NUREG-0797
Supplement No. 2

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

January 1982

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ABSTRACT

Supplement No. 2 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 since Supplement No. 1 was issued in October 1981, and addresses changes to the SER and Supplement No. 1 which have resulted from the receipt of additional information from the applicant. This document also addresses those items that are identified in the Advisory Committee on Reactor Safeguards letter dated November 17, 1981.

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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 operating licenses 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.

In October 1981, Supplement No. 1 to the SER was issued providing the NRC staff's evaluation of the outstanding and confirmatory issues that had been resolved to that date.

At its 259th meeting on November 13, 1981, the Advisory Committee on Reactor Safeguards completed its review of the application. The Committee in its November 17, 1981 letter to Chairman Palladino of the NRC concluded that if due consideration is given to the recommendations mentioned in its letter, and subject to the satisfactory completion of construction, staffing and preoperational testing, there is reasonable assurance that the Comanche Peak Steam Electric Station Units 1 and 2 can be operated at power levels up to 3425 MWt without undue risk to the health and safety of the public.

This document is SER Supplement No. 2 (this supplement or SSER #2). The purpose of this supplement to the SER is to provide the staff's evaluation of the outstanding and confirmatory issues that have been resolved since Supplement No. 1 (SSER #1) was issued in October 1981, to address changes to the SER and SSER #1 which have resulted from the receipt of additional information from the applicant, and to address those recommendations that are contained in the Advisory Committee on Reactor Safeguards letter of November 17, 1981. That letter is included as Appendix F to this supplement. The staff's report on the recommendations in the Committee's letter is given in Section 18 of this supplement.

By Amendments 28 and 29 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 listed in SER Supplement No. 1.

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 Conditions. 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.

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

Copies of this supplement are available for inspection at the NRC Public Document Room, 1717 H Street, NW., 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.

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1.7 Summary of Outstanding Issues

As a result of the staff's 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 SSER #1 and this supplement. The outstanding issues remaining in the staff operating license review are listed below, with the appropriate section numbers in the SER, SSER #1 or this supplement noted that describe the issue. The staff will complete its review of these items before an 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 #1 Section 3.10)

(3) Environmental qualification of safety-related electrical equipment (SER Section 3.11 and SSER #1 Section 3.11)

(4) Preservice and inservice inspection program for compliance with 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) Fire protection program

   (a) Alternate safe shutdown system (SER Sections 7.4.2, 8.4.6 and 9.5.1.5, and SSER #1 Section 9.5.1.8)

   (b) Conformance with Appendix R (SER Sections 9.5.1 and 9.5.1.7, and SSER #1 Sections 9.5.1.7 and 9.5.1.8)

(8) TMI Action Plan (SER Section 22 and SSER #1 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.F.1	Additional accident monitoring instrumentation, attachments 1, 2, and 3 (SER)
II.F.2	Instrumentation for detection of inadequate core cooling (SER and SSER #2 Section 18 Item 7)
III.A.1.2	Upgrading emergency support facilities (SER and SSER #1)
III.A.2	Improving licensee emergency preparedness, long term (SER and SSER #1)
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's review, there are a few items which essentially have 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, SSER #1 and SSER #2 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 asymmetric 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 #1 Section 3.9.3.1)

(5) Periodic leak testing of pressure isolation valves (SSER #1 Section 3.9.6)

(6) Staff review of PAD-3.3 computer program incomplete (SER Sections 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 #1 Sections 5.3.1.2, 5.3.1.3, 5.3.2, and 5.3.3)

(8) Control and protection system interaction for feedwater line isolation actuation (SSER #2 Section 7.3.1.5)

(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 #1 Section 9.1.4)

(12) Documentation of applicant's commitments on fire protection (SSER #1 Section 9.5)

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

(14) TMI Action Plan (SER Section 22 and SSER #1 Section 22)

I.C.2 Shift and relief turnover procedures (SER and SSER #1)

I.C.5 Procedures for feedback of operating experience to plant staff (SER and SSER #1)

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 #1)

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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 #1)

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II.K.3.1 Installation and testing of automatic PORV isolation system (SER and SSER #1)

II.K.3.2 Report on overall safety effect of PORV isolation system (SER and SSER #1)

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 #1)

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 its supplements, 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 #1 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 #1 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 for operating shifts, these augmentation personnel must remain on shift until the plant attains a 100 percent power level (SER Section 13.1.2.1, SSER #1 Section 13.1.2.1, and SSER #2 Section 18 Item 2).

(7) TMI Action Plan

II.B.3 Postaccident sampling capability (SER and SSER #2 Section 22).

7 INSTRUMENTATION AND CONTROLS

7.3 Engineered-Safety-Features Actuation System

7.3.1 Description

7.3.1.5 Feedwater Line Isolation Actuation

As a result of the staff's review of current operating license applications for Westinghouse plants, a concern about control and protection systems interaction has been identified which was not addressed in the earlier review. The concern relates to steam generator level measurement channels which are used for both protection and control. The protection system initiates feedwater isolation when 2 of 3 channels indicate high level. Since one of these level channels is used for control, the failure of this channel would result in steam generator overfill by action of the level control system. As a consequence of this failure, the two remaining channels must operate to provide the required protective action. This does not satisfy the requirements for control and protection system interaction in that the remaining channels of the protection system must satisfy the single failure criterion. Therefore, the staff has requested that the applicant respond to this concern and the staff will address its resolution in a future supplement to the SER.

7.3.2 Resolution of Concerns Related to ESFAS

7.3.2.1 Loss of Safety Function After Reset

In the SER, the staff indicated that the logic to initiate containment ventilation isolation uses a radiation high-level signal and an ESFAS signal input to a retentive memory with actuation block gate. Thus a reset of a high-radiation signal would block the ESFAS input signal. In Amendment 29 of the FSAR, the applicant provided a design change to address this issue. The high-radiation signal was changed to a pulsed signal. If the gate was set because of a high-radiation signal, resetting the gate after the pulse would not invoke the actuation block feature. and thus it would initiate protective action on a subsequent ESFAS signal. The staff finds that the applicant's modification is acceptable, and therefore, this issue is resolved.

7.7 Control Systems Not Required for Safety

7.7.2 Conclusions

In the SER it was noted that the staff had requested a review by the applicant to determine whether the harsh environments associated with high-energy line breaks might cause control system malfunctions and result in consequences more severe than those calculated in the Chapter 15 analyses or beyond the capability of operators or safety systems.

The applicant has performed a review to evaluate the impact of control system failures as a consequence of harsh environments due to high-energy line breaks. Based on this review, the applicant concludes that the following failures could occur and that the consequences of these failures are not bounded by the safety analysis provided in Chapter 15 of the FSAR.

(1) Failure of the power range neutron detectors resulting in rod withdrawal.

(2) Failure of pressure transmitters, wiring or solenoid valves for the control of pressurizer pressure causing the PORVs to open.

(3) Failure of steam generator level and steam flow transmitters resulting in reduced steam generator inventory.

(4) Failure of steam generator pressure transmitters causing the steam generator PORVs to open.

(5) Failure of steam generator PORV electro/pneumatic transducers causing the PORVs to open.

The applicant has provided a commitment to relocate the steam generator electro/pneumatic transducers so that they will not be subjected to a harsh post-accident environment. The remaining items noted above will be environmentally qualified to preclude their failure as a consequence to high energy line breaks.

Based on the staff's review of the applicant's analysis of consequential failure of control system components due to high-energy line breaks and the applicant's commitment to take appropriate action to preclude unacceptable failures, the staff considers this matter resolved.

15 ACCIDENT ANALYSIS

15.4 Radiological Consequences of Design-Basis Accidents

15.4.8 Fuel Handling Accident

In the SER (Section 9.4.2), the staff noted that the fuel handling building ventilation system is normally aligned to the ESF units during fuel handling operations. However, a single failure of the dampers in the exhaust system could isolate the fuel building ventilation from the ESF exhaust filtration units or divert the flow to the normal exhaust units. In either case, an unfiltered release of radioactivity could result during a fuel handling accident.

The filtration system in the fuel storage facility is not fully ESF grade; however, the applicant has shown that with proper (manual or automatic) operation of the exhaust dampers prior to fuel handling operations, ESF-grade filtration can be achieved.

In the SER (Section 15.4.8), the radiological consequences of the fuel handling accident were evaluated based upon the assumption that the fuel building exhaust was filtered by the ESF filter units. The staff has reanalyzed the fuel handling accident using the more conservative assumption that the ESF filtration units are not available during the course of the accident. In the reevaluation of the fuel handling accident, the assumptions used by the staff are based on Regulatory Guide 1.25 and are substantially the same as those used by the applicant. In the staff's dose calculations, the same input values were used for the two areas (inside containment and in the fuel handling building) where the fuel handling accident could occur. The staff's model of the accident assumes that a fuel assembly was dropped in the fuel pool during refueling operations and that all of the fuel rods in the assembly were damaged, releasing radioactive materials contained within the fuel rod gaps into the pool. The radioactive material that escaped from the pool was assumed to be released over a two-hour period to the environment without filtration. The list of assumptions and parameters used in the analysis are given in Table 15.8. The offsite doses are shown in Table 15.9.

In summary, even without taking credit for filtration, the potential doses from the postulated fuel handling accidents are well within the guideline values given in 10 CFR Part 100. Therefore, the staff concludes that the system design is acceptable in mitigating the consequences of fuel handling accidents.

Table 15.8 Fuel handling accident dose assumptions

Element	Value
Power level, MWt	3565
Peaking factor	1.65
Number of rods damaged	264
Number of rods in core	50,913
Shutdown time, hr	100
Relative concentration values, sec/m^3	
0-2 hr at the exclusion area boundary	1.5×10^{-4}
0-8 hr at the low population zone boundary	2.3×10^{-5}
8-24 hr at the low population zone boundary	1.5×10^{-5}
24-96 hr at the low population zone boundary	6.0×10^{-6}
96-720 hr at the low population zone boundary	1.6×10^{-6}

Table 15.9 Radiological consequences of fuel handling accident

Accident and exposure time	Exclusion area boundary[1] dose (rem)		Low-population zone boundary[2,3] dose (rem)	
	Thyroid	Whole body	Thyroid	Whole body
Fuel handling accident				
0-2 hr	42	2.8		
0-8 hr			6.5	1

[1]Exclusion area distance: 2206m

[2]Low-population zone distance: 6640m

[3]The LPZ doses after 8 hr were determined to be negligible.

18 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

On June 29, 1981, a subcommittee of the Advisory Committee on Reactor Safeguards (Committee or ACRS) met with representatives of the applicant and the NRC staff to consider the applicant's application for a license to operate the Comanche Peak Steam Electric Station, Units 1 and 2. The meeting was held in the Dallas/Fort Worth area following a tour of the facility by the subcommittee members on the morning of that same date. An additional subcommittee meeting was held in Washington, D.C. on November 11, 1981. On November 13, 1981, at its 259th meeting, the full Committee met with representatives of the applicant and NRC staff to consider the application. The Committee identified a number of items that it recommended should be considered by the applicant and the NRC staff, and stated that if due consideration is given to these recommendations and subject to satisfactory completion of construction, staffing and preoperational testing, there is reasonable assurance that Comanche Peak Steam Electric Station, Units 1 and 2, can be operated at power levels up to 3425 MWt without undue risk to the health and safety of the public.

The Committee's letter from J. Carson Mark to Nunzio J. Palladino dated November 17, 1981 is included as Appendix F to this supplement.

The purpose of this section is to respond to the items identified in the Committee's November 17, 1981 letter. In the following discussion, each item identified by the Committee is set forth, followed by the NRC staff's report on each such recommendation.

Item 1

The Reactor Protection System will use N-16 gamma radiation detectors to provide a signal for reactor trip. Because this system has not been proven in commercial applications, we recommend that the NRC staff closely follow its implementation and operation. The Committee wishes to be kept informed.

Response

The November 17, 1981 letter from the ACRS, reporting on the Comanche Peak reactors, requested that the NRC staff closely follow the implementation and operation of the N-16 gamma detection systems which will provide input to the reactor protection system. In response to this request, the NRC staff will review the applicant's program for the initial testing of the instrument, review the results of this testing, require that the applicant notify the staff of significant problems that occur, and request that the applicant prepare an evaluation of the instrument's performance after some period of operation (e.g., at first refueling). The NRC staff will then prepare a summary report on the results of the surveillance for submission to the ACRS.

Item 2

This is the first commercial nuclear power plant to be operated by TUGCO and the first in the State of Texas. The Committee's review included consideration of the management organization and capability and the operator training program. The training program is well planned and comprehensive, and includes simulator training at other facilities. We were favorably impressed with the training program, general competence, and responsive attitude of the utility's operating organization. Nevertheless, there is a significant lack of hands-on experience with large commercial nuclear power plants that will only be corrected by the operation of the Comanche Peak Plant. The NRC staff is requiring the utility to strengthen its own organization with on-shift personnel having experience with large commercial PWR operations until suitable experience has been developed by the operating staff. We endorse the NRC staff requirement but recommend that attainment of 100% rated power should not be the only consideration in determining that operational proficiency has been achieved.

Response

In Section 13.1.2.1 of Supplement No. 1 to the Safety Evaluation Report, the NRC staff stated that the applicant has committed that at least one member of each operating shift crew would have previous experience (at least one year) as a licensed operator at a commercial pressurized water reactor nuclear power plant. The NRC staff noted its acceptance of this commitment with the proviso that, if augmentation personnel were used to meet the commitment, they should stay on shift until the plant attained a 100% power level.

In its letter of November 17, 1981, reporting on the Comanche Peak application, the ACRS endorsed the requirement to have operating experience available on shift, but recommended that attainment of 100% power should not be the only consideration in determining that operational proficiency (by the operating staff) has been achieved. By letter to H. Denton dated December 18, 1981, the applicant stated its intention to keep the extra experienced personnel on shift as long as it feels they enhance the plant operating capability.

The NRC staff agrees that attainment of 100% power should not be the sole criterion for release of experienced personnel from shift duties. However, by the time the plant has reached the 100% power level, generally for PWRs about a year from startup, the plant operators will have gained considerable experience taking the plant through the various start-up evolutions. Only limited additional experience is gained during steady-state operation at power. The staff anticipates that the plant operating staff should attain sufficient operating proficiency to continue safe plant operation without the assistance of augmentation personnel by the time the 100% power level is reached. However, should this not be the case, the staff will assure that the applicant retains the augmentation personnel on shift until the NRC staff believes that the operating staff is sufficiently proficient.

Item 3

The Committee also recommends that the operating organization establish a list of technological matters which may have to be faced in future operation of the nuclear plant and identify sources of skilled personnel and expertise that ought to be available to address these matters when needed. The Committee wishes to be kept informed.

Response

In its letter of December 18, 1981, the applicant stated that it has compiled lists of the types of technical expertise that will be needed for plant support, that it is continuing to update these lists, and that it is engaged in an effort to ensure that the necessary technical resources will be available to meet these needs. The applicant stated that it will keep the NRC staff informed regarding these matters. The NRC staff, in turn, will keep the ACRS informed.

Item 4

The Station Operations Review Committee, the Independent Safety Engineering Group, and the Operations Review Group should include personnel from outside the operating organization who are experienced in the operational management of large PWRs and related technology as well as other independent advisors with mature judgment about public safety matters.

Response

The applicant responded to this recommendation in its letter of December 18, 1981. The applicant noted that the Station Operations Review Committee (SORC) is composed of members of the station operations organization under the management and direction of the Manager, Plant Operations. The SORC membership includes persons, such as the Operations Quality Assurance Supervisor and the Radiation Protection Engineer, who do not have any direct responsibility for plant operations and who thus provide for a significant degree of independence from the pressures of station operation. The applicant stated its opinion that having persons who are not members of the operating organization serve as regular members of the SORC would adversely affect the SORC's ability to meet on short notice and provide the required timely and continuing monitoring of operating activities.

The NRC staff notes that the SORC membership for Comanche Peak meets the guidelines of Section 4.4 of ANSI N18.7-1976/ANS 3.2 as endorsed by Regulatory Guide 1.33 for review activities of the onsite operating organization. On that basis, the NRC staff concludes the membership of the SORC is in accordance with Section 13.4 of the Standard Review Plan and, therefore, is acceptable.

As regards the membership of the Independent Safety Engineering Group (ISEG), the applicant states that this group will be staffed with degreed engineers with training and experience in various engineering fields. They will be employees of the applicant, but will not be members of the operating organization.

The NRC staff notes that the ISEG is a full-time group of engineers, located onsite, but reporting offsite outside the normal operational chain. In SSER #1, Section 13.4.3, the NRC staff found the applicant's commitments concerning the organization of the ISEG and the qualifications of the ISEG members to be acceptable. They are in accordance with Section 13.4 of the Standard Review Plan.

As regards the Operations Review Committee (ORC), the applicant stated in its December 18, 1981 letter that the majority of the members of this committee would be independent of the operating organization. At least one regular member of the ORC will not be an employee of the applicant. Other persons with specific expertise may be called upon from time to time to serve on the ORC.

The NRC staff concludes that the proposed membership of the ORC meets the requirements of Section 4.3 of ANSI N18.7-1976/ANS 3.2 and is in accordance with Section 13.4 of the Standard Review Plan and, therefore, is acceptable.

Item 5

TUGCO should expand its studies on systems interaction and probabilistic assessment so that it will have a better understanding of the Comanche Peak nuclear systems.

Response

The applicant responded to this recommendation in its letter of December 18, 1981, stating that the Comanche Peak staff is engaged in and will continue to expand its studies on systems interaction and probabilistic risk assessment.

The NRC staff notes that the matter of systems interaction in nuclear power plants was discussed as Unresolved Safety Issue A-17 in Appendix C of the SER. The SER described the earlier studies performed on this matter by Sandia Laboratories and more recent NRC staff efforts to develop systematic ways to conduct a separate analysis for adverse systems interactions. At this time, there are no regulatory requirements for such an analysis. The resolution of issue A-17 may lead to a requirement. To date, the applicant has not described a comprehensive program that separately analyzes all structures, systems, and components important to safety for the three categories of adverse systems interactions (spacially coupled, functionally coupled, and humanly coupled). Following the resolution of issue A-17, the NRC staff will determine whether the applicant must perform further evaluations for adverse systems interactions. In the interim, as discussed in Appendix C to the SER, the staff concludes that there is reasonable assurance that Comanche Peak can be operated before final resolution of this generic issue without endangering the health and safety of the public.

The NRC is making increasing use of Probabilistic Risk Analysis (PRA) skills in the regulation of nuclear power plant safety. The goal of the NRC in using PRA is to gain a better understanding of the unique safety aspects of these plants. The staff is considering the requirement of PRA studies for all operating nuclear plants on a phased schedule over the coming years. When regulatory requirements for a plant PRA are defined, Comanche Peak will be included within the requirement.

Item 6

Other issues have been identified as Outstanding Issues, License Conditions, and Confirmatory Issues in the Staff's Safety Evaluation Report supplement dated October 1981. The ACRS is satisfied with the progress on these topics and believes that they should be resolved in a manner satisfactory to the NRC staff.

Response

The applicant responded to this item in its letter of December 18, 1981, by stating that it is working with the staff to resolve all outstanding issues, and that the resolutions are in large part contingent upon completing several studies. The staff will require that all outstanding issues are satisfactorily resolved prior to the issuance of the operating license.

Item 7

TUGCO is evaluating potential methods of providing instrumentation for detection of inadequate core cooling as discussed in the ACRS letter to the Executive Director for Operations dated June 9, 1981. The Committee believes that this equipment should not be installed until it is well established that the instruments will provide reliable information of significant value beyond that provided by the instrumentation which is already installed.

Response

The applicant responded to this comment in its letter of December 18, 1981 by stating that it agrees with the above statement.

The NRC staff comments as follows on this matter. The ACRS, in its letter of November 17, 1981 to Chairman Palladino, "Report on Comanche Peak Steam Electric Station Units 1 and 2," stated as follows:

(1) the Texas Utilities Generating Company (TUGCO) is evaluating potential methods of providing instrumentation for detection of ICC as discussed in the ACRS letter to the Executive Director for Operations dated June 9, 1981;

(2) .ie Committee believes that this equipment should not be installed until it is well established that the instruments will provide reliable information of significant value beyond that provided by the instrumentation which is already installed.

With respect to the first item, in a memorandum for Chairman J. Carson Mark from William J. Dircks, Executive Director for Operations, dated July 10, 1981, the staff responded to the concerns expressed in the ACRS letter of June 9, 1981. The ACRS expressed the same concern in its review of the license application for the St. Lucie Plant, Unit 2. The staff has provided a more detailed response to the concerns of the ACRS in the staff's evaluation of the TMI Action Plan, Item II.F.2, Instrumentation for Detection of Inadequate Core Cooling, relating to the St. Lucie Plant, Unit 2; see Section 22 of the St. Lucie Unit 2 SSER No. 1, NUREG-0843, December 1981.

With respect to the second item, the staff believes that design and testing of instrumentation available from Westinghouse and Combustion Engineering has progressed to an extent that establishes that their proposed systems for detection of ICC, including level monitoring instrumentation, will provide reliable information of significant value. Therefore, an acceptable design and installation schedule for level monitoring instrumentation will be required for an operating license. The staff has not received additional information on this instrumentation since the SER was issued in July 1981. Therefore, the status of the staff review as presented in Item II.F.2, Section 22 of the SER remains unchanged.

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" statement on the resolution or status as it relates to Comanche Peak. This supplement provides only an addition to the sections entitled "Discussion and Conclusions."

II.B.2 Plant Shielding to Provide Access to Vital Areas for Post-Accident Operations

The applicant reviewed the shielding design for Comanche Peak to evaluate the ability to have access to vital areas necessary to operate essential systems required after a LOCA with significant core damage.

The systems analyzed as sources of radiation that would be designed to function after an accident included: reactor containment, the emergency core cooling system, the containment spray system, the sampling system, the residual heat removal system, the chemical and volume control system, and the gas waste processing system. Dose rate calculations were performed for the areas of these systems using the postulated post-accident assumptions of Regulatory Guide 1.4 and TID-14844 to develop source terms, and computer codes that calculate dose rates from these source terms and respective geometries.

Evaluation of the shielding included direct and scattered radiation from containment, from containment leakage radiation streaming through penetrations, and from piping and system components. Dose rate zone maps, as a function of time following an accident, were provided by detailed layout drawings for each of the areas of interest. Post-accident radiation levels in the Control Room Complex and the Technical Support Center, which are vital areas, are acceptable for continuous occupancy in accordance with NUREG-0737 and the requirements of GDC-19.

Limited access to some areas, such as the post-accident sample panel and the hot laboratory, may be required on an irregular basis. To meet the requirements of post-accident sampling, a new system is being added incorporating additional shielding and/or remote operation capability. Additionally, a shielded hot cell will be added to the hot laboratory where post-accident samples will be handled and analyzed. Limited access to perform surveillance and inspection will also be provided for emergency power supplies (diesel-generators and batteries) and instrument panels among other areas.

On the basis of its review of the applicant's radiation and shielding design review for vital area access and the applicant's commitment to incorporate additional shielding and/or remote operation capability, the NRC staff concludes that the applicant has provided for adequate access to vital areas in accordance with Item II.B.2 of NUREG-0737. The NRC staff considers this matter resolved.

II.B.3 Post-Accident Sampling Capability

In Amendment 27 to the FSAR, the applicant has committed to meet the requirements of item II.B.3 in NUREG-0737 by installing a post-accident sampling system which will become operational prior to reactor operation above 5% power level. The system consists of sample lines, mobile cask, transport cart, and one sample station for each unit. The post-accident sampling system will be capable of permitting plant operators to obtain and analyze samples within three hours from the time a decision is made to take a sample, without incurring radiation exposure to any individual exceeding the requirements of General Design Criterion 19 in Appendix A to 10 CFR Part 50, assuming the source terms given in Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors." However, the applicant has not submitted a summary of person-motion studies on shielding design to verify compliance with the radiation exposure requirements.

Full strength and diluted grab samples will be obtained from the reactor coolant, containment atmosphere, and containment sump. On-site radiological and chemical analysis capability will be available in the common plant hot laboratory to measure (1) radionuclides (noble gases, iodine and cesium species, and non-volatile isotopes) in the reactor coolant and containment atmosphere, that are indicators of the degree of reactor core damage; (2) hydrogen and oxygen levels in the containment atmosphere; and (3) dissolved hydrogen and oxygen, pH, boron, and chloride concentrations in liquids. The type of samples and ranges of measurement for gross radioactivity and chemical analyses are in accordance with the requirements of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," revision 2. However, the applicant has not submitted a summary of procedures for sample analyses, the accuracy and sensitivity of these analyses in an accident and radiation environment, or a procedure relating radionuclides to estimated reactor core damage.

The applicant indicated that chlorides can be analyzed within four days following an accident. Use of the post-accident sampling system does not require an isolated auxiliary system to be operational. In-line monitoring is not used for post-accident sampling.

Provisions are available for purging the sample lines to reduce plateout in the sample lines, to minimize sample loss or distortion, to prevent loss of sampling capability by loose materials in the reactor coolant system or containment, and to restrict flow to limit reactor coolant loss from sample line rupture by use of the external containment isolation valve. The containment atmosphere sample lines are heat traced in order to assure that samples reaching the sample panel will be representative. Gaseous sample lines are flushed back to the containment. Liquid sample lines are flushed to the reactor coolant drain tank inside the containment. The sample station has charcoal absorbers and high-efficient particulate air filters to collect airborne radioactivity within the sample panel.

The applicant has committed to meet the training criteria for plant operators set forth in letter dated March 28, 1980, by H. R. Denton on "Qualifications of Reactor Operators." However, the applicant has not indicated the frequency of training of plant operating personnel for post-accident sampling.

Based on the above evaluation, the NRC staff has determined that the proposed post-accident sampling system, when it is installed and becomes operational, will partially meet the requirements of items II.B.3 in NUREG-0737. However, in addition to the deficiencies noted above, the applicant has not provided sufficient information for the staff to complete its review on restriction of background radiation for sample analysis; analysis of dissolved gases in unpressurized samples; operability of the system assuming loss of off-site power; environmental qualification of isolation valves that are not accessible for repair after an accident; capability of analyzing dissolved oxygen at less than 0.1 ppm; and testing frequency and type of testing of the post-accident sampling system.

Implementation of all the requirements of item II.B.3 in NUREG-0737 is not necessary prior to low power operation, because only small quantities of radionuclide inventory will exist in the reactor coolant system, and therefore will not affect the health and safety of the public. Prior to exceeding 5% power operation, the applicant must demonstrate the capability to promptly obtain reactor coolant samples in the event of an accident in which there is core damage, consistent with the conditions stated below.

1.a. Provide a summary of person-motion studies on shielding requirements, to verify that post-accident sampling will not incur radiation exposure to any individual exceeding the requirements of General Design Criterion 19 in Appendix A to 10 CFR Part 50. The studies should include handling, transporting, and analyzing of both liquid and gaseous samples.

1.b. Describe the procedures for on site radiological and chemical analyses and provide the accuracy and sensitivity of these analyses in an accident chemistry and radiation environment, i.e., presence of large amounts of fission products and a high radiation field in the samples.

1.c. Verify that provisions are available to restrict background radiation levels such that the sample analyses will provide results with a range of accuracy within a factor of 2.

1.d. Verify that total dissolved gases or hydrogen in the reactor coolant can be measured in unpressurized samples.

2. Verify that all electrically powered components associated with post-accident sampling are capable of being supplied with power and operated, within thirty minutes of an accident in which there is core degradation, assuming loss of offsite power.

3. Verify that valves which are not accessible for repair after an accident are environmentally qualified for the conditions in which they must operate.

4. Provide a procedure for relating radionuclide gaseous and ionic species to estimate reactor core damage.

5. Provide a method for verifying that reactor coolant dissolved oxygen is at less than 0.1 ppm if reactor coolant chlorides are determined to be greater than 0.15 ppm.

6. Provide information on (a) testing frequency and type of testing to ensure long term operability of the post-accident sampling system and (b) operator training frequency for post-accident sampling.

In addition to the above licensing conditions, the NRC staff is conducting a generic review of accuracy and sensitivity for analytical procedures to be used for post-accident analysis. The NRC staff will require that the applicant submit data supporting the applicability of each selected analytical chemistry procedure, along with documentation demonstrating compliance with the licensing conditions four months prior to exceeding 5% power operation, but review and approval of these procedures will not be a condition for full power operation. In the event its generic review determines a specific procedure is unacceptable, the NRC staff will require the applicant to make modifications as determined by its generic review.

The operating license will be conditioned for the items stated above.

APPENDIX A

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

October 8, 1981	Letter from applicant concerning relief request for inservice testing of pumps and valves.
October 15, 1981	Letter from applicant concerning functional capability of ASME Code Class 2 and 3 stainless steel elbows.
October 16, 1981	Letter to applicant transmitting two copies of Supplement No. 1 to the Safety Evaluation Report. An additional twenty copies will be forwarded when they have been delivered by the NRC contractor-printer.
October 21, 1981	Letter to applicant concerning Appendix R of 10 CFR Part 50 - Fire Protection Rule.
October 21, 1981	Letter from applicant concerning NUREG-0612, Control of Heavy Loads.
October 22, 1981	Letter from applicant concerning containment ventilation isolation signal blocked by reset.
October 23, 1981	Letter to applicant transmitting 20 copies of Supplement No. 1 to the Safety Evaluation Report related to the operation of Comanche Peak.
October 26, 1981	Letter from applicant transmitting Amendment 28 to the Final Safety Analysis Report.
October 26, 1981	Letter from applicant transmitting a certificate of service for Amendment No. 28 to the Final Safety Analysis Report.
October 26, 1981	Letter from applicant concerning long term operability of deep draft pumps.
October 29, 1981	Letter from applicant concerning schedule for Comanche Peak.
November 2, 1981	Letter from applicant concerning commitment on low temperature overpressure control logic.
November 9, 1981	Letter from applicant concerning outstanding items response schedule.
November 10, 1981	Letter from applicant transmitting Revision 2 drawings to the Physical Security Plan.

November 10, 1981	Letter to applicant concerning storage of low-level radioactive wastes at power reactor sites (Generic Letter 81-38).
November 11, 1981	ACRS subcommittee meeting on Comanche Peak.
November 13, 1981	ACRS full committee meeting on Comanche Peak.
November 17, 1981	ACRS Report on Comanche Peak, Units 1 and 2, issued.
November 18, 1981	Letter to applicant transmitting the ACRS Report for Comanche Peak.
November 18, 1981	Letter from applicant confirming hand delivery of a test report "Fire Qualification Test of a Protective Envelope System."
November 25, 1981	Letter from applicant concerning confirmatory items response schedule.
December 1, 1981	Letter to applicant concerning Comanche Peak Tray Fire Barrier Evaluation.
December 2, 1981	Letter to applicant concerning response to ACRS Report.
December 3, 1981	Letter from applicant concerning Final Safety Analysis Report Section II.B.2; Plant Shielding.
December 7, 1981	Letter from applicant transmitting Southwest Research Institute Report, "Fire Qualification Test of a Protective Envelope System."
December 8, 1981	Meeting with applicant on transfer of containment spray from injection to recirculation mode. (Summary issued December 14, 1981)
December 16, 1981	Letter to applicant concerning qualifications of reactor operators-license examinations (Generic Letter 81-40).
December 18, 1981	Letter from applicant concerning response to ACRS letter.
December 18, 1981	Letter from applicant transmitting Amendment 29 to the Final Safety Analysis Report.
December 18, 1981	Letter from applicant transmitting a table providing the description of Amendment 29.
January 6, 1982	Letter from applicant concerning ECCS analysis for large break LOCA assuming no single failure.
January 7, 1982	Letter to applicant requesting additional information for the Instrumentation and Control Systems Branch concerning the steam generator level control.

January 11, 1982	Letter from applicant concerning response schedule for transfer of containment spray from injection to recirculation mode.
January 13, 1982	Letter to applicant providing guidance on resolution of preservice inspection program.
January 13, 1982	Letter from appicant concerning change to FSAR relating to offsite power sources.

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) Code of Federal Regulations, 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 29* and other letters and documents filed by the applicant in support of this application.

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

American National Standards Institute
American Nuclear Society

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

1.4 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors."

1.25 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

1.97 "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."

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

NUREG-0737 "Clarification of TMI Action Plan Requirements," November 1980.

NUREG-0843 "Safety Evaluation Report Related to the Operation (Supplement No. 1) of St. Lucie Plant, Unit No. 2," December 1981.

TID-14844 "Calculation of Distance Factors for Power and Test Reactor Sites," March 23, 1962.

APPENDIX D

LIST OF PRINCIPAL CONTRIBUTORS

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

NAME	TITLE	BRANCH
Block, Seymour	Sr. Health Physicist	Radiological Assessment
Brooks, Walter L.	Sr. Reactor Physicist	Core Performance
Coffman, Franklin D.	Section Leader. Systems Interaction	Reliability and Risk Assessment
Crocker, Lawrence P.	Sr. Nuc. Engr., Mgmt. Systs	Licensee Qualification
Dempsey, Kenneth C.	Nuclear Engr.	Accident Evaluation
Dunning, Thomas G.	Section Leader	Instrumentation and Control Systems
Huang, Tai L.	Systems Analyst	Core Performance
Li, Hulbert C.	Sr. Reactor Engr.	Instrumentation and Control Systems
Wing, James	Sr. Chemical Engr.	Chemical Engineering

APPENDIX E

ERRATA TO COMANCHE PEAK SAFETY EVALUATION REPORT AND SUPPLEMENT NO. 1

The following errata for the Comanche Peak SER are supplementary to the errata presented in Appendix E of the SSER #1.

Page 2-12, Line 18:	Change "(2206 ft)" to "(2206m)"
Page 8-4, Line 35:	Change "volage" to "voltage"
Page 12-4, Line 32	Change "ANISN, SDC" to "COSACS, G^3"
Page 15-22, Line 4:	Change "Leaking" to "Peaking"
Page 17-4, Line 8:	Change "Revision 1" to "Revision 2"
Page 17-4, Line 9:	Change "January 1977" to "February 1978"
Page 17-5, Line 2:	Change "1.44" to "1.146"
Page 17-6, Lines 12 and 13:	Change "Executive Vice-President and General Manager" to "Vice-President, Nuclear"
Page 20-3, Table 20.1, Line 3:	On the line entitled Capacity Factor, %: delete "1983 1984 1985 1986 1987 1988 1989"
Page 20-3, Table 20.1, Line 4:	On the line entitled Unit 1:, add above the respective columns "1983 1984 1985 1986 1987 1988 1989"
Page 20-3, Table 20.1, Line 8:	On the line entitled Unit 2:, add above the respective columns "1985 1986 1987 1988 1989 1990 1991"
Page 22-33, Line 2:	Change "is" to "if"
Page 22-79, Line 23:	Change "needeed" to "needed"
Page A-2, Line 29:	Change "Environmntal" to "Environmental"
Page C-12, Line 41:	Change "reagrding" to "regarding"
Page C-14, Line 43:	Change "amont" to "among"

The following errata apply to Supplement No. 1 to the Comanche Peak SER.

Page 1-1, Line 21:	Change "Conditons" to "Conditions"
Page 1-1, Line 34:	Change "NUREG-0775)" to "(NUREG-0775)"
Page 1-3, Line 15:	Change "remote" to "safe"
Page 1-4, Line 27:	Change "7.4.5" to "7.5.4"
Page 3-2, Line 2:	Change "104,000" to "24,636"
Page 3-2, Line 5:	Insert following "___ area of the" "components important to safety listed in FSAR Table 3.5-7. These components are located within the"
Page 3-2, Line 15:	Change "10^{-7}" to "10^{-6}"
Page 5-1, Lines 8 and 9:	Change "in its" to "inoperable and therefore depressurized. Consequently, the letdown line air-operated isolation valves may also be assumed to be in a"
Page 5-3, Line 2:	Change "evauation" to "evaluation"
Page 5-4, Line 36:	Change "Insevice" to "Inservice"
Page 13-3, Line 24:	Change "19" to "29"
Page 13-3, Line 43:	Change "nuclear plant" tc "engineering"
Page 22-3, Line 5:	Change "nuclear plant" to "engineering"
Page 22-4, Line 30:	Change "SSE-IN" to "SEE-IN"
Page 22-6, Line 44	Change "WCAP-9304" to WCAP-9804"
Page 22-7, Line 6	Change WCAP-9304" to WCAP-9804"
Page A-2, Line 30:	Change "Upgradeed" to "Upgraded"

APPENDIX F

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

November 17, 1981

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: REPORT ON COMANCHE PEAK STEAM ELECTRIC STATION UNITS 1 AND 2

Dear Dr. Palladino:

During its 259th meeting, November 12-14, 1981, the Advisory Committee on Reactor Safeguards reviewed the application of the Texas Utilities Generating Company (TUGCO), Dallas Power and Light Company, Texas Electric Service Company, Texas Power and Light Company, Texas Municipal Power Agency, Brazos Electric Power Cooperative, Inc. and Tex-La Electric Cooperative for a license to operate the Comanche Peak Steam Electric Station Units 1 and 2. The Units are to be operated by the Texas Utilities Generating Company. A Subcommittee meeting was held in the Dallas/Fort Worth area on June 29, 1981 to consider this project. A tour of the facility was made by Subcommittee members on June 29, 1981. An additional Subcommittee meeting was held in Washington, D.C. on November 11, 1981. During its review, the Committee had the benefit of discussions with representatives of the Applicant and the NRC Staff. The Committee also had the benefit of the documents listed. The Committee commented on the construction permit application for this station in its report dated October 18, 1974 to AEC Chairman Dixie Lee Ray.

The Comanche Peak Station is located in Somerville County in North Central Texas about 40 miles southwest of Fort Worth, Texas, the nearest city having a population in excess of 25,000 persons.

Each Comanche Peak Unit is equipped with a Westinghouse pressurized water reactor having a rated power level of 3425 MWt. Each unit is housed in a steel-lined, reinforced concrete, dry containment building with a design pressure of 50 psig.

The Reactor Protection System will use N-16 gamma radiation detectors to provide a signal for reactor trip. Because this system has not been proven in commercial applications, we recommend that the NRC Staff closely follow its implementation and operation. The Committee wishes to be kept informed.

This is the first commercial nuclear power plant to be operated by TUGCO and the first in the state of Texas. The Committee's review included consideration of the management organization and capability and the operator training program. The training program is well planned and comprehensive, and includes simulator training at other facilities. We were favorably impressed with the training program, general competence, and responsive attitude of the utility's operating organization. Nevertheless, there is a significant lack of hands-on experience with large commercial nuclear power plants that will only be corrected by the operation of the Comanche Peak Plant. The NRC Staff is requiring the utility to strengthen its own organization with on-shift personnel having experience with large commercial PWR operations until suitable experience has been developed by the operating staff. We endorse the NRC Staff requirement but recommend that attainment of 100% rated power should not be the only consideration in determining that operational proficiency has been achieved.

The Committee also recommends that the operating organization establish a list of technological matters which may have to be faced in future operation of the nuclear plant and identify sources of skilled personnel and expertise that ought to be available to address these matters when needed. The Committee wishes to be kept informed.

The Station Operations Review Committee, the Independent Safety Engineering Group, and the Operations Review Group should include personnel from outside the operating organization who are experienced in the operational management of large PWRs and related technology as well as other independent advisors with mature judgment about public safety matters.

TUGCO should expand its studies on systems interaction and probabilistic assessment so that it will have a better understanding of the Comanche Peak nuclear systems.

Other issues have been identified as Outstanding Issues, License Conditions, and Confirmatory Issues in the Staff's Safety Evaluation report supplement dated October 1981. The ACRS is satisfied with the progress on these topics and believes that they should be resolved in a manner satisfactory to the NRC Staff.

TUGCO is evaluating potential methods of providing instrumentation for detection of inadequate core cooling as discussed in the ACRS letter to the Executive Director for Operations dated June 9, 1981. The Committee believes that this equipment should not be installed until it is well established that the instruments will provide reliable information of significant value beyond that provided by the instrumentation which is already installed.

We believe that if due consideration is given to the recommendations above, and subject to satisfactory completion of construction, staffing and preoperational testing, there is reasonable assurance that Comanche Peak Steam

Electric Station Units 1 and 2 can be operated at power levels up to 3425 MWt without undue risk to the health and safety of the public.

Sincerely yours,,

J. Carson Mark

J. Carson Mark
Chairman

References:

1. "Final Safety Analysis Report for the Comanche Peak Steam Electric Station Units 1 and 2," including Amendments 1 through 23.
2. U.S. Nuclear Regulatory Commission "Safety Evaluation Report related to the Operation of Comanche Peak Steam Electric Station, Units 1 and 2," USNRC Report NUREG-0797, dated July 1981 and Supplement No. 1 dated October 1981.
3. Letter from Citizens for Fair Utility Regulation To S. Duraiswamy, ACRS, regarding the licensing of Comanche Peak, dated July 18, 1981

NRC FORM 335
(7-77)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by DDC)
NUREG-0797
Supplement No. 2

2. (Leave blank)

3. RECIPIENT'S ACCESSION NO.

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

5. DATE REPORT COMPLETED
MONTH: JANUARY | YEAR: 1982

DATE REPORT ISSUED
MONTH: JANUARY | YEAR: 1982

6. (Leave blank)

7. AUTHOR(S)

8. (Leave blank)

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

10. PROJECT/TASK/WORK UNIT NO.

11. CONTRACT NO.

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Same as 9. above

13. TYPE OF REPORT
Technical Report - Safety Evaluation

PERIOD COVERED (Inclusive dates)
November 1981 - January 1982

14. (Leave blank)

15. SUPPLEMENTARY NOTES
Docket Nos. 50-445 and 50-446

16. ABSTRACT (200 words or less)

Supplement No. 2 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 since Supplement No. 1 was issued in October 1981 and addresses changes to the SER and Supplement No. 1, which have resulted from the receipt of additional information from the applicant. This document also addresses those items that are identified in the Advisory Committee on Reactor Safeguards letter, dated November 17, 1981.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
Unlimited

19. SECURITY CLASS (This report)
UNCLASSIFIED

20. SECURITY CLASS (This page)
UNCLASSIFIED

21. NO. OF PAGES

22. PRICE
$

NRC FORM 335 (7-77)

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

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SER RELATED TO THE OPERATION OF
COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2

JANUARY 1982