.

A description of the EPRI/NSAC test program was provided to NRC by EPRI in December 1979 and an updated revision of the program description was provided in July 1980. The staff has reviewed these descriptions and is in agreement that the NUREG-0737 technical requirements for relief and safety valves and associated piping and supports can be met on satisfactory completion of the test program.

By letter dated July 1, 1981, from R. C. Youngdahl to Harold R. Denton, the PWR Owners Group reported on the status of the EPRI PWR safety and relief valve test program to date and requested an extension of the completion dates specified in NUREG-0737. The Owners Group stated their intention to develop an expanded test matrix in order to obtain more information with respect to the effects of inlet piping configurations and adjustments of ring settings on safety valve operation.

On July 17, 1981, the staff met with EPRI and the PWR Owners Group representatives to review the status of the test program and discuss the expanded test matrix. Although the exact number of additional tests will have to be determined as the program progresses, the test program managers estimated that it could take from four to eight months longer than the original test completion date of July 1, 1981, to complete the expanded test program.

The PORV test results reported to date indicate that, while the initial screening criteria were not met in some cases, all PORVs tested will function in the primary mode (pressure relief) as required. Additional PORV tests are being planned to evaluate the effect of variable water seal temperature on valve closure times.

The safety valve test results indicate a need for additional information regarding the effects of inlet piping configuration and adjusting ring settings on safety valve operation.

Based on its review of the EPRI test program to date, the staff has concluded that the program represents a full responsive effort to meet Commission requirements and that the additional testing program will provide needed information to assure that the technical requirements of Item II.D.1 of NUREG-0737 will be met. Since testing to date has not uncovered problems with safety and relief valves which indicate the potential for occurrences that fall outside the safety evaluation of operating plants, the staff believes that good cause has been shown to extend the NUREG-0737 completion date for PORV and safety valve testing so that the extended EPRI program may be carried to completion on an orderly basis. The latest estimated test completion date is March 31, 1982.

The staff, in a memorandum to the NRC Commissioners, SECY 81-491 dated August 17, 1981 summarized the results of the EPRI/NSAC program and requested Commission approval to extend the schedule for plant specific submittal of PWR valve test program results to July 1, 1982.

Subsequently, Commission approval of the extended schedule was provided. The applicant in Amendments 27 and 28 to the FSAR has committed to the requirements of this item to the extent practicable at this time. The applicant is participating in the EPRI/PWR safety and relief valve performance verification test program and is monitoring this program to assure that the test results apply to the plant specific valves and associated piping and supports. Regarding the NUREG-0737 requirement for verification of PORV block valve functionability, this topic is under continuing discussion between PWR utilities and the NRC staff. The applicant has committed to abide by whatever final conclusions are reached between the NRC and the PWR Owners Group utilities. The staff believes that these commitments provide adequate assurance that the requirements for performance testing of the relief valves, safety valves, block valves, and associated piping will be satisfied.

The basis for accepting this commitment is the staff's review to date of the EPRI/NSAC relief and safety valve test program and its continued review of this program to confirm that it is acceptable for the Comanche Peak plant specific design. The staff will report the final results of its review in a future supplement to the SER.

II.K.2.13 Thermal Mechanical Report--Effect of High-Pressure Injection on Vessel Integrity for Small-Break Loss-of-Coolant Accident With No Auxiliary Feedwater

The staff has reexamined this item and concludes that there is no reason that it should not be considered confirmatory. Therefore, the staff has shifted its classification from an outstanding issue (Section 1.7) to a confirmatory issue (Section 1.8).

II.K.2.17 Potential for Voiding in the Reactor Coolant System During Transients

The staff has reexamined this item and concludes that there is no reason that is should not be considered confirmatory. Therefore, the staff has shifted its classification from an outstanding issue (Section 1.7) to a confirmatory issue (Section 1.8).

II.K.3.1 Installation and Testing of Automated Power-Operated Relief Valve Isolation System

In Amendment 22 to the FSAR, the applicant referenced a Westinghouse Topical Report (WCAP-9304) to address this issue. The report was submitted on March 15, 1981 and is under staff review. If modifications are indicated by the staff review of WCAP-9804, the staff will require that Comanche Peak be modified accordingly (see Item II.K.3.2).

II.K.3.2 Report on Overall Safety Effect of Power-Operated Relief Valve Isolation System

In Amendment 22 to the FSAR, the applicant referenced a Westinghouse Topical Report (WCAP-9304) to address this issue. The report was submitted on March 15, 1981 and is under staff review. This schedule is consistent with the requirements of NUREG-0737 and the staff finds this acceptable (see Item II.K.3.1).

II.K.3.11 Justification of Use of Certain PORVs

The staff has reexamined this item and concludes that there is no reason that it should not be considered confirmatory. Therefore, the staff has shifted its classification from an outstanding issue (Section 1.7) to a confirmatory issue (Section 1.8).

III.A.1.1 Upgrade Emergency Preparedness

This item has been superseded by new rulemaking for emergency planning. Regulatory Guide 1.101 Revision 1, and NUREG-75/111 have been superseded by NUREG-0654 and NUREG-0696. The staff will complete its review of the applicant's Emergency Plan under TMI Action Item III.A.2.

III.A.1.2 Upgrade Emergency Support Facilities

On August 24, 1981, the applicant submitted a design description of the Comanche Peak upgraded emergency response facilities in response to Generic Letter 81-10. This information is under review by the staff. The results of that review will be reported in a future supplement to the SER.

III.A.2 Improving Licensee Emergency Preparedness--Long Term

As noted in Section 13.3 of this supplement, the staff's review of the applicant's submittal dated August 24, 1981, determined that a number of questions remained unresolved. A meeting is scheduled with the applicant to discuss these matters. Further, the staff has not received the State of Texas Emergency Plan, which is expected in early 1982. The staff cannot complete its evaluation until the state plan is available. As stated above, review of the applicant's Emergency Plan will be completed as part of this item.

APPENDIX A

CONTINUATION OF CHRONOLOGY OF NRC STAFF RADIOLOGICAL SAFETY REVIEW OF COMANCHE PEAK

July 9, 1981	Letter from applicant concerning Physical Security Plan.
July 10, 1981	Letter from applicant transmitting draft responses to Questions 251.1 through 251.4.
July 13, 1981	Letter from applicant concerning Comanche Peak schedule.
July 14, 1981	Letter to applicant transmitting two copies of the SER on Comanche Peak.
July 14-17, 1981	Site visit by Power Systems Branch reviewer and consultant.
July 15, 1981	Meeting concerning the environmental qualification program for safety-related electrical equipment.
July 17, 1981	Letter to applicant concerning Comanche Peak Review Schedule.
July 17, 1981	Letter from applicant transmitting Amendment No. 24 to the FSAR.
July 20, 1981	Letter to applicant concerning Prompt Notification in the Event of an Emergency.
July 23, 1981	Letter to applicant transmitting review comments on Emergency Plan.
July 23, 1981	Letter from applicant transmitting an advance copy of the response to Question 032.108.
July 24, 1981	Letter to applicant transmitting 20 copies of NUREG-0797, SER for Comanche Peak.
July 27, 1981	Letter from applicant concerning Power Systems Branch Site Visit - Separation at Hot Shutdown Panel.
July 31, 1981	Letter to applicant concerning Steam Generator Overfill (Generic Letter 81-28) - formerly issued as 81-16 on July 1, 1981.
July 31, 1981	Letter from applicant transmitting clarification and supplemental information on several topics concerning conduct of operations at Comanche Peak.

August 3, 1981	Letter from applicant concerning a response to Generic Letter 81-01.
August 4, 1981	Meeting concerning seismic qualification of safety-related equipment.
August 5, 1981	Letter to applicant concerning withdrawal of Power Systems Branch Question 040.139.
August 5, 1981	Letter from applicant concerning open items response schedule.
August 7, 1981	Letter from applicant transmitting Amendment 25 to the FSAR.
August 7, 1981	Letter to applicant concerning simulator examination (Generic Letter 81-29).
August 7, 1981	Letter from applicant concerning control of heavy loads: NUREG-0612.
August 10, 1981	Letter to applicant concerning Physical Security Plan.
August 14, 1981	Letter from applicant concerning consequences of a postulated loss of dc bus coupled with a single failure disabling a PORV allowing a cold overpressurization event.
August 14, 1981	Letter from applicant concerning SSI sedimentation monitoring.
August 18, 1981	Letter from applicant concerning the Physical Security Plan.
August 19, 1981	Letter from applicant concerning proposed secondary water chemistry monitoring program, revision 2.
August 24, 1981	Letter from applicant transmitting a response to NRC Question 400.4.
August 24, 1981	Letter from applicant concerning backfill and bedding liquification potential.
August 24, 1981	Letter from applicant transmitting Amendment 26 to the FSAR.
August 24, 1981	Letter from applicant concerning Upgradeed Emergency Response Facilities.
August 24, 1981	Letter to applicant concerning long term operability of deep draft pumps.
August 24, 1981	Letter from applicant concerning Emergency Plan discrepancies.
August 28, 1981	Letter from applicant transmitting response to fire protection open items.

September 2-4, 1981	Representatives from NRC and TUGCO meet in Arlington and Dallas, Texas, and at Comanche Peak to perform an audit on the proposed organization for operation of Comanche Peak.
September 4, 1981	Letter from applicant advising that a revised estimated completion date for Unit 1 will be determined within the next several months. The completion of Unit 1 prior to June 1982 is unlikely.
September 11, 1981	Letter to applicant concerning clarification on the discussion held on July 30, 1981 concerning requirements for a safety parameter display system.
September 14, 1981	Letter from applicant concerning reactor vessel upper head temperature.
September 15, 1981	Letter from applicant transmitting responses to informal NRC questions on ECCS switchover.
September 16, 1981	Letter from applicant transmitting additional information concerning request for amendment to the construction permits.
September 21, 1981	Letter from applicant responding to Generic Letter 81-21 concerning procedural and training requirements to avoid voiding during natural circulation cooldown.
September 24, 1981	Letter to applicant transmitting two copies of the FES. An additional 20 copies will be forwarded when they have been delivered by the NRC printer-contractor.
September 24, 1981	Letter from applicant transmitting Revision 2 to the Physical Security Plan.
September 29, 1981	Letter to applicant concerning a revised schedule for completion of TMI Action Plan Item II.D.1. (Generic Letter 81-36).
September 29, 1981	Letter from applicant documenting items discussed during the September 2-4, 1981 NRC audit team visit.
September 30, 1981	Letter to applicant transmitting Amendment No. 4 to Construction Permits CPPR-126 and CPPR-127 to transfer ownership shares and to add a co-owner, Tex-La.
October 1, 1981	Letter from applicant transmitting confirmatory analysis of seismic Category I buildings.
October 2, 1981	Letter to applicant transmitting 20 copies of the FES.
October 2, 1981	Letter from applicant transmitting Amendment 27 to the FSAR.
October 5, 1981	Letter from applicant transmitting response to question on future fuel cycles.

October 5, 1981	Letter from applicant concerning LP turbine disks.
October 6, 1981	Letter from applicant concerning functional capability of ASME Code Class 2 and 3 stainless steel elbows.
October 6, 1981	Letter to applicant requesting an update of the Comanche Peak Application for Operating License.
October 7, 1981	Letter to applicant requesting additional information - Quality Assurance Branch and Core Performance Branch.
October 8, 1981	Letter form applicant concerning control of heavy loads.

APPENDIX B

BIBLIOGRAPHY

In addition to the references cited at the end of specific chapters, the documents listed below were used in the preparation of this report. Those marked with an asterisk are available for inspection and copying for a fee in the NRC Public Document Room, 1717 H Street, NW, Washington, DC 20555. Those marked with two asterisks are available for purchase from the NRC/GPO Sales Program, Washington, DC 20555, and/ or the National Technical Information Service, Springfield, VA 22161. Except as specifically noted, all other documents are available through public technical libraries.

(1) <u>Code of Federal Regulations</u>, Title 10, "Energy," including the General Design Criteria in Appendix A to 10 CFR Part 50.

(2) "Final Safety Analysis Report for Comanche Peak Steam Electric Station Units 1 and 2," including Amendments through 27* and other letters and documents filed by the applicant in support of this application.

(3) Westinghouse Electric Company Topical Reports*

(4) Industry Codes and Standards, including those of the following organizations:

American National Standards Institute
American Nuclear Society
American Society of Testing Materials

(5) U.S. Nuclear Regulatory Commission Regulatory Guides, including the following:**

1.33 "Quality Assurance Program Requirements (Operations)."

1.75 "Physical Independence of Electric Systems."

1.101 "Emergency Planning for Nuclear Power Plants."

1.115 "Protection Against Low-Trajectory Turbine Missile."

1.127 "Inspection of Water-Control Structures Associated with Nuclear Power Plants."

(6) U.S. Nuclear Regulatory Commission Reports, including the following:

NUREG-75/087 (now published as NUREG-0800) "Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants -- LWR Edition," December 1975 (includes Branch Technical Position papers).

NUREG-75/111 "Guide and Checklist for the Development and Evaluation of State and Local Government Radiological Emergency Response Plans," October 1975.

NUREG-0612	"Control of Heavy Loads at Nuclear Power Plants, Resolution of Generic Technical Activity A-36," July 1980.
NUREG-0737	"Clarification of TMI Action Plan Requirements," November 1980.
NUREG-0588	"Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," December 1979.
NUREG-0654	"Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," January 1980.
NUREG-0696	"Functional Criteria for Emergency Response Facilities," February 1981.

APPENDIX C

NUCLEAR REGULATORY COMMISSION (NRC)
UNRESOLVED SAFETY ISSUES

C.5 Discussion of Tasks as They Relate to Comanche Peak Units 1 and 2

A-47 Safety Implications of Control Systems:

The SER noted that the staff had requested additional information on this matter as described in SER Section 7.7.2. In Section 7.7.2 of this supplement, the staff noted that the applicant had submitted a response. The staff reviewed the bases for the applicant's response and concluded that, with reasonable assurance, the consequences of single failures within the control systems are bounded by analyses in Chapter 15 of the FSAR. On that basis, the staff finds the applicant's study acceptable and its concerns on Comanche Peak resolved.

APPENDIX D

LIST OF PRINCIPAL CONTRIBUTORS

The following staff members of the NRC were principal contributors for this supplement.

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APPENDIX D (Continued)

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APPENDIX E

ERRATA TO COMANCHE PEAK SAFETY EVALUATION REPORT

Page 1-7, Line 35:	Change "Preservice and inservice" to "Inservice"
Page 1-7, Line 36:	Change "(Sections 3.9.2.1 and 3.9.6)" to "(Section 3.9.6)"
Page 1-8, Line 5 & 6:	Change "(Sections 4.2.1 and 4.2.5)" to "(Sections 4.2.1, 4.2.5 and 4.3.2.3)"
Page 1-8, Lines 16 & 17:	Change "(Sections 5.3.1.2 and 5.3.3)" to "(Sections 5.3.1.2, 5.3.2 and 5.3.3)"
Page 1-8, Line 28:	Change "Environmental qualification of control systems for" to "Consequences of control system failures from"
Page 1-10, Line 1:	Change "on" to "of"
Page 1-10, Line 10:	Change "licensee preparedness" to "licensee emergency preparedness"
Page 1-10, Lines 17 & 18:	Delete: "III.D.3.3 Improved inplant iodine instrumentation under accident conditions"
Page 2-39, Lines 10 & 11:	Delete: ", except as identified in Section 2.5.6.5 and 2.5.6.6 above"
Page 2-39, Lines 40 & 41:	Delete: "This conclusion is subject to satisfactory resolution of the outstanding issues discussed above."
Page 6-16, Line 16:	Change: "provded" to "provided"
Page 7-14, Line 30:	Change: "increase" to "decrease"
Page 7-23, Lines 19 & 20:	Change: "If transfers are made only to take direct control over a given piece of equipment, this would place the equipment in a mode of operation to ensure" to "But transfers are made only to take direct control over a given piece of equipment, that is to place the equipment in a mode of operation to effect"
Page 22-35, Line 35:	Change "descritpion" to "description"
Page 23-1, Line 22:	Change "technically qualified" to "technically and financially qualified"
Page A-23, Line 11:	Change "the FSAR." to "the FSAR and Revision 1 to the Emergency Plan."

Page A-26, Line 10: After this line, insert:

June 3, 1981 Letter from applicant responding to request for information made at May 28, 1981 meeting on conduct of operations and corporate management.

Page A-27, Line 22: Change "exhuast" to "exhaust"

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NRC FORM 335 (7-77) U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET**	1. REPORT NUMBER (Assigned by DDC) NUREG-0797 Supplement No. 1
4. TITLE AND SUBTITLE (Add Volume No., if appropriate) Safety Evaluation Report related to the operation of Comanche Peak Steam Electric Station, Units 1 and 2	2. (Leave blank)
	3. RECIPIENT'S ACCESSION NO.
7. AUTHOR(S)	5. DATE REPORT COMPLETED MONTH October YEAR 1981
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) U. S. Nuclear Regulatory Commission Office of Nuclear Reactor Regulation Washington, DC 20555	DATE REPORT ISSUED MONTH October YEAR 1981
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	11. CONTRACT NO.
13. TYPE OF REPORT Technical Report - Safety Evaluation Report	PERIOD COVERED (Inclusive dates) July 1981 - October 15, 1981
15. SUPPLEMENTARY NOTES Docket Nos. 50-445 and 50-446	14. (Leave blank)

16. ABSTRACT (200 words or less)

Supplement No. 1 to the Safety Evaluation Report related to the operation of the Comanche Peak Steam Electric Station, Units 1 and 2, has been prepared by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. The facility is located in Somervell County, Texas. Subject to favorable resolution of the items identified in this supplement, the staff concludes that the facility can be operated by the applicant without endangering the health and safety of the public. This document provides the NRC staff's evaluation of the outstanding and confirmatory issues that have been resolved, and addresses changes to the SER which have resulted from the receipt of additional information from the applicant.

17. KEY WORDS AND DOCUMENT ANALYSIS

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17b. IDENTIFIERS/OPEN-ENDED TERMS

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SER RELATED TO THE OPERATION OF
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OCTOBER 1981