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TEXAS UTILITIES GENERATING COMPANY

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

June 5, 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. 4 AND 5

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

Dear Mr. Denton:

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

The fourth test proposed for deferment concerns the preoperational testing of turbine driven auxiliary feedwater pump inlet steam supply line check valves and drain pot level valve. The fifth test proposed for deferment concerns the preoperatioanl testing of the Reactor Coolant Pump Seal Performance. As noted in the attachments, our evaluation indicates that deferral of these items do not constitute unreviewed safety questions and does not require any Technical Specification exceptions. We request your concurrence with our proposal to defer these tests until after 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.

Respectfully,

Billy R Clements

BRC/grr
Attachment

B001
1/1

Deferred Preoperational Testing
Turbine Driven Auxiliary Feedwater Pump Steam Supply
Line Check Valve and Drain Pot Level Control Valve

DESCRIPTION AND SCOPE

During the conduct of Preoperational Tests 1CP-PT-37-03, "Auxiliary Feedwater Turbine-Driven Pump Test", section 7.8, Cold Start of the Turbine Driven Pump, steam line inlet supply drain pot level control valve 1LV-2383, failed to shut after condensate was drained from the steam line piping. This failure was due to a level switch sticking shut. Furthermore, an inspection of the steam line inlet supply check valves, 1MS-142 and 1MS-143, revealed that the disks were eroded and bent.

The level switch associated with drain valve 1LV-2382 has since been replaced. The subject check valves have been modified by Borg-Warner. In order to close out this preoperational test, proper drain pot drain valve operation must be verified and auxiliary feedwater pump reliability verified by conducting five consecutive cold quick starts.

It is our plan to conduct the above testing after fuel load, but prior to criticality when the next plant heat-up is expected to occur. This testing is to be performed during Mode 3, but prior to entering Mode 2 as required by section 3/4.7.1.2 of the present CPSES Technical Specifications. This testing has been incorporated into initial startup Procedure ISU-206A, "Auxiliary Feedwater System Performance". This test procedure has been approved by the Station Operations Review Committee.

SUMMARY SAFETY EVALUATION

A review of the 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 a change to the draft CPSES Technical Specifications. Qualitative evaluations of the appropriate Chapter 15 events provided the bases for the conclusion that no technical specification exceptions are required and no unreviewed safety questions exist, provided that the subject testing is conducted during Mode 3. Incidents from FSAR Chapter 15 considered in this evaluation include:

15.1 Inadvertent Opening of a Steam Generator Relief or Safety Valve

15.1 Steam System Piping Failure

15.2 Loss of Non-Emergency AC Power to the Station Auxiliaries

15.2 Loss of Normal Feedwater Flow

15.2 Feedwater System Pipe Break

15.6 Steam Generator Tube Failure

15.6 Loss of Coolant Accidents from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary

The appropriate Technical Specification for this system is Section 3.7.1.2. The limiting condition for operation requires that all auxiliary feedwater system pumps and their associated flow paths be operable during Modes 1, 2 and 3. Surveillance requirement 4.7.1.2 permits entry into Mode 3 to demonstrate operability of the auxiliary feedwater pumps. Therefore, 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 Reactor Coolant Pump Seal Performance

DESCRIPTION AND SCOPE

During the conduct of preoperational tests 1CP-PT-55-09, "Reactor Coolant Pump Test", and 1CP-PT-49-02, "Seal Water and Letdown Flow Performance", several test deficiencies were identified relating to the performance of RCP seals. A modification has since been completed on the seal housing of each RCP which should correct the subject test deficiencies. The testing required to close out this preoperational test include:

a) Verification of proper RCP seal injection and leakoff flows prior to and after starting of RCP's.

b) Verification of proper RCP seal injection and leakoff flows at RCS normal operating temperature and pressure.

The minimum RCS pressure at which a RCP can be started is 400 psig. It is our plan to conduct the above testing after fuel load but prior to initial criticality, when the next heatup is expected to occur. The above testing would be conducted during modes 3 and 4 and would verify proper seal injection and leakoff flows of each RCP. This testing will be incorporated into appropriate Initial Startup Test Procedures.

These procedures are all reviewed and approved by the 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 a change to the draft CPSES Technical Specifications. Qualitative evaluation of the appropriate Chapter 15 events provided the bases to the conclusion that no Technical Specification exceptions are required and no unreviewed safety question exists. The decrease in reactor coolant system flow accidents discussed in Chapter 15 are postulated to occur with the reactor operating at the upper limits of the maximum guaranteed steady state thermal power. The FSAR goes on to conclude that even in the unlikely event of a reactor coolant pump shaft seizure (which is the worst postulated accident of this category), the integrity of the primary coolant system is not endangered. Therefore, with the commitment to demonstrate RCS flow prior to criticality, deferment of reactor coolant pump seal retesting until after fuel load will not adversely impact the safety analysis.

Technical Specification 3.4.6.2 requires that the RCS leakage be limited to 40 gpm controlled leakage at a RCS pressure of 2235 $\pm$ 20 psig. Although the surveillance testing requirements of 4.4.6.2.1.c do not apply for entry into mode 3 or 4, the retesting requirements and plant specific operating procedures for operating the reactor coolant pumps along with the surveillance requirements for modes 1 and 2 will ensure that the leakage criteria is met.