Docket
P1-22

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

January 29, 1993

See Rpt
(Tech Spec)

Docket No. 50-445

Mr. William J. Cahill, Jr.
Group Vice President
TU Electric
400 North Olive Street, L.B. 81
Dallas, Texas 75201

Dear Mr. Cahill:

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION, UNIT 1 - AMENDMENT NO. 14 TO FACILITY OPERATING LICENSE NO. NPF-87 (TAC NOS. M80210 AND M84709)

The Commission has issued the enclosed Amendment No. 14 to Facility Operating License No. NPF-87 for the Comanche Peak Steam Electric Station, Unit 1. The amendment consists of changes to the Technical Specifications (TS) in response to your applications dated April 2, 1991, and August 31, 1992. These applications were supplemented by letters dated October 29, 1992, and December 14, 1992.

This amendment modifies the Appendix A Technical Specifications by: (1) adding action requirements for the engineered safety features actuation system instrumentation loss of power function, (2) changing the Unit 1 TS to the Combined TS for Units 1 and 2, (3) revising the TS for the station service water system to reflect two operational units, (4) removing the option of performing a containment reduced pressure test in lieu of a containment peak pressure test during preoperational, periodic and supplemental tests, (5) revising the safety limits and limiting conditions for operation related to departure from nucleate boiling ratio to make them applicable to both Unit 1 and 2, (6) changing the pressure/temperature limits for both Unit 1 and 2 and making the new limits applicable for 16 effective full power years, (7) revising the heatup/cooldown curves and power operated relief valve setpoints for low temperature overpressure protection, and (8) adding a TS and associated bases for the feedwater isolation valve pressure/temperature limit.

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A copy of our related safety evaluation is enclosed. The Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,
Original Signed By

Thomas A. Bergman, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 14 to NPF-87
2. Safety Evaluation

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Name	EPeyton	TBergman:nb BEH	BHolian BEH	RSchaaf BEH	DSkay	
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

TEXAS UTILITIES ELECTRIC COMPANY, ET AL.*

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT 1

DOCKET NO. 50-445

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 14
License No. NPF-87

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Texas Utilities Electric Company (TU Electric) acting for itself and as agent for Texas Municipal Power Agency (licensees) dated April 2, 1991, and August 31, 1992, as supplemented by letters dated October 29, 1992, and December 14, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public;

   E. The issuance of this license amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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* The current owners of the Comanche Peak Steam Electric Station are: Texas Utilities Electric Company and Texas Municipal Power Agency. Transfer of ownership from Texas Municipal Power Agency to Texas Utilities Electric Company was previously authorized by Amendment No. 9 to Construction Permit CPPR-126 on August 25, 1988 to take place in 10 installments as set forth in the Agreement attached to the application for Amendment dated March 4, 1988. At the completion thereof, Texas Municipal Power Agency will no longer retain any ownership interest.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2. C.(2) of Facility Operating License No. NPF-87 is hereby amended to read as follows:

   2. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 14, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and to be implemented upon issuance of the low power license for Unit 2.

FOR THE NUCLEAR REGULATORY COMMISSION

Suzanne C. Black

Suzanne C. Black, Director
Project Directorate IV-2
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
See next page

Date of Issuance: January 29, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 14

FACILITY OPERATING LICENSE NO. NPF-87

DOCKET NO. 50-445

Remove, in their entirety, the existing Technical Specifications for Unit 1 (NUREG-1399) and replace with the Combined Technical Specifications (NUREG-1468) issued with the low power license for Unit 2.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 14 TO FACILITY OPERATING LICENSE NO. NPF-87

TEXAS UTILITIES ELECTRIC COMPANY, ET AL.

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT 1

DOCKET NO. 50-445

1.0 INTRODUCTION

By applications dated April 2, 1991, and August 31, 1992, supplemented by letters dated October 29, 1992, and December 14, 1992, Texas Utilities Electric Company (TU Electric or the licensee) requested changes to the Technical Specifications (Appendix A to Facility Operating License No. NPF-87) for the Comanche Peak Steam Electric Station Unit 1. The proposed changes would modify the technical specifications (TS) by: (1) adding action requirements for the engineered safety features actuation system (ESFAS) instrumentation loss of power function, (2) changing the Unit 1 TS to the Combined TS for Units 1 and 2, (3) revising the TS for the station service water system (SSWS) to reflect two operational units, (4) removing the option of performing a containment reduced pressure test in lieu of a containment peak pressure test during preoperational, periodic and supplemental tests, (5) revising the safety limits and limiting conditions for operation related to departure from nucleate boiling ratio (DNBR) to make them applicable to both Unit 1 and 2, (6) changing the pressure/temperature (P/T) limits for both Unit 1 and 2 and making the new limits applicable for 16 effective full power years (EFPY), (7) revising the heatup/cooldown curves and power operated relief valve (PORV) setpoints for low temperature overpressure protection, and (8) adding a TS and associated bases for the feedwater isolation valve (FWIV) P/T limit. Each of these changes are addressed separately below.

2.0 EVALUATION

2.1 Action Statements for ESFAS Instrumentation Loss of Power Function

Background

By letter dated April 2, 1991, TU Electric requested a revision to TS Table 3.3-2, "Engineered Safety Features Actuation System Instrumentation," for CPSES. The licensee proposed the addition of action statements to Table 3.3-2 Item 8, "Loss of Power (6.9kV and 480Vac Safeguard System Undervoltage)." The additional action statements will provide direction in the event that two undervoltage relay instrumentation channels on a bus become inoperable. The licensee also proposed to change the number of the action statements for Table 3.3-2, Item 6e, "Loss of All Main Feedwater Pumps and Start of All Motor Driven Pumps." The staff has reviewed the licensee's submittal and the additional related information and provides the following evaluation.

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Evaluation

The CPSES undervoltage protection scheme consists of six distinct functional groups of relays for each train as shown in Item 8 of the CPSES TS Table 3.3-2. The primary functions of these relays are as follows:

- 6.9 kV Preferred Offsite Source Undervoltage

  Upon detection of an undervoltage condition, these relays cause the preferred offsite feeder breaker to the Class 1E safety busses to open.

- 6.9 kV Alternate Offsite Source Undervoltage

  Upon detection of an undervoltage condition, these relays either prevent the alternate offsite feeder breaker to the Class 1E safety busses from closing or open the feeder breaker if closed.

- 6.9 kV Bus Undervoltage

  Upon detection of an undervoltage condition on the safety bus, these relays cause load shedding of large motors, and the diesel generator to start.

- 6.9 kV Degraded Voltage

  Upon detection of a degraded voltage condition on a 6.9 kV safety bus, these relays cause the preferred offsite source breaker to the Class 1E safety busses to open and the alternate offsite source breaker to trip if it fails to establish the bus voltage. This in turn will actuate the bus undervoltage relays (item 3 above) to initiate load shedding, diesel start, etc.

- 480 Vac Degraded Voltage

  These relays sense voltage on the 480 Vac safety buses and initiate similar actions as the 6.9 kV degraded voltage relay.

- 480 Vac Low Grid Undervoltage

  These relays sense voltage on 480 Vac safety bus and initiate similar actions as the 6.9 kV degraded voltage relay.

Each of the above groups of protection consist of two sensing relays per bus that provide input to a two-out-of-two logic channel. Currently, Action Statement 23 in Table 3.3-2 states that if a channel is determined to be inoperable it must be placed in the trip condition within six hours. This will permit the other relay channel to actuate and make up the logic for the undervoltage protection scheme. However, Table 3.3-2 does not address a situation where there is a loss of two channels on the same bus. If this condition were to occur the licensee would be required to follow the instruction in Specification 3.0.3. This specification requires that

the licensee restore at least one channel to operable status in one hour or initiate the following actions:

a. At least hot standby within the next six hours,

b. At least hot shutdown within the following six hours, and

c. At least cold shutdown within the subsequent 24 hours.

The licensee considers this to be unduly restrictive since TS 3.8.1.1 for inoperable A.C. sources and TS 3.8.3.1. for inoperable A.C. emergency busses allow greater outage time for the loss of a bus or source than TS 3.0.3 does for the loss of undervoltage monitoring on a bus or source. The licensee proposes the addition of action statements 23a, 23b, 23c, 23d and 23e to Table 3.3-2, to address the situation where there is a loss of two channels of undervoltage relays on one bus.

- Addition of Action Statement 23a

  Action Statement 23a retains the original actions contained in Action Statement 23 for the loss of one relay channel per bus.

- Addition of Action Statements 23b and 23c

  Action Statements 23b and 23c require that the preferred or alternate power source be declared inoperable if both the associated relay channels on one bus become inoperable and take the actions required in TS 3.8.1.1 for the loss of one offsite source of power.

- Addition of Action Statement 23d

  Action Statement 23d requires that the 6.9 kV bus be declared inoperable if both of the associated relay channels on one bus become inoperable and take the actions required in TS 3.8.3.1 for the loss of one emergency bus.

- Addition of Action Statement 23e

  Action Statement 23e requires that both the preferred and alternate offsite sources be declared inoperable and take the actions required in TS 3.8.1.1 for the loss of two offsite sources.

The above changes to the action statements for the undervoltage relays ensure that in the event that both relays become inoperable, the monitored function is declared inoperable and the limiting condition of operation of TS 3.8.1.1 or 3.8.3.1 are entered. Entering Section 3.8 of the TS instead of 3.0.3 is less restrictive and precludes prematurely changing plant operating modes.

Summary

The staff agrees that the TS Section 3.0.3 is unduly restrictive for the loss of two relay channels and the out-of-service time for inoperable relay channels should be more consistent with the outage time established for the loss of an offsite power source or emergency bus as provided in TS 3.8. The proposed changes do not increase the out-of-service time for inoperable equipment any more than already established in TS 3.8 and, therefore, do not cause a significant reduction in safety margin. The proposed changes do not involve system hardware or circuit changes and do not change the manner in which the undervoltage relay channels operate. Therefore, the staff finds the above proposed changes to be acceptable.

2.2 Change the Unit 1 TS to the Combined TS for Units 1 and 2

Background

The Comanche Peak Steam Electric Station Unit 1 operating license includes the TS for the operation of Unit 1. At the time Unit 2 receives an operating license, TU Electric will receive TS that are applicable for both units, i.e., Combined TS. To implement the Combined TS on Unit 1, the Unit 1 license requires an amendment. In general, the proposed changes add Unit 2 specific requirements and provide clarifications necessary to determine unit applicability. No hardware or operational changes are being made as a result of this amendment.

Evaluation

The section of the proposed amendment regarding the surveillance requirements for Reactor Coolant System Pressure Isolation Valves replaces a reference to paragraph IWV-3427(b) of Section XI of the ASME Boiler and Pressure Vessel Code (the Code) with a reference to Specification 4.0.5. This change will make the requirement consistent with other Code requirements in the TS.

The latest revision to the Inservice Testing Program for Units 1 and 2 was developed using the 1989 Edition of the Code, which no longer contains paragraph IWV-3427(b), but references ASME/ANSI OM Standard Part 10. The NRC provided approval for use of the 1989 Edition of the Code in Supplement 26 to NUREG-0797, "Safety Evaluation Report relating to the operation of Comanche Peak Steam Electric Station," which is being issued with the low-power license for Unit 2. The approval for Unit 1 was provided by letter dated January 29, 1993. Adoption of the 1989 Code Edition provides an equivalent level of assurance of operability as was provided by previously approved Code editions and addenda.

Three changes were proposed regarding the administrative section of the TS. The first change makes the term "unit" plural, in reference to the plant manager being responsible for the operation of both units. The second change delineates the required minimum shift crew composition for two units with a common control room, as is the case at CPSES. The changes associated with

this request do not impact safety in that assurance is provided that adequate operations personnel will be available in order to respond appropriately to accident situations. The third change replaces the term "plant," which refers to both units, with "unit" when delineating the requirements related to submittal of the unit-specific Startup Report.

Summary

On the basis that the changes listed above are primarily administrative in nature and do not make any hardware or operational changes, the staff finds the above proposed changes acceptable.

2.3 Revision to the Station Service Water System Technical Specification

Background

Generic Letter (GL) 91-13, "Request for Information Related to the Resolution of Generic Issue 130, "Essential Service Water System Failures at Multi-Unit Sites," Pursuant to 10 CFR 50.54(f)" dated September 19, 1991, proposed technical specification changes to enhance the availability of the essential service water system. TU Electric's response (dated March 16, 1992), committed to propose a revision to the CPSES technical specifications and their bases to address the concerns of GL 91-13. Currently, station service water system operability requirements are included in the CPSES technical specifications and technical requirements manual (TRM).

The existing CPSES Technical Specification 3/4.7.4 is consistent with the Westinghouse Standard Technical Specifications. Technical Specification 6.8 covers the implementation of the TRM, which includes Technical Requirement 3.2, "Station Service Water System Operability Criteria."

The TRM 3.2 currently requires a Unit 2 service water pump to be available to support Unit 1 operation whenever Unit 1 is in Modes 1, 2, 3 and 4. If this condition is not satisfied, a Unit 2 pump must be restored to available status within 7 days or Unit 1 must be shut down. If only one Unit 1 service water pump is operable and neither Unit 2 service water pump is available, immediate action to restore at least one additional pump is required. Verification of Unit 2 pump availability includes an energized bus (once per day), cross-connect valve testing (quarterly) and monthly pump runs of at least 15 minutes. This technical requirement was implemented to improve station service water system (SSWS) reliability based on a generic probabilistic assessment of plants with two full capacity service water pumps and NRC Information Notice No. 86-11.

This proposed change to the CPSES technical specifications is provided to assure the required operability of SSWS in each unit while improving the overall availability of SSWS by providing for cross-connects between the units. The present technical specification and technical requirement were appropriate while Unit 1 was operating and Unit 2 was under construction. A

revised specification is needed for two operating units. The TRM section on SSWS will be superseded by this technical specification change.

Discussion

CPSES has two 100 percent capacity SSWS pumps per unit. These four service water pumps have crosstie capability such that any service water pump may supply any other service water pump's cooling loads. The unit crosstie piping is ASME Class 3 and contains five manual gear operated butterfly valves. Each train's crosstie isolation valve is maintained in the normally closed position. The Unit 1-Unit 2 crosstie valve is locked closed and will normally be maintained in the locked closed position during two unit operation. To establish a crosstie between the units, three of these valves, including the Unit 1-Unit 2 crosstie valve, must be opened.

The cross-connect valves are manual, gear operated butterfly valves with rubber seats that are not prone to binding due to differential pressure, galvanic corrosion, or hydraulic blocking above the disc. Although testing is not required by ASME Section XI, quarterly full stroke testing of these valves is consistent with GL 91-13 and the ASME Section XI requirement for Category A and B valves.

The proposed change to TS 3/4.7.4 adds the requirement for a cross-connect between the station service water systems to be operable when either or both units are in Mode 1, 2, 3, or 4. It also adds the requirement for a minimum of one station service water pump to be operable in the shutdown unit in order to support the other unit in the event of a loss of essential service water event in a unit in Mode 1, 2, 3 or 4.

Evaluation

The proposed technical specification is consistent with the GL 91-13 draft specification except as follows:

- The proposed TS splits the requirements into two specifications so that a shutdown unit can satisfy the limiting condition for operation (LCO) without entering the associated action statement. Proposed TS 3/4.7.4 is to be divided into 3/4 7.4.1 for both units in Mode 1, 2, 3, or 4, and 3/4.7.4.2 for only one unit in Mode 1, 2, 3, or 4 and the other unit in Mode 5, 6 or defueled. This change is administrative and does not change the intent of the GL 91-13 SSWS operability requirements; therefore, it is acceptable.

- The addition of the "defueled" mode to cover standard refueling practices and the plant status prior to Unit 2 fuel load. GL 91-13 did not address one unit in Mode 1, 2, 3, or 4 and one unit in the defueled mode. Appropriate operability requirements are maintained (i.e., the Mode 5 and 6 requirements), therefore, this change is acceptable.

- The proposed TS contains a seven day allowed outage time (AOT) for the service water pump in the shutdown unit and the cross-connect valves to allow for appropriate maintenance during refueling/maintenance outages. (The GL 91-13 sample TS contained a 72 hour AOT for the service water pump in the shutdown unit and the cross-connect valves.)

The crosstie capability requires the closure or throttling of the discharge isolation valve for the cross-connected service water pump. The licensee stated in their submittal that CPSES procedures require that both pumps be declared inoperable whenever all the cross-connect valves between them are open. This action would in effect put the licensee in the GL 91-13 recommended 72-hour LCO. Therefore, although a seven day period is allowed for pump or valve maintenance, if the cross-connects are opened the operating unit would be placed in a 72-hour LCO. This situation was not described in GL 91-13. The TS bases for the proposed TS describe the controls over the cross-connects for CPSES.

Additionally, the licensee performed a probabilistic risk assessment (PRA) calculation to determine the change in total core damage frequency due to the AOT variation from 72 hours to 7 days. The licensee submitted the conclusion from their review, which showed that the impact of the change in AOT from 72 hours to 7 days on the calculated core damage frequency is insignificant (the equipment unavailability due to the increase in AOT is insignificant and, therefore, it has relatively no impact on the total core damage frequency). These results were reviewed onsite during an audit on January 7, 1993. The results show that the unavailability of the cross-connect between the units is dominated by human error scenarios. The impact on the calculated core damage frequency is less than one percent.

Finally, the seven day AOT for the cross-connect valves and the supporting service water pump from a shutdown unit is consistent with both the CPSES TRM and similar TS granted by the staff.

Therefore, on the basis of the site-specific CPSES design with the above mentioned procedural controls over the use of the cross-connect valves, the results of the PRA calculation for CPSES which shows an insignificant change in the equipment unavailability due to the increased AOT, and the existence of similar TS for similar designed plants, the 7 day LCO is acceptable.

In the submittal the licensee provided a failure mode and effects analysis for loss of service water pump events under the provisions of the proposed TS changes. The table described the LCO and actions that would apply for typical modes and conditions. The table was reviewed and found to be comprehensive. Additionally, a table was provided which listed the unit cross-connect valves. The table was reviewed and was found to provide a useful matrix which ensures that the appropriate cross-connects are closed to assure train operability, unit separation, and compliance with GDC-5 and GDC-44.

There are no surveillance testing requirements which would require the cross-connect valves to be open for pump testing in accordance with TS 4.0.5. However, GL 91-13 notes that the guidance contained in GL 89-13 and Supplement 1 should be considered. Therefore, the licensee stated in their submittal that CPSES procedures will implement periodic flushing of the cross-connect. The frequency of this flushing will be in accordance with GL 89-13, Supplement 1. This is acceptable.

The TS bases were also expanded to better define cross-connect operability: "A cross-connect valve is OPERABLE if it can be cycled or is locked open. A valve that cannot be demonstrated OPERABLE by cycling is considered inoperable until the valve is surveilled in the locked open position. However, at least one cross-connect valve between units is required to be maintained closed in accordance with GDC-5 unless required for flushing or due to total loss of SSWS pumps for either unit." This description maintains the GL 91-13 description and clarifies operability when the valve can be appropriately locked open. This change is acceptable.

Summary

The proposed TS include all the requirements included in the existing technical specification. In addition, the proposed specification adds new requirements to improve the SSWS availability in the event that all SSWS is lost on one unit. Therefore, these changes are acceptable.

2.4 Removal of Option to Perform Containment Reduced Pressure Test

Background

This change removes the option of performing a containment reduced pressure test in lieu of a containment peak pressure test during preoperational, periodic, and supplemental tests. Section III.A.4.a of Appendix J to 10 CFR 50 allows an overall integrated leak rate test (ILRT) to be performed at less than or equal to 50 percent of the peak accident pressure provided that a correlation can be made between the reduced pressure ILRT and the peak pressure ILRT. This reduced pressure test is initially performed during preoperational startup tests in order to apply this correlation for future periodic ILRTs. However, because experimental data does not support a clear correlation, the licensee has proposed eliminating the performance of reduced pressure ILRTs and satisfying the requirements of 10 CFR 50 Appendix J by conducting peak pressure tests only. The change would remove all references to the reduced pressure ILRT in LCO 3.6.1.2, Surveillance 4.6.1.2, and related bases.

Evaluation

The review of data collected by the American Nuclear Insurers (ANI) for 50 pairs of pre-operational ILRTs conducted at both peak and reduced pressure does not support a clear correlation between reduced pressure and peak leak rates. In addition, the Unit 1 containment reduced pressure test results do

not give a satisfactory correlation. Therefore, any relationship between leakage rates determined during pre-operational testing cannot be reasonably presumed to exist for extended periods of time. The ANI proposed draft position indicates that testing and failure experience has shown that some leakage testing failures were clearly due to the inception of leaks brought on by increasing pressure. These failures demonstrate that conducting a successful leakage rate test at reduced pressure cannot assure that leakage integrity will be maintained at peak pressure. It is not clear how the results of reduced pressure leakage tests can be used with confidence to demonstrate that the leakage rate at peak pressure will be less than the maximum allowed by the technical specification for peak pressure conditions.

The intent of conducting an initial reduced pressure test is to allow development of a correlation between reduced pressure and peak pressure leak rates. Because this correlation cannot be clearly established the option of performing reduced pressure tests should be eliminated. Performance of the pre-operational ILRT at peak design pressure for the Unit 2 containment structure satisfies the requirements of 10 CFR 50 Appendix J. The proposed TS change would also require that periodic ILRTs be conducted at peak design pressure. Because other components in the pressure boundary are designed to withstand the pressure introduced in a peak pressure ILRT, these components will not be adversely affected.

Summary

Because an accurate correlation between reduced pressure and peak pressure ILRTs cannot be assured, the elimination of a containment reduced pressure test during preoperational, periodic, and supplemental tests is acceptable.

2.5 Revised Safety Limits and LCOs for Operation Related to DNBR

Background

These changes revise the safety limits and limiting conditions for operation related to DNBR to make them applicable to both CPSES Unit 1 and 2. Because the designs of the reactor cores were done with different departure from nucleate boiling (DNB) correlations and analysis methodologies, the DNBR safety analysis limit value is calculated at 1.49 for Unit 1 and 1.30 for Unit 2. The Unit 1 core was designed using the W-3 DNB correlation and the Westinghouse Standard Thermal Design Procedure (STDP). Unit 2 was designed using the WRB-1 DNB correlation and the Westinghouse Improved Thermal Design Procedure (ITDP). The two units also have different nuclear enthalpy rise hot channel factor multipliers.

The specific changes include the addition of a Unit 2 reactor core safety limit figure, Figure 2.1-1b, which is different from Unit 1 due to a differing DNB correlation and analysis methodology. LCO 3.2.5 is revised to add the allowable Unit 2 DNB related parameters (Tavg, pressurizer pressure, and reactor coolant system flow). The bases would be changed to replace Unit 1 specific values and discussions with discussions that apply to both units.

An additional change to Section 6.9.1.5 replaces the addressee of the monthly operating reports to be in agreement with 10 CFR 50.4.

Evaluation

These changes update the technical specifications to reflect the differences in DNB methodologies and thermal design procedures between Units 1 and 2. These changes do not affect the Unit 1 analysis but rather replace Unit 1 specific equations with equations that apply to both units. The specific values for each unit will be included in the core operating limits report as necessary.

Summary

The above changes are editorial or included for clarification. Therefore, the proposed changes are acceptable.

2.6 Change the P/T Limits and Make Applicable for 16 EFPY

Background

On October 29, 1992, the licensee requested permission to revise the pressure/temperature (P/T) limits in Section 3.6 of the CPSES Unit 1 technical specifications. The proposed revision was based on the test results of reactor vessel material specimens and neutron flux monitors in surveillance capsule U which was removed during the first refueling outage in August 1991. The licensee proposed to use the Unit 1 limits for Unit 2 operation and to incorporate the limits into the combined CPSES Unit 1 and 2 technical specifications. The P/T limits were requested for 16 effective full power years.

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); Regulatory Guide (RG) 1.99, Revision 2; Standard Review Plan (SRP) Section 5.3.2; and GL 88-11. GL 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," recommends RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Rev. 2, be used in calculating P/T limits, unless the use of different methods can be justified.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide TS for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the technical specifications. The P/T limits are among the limiting conditions of operation in the TS for all commercial nuclear plants in the U.S. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART). GL 88-11 requested that licensees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Evaluation

The staff evaluated the effect of neutron irradiation embrittlement on each reactor vessel beltline material in both CPSES Units 1 and 2. For Unit 1, the staff determined that the limiting material at 16 EFPY is the lower shell plate, R1108-1, with 0.08% copper, 0.64% nickel, and initial RT_{ndt} of 0°F. Using RG 1.99, Revision 2, the staff calculated the limiting ARTs for R1108-1 to be 83.6 °F at 1/4T (T= reactor beltline wall thickness) and 69.1°F at 3/4T based on a neutron fluence of 1.8E19 n/cm^2.

For Unit 2, the staff determined that the limiting material is the intermediate shell plate R3807-2, with 0.06% copper, 0.64% nickel, and an initial RT_{ndt} of 10°F. The staff calculated the limiting ARTs to be 81°F at 1/4T and 60.9°F at 3/4T based on a neutron fluence of 1.8E19 n/cm^2.

The licensee's ARTs for both units are about the same as the staff calculated values and are acceptable. The licensee used the limiting ARTs in Unit 1 to construct the proposed P/T limits because the limiting Unit 1 ARTs are higher than the limiting Unit 2 ARTs. The staff finds this practice acceptable. The staff substituted the ARTs of 84°F and 69.1°F into equations in SRP 5.3.2. and verified that the proposed P/T limits meet paragraphs IV.A.2 and V.A. of Appendix G to 10 CFR Part 50.

In addition to beltline materials, paragraph IV.A.2 of Appendix G also imposes P/T limits to meet the temperature requirement for the reactor vessel closure flange. Paragraph IV.A.2 states that when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the reactor closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120 °F for normal operation and by 90 °F for hydrostatic pressure tests and leak tests. The reference temperature for the reactor closure head flange in both units is 40 °F. Based on 40 °F, the staff has determined that the proposed P/T limits satisfy the temperature requirement for the closure head flange in paragraph IV.A.2 of Appendix G.

Summary

The staff concludes that the proposed P/T limits for heatup, cooldown, leak test, and criticality are valid through 16 EFPY because the limits conform to Appendix G to 10 CFR Part 50. The proposed P/T limits also satisfy GL 88-11 because the licensee used the method in RG 1.99, Rev. 2, to construct the limits. Hence, the proposed P/T limits may be incorporated into the combined Comanche Peak Unit 1 and 2 Technical Specifications.

2.7 Revised Heatup/Cooldown Curves and PORV Setpoints for LTOP

Background

The reactor coolant system (RCS) pressure-temperature limits during plant heatup and cooldown for Comanche Peak Steam Electric Station are specified in TS Figures 3.4-2 and 3.4-3 and are applicable up to 16 effective full power years (EFPY). The pressurizer power operated relief valve setpoints for overpressure mitigation applicable up to 16 EFPY are specified in TS Figure 3.4-4.

By letters dated October 29, 1992 and December 14, 1992, the licensee proposed revisions to the P/T curves in TS Figures 3.4-2 (for heatup) and 3.4-3 (for cooldown) and a change in the TS Figure 3.4-4 (for PORV setpoints). The proposed changes incorporated the corrections for nonconservatisms in the original analysis. The revised TS values were provided by Westinghouse in its letter to the licensee dated September 2, 1992. The new PORV setpoints provide more conservative protection against a potential overpressurization during low temperature operation condition.

Evaluation

In the current TS, when the temperature of RCS cold leg is less than or equal to 350°F, low temperature overpressure protection is provided by either of the pressurizer PORVs with lift settings specified in TS Figure 3.4-4. These settings were developed to avoid RCS pressures from exceeding either the reactor vessel Appendix G limit or 800 psig at the PORVs, whichever is the more limiting. The 800 psig was included to limit the loads on the PORV discharge piping and supports when the PORVs cycle open and closed during a subcooled water discharge as could occur during the design basis cold overpressure events. For RCS temperatures below 237°F, a single PORV setpoint of 560 psig is specified to protect against Appendix G limits. For RCS temperature above 237°F, variable PORV settings below 730 psig is specified to protect against 800 psig at PORVs as limiting concern. Each PORV has adequate relieving capacity to protect the RCS against either P/T limits and 800 psig limit for PORV discharge piping when the transient is limited to either (1) the start of an idle RCP with secondary water temperature in the steam generator less than or equal to 50°F above the RCS cold leg temperature or (2) the mass addition transient due to 2 charging pumps in operation with full flow into the RCS, assuming the letdown line is isolated.

By letter dated October 29, 1992, the licensee proposed a change in the PORV setting from 560 psig to 521 psig for RCS temperature below 237°F. This change is proposed by the licensee as a corrective action in response to a nonconservatism that was discovered by the licensee recently. This nonconservatism was not included in the pressure difference between the wide range pressure transmitter and the pressure corresponding to the reactor vessel location where the Appendix G limit is applicable. This pressure difference effectively results in the pressure in the reactor vessel to be greater than that seen by the wide range pressure transmitters and therefore potentially resulting in violation of the Appendix G limit. On the basis of the calculation performed by Westinghouse, a reduction of 50 psi in the pressure limit is needed in the P/T curves specified in TS Figure 3.4-2 and 3.4-3 and a reduction in 39 psi in the PORV setpoint for RCS temperature below 237°F is needed in order to protect against the revised P/T curves. The PORV setpoint change of 39 psi is developed using more realistic assumptions in its recent calculation. The licensee proposed changes of TS are based on the above calculations. The PORV settings for RCS temperature above 237°F do not require a change since they are not effected by the change of Appendix G limit.

Summary

The licensee's proposed changes in TS Figures 3.4-2, 3.4-2 and 3.4-4 reflect the changes discussed above. The staff has reviewed the licensee's submittals and finds that the changes are based on applicable regulatory guidance in Standard Review Plan 5.2.2 (Revision 2), are reasonably conservative and are acceptable.

2.8 Add Technical Specification for FWIV Pressure/Temperature Limits

Background

The CPSES Unit 1 technical specifications do not currently include limiting conditions for operation, surveillance requirements, or bases for the main feedwater isolation valve pressure/temperature limit (P/T). The P/T limit for the FWIVs is currently administratively controlled through the technical requirements manual. The licensee proposed an additional TS to move the TRM requirements to the TS to be consistent with other P/T limits, (e.g., the RCS in TS 3/4.4.8).

Description

The proposed change adds TS 3/4.7.13 for the FWIV pressure/temperature limit to the CPSES Unit 1 technical specifications. The change assures that appropriate valve temperatures are maintained before and while these valves are pressurized above 675 psig. The change to the TS consists of the FWIV limiting condition for operation, surveillance requirements, and basis for the TS. A footnote has been added to TS 3/4.4.9, "Structural Integrity," that identifies the applicable P/T limit TS for specific components.

Evaluation

The proposed TS provides explicit identification of the P/T limitation for FWIVs in a manner consistent with the existing TS for other similar pressure/temperature limits. The proposed TS does not alter any of the assumptions used in the safety analysis for CPSES Unit 1. The TS assures that the capability of the FWIVs to perform their intended function under normal and accident conditions is maintained by ensuring suitable temperatures are maintained to prevent flaws from propagating in the valve body. The proposed surveillance requirement enhances the reliability of the valves by requiring a periodic surveillance during modes of operation and other conditions where valve integrity could be challenged by pressure/temperature changes.

Summary

On the basis that the proposed TS is consistent with comparable TS for other P/T limit TS, is primarily administrative in that the requirements in the proposed TS are already administratively controlled in the TRM, and there is no adverse impact on safety, the proposed TS is acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Texas State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (57 FR 55595, 57 FR 55596, 57 FR 55597, 57 FR 60544, and 57 FR 61121). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the

public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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