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Log # TXX-99195
File # 10010
Ref. # 10CFR50.36

August 13, 1999

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NOS. 50-445 AND 50-446
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION ON LICENSE AMENDMENT REQUEST 98-010
(TAC Nos. MA4436 and MA4437)

REF: TXU Electric[1] letter, logged TXX-98265, from C. L. Terry to the NRC dated December 21, 1998

Gentlemen:

In the referenced letter, TXU Electric submitted a request to amend the CPSES Unit 1 Operating License (NPF-87) and CPSES Unit 2 Operating License (NPF-89) by incorporating changes into the CPSES Units 1 and 2 Technical Specifications and the CPSES Unit 2 Operating License to increase the licensed power for operation of CPSES Unit 2 to 3445 MWth; an increase of approximately 1%. Per telephone conversation with NRR on August 5, 1999, TXU Electric received a request to provide the attached additional information regarding License Amendment Request 98-010. Attachment 1 is the affidavit for this information supporting License Amendment Request 98-010. Attachment 2 provides our response to the information requested.

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1 TXU Electric was formerly TU Electric. A license amendment request (LAR 99-003) was submitted per TXX-99122, dated May 14, 1999, to revise the company name contained in the CPSES operating licenses.

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If you have any questions regarding the attached information, please contact Mr. J. D. Seawright at (254) 897-0140.

This communication contains no new licensing basis commitments regarding CPSES Units 1 and 2.

Sincerely,

C. L. Terry

C. L. Terry

By: Roger D. Walker

Roger D. Walker
Regulatory Affairs Manager

JDS/jds
Attachments

c - E. W. Merschoff, Region IV
J. I. Tapia, Region IV
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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

| | | |
|---|---|---|
| In the Matter of | ) | |
| | ) | |
| TXU Electric | ) | Docket Nos. 50-445 |
| | ) | 50-446 |
| (Comanche Peak Steam Electric | ) | License Nos. NPF-87 |
| Station, Units 1 & 2) | ) | NPF-89 |

AFFIDAVIT

Roger D. Walker, Jr. being duly sworn, hereby deposes and says that he is the Regulatory Affairs Manager of TXU Electric, the licensee herein; that he is duly authorized to sign and file with the Nuclear Regulatory Commission this Request for Additional Information regarding License Amendment Request 98-010; that he is familiar with the content thereof; and that the matters set forth therein are true and correct to the best of his knowledge, information and belief.

Roger D. Walker

Roger D. Walker
Regulatory Affairs Manager

STATE OF TEXAS )
)
COUNTY OF Johnson )

Subscribed and sworn to before me, on this 13th day of August, 1999.

Carolyn L Cosentino
Notary Public

NOTARY PUBLIC STATE OF TEXAS
CAROLYN L. COSENTINO
Notary Public, State of Texas
My Comm. Expires 05/07/02

# RESPONSE TO NRC REQUEST FOR INFORMATION

**Question 1:**

**Provide a comparison of the relevant acceptance criterion to the appropriate design limit (e.g., DNBR, RCS pressure) for each of the following safety analyses:**

**15.4.2 Uncontrolled RCCA withdrawal from power**
**15.4.7 Misloaded fuel assembly**
**15.4.8 Rod Ejection**
**15.4.3 Dropped RCCA**

Response:

The system analyses for the above events were performed in accordance with the NRC approved methodologies described in Technical Specification 5.6.5, Item 14. Information contained in Item 13 provides additional information. Where necessary, the comparison against the DNBR limit was performed as described in TS 5.6.5 Items 10, 11, and 12. Using these NRC-approved methods and considering operation at a Rated Thermal Power of up to 3445 MWth, compliance with all relevant event acceptance criteria was demonstrated for the Unit 2 Cycle 5 core configuration.

As described in the response to Question 25 of the RAIs contained in Technical Specification 5.6.5, Item 13, the relevant acceptance criterion for the uncontrolled rod withdrawal at power event is compliance with the DNBR limit. For the analysis performed to support Unit 2 Cycle 5 operation, the full power cases were analyzed at a power level of 102% of 3445 MWth, which is greater than the required value of 102% of 3411 MWth, or 101% of 3445 MWth. The assumed initial power level is the licensed core thermal power plus an allowance of 2% of the initial power to account for power measurement uncertainties. The initial power assumed for this specific calculation was 3445 MWth (plus the 2% uncertainty), which bounds the licensed rated power of 3411 MWth. The calculated minimum DNBR for this case is 1.365 (including any effects attributed to mixed cores and the lower plenum flow anomaly) which is greater than the DNBR limit value of 1.16.

The relevant acceptance criterion for the misloaded fuel assembly analysis is compliance with the DNBR limit. This event is analyzed by calculating a maximum allowable value of the nuclear enthalpy rise hot channel factor ($F_{\Delta H}$) such that the DNBR acceptance limit is just met. The thermal-hydraulic conditions used for this determination is 102% RTP where RTP is 3411 MWth, and the 2% RTP allowance is provided to account for power measurement uncertainties. A reactor physics calculation is then performed to ensure that

for the spectrum of potential misloaded assemblies identified, the resulting nuclear enthalpy rise hot channel factor is less than the maximum allowable value. Although case-specific DNBR calculations are not performed, compliance with the DNBR acceptance criterion is assured through compliance with the maximum allowable value of $F_{\Delta H}$.

The rod ejection event is analyzed to ensure compliance with the guidelines of 10CFR100. The source term for this analysis is based on assumptions concerning the integrity of the fuel rods which are confirmed to remain valid on a cycle-specific basis. The source term is based on assumptions of 10% fuel failures and 0.25% fuel melt. Fuel failures are assumed to occur if the DNBR limit is exceeded; fuel melt is assumed to occur if the peak centerline fuel temperature exceeds 4700°F. An additional criterion is that the fuel remains in a coolable geometry. Compliance with an average fuel pellet enthalpy limit of 280 cal/gm is used to ensure that no fuel dispersion occurs and the a coolable geometry is maintained. The full power scenarios are analyzed at an initial power of 100% RTP (i.e., 3411 MWth) plus an allowance of 2% RTP to account for power measurement uncertainties. For both the beginning of life and end of life full power cases for Unit 2 Cycle 5, the peak average fuel enthalpy is calculated to be 152 cal/gm which is less than the limit of 280 cal/gm. To ensure compliance with the assumptions on fuel failures and fuel melt, maximum allowable values of the nuclear enthalpy rise hot channel factor ($F_{\Delta H}$) and heat flux hot channel factor ($F_Q$) are calculated such that the respective limits of DNBR and fuel centerline temperature are just met. Reactor physics calculations are then performed to evaluate the distribution of peaking factors in the core for the spectrum of potential ejected RCCAs. A pin census is then performed to calculate the percentage of the core that exceeds the relevant peaking factors.

The dropped rod event is analyzed to demonstrate compliance with the DNBR acceptance limit. For Unit 2 Cycle 5, this analysis was performed using the statistical combination of uncertainties (SCU) method described in Technical Specification 5.6.5, Item 14. Using this method, the system analyses are assumed to be initiated from nominal, full power conditions. The uncertainties in the initial conditions, (in this case, power, pressure, temperature, and $F_{\Delta H}$) are combined statistically and included in the DNBR limit. As such, separate analyses were required to address operation at a Rated Thermal Power of 3411 MWth with an allowance of 2% RTP to account for the power calorimetric uncertainty, and operation at a Rated Thermal Power of 3445 MWth with an allowance of 1% RTP to account for the smaller power calorimetric uncertainty. Intuitively, one would expect the latter case to be limiting, since more of the initial power is considered in a deterministic, rather than statistical, manner. Such is the case for the Unit 2 Cycle 5 analyses. Several bounding assumptions from prior cycles (e.g., the axial power shape) were retained in the Unit 2 Cycle 5 - specific analysis. In other words, the analysis was more conservative than required by the NRC-approved methodology. The

minimum DNBR was calculated to be approximately the same as the cycle-specific SCU DNBR limit of 1.336. Because the purpose of the analysis was satisfied (i.e., the DNBR limit was assured of being greater than the limit value), additional cycle-specific analyses were not performed.

Using the approved methods listed in the Technical Specification 5.6.5 and considering operation at a Rated Thermal Power of up to 3445 MWth, compliance with all relevant event acceptance criteria was demonstrated for the Unit 2 Cycle 5 core configuration.

**Question 2:**

**The topical report detailing the analysis of an inadvertent boron dilution event (RXE-91-002-A) indicates that the analysis assumed a power level of 100 percent. Discuss the sensitivity of the analysis results to initial power level. Summarize the methods and results of any supporting sensitivity analysis and provide references.**

Response:

Using the NRC-approved methods described in Technical Specification 5.6.5, Items 14 and 18, the inadvertent boron dilution event is analyzed to demonstrate that sufficient time is available for the reactor operators to take appropriate mitigative actions after an alarm has been initiated. The required time, 15 minutes, is the same for all events regardless of the Mode in which the event is assumed to be initiated. For the MODE 1 analysis, the initiating alarm is either a rod insertion limit alarm (if the rods are in automatic) or a reactor trip (probably on overtemperature, although the exact trip function is unimportant). The important point is that after the operator first receives an alarm, the available shutdown margin is at least as large as the required shutdown margin. For a given burnup and coolant temperature, a larger value of the initial boron concentration results in a quicker reduction in the RCS boron concentration, and hence, a faster erosion of the shutdown margin. Following the reactor trip from power operations, the fluid conditions will be equivalent to hot zero power conditions (Mode 3). Because of the moderator, Doppler fuel temperature and flux redistribution reactivity feedback effects, the initial boron concentrations at hot zero power conditions are higher than at hot full power; therefore, the hot zero power analysis will always be more limiting. Thus, the initial power level assumed for the at-power analysis is insignificant.

**Question 3:**

**Discuss the sensitivity of the analysis results to initial power level for the SG tube rupture event. Summarize the methods and results of any supporting sensitivity analysis and provide references.**

Response:

Using the NRC-approved methods described in Technical Specification 5.6.5, Item 16, the SGTR event is analyzed to demonstrate that the calculated dose consequences satisfy the guidelines of 10CFR100. The SGTR event is first analyzed to ensure that the ruptured SG does not completely fill with fluid prior to the time the reactor operators terminate the primary-to-secondary break flow. Assuming success, the single failure scenario that results in the largest radiological dose consequences is the failure to close the atmospheric relief valve on the ruptured steam generator steam line. The source term used for the radiological dose consequence evaluation is based on operation at a power level of 104.5% of 3411 MWth. The mass releases used in the radiological dose consequence evaluation are dominated by the blowdown of the fluid in the ruptured steam generator through the failed-to-close atmospheric relief valve. The primary-to-secondary leak rate during the event is also relatively important. Because of the rapid depressurization of the ruptured SG, the time-dependent mass release is insensitive to small changes in the assumed initial power level. This insensitivity was first identified during the development of the analyses supporting the topical report described in TS 5.6.5, Item 16. While investigating the effects of the proposed 1% uprate for Unit 2, additional calculations were performed at the uprated power. The calculated mass releases for the cases analyzed at the uprated power level are essentially indistinguishable from the original mass releases. Because the mass releases are unchanged and the radiological source term remains bounding, it is concluded that the results of the SGTR event are insensitive to changes related to the proposed power uprate.

**Question 4:**

**CPSES technical specifications contain a surveillance requirement (3.3.1.2) requiring that power levels measured by nuclear instruments and by the N-16 monitoring system be checked to within 2% of the daily calorimetric. Explain why this surveillance requirement is not being modified to require that the readings be within 1% of the calorimetric.**

Response:

The uncertainty associated with the accuracy of the plant calorimetric measurement is considered in the plant safety analyses. It is this uncertainty that can be reduced through the use of the improved LEFM instrumentation.

Technical Specification Surveillance Requirement (SR) 3.3.1.2 is a requirement for the renormalization of the NIS and N-16 power indications if the allowed deviation (±2% RTP) between the power calculated through a plant calorimetric measurement and the NIS and N-16 indicated power is exceeded. This deviation is explicitly considered in the uncertainty analyses of those reactor trip functions that are based on either of these

instruments.

SR 3.3.1.2 is required to be performed every 24 hours (daily). At that time, the NIS and N-16 power indications must be normalized to indicate within at least ± 2% RTP of the calorimetric measurement. The plant may then be run for the next 24 hour period, using these normalized NIS and N-16 power indications, such that the calorimetric power does not exceed 100% RTP. Although the calorimetric power indication may be monitored continuously for control of the unit power, the calorimetric power indication is not required to be consulted again until the daily calorimetric comparisons of the NIS and N-16 power indications are performed.

Procedural guidance is provided for operation of the plant in a manner consistent with the calorimetric measurement, even if the NIS and N-16 indications are within ±2% RTP and are not renormalized. For example, if the calorimetric measurement results indicate a thermal power of 100.5% RTP and the NIS Power Range channels indicate 100.0% RTP, the operator will reduce power to achieve a calorimetric thermal power of 100% RTP with the corresponding NIS-indicated power of 99.5% RTP. This action will ensure operation consistent with the Operating License. Conversely, operation at an NIS-indicated power of greater than 100% RTP is prohibited. This latter restriction is the basis for administrative guidance in which much smaller deviations between NIS and N-16 power indications and the calorimetric power indication are maintained.

The NRC monitors compliance with the Rated Thermal Power limit through Inspection Procedure 61706 (7/14/86), which allows operation in excess of 100% RTP for short periods of time. This allowance prevents any long term or systematic violations of the Operation License, but reflects the fact that a PWR, which follows load naturally, can have transients that result in 100% RTP being exceeded. This guidance also explicitly allows operation at 100% RTP indicated (calorimetric) power without forcing operation at a slightly reduced power level to ensure the Operating License is not inadvertently violated.

In summary, the uncertainty associated with the power calorimetric measurement is explicitly considered in the accident analyses. The allowed deviations between the power calorimetric measurement and the NIS and N-16 power indications are explicitly considered in the relevant setpoint uncertainty analyses.

**Question 5:**

**In response to a previous request for additional information the revised overpower N-16 allowable value of 113.5% of rated thermal power was defended as having been derived based on WCAP-12123 methods. Provide the detailed calculation showing how the allowable value for the N-16 overpower trip was determined.**

Response:

As more fully described in WCAP-12123, the total allowance is defined as the difference between the safety analysis limit and the nominal trip setpoint, expressed as a percentage of the N-16 instrument span. Appropriate uncertainties are combined using the standard square root of the sum of the squares methodology to determine the channel statistical allowance. Obviously, the total allowance must be larger than the channel statistical allowance. The "trigger" used to determine the Allowable Value is the algebraic sum of the uncertainties associated with the rack calibration and drift to the extent that there is sufficient margin between the total allowance and the channel statistical allowance. The Allowable Value is then determined to be the nominal trip setpoint plus the trigger value.

The detailed calculations are available for inspection at CPSES.

While preparing the response to this question, an error was discovered in the setpoint uncertainty calculation for the overpower and power range neutron flux - high reactor trip functions. This error only affects the proposed increase in the Rated Thermal Power to 3445 MWth. Based on the corrected calculations, the following setpoints are required for operation at 3445 MWth, where all powers are expressed as a percentage of 3445 MWth:

| Function | Safety Analysis Limit | Nominal Trip Setpoint | Allowable Value |
|---|---|---|---|
| Power Range Neutron Flux - High | 116.8% | 109% | 111.1% |
| Overpower N-16 | 116.8% | 110% | 113.4% |

The License Amendment Request will be revised to reflect these changes.