NRC PDR

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

JUN 8 1979

Docket Nos. 50-445
and 50-446

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

Dear Mr. Gary:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2

Enclosed is a request for additional information which we require to complete our evaluation of your application for operating licenses for Comanche Peak Steam Electric Station, Units 1 and 2. This request for additional information is the result of our review by the Mechanical Engineering Branch of the information in your FSAR. At this time we have not completed the review of the information submitted in FSAR Amendment 5, and our continuing review may result in additional requests or NRC staff positions by the Mechanical Engineering Branch. Please amend your FSAR to include the information requested in the Enclosure.

Your schedule for responding to the enclosed request for additional information should be submitted within three weeks. Based on your schedule for response and our workload, we will determine any licensing review schedule adjustments and inform you of any significant changes.

Sincerely,

Robert L. Baer

Robert L. Baer, Chief
Light Water Reactors Branch No. 2
Division of Project Management

Enclosure:
Request for Additional
Information

ccs w/enclosure:
See next page

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ccs:
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Pittsburgh, Pennsylvania 15230

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ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION
COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2
TEXAS UTILITIES GENERATING COMPANY
DOCKET NOS. 50-445/50-446

110.0 MECHANICAL ENGINEERING BRANCH

112.25
(3.9.3)
(3.9.2)

Previous analyses for other nuclear plants have shown that certain reactor system components and their supports may be subjected to previously underestimated asymmetric loads under the conditions that result from the postulation of ruptures of the reactor coolant piping at various locations. It is therefore necessary to reassess the capability of these reactor system components to assure that the calculated dynamic asymmetric loads resulting from these postulated pipe ruptures wil be within the bounds necessary to provide high assurance that the reactor can be brought safely to a cold shutdown conditions. The reactor system components that require reassessment shall include:

a. Reactor pressure vessel

b. Core support and other reactor internals

c. Control rod drives

d. ECCS piping that is attached to the primary coolant piping.

e. Primary coolant piping

f. Reactor vessel, steam generator, pressurizer, and pump supports.

The following information should be included in the FSAR about the effects of postulated asymmetric LOCA loads on the above mentioned reactor system components and the various cavity structures.

1. Provide arrangement drawings of the reactor vessel support systems in sufficient detail to show the geometry of all principal elements and materials of construction.

2. If a plant-specific analysis will not be submitted for your plant, provide supporting information to demonstrate that the generic plant analysis under consideration adequately bounds the postulated accidents at your facility. Include a comparison of the geometric, structural, mechanical and thermal-hydraulic smilarities between your facility and the case analyzed. Discuss the effects of any differences.

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3. Consider all postulated breaks in the reactor coolant piping system, including the following locations:

    a. Reactor vessel hot and cold leg nozzle to piping terminal ends.

    b. Pump suction and discharge nozzles to piping terminal ends.

    c. Steam generator inlet and outlet nozzles to piping terminal ends. 1/

    Provide an assessment of the effects of asymmetric pressure differentials 2/ on the systems and components listed above in combination with all external loadings including safe shutdown earthquake loads and other faulted condition loads for the postulated breaks described above. This assessment may utilize the following mechanistic effects as applicable:

    a. limited displacement break areas

    b. fluid-structure interaction

    c. actual time-dependent forcing function

    d. reactor support stiffness

    e. break opening times.

4. If the results of the assessment in item 3. above indicates loads leading to inelastic action in these systems or displacement exceeding previous design limits, provide an evaluation of the inelastic behavior (including strain hardening) of the material used in the system design and the effect on the load transmitted to the backup structures to which these systems are attached.

1/ Postulated steam line breaks may control the design of certain steam generator supports and therefore must also be considered in support design.

2/ Blowdown jet forces at the location of the rupture (reaction forces), transient differential pressures in the annular region between the component and the wall, and transient differential pressures across the core barrel within the reactor vessel.

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5. For all analyses performed, include the method of analysis, the structural and hydraulic computer code employed, drawings of the models employed, and comparisons of the calculated to allowable stresses and strains or deflections with a basis for the allowable values.

6. Demonstrate that active components will peform their safety function when subjected to the combined loads resulting from the loss-of-coolant accident and the safe shutdown earthquake.

7. Demonstrate the functional capability of any essential piping when subjected to the combined loads resulting from the loss-of-coolant accident and the safe shutdown earthquake.

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