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TEXAS UTILITIES GENERATING COMPANY

SKYWAY TOWER • 400 NORTH OLIVE STREET, L.B. 81 • DALLAS, TEXAS 75201

May 29, 1984

BILLY R. CLEMENTS
VICE PRESIDENT, NUCLEAR OPERATIONS

Mr. Harold R. Denton
Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION
DOCKET NOS. 50-445 AND 50-446
DEFERRED PREOPERATIONAL TESTING
ITEM NOS. 2 AND 3

REF: Letter to Mr. Harold R. Denton from Mr. B. R. Clements
dated May 14, 1984

Dear Mr. Denton:

Per our commitment in the referenced letter, we are submitting for NRC staff review and concurrence a description and summary evaluation of two tests proposed for deferment.

The tests proposed for deferment concern the preoperational testing of Safety Injection System Check Valve Leakage and Containment Cooling Systems. As noted in the attachment, our evaluation indicates that deferral of these items does not constitute an unreviewed safety question and does not involve change to the Technical Specifications. We request your concurrence with our proposal to defer these tests until fuel load but prior to initial criticality.

If you have any questions concerning this request, please contact me to arrange a meeting with the appropriate members of my staff.

Sincerely

Billy R Clements

BRC/grr

B001
1/1

Deferred Preoperational Testing of Containment Cooling Systems

Description and Scope

During the conduct of Preoperational Test, 1CP-PT-45-06, "Containment and Pump Room Cooler Performance Test," several test deficiencies were identified involving the failure of the Containment Cooling System to maintain sufficient cooling to various areas inside of Containment. The following is a list of the affected areas:

1. Reactor Vessel Supports and Neutron Detector Well Area
2. Pressurizer Compartments
3. Steam Generator Compartments 1, 2, 3 and 4 areas

Modifications are currently in progress which should assure that the Containment cooling deficiencies at Comanche Peak are corrected. The post fuel load testing required to close out this preoperational test item include:

1. Performance of area temperature surveillances as required by current CPSES Technical Specification 4.7.13 and 4.6.1.5.

2. Performance of a local temperature survey of the following areas:
   a. Reactor Vessel Support and Detector Wells
   b. Pressurizer Compartment
   c. Steam Generator Compartment 1, 2, 3 and 4.

It is our plan to conduct the above testing after fuel load when the next plant heatup is expected to occur. The majority of this testing would be performed as part of the Station Surveillance Test Program utilizing Procedure OPT-102, "Operations Shiftly Routine Tests" as required by 4.6.1.5 and 4.7.13 of the present CPSES Technical Specifications. The remainder of the testing has been incorporated into appropriate initial startup test program procedures. These test and surveillance procedures have all been reviewed and approved by our Station Operations Review Committee.

Summary Safety Evaluation:

A review of this deferred item was conducted per 10CFR 50.59. This review was performed to determine if deferral of this preoperational testing would constitute an unreviewed safety question or require a change to CPSES Technical Specifications. Qualitative evaluation of the appropriate chapter 15 events in the CPSES Safety Analysis Report provided the bases for the conclusion that no technical specification exceptions are required and no unreviewed safety questions exist due to this preoperational test deferral.

The Neutron Detector Well and Reactor Vessel Support Cooling System is designed to prevent the neutron detectors from exceeding their temperature limitations which are 135^{o}F with possible excursions to 175^{o}F. It also supplies cooling for reactor shield wall concrete and nozzle supports. No credit is taken for this system in the design basis accident.

The steam generator compartments and pressurizer compartments are cooled by the containment recirculation and cooling system. The system is designed to maintain the containment ambient temperature below 120°F during normal operation. The 120°F limitation ensures that the environmental conditions assumed for qualification of safety related equipment are maintained. No credit is taken for this system in the design basis accident.

Technical Specification 3.6.1.5 requires that the primary containment average air temperature not be allowed to exceed 120°F in modes 1, 2, 3 and 4. The surveillance testing of requirement 4.6.1.5 will ensure that the temperature criteria is met during plant operation after fuel load.

Technical Specification 3.7.13 requires that the Containment Detector Well area temperature be maintained below 135°F whenever equipment in the effected area is required to be operable. Since the source range nuclear instruments are required for fuel loading, mode 6 operability is required. The surveillance testing of requirement 4.7.13 will ensure that temperature criteria are met for all applicable modes of operation.

Since no adverse effects are associated with the deferral of this item, this activity is submitted and recommended for deferral until after fuel load of Unit 1.

Deferred Preoperational Testing of
Safety Injection System Check Valve Leakage

Description and Scope

During the conduct of Preoperational Test, 1CP-PT-57-09, "Check Valve and Hot Functional Safety Injection", a number of check valves were determined to leak in excess of their acceptance criteria. The following is a list of the subject valves:

1-8949A
1-8956A,B,D
1SI-8948A,B,C,D
1SI-8905B

Each of the above valves have subsequently been either repaired or replaced. In order to close out this preoperational test, the subject valves must be leakage checked at Reactor Coolant System operating temperature and pressure.

It is our plan to conduct the above testing after fuel load but prior to initial criticality when the next plant heatup is expected to occur. This testing is to be performed during Mode 3 prior to entering Mode 2 as required by section 3/4.3.4.6.2 of the present CPSES Technical Specifications. This testing will be performed as part of the Station Surveillance Test Program utilizing procedure EGT-712A, "Reactor Coolant System Pressure Boundary Isolation Valve Leakage Testing". This test procedure has been reviewed and approved by our Station Operations Review Committee.

Summary Safety Evaluation

A review of this deferred item was conducted per 10CFR50.59. This review was performed to determine if deferral of this preoperational testing would constitute an unreviewed safety question or require change to the CPSES Technical Specifications. Qualitative evaluation of the appropriate Chapter 15 events of the CPSES Safety Analysis Reports provided the bases for the conclusion that no technical specification exceptions are required and no unreviewed safety questions exist due to this preoperational test deferral.

The check valves in question are primary pressure boundary isolation valves. Their function is to isolate the relatively lower pressure portions of the ECCS systems from the higher pressure primary system. This precludes the potential for an intersystem loss of coolant accident which effectively bypasses the containment boundary. For this function, these valves are passive components and are not considered to mitigate any of the analyzed events in the CPSES safety analysis.

Technical Specification 3.4.6.2f requires the measured leakage from any Reactor Coolant System (RCS) Pressure Isolation Valve be less than or equal to 1 gpm at an RCS pressure of 2235 $\pm$ 20 psig. Surveillance Requirement 4.4.6.2.2 waives Specification 4.0.4 and thus permits entry into modes 4 and 3 to establish proper conditions for testing. Therefore, testing the deficient valves in mode 3 prior to entering mode 2 will satisfy the Technical Specification Requirements.

Since no adverse effects are associated with the deferral of this item, this activity is submitted and recommended for deferral until after fuel load of Unit 1.