Docket Nos. 50-445/446

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

MAY 17 1984

Dear Mr. Spence:

Subject: Transmittal of Proposed Supplement to Appendix C of the SER for Comanche Peak Steam Electric Station (Units 1 and 2)

Enclosed is an update to Appendix C of the Comanche Peak SER (NUREG-0797) regarding Unresolved Safety Issues (USI), which we propose to incorporate in the next SER supplement. The enclosed supplement provides the current status in the resolution of USI A-49 (Pressurized Thermal Shock) added to Appendix C by SER Supplement No. 3 issued in March 1983. It also provides a summary of the following USI's which have been resolved since the SER was issued in July 1981, and USI's which are no longer deemed applicable to Comanche Peak:

USI's Resolved:

A-1, Water Hammer
A-9, Anticipated Transients Without Scram
A-11, Reactor Vessel Materials Toughness
A-12, Fracture Toughness of PWR Steam Generator and Reactor Coolant Pump Supports

USI's No Longer Applicable:

A-46, Seismic Qualification of Equipment in Operating Plants
A-48, Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

Should you have any questions concerning the enclosed SER Appendix C supplement, please direct them to Mr. John J. Stefano of my staff.

Sincerely,

ORIGINAL SIGNED BY:

B. J. Youngblood, Chief
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Division of Licensing

Enclosure: As stated

cc: See next page

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COMANCHE PEAK

MAY 17 1984

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Supplement to Appendix C

A-49 PRESSURIZED THERMAL SHOCK

The issue of pressurized thermal shock (PTS) arises because in pressurized water reactors (PWRs) transients and accidents can occur that result in severe overcooling (thermal shock) of the reactor pressure vessel, concurrent with or followed by repressurization. In these PTS events, rapid cooling of the reactor vessel internal surface results in thermal stress with a maximum tensile stress at the inside surface of the vessel. The magnitude of the thermal stress depends on the temperature profile across the reactor vessel wall as a function of time. The effects of this thermal stress are compounded by pressure stresses.

Severe reactor system overcooling events simultaneous with or followed by pressurization of the reactor vessel (PTS events) can result from a variety of causes. These include system transients, some of which are initiated by instrumentation and control systems malfunctions (including stuck open valves in either the primary or secondary system), and postulated accidents such as small break loss-of-coolant accidents (LOCAs), main steam line breaks (MSLBs), and feedwater line breaks.

The PTS issue is a concern for PWRs only after the reactor vessel has lost its fracture toughness properties and is embrittled by neutron irradiation. The standards and regulatory requirements to which the Comanche Peak reactor vessel was designed and fabricated are described in Section 5.3 of the SER.

As long as the fracture resistance of the reactor vessel material is relatively high, overcooling events are not expected to cause vessel failure. However, the fracture resistance of reactor vessel materials decreases with exposure to fast neutrons during the life of a nuclear power plant. The rate of decrease is dependent on the metallurgical composition of the vessel walls and welds. If the fracture resistance of the vessel has been reduced sufficiently by neutron irradiation, severe overcooling events could cause propagation of small flaws that might exist near the inner surface. The assumed initial flaw might be enlarged into a crack through the vessel wall of sufficient extent to threaten vessel integrity and, therefore, core cooling capability.

For the reactor pressure vessel to fail and constitute a risk to public health and safety, a number of contributing factors must be present. These factors are (1) a reactor vessel flaw of sufficient size to initiate and propagate; (2) a level of irradiation (fluence) and material properties and composition sufficient to cause significant embrittlement (the exact fluence depends on materials present; i.e., high copper content causes embrittlement to occur more rapidly); (3) a severe overcooling transient with pressurization; and (4) the crack resulting from the propagation of initial cracks must be of such size and location that the vessel fails.

As a result of the evaluation of the PTS issue, the staff recommended to the Commission in SECY-82-465 (November 23, 1982) actions to prevent PTS events in operating reactors. The Commission accepted the staff recommendations and directed the staff to develop a Notice of Proposed Rulemaking that would establish an RT_{NDT} screening criterion (below which PTS risk is considered acceptable), require licensees to submit present and projected values of RT_{NDT}, require early analysis and implementation of such flux reduction programs as are reasonably practicable to avoid reaching the screening criterion and require plant-specific PTS safety analysis before plants are within three calendar years of reaching the screening criterion including analyses of proposed alternatives to minimize the PTS problem.

Such a proposed rule has been published for public comment (Federal Register, February 7, 1984) by the staff. We believe that the Comanche Peak plant could easily meet the requirements of the proposed rule.

On the basis of the above consideration, the staff concludes that the Comanche Peak facility can be operated before complete resolution of this issue and completion of the proposed rulemaking without undue risk to the health and safety of the public.

SUMMARY OF THE STATUS CHANGES OF THE
UNRESOLVED SAFETY ISSUES IDENTIFIED IN APPENDIX C FOR COMANCHE PEAK

Task	Status
A-1, Water Hammer	This issue has been resolved by issuance of NUREG-0927, Rev. 1, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants."
A-9, Anticipated Transients Without Scram	The technical findings for this issue have been published in NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," Vol. 4. A proposed rule based on this work plus additional analysis was published for comment. The comments received were addressed and a final rule was affirmed by the Commission in November 1983. However, there has been further discussion among the Commissioners regarding the specific quality assurance requirements for the ATWS mitigating equipment and therefore the final rule has not yet been published.
A-11, Reactor Vessel Materials Toughness	This issue has been resolved by issuance of NUREG-0744, "Resolution of the Task A-11, Reactor Vessel Materials Toughness Safety Issue," Vols. I and II, Revision 1.

SUMMARY OF THE STATUS CHANGES OF THE
UNRESOLVED SAFETY ISSUES IDENTIFIED IN APPENDIX C FOR COMANCHE PEAK

Task	Status
A-12, Fracture Toughness of PWR Steam Generator and Reactor Coolant Pump Supports	This issue has been resolved by issuance of NUREG-0577, Rev. 1, "Potential for Low Fracture Toughness and Lamellar Tearing in PWR Steam Generator and Reactor Coolant Pump Supports." A detailed risk analysis and value/impact study for this issue resulted in a conclusion that no backfit requirements to operating reactors or near term operating license applications were required. A proposed new Standard Review Review Plan Section 5.3.4 has been prepared for implementation on new Construction Permit (CP) and Preliminary Design Approval (PDA) applications only, after review and resolution of public comments and its issuance in final form.
A-46, Seismic Qualification of Equipment in Operating Plants	The scope of Task A-46 is limited to dealing with seismic qualification of equipment in operating plants. In addition, Comanche Peak was designed on the basis of current seismic design criteria, and commitments for seismic equipment qualification are in accordance with the latest codes and standards. Therefore, the issue related to Task A-46 is not applicable for Comancne Peak.

SUMMARY OF THE STATUS CHANGES OF THE
UNRESOLVED SAFETY ISSUES IDENTIFIED IN APPENDIX C FOR COMANCHE PEAK

Task	Status
A-48, Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	This issue is limited to plants with pressure suppression containments, i.e., ice condenser for PWR plant and Mark I, II, and III containments for BWR plants. The containment for Comanche Peak is a large dry containment. Therefore, this issue is not applicable to Comanche Peak.