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William J. Cahill, Jr.
Group Vice President

May 28, 1993

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
SUBMITTAL OF LICENSE AMENDMENT REQUEST 93-004
RELOAD ANALYSES

Gentlemen:

Pursuant to 10CFR50.90, TU Electric hereby requests an amendment to the CPSES Unit 1 Operating License (NPF-87) and CPSES Unit 2 Operating License (NPF-89) by incorporating the attached changes into the CPSES Units 1 and 2 Technical Specifications. These changes apply equally to CPSES Units 1 and 2 except where a specific unit is indicated.

The current CPSES Technical Specifications only permit reload analyses using Westinghouse methodologies. TU Electric has developed in-house analysis methodologies for CPSES 1 and 2 reload analyses. These methodologies were submitted to the NRC for review and approval. As discussed in Attachment 2, a number of these methodologies have been approved. Based on recent discussions with the NRC staff, the remaining submittals are in the final stages of NRC review and, that for the purposes of this License Amendment Request, may be treated as acceptable for reload analyses.

TU Electric proposes to use these in-house reload analysis methodologies, beginning with CPSES Unit 1, Cycle 4. The reload analyses for CPSES Unit 1, Cycle 4 result in revised core safety limit curves and revised N-16 Overtemperature reactor trip setpoints. In addition, the minimum required Reactor Coolant System (RCS) flow is increased and a previously imposed penalty on pressurizer pressure uncertainty is removed. Finally, an operational enhancement is included in the treatment of the uncertainty allowance for the N-16 power indication.

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TU Electric requests approval of this proposed license amendment by November 1, 1993, with implementation of the Technical Specification changes to occur within 30 days after NRC approval.

In accordance with 10CFR50.91(b), TU Electric is providing the State of Texas with a copy of this proposed amendment.

Should you have any questions, please contact Mr. Bob Dacko at (214) 812-8228.

Sincerely,

William J. Cahill, Jr.
Group Vice President, Nuclear

By: Charles L Terry
Charles L. Terry
Vice President of Nuclear
Engineering and Support

BSD
Attachments: 1. Affidavit
2. Description and Assessment
3. Affected Technical Specification page (NUREG-1468)

c - Mr. J. L. Milhoan, Region IV
Mr. B. E. Holian, NRR
Mr. T. A. Bergman, NRR
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Mr. D. K. Lacker
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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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

AFFIDAVIT

Charles L. Terry being duly sworn, hereby deposes and says that he is Vice President of Nuclear Engineering and Support for TU Electric, the licensee herein; that he is duly authorized to sign and file with the Nuclear Regulatory Commission this License Amendment Request 93-004; 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.

Charles L Terry

Charles L. Terry
Vice President of Nuclear
Engineering and Support

STATE OF TEXAS)
)
COUNTY OF Dallas)

Subscribed and sworn to before me, on this 28th day of May.

Lacrescia Albertson
Notary Public

NOTARY PUBLIC STATE OF TEXAS
Lacrescia Albertson
Notary Public, State of Texas
My Comm. Expires 11/11/96

DESCRIPTION AND ASSESSMENT

I. BACKGROUND

The fuel supplier for Comanche Peak Steam Electric Station (CPSES) Unit 1, has changed. Beginning with Cycle 4, Siemens Power Corporation (SPC) will supply the nuclear fuel assemblies. For Cycle 4, the Siemens fuel assemblies will be co-resident with the existing Westinghouse Standard Fuel Assemblies. For CPSES Unit 2, the fuel supplier is scheduled to change from Westinghouse to Siemens beginning with Cycle 3.

TU Electric has developed in-house analysis methodologies for the CPSES Units 1 and 2 reload analyses. In this Technical Specification change, TU Electric proposes using these in-house reload analysis methodologies for CPSES Unit 1, Cycle 4 to demonstrate that all applicable limits of the safety analysis are met. These methodologies are scheduled to be approved by the NRC prior to the start of Unit 1, Cycle 4. One result of the Unit 1, Cycle 4 reload analyses will be new Technical Specification core safety limits.

In order to enhance the DNB-related analyses of the mixed core configuration using the TU Electric methodologies, the Thermal Design Flow will be increased. Currently, the actual RCS flow is approximately 7.9% higher than the Thermal Design Flow rate assumed in the CPSES Unit 1, Cycle 3 accident analyses. For CPSES Unit 1, Cycle 4, TU Electric proposes crediting 3.5% of this flow in the accident analyses, resulting in the definition of a higher RCS Thermal Design Flow rate. Correspondingly, the Technical Specification minimum measured RCS flow requirement will also be higher.

The CPSES Unit 1 safety analysis was previously assessed a penalty on the pressurizer pressure uncertainty associated with the Barton 763 pressure transmitters. The penalty was due to the non-repeatability of the transmitters at high temperatures. Because the transmitters have now been refurbished by the vendor, the penalty is no longer necessary and will be removed from the setpoint determination.

With new Unit 1, Cycle 4 core safety limits, the Overtemperature N-16 reactor trip setpoints must be recalculated to ensure that the new core safety limits are met. The recalculation of the Overtemperature N-16 reactor trip setpoint provides an opportunity to add an operational enhancement. The Technical Specifications require the readjustment of indicated N-16 power if the N-16 power indication differs by more than ± 2 percent of rated thermal power determined by the daily power calorimetric measurement. Currently, the sensor measurement and test equipment (SMTE) allowance for the N-16 power indication is subtracted directly from the allowable power difference. This reduces the allowed tolerance between the indicated N-16 power and the calorimetric power and results in an unnecessarily high N-16 readjustment frequency. To reduce this readjustment frequency, the SMTE allowance associated with the indicated N-16 power will be included in the channel statistical allowance of the statistical setpoint studies for N-16.

These changes to the plant Technical Specifications are specific to CPSES and are needed to support Unit 1, Cycle 4 operation.

II. DESCRIPTION OF TECHNICAL SPECIFICATION CHANGE REQUEST

The following specific Technical Specification (TS) changes are proposed:

A. The following analytical methods will be added to 6.9.1.6b to determine core operating limits for Units 1 and 2:

- RXE-90-006-P (Power Distribution Control and Overtemperature and Overpower Trip Functions),
- RXE-88-102-P (Departure from Nucleate Boiling (DNB) Correlation),
- RXE-88-102-P Supplement 1, (Application of DNB Correlation to SPC Fuel),
- RXE-89-002-P (VIPRE-01, Core Thermal Hydraulics),
- RXE-91-001 (Transient Analysis),
- RXE-91-002 (Reactivity Anomaly Events),
- RXE-90-007 (Large Break Loss of Coolant Accident (LBLOCA)), and,
- TXX-88306 (Steam Generator Tube Rupture (SGTR) Analysis)

In addition, the parenthetical reference to the specifications contained in the current References 6), 7), 8), 9), 10) and 11) are deleted. These accident analysis methodologies are used to demonstrate that all applicable limits of the safety analyses are met but are not used for the actual determination of a core operating limit. Only methodologies that are used to calculate core operating limits will reference the specification associated with that limit.

B. The increase in Thermal Design Flow will result in a change to Table 2.2-1, Item 12.a, the "**" footnote. The Unit 1 loop design flow will be increased from 95,700 gpm to 99,050 gpm. The minimum indicated RCS flow will be increased from 389,700 gpm to 403,400 gpm in TS 3.2.5. Unit 2 is not affected by this change.

C. The refurbishment of the CPSES Unit 1 pressurizer pressure transmitters and resultant improvement in repeatability allows an increase in the minimum pressurizer pressure value. In TS 3.2.5 and the BASES for TS 3/4.2.5, the minimum indicated pressurizer pressure value will be increased from 2207 psig to 2219 psig. Likewise, the analytical limit, with allowance for measurement uncertainty, will be increased in TS BASES 3/4.2.5 from 2193 psig to 2205 psig. Unit 2 is not affected by this change.

D. Using the methodologies and changes in "A", "B" and "C" above, calculations and analyses have been performed to identify the new core safety limit curves for Unit 1. TS Figure 2.1-1 will be revised to replace the old curves with the new core safety limit curves. In TS BASES 3/4.2.2 and 3/4.2.3, the DNBR generic margin will increase from 9.1% to 18.1% for Unit 1. The margin discussion for Unit 1 will be revised by replacing the specific breakdown presently provided with a discussion similar to that provided for Unit 2.

E. Using (1) the new core safety limit curves from "D" above, (2) the new methodologies from "A" above, and (3) the application of the N-16 power sensor measurement and test equipment (SMTE) allowance to the channel statistical allowance, calculations and analyses have been performed to determine new N-16 related setpoint values and parameters for Unit 1 as noted below:

- In TS Table 2.2-1, Item 7a, for Overtemperature N-16, Total Allowance (TA) will increase from 5.8 to 10.53; "Z" will increase from 3.65 to 6.70; and Sensor Error (S) will change from 1.2% span for ΔT(RTDs) and 0.8% for pressurizer pressure to 1.0% span for N-16 power monitor, 1.10% for T_c RTDs, and 0.76% for pressurizer pressure sensors.

- In TS Table 2.2-1, Item 8 for Overpower N-16, "Z" will increase from 1.93 to 2.05; Sensor Error (S) will increase from zero to 1.0% span for N-16 power monitor and 0.05% for T_c RTDs; and Allowable Value will decrease from 115.1% of rated thermal power to 114.5% of rated thermal power.

- In TS Table 2.2-1, Note 1 for the Overtemperature N-16 Trip Setpoint, the following Terms will be changed as noted:

  - $T_c°$ from 559.6°F to 560.5°F
  - K_1 from 1.078 to 1.150
  - K_2 from 0.00948/°F to 0.0134/°F
  - K_3 from 0.000494/psig to 0.000719/psig
  - q_t-q_b range from -35% and +10% to -65% and +4%
  - Overtemperature N-16 setpoint reduction from 1.22% to 1.81% for each percent that the magnitude of q_t-q_b exceeds -65%
  - Overtemperature N-16 setpoint reduction from 1.40% to 2.26% for each percent that the magnitude of q_t-q_b exceeds +4%

- In TS Table 2.2-1, Note 2, for the Overtemperature N-16 Allowable Value, the maximum amount by which the Trip Setpoint is allowed to exceed the computed Trip Setpoint, is increased from 1.8% to 3.51%.

In summary, the license amendment request includes the additional changes proposed to support CPSES Unit 1, Cycle 4. These changes will accomplish the following: (A) identify the additional analysis methods that will be approved by the NRC for Units 1 and 2; (B) increase the Thermal Design Flow for Unit 1; (C) increase the minimum pressurizer indicated pressure values for Unit 1; (D) provide new Unit 1 core safety limit curves and (E) provide new Unit 1 N-16 related setpoint values which include applying the indicated N-16 power SMTE uncertainty to the channel statistical allowance.

III. ANALYSIS

TU Electric uses NRC approved reload analysis methodologies for CPSES Units 1 and 2 to determine the core safety limits and to meet the applicable limits of the safety analyses. TU Electric will use the departure from nucleate boiling (DNB) correlation, TUE-1, for performing the DNB-related analyses. The TUE-1 DNB correlation has been approved by the NRC for use with Westinghouse and Siemens fuel, as well as in the mixed core configuration of Westinghouse standard fuel assemblies and Siemens fuel assemblies which will be co-resident in the core of CPSES Unit 1 during Cycle 4.

The core safety limit curves are determined to insure that protective actions will be initiated to prevent the core from exceeding the minimum Departure from Nucleate Boiling Ratio (DNBR) limit and to prevent the core exit fluid conditions from reaching saturated conditions. Because a different DNB correlation, TUE-1, is to be used for the CPSES Unit 1, Cycle 4 core configuration, new core safety limits have been calculated.

In addition to the analysis of the core safety limits and the DNB related parameters for the Unit 1, Cycle 4 core configuration (including revised Overtemperature N-16 setpoint equation coefficients), TU Electric also intends to:

1) increase the Reactor Coolant System (RCS) Thermal Design Flow rate,
2) remove the bias on the system pressure uncertainty due to the thermal non-repeatability of the Barton 763 pressure transmitters used to indicate pressurizer pressure, and
3) provide an allowance for the normalization of the N-16 power to the daily plant calorimetric measurement in the statistical setpoint study.

The safety implications of these changes are described below.

A. Incorporate TU Electric's topical reports which have been approved by the NRC.

The referenced methodologies in Section 6.9.1.6b are expanded to include methodologies developed in-house by TU Electric for the performance of core reload licensing analyses. These methodologies can be applied to both CPSES Units 1 and 2, subject to the constraints of the applicable Safety Evaluation Reports (SERs). The reload analysis methodologies have been, or will be, approved by the NRC and can be used to support CPSES Unit 1, Cycle 4 operation. For CPSES Unit 1, Cycle 4, these methodologies will be used to determine the core safety limits and perform the DNB-related portion of the safety analyses. These methodologies will ensure that all applicable limits of the safety analyses are met for the reload core configuration.

The following documents have been approved:

DOCUMENT	APPROVAL DOCUMENT
RXE-88-102-P	NRC SER dated June 11, 1992
RXE-88-102-P Sup. 1	NRC SER dated June 11, 1992
RXE-91-002	NRC SER dated January 19, 1993
RXE-90-007	NRC SER dated April 26, 1993
SGTR analysis	SSER 26

The following documents are under NRC review:

DOCUMENT	SUBMITTAL DOCUMENT(S)
RXE-90-006-P	TXX-91074 dated February 28, 1991
RXE-89-002-P	TXX-89441 dated June 30, 1989
RXE-91-001	TXX-91054 dated February 28, 1991

Upon NRC approval, adding these documents to TS 6.9.1.6b is an administrative change, as the safety implications have already been addressed.

B. Increase in the Unit 1 Thermal Design Flow

Using NRC approved methodologies developed by TU Electric for determining core safety limits, a model of the CPSES Unit 1 mixed core configuration was developed to accurately account for the effects of the different co-resident fuel assembly designs. The Thermal Design Flow (TDF) rate was increased by 3.5% to meet all applicable limits of the safety analysis.

Increasing the TDF rate by 3.5% (from 95,700 gpm per loop to 99,050 gpm per loop) is acceptable because there is approximately a 7.9% difference that currently exists between the actual measured Reactor Coolant System (RCS) flow rate and the TDF rate assumed in CPSES Unit 1, Cycle 3 safety analyses. For CPSES Unit 1, Cycle 4, 3.5% of this difference will be credited in the accident analyses, resulting in the definition of a higher TDF rate. The remaining difference, approximately 4.4%, is sufficient to account for all uncertainties associated with measuring the RCS flow rate (1.8% measurement and 0.5% for the effects of the lower plenum flow anomaly) and the increased RCS flow resistance due to a full core of SPC fuel assemblies. Meeting the minimum measured flow requirement in Technical Specification 3.2.5c will ensure that the TDF rate assumed in the CPSES Unit 1, Cycle 4 safety analyses is valid. Increasing the assumed TDF rate has no impact on the actual measured RCS flow rate.

The loop design (thermal design) flow rate, footnote ** of Table 2.2-1 relating to Item 12 - Reactor Coolant Flow-Low, is changed. The low flow trip setpoint as a percentage of the loop thermal design flow rate is unchanged.

The proposed change to the CPSES Unit 1, Cycle 4 total RCS TDF rate necessitates a change to the minimum indicated total RCS flow rate (from 389,700 gpm to 403,400 gpm, see Technical Specification 3.2.5c) because of the relationship between the TDF flow rate assumed in the safety analyses and the minimum required indicated RCS flow rate. The thermal design flow rate plus measurement uncertainties establishes the Technical Specification minimum RCS flow requirement, thereby ensuring that the TDF rate assumed in the safety analyses remains valid.

Based on previous cycles, sufficient flow exists to ensure that the actual measured RCS flow will exceed the minimum required RCS flow, including allowances for measurement uncertainty and the increased flow resistance corresponding to a full core of Siemens fuel assemblies. ınerefore, sufficient flow exists to ensure that the TDF rate assumed in the safety analyses remains valid. The 1.8% RCS flow measurement uncertainty indicated in footnote ** of Technical Specification 3.2.5 remains valid. The proposed indicated RCS flow requirement limit is calculated by adding the RCS flow uncertainty to the proposed TDF rate for CPSES Unit 1, Cycle 4.

C. Increase in the Unit 1 Minimum Pressurizer Pressure

Technical Specification 3.2.5b and BASES 3/4.2.5 establishes the minimum indicated and the safety analyses values for the Unit 1 pressurizer pressure. CPSES Unit 1 was assessed a penalty on the pressurizer pressure uncertainty associated with the Barton 763 transmitters which provide indication of pressurizer pressure. The penalty was due to the non-repeatability of the transmitters at high temperatures. The penalty was assessed in the safety analyses value for pressurizer pressure which was decreased by the amount of the penalty. TU Electric had all of the Barton 763 pressurizer pressure transmitters refurbished by the vendor prior to initial fuel load. Consequently, the penalty (-12 psi, treated as a bias on pressurizer pressure uncertainty) was no longer required. The removal of the penalty allows TU Electric to raise the analytical limit for pressurizer pressure for the safety analyses (from 2193 psig to 2205 psig). Consequently, the minimum required indicated value, which never included the non-repeatability penalty, is also higher (increasing from 2207 psig to 2219 psig). Removal of the penalty results in the same safety analyses analytical limit for minimum pressurizer pressure as Unit 2 and eliminates a Unit 1/Unit 2 difference. Increasing the minimum pressurizer pressure in the safety analyses has no impact on the normal pressurizer pressure control range.

Limits are placed on the DNB-related parameters to assure that they are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits on pressurizer pressure are consistent with the FSAR initial condition assumptions apnd have been analytically demonstrated adequate for Unit 1, Cycle 4 to maintain a minimum DNBR at or above the safety analysis limit value throughout each analyzed transient.

D. Revision to the Unit 1 Core Safety Limits

The fuel supplier for Comanche Peak Steam Electric Station (CPSES) Unit 1, Cycle 4, and several subsequent CPSES Unit 1 cycles, is different from the current supplier. Beginning with Cycle 4, Siemens Power Corporation (SPC) will supply the nuclear fuel assemblies for Unit 1. During Cycle 4, the Siemens fuel assemblies will be co-resident with existing Westinghouse Standard Fuel Assemblies.

TU Electric has used in-house reload analysis methodologies to determine the core safety limits and to meet applicable limits of the safety analyses for CPSES Unit 1, Cycle 4.

In conjunction with the above methodologies, TU Electric will also use the DNB correlation TUE-1 which has been approved by the NRC for performing DNB-related analyses. This correlation has also been approved by the NRC for the core configuration of Westinghouse standard fuel assemblies and Siemens fuel assemblies, including a mixture of these fuels which will be co-resident in the core of CPSES Unit 1 during Cycle 4. The TUE-1 correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence level that DNB will not occur when the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. Margin has been maintained in the design by meeting safety analysis DNBR limits in performing safety analyses.

The in-house methodologies used by TU Electric to determine the core safety limits are wholly consistent with and represent no change to the Technical Specification 2.1 BASES for Safety Limits.

With NRC approved TU Electric methodologies for determining core safety limits, an increase in the assumed RCS Thermal Design Flow rate, an increase in the minimum assumed pressurizer pressure, and a safety analysis DNBR limit based on the NRC approved TUE-1 DNB correlation, the core safety limits for CPSES Unit 1, Cycle 4 (Technical Specification 2.1, Figure 2.2-1a) have been determined. The core safety limits curves are the loci of points of thermal power, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the safety analysis limit value, and the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid (i. e., no core exit boiling occurs).

The Technical Specification BASES (3/4.2.2 and 3/4.2.3) description of DNBR generic margin is revised due to the change from the W-3 R-grid Critical Heat Flux (CHF) correlation (Westinghouse methodology) to the TUE-1 Departure from Nucleate Boiling (DNB) correlation (TU Electric methodology) for the Unit 1, Cycle 4 DNB analyses. The generic margin was established for these two correlations by different methods.

The current method of allocating the DNBR generic margin for Unit 1 quantifies the change in the DNBR predicted by the W-3 R-grid CHF correlation due to various modeling conservatisms. The total change in the DNBR due to the selected modeling conservatisms is then presented as a percent of the calculated DNBR. This approach is used by Westinghouse in arriving at the 9.1% DNBR generic margin for Unit 1. Westinghouse determined the impact on the calculated DNBR as a result of the use of conservative values for the grid spacing correction coefficient and the thermal diffusion coefficient and as a result of modeling pitch reduction in the hot subchannels. Westinghouse then credited the change in DNBR due to these modeling conservatisms as well as their use of a conservative DNBR multiplier and DNBR design limit to calculate the DNBR generic margin.

The method of allocating the DNBR generic margin used by TU Electric for Unit 1 is similar to the method used by Westinghouse in allocating the DNBR generic margin for Unit 2 for which the WRB-1 CHF correlation is used. This method simply sets a DNBR limit to be utilized in the safety analyses (i.e., the DNBR safety analysis limit) above the 95/95 DNBR correlation limit (i.e., the DNBR design limit) by an amount which will be used to offset known and potential DNBR penalties. The amount by which the DNBR safety analysis limit exceeds the DNBR design limit is determined based on evaluations of the known DNBR penalties to provide assurance that sufficient margin will exist in the analysis results to offset these penalties. The TU Electric method of allocating DNBR generic margin results in a generic margin of 18.1% for CPSES Unit 1, Cycle 4 above the TUE-1 95/95 DNBR correlation limit. The TU Electric DNB analysis methods and TUE-1 DNB correlation have been approved by the NRC or are presently under review.

E. Revision to Unit 1 Overtemperature and Overpower N-16 Reactor Trip Setpoints, Parameters and Coefficients

The Reactor Trip System setpoint limits specified in Technical Specification 2.2, Table 2.2-1 are the nominal values at which the reactor trips are set for each functional trip. The trip setpoints have been selected to ensure that the core and Reactor Coolant System (RCS) are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences. The Overtemperature and Overpower N-16 trip setpoints are reactor trips which help protect the core and RCS from exceeding their safety limits.

The Overtemperature N-16 trip provides core protection to prevent DNB and core exit saturation for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that: the transient is slow with respect to piping delays from the core to the N-16 detectors; the pressure is within the range between the Pressurizer High and Low pressure reactor trip setpoints; and the power is less than the Overpower N-16 trip setpoint. The Overtemperature N-16 setpoint is automatically varied with coolant temperature, pressurizer pressure, and axial power distribution.

With a normal operation axial power distribution, the Overtemperature N-16 reactor trip limit is always below the core safety limit. If the axial flux difference is greater than design, as indicated by the difference between top and bottom power range neutron flux detectors, the Overtemperature N-16 reactor trip setpoint is automatically reduced according to the notations (Note 1) in Technical Specification 2.2, Table 2.2-1. This provides protection consistent with the core safety limits.

Because the core safety limits have changed for CPSES Unit 1, Cycle 4, the Overtemperature N-16 reactor trip setpoint must be recalculated to ensure that the core safety limits remain protected by this reactor trip function. This calculation has been performed in accordance with the methods developed by TU Electric and is consistent with the BASES (BASES 2.2.1) for the Overtemperature N-16 reactor trip.

The Overtemperature N-16 reactor trip setpoint calculation includes the calculation of the K_1, K_2, K_3 and $f_1(\Delta q)$ coefficients for the equation shown in Technical Specification 2.2, Table 2.2-1, Note 1. The $f_1(\Delta q)$ terms (the range for qt-qb and the Overtemperature reductions when exceeding that range) are a function of axial flux difference and account for variations in the core axial power distributions. The K_1, K_2, and K_3 safety settings are determined assuming a fixed reference (normal operations) axial power distribution; then, the compensation terms $f_1(\Delta q)$ are determined accounting for variations in the axial power distribution during accident conditions. The combination of these parameters in the Overtemperature N-16 reactor trip setpoint equation is designed to provide core safety limit protection by preventing DNB and core exit saturation for all combinations of pressure, power, coolant temperature, and axial power distribution.

The value of T_c° (reference cold leg temperature at rated thermal power) for the Overtemperature N-16 trip setpoint equation in Technical Specification 2.2, Table 2.2-1, Note 1 is also changed. Due to the increase in Reactor Coolant System Thermal Design Flow rate (see Section III.B, above), the ΔT across the reactor vessel must decrease in order to maintain the same core power and reactor vessel average temperature. Performing an energy balance at rated thermal power with the higher thermal design flow rate, a new value of T_c° is determined.

Once the safety analysis values for the Overtemperature N-16 reactor trip setpoint have been determined, the instrumentation trip setpoints are determined. These trip set points are defined by the Total Allowance (TA), Z, Sensor Errors (S), Trip Setpoint and Allowable Value, in Technical Specification Table 2.2-1. The methodology to derive the Overtemperature N-16 reactor trip setpoints in Table 2.2-1 is based upon a statistical combination of all of the uncertainties in the channels to arrive at a total uncertainty. Sensor and rack instrumentation used in these channels are expected to be capable of operating within the allowances of the uncertainty magnitudes. The total uncertainty plus additional margin is applied in a conservative direction to the safety analysis trip setpoint value to arrive at the nominal trip setpoint value provided in TS Table 2.2-1. Because the safety analysis value for the Overtemperature N-16 reactor trip setpoint is changed, the nominal and allowable values also change. However, they are still calculated in a manner which is consistent with the current values.

In determining the CPSES Unit 1, Cycle 4 Overtemperature N-16 Reactor Trip System instrumentation trip setpoint, an operational enhancement is added. Technical Specification 4.3.1.1 (Note 2 to Table 4.3-1) requires that the indicated N-16 power be readjusted if the indicated N-16 power differs by more than ± 2 percent of rated thermal power (RTP) from the power calculated from the daily power calorimetric measurement. Currently, the sensor measurement and test equipment (SMTE) allowance for the indicated N-16 power ($\pm 1.5\%$ of RTP) is subtracted from the allowable ± 2 percent of RTP difference. This reduces the allowed tolerance between the N-16 power indication and the calorimetric power to ± 0.5 percent of RTP. Treating the indicated N-16 power SMTE in this way results in an unnecessarily high N-16 readjustment frequency. Because readjustment requires entry into the Westinghouse 7300 process cabinets, which increases the potential for personnel errors, it is desirable to minimize the required frequency of the readjustments. To reduce this readjustment frequency, the SMTE allowance associated with the indicated N-16 power will be included in the channel statistical allowance calculation of the Overtemperature N-16 reactor trip setpoint (which uses the N-16 power signal) instead of being subtracted from the allowable power difference. The inclusion of the additional SMTE uncertainty will increase the total statistical combination of all uncertainties associated with the channel. The increase in the channel total uncertainty is accounted for in the determination of the nominal setpoint presented in Table 2.2-1 for the Overtemperature N-16 reactor trip function. The change in the "S" term only affects the determination of channel operability and has no effect on the nominal or allowable setpoints presented in the table.

The change to include the indicated N-16 power SMTE in the statistical treatment of the nominal Overtemperature N-16 reactor trip setpoint is acceptable because the Overtemperature N-16 measurements continue to be made with an acceptable level of accuracy which will assure that the assumptions in the accident analyses are valid. Further, the change will reduce the required frequency of N-16 power readjustment which in turn will reduce the potential for personnel error when working with sensitive process equipment. This change will also make the Unit 1 requirements consistent with Unit 2.

The CPSES Unit 1, Cycle 4 Overtemperature N-16 reactor trip setpoints are also sufficiently high such that the operational effects of the upper plenum flow anomaly on turbine runbacks or reactor trips will be minimized; thereby reducing the potential for challenges to the plant safety systems.

Since the N-16 signal is also part of the Overpower N-16 Reactor Trip Setpoint, Overpower N-16 Reactor Trip Setpoint values for Total Allowance (TA), Z, Sensor Error (S), and Allowable Value (AV) were recalculated to include the SMTE allowance discussed above. This change will make the Unit 1 Overpower N-16 reactor trip function calibration procedures consistent with CPSES Unit 2 and with the CPSES Unit 1 Overtemperature N-16 reactor trip setpoint calculation. Further, this change will eliminate Unit 1/Unit 2 differences.

No change to the safety analysis value of the Overpower N-16 reactor setpoint occurred and instrument uncertainties are properly accounted for in determining the trip instrumentation values of TA, Z, S, and AV.

SUMMARY

To summarize, TU Electric proposes using its in-house, NRC approved reload analysis methodologies to determine the core safety limits and perform DNB-related analyses for the mixed core configuration in CPSES Unit 1, Cycle 4. As a result of the new core safety limits, the Overtemperature N-16 trip setpoints are being recalculated. In performing these analyses, the Reactor Coolant System Thermal Design Flow rate is increased and the bias on the system pressure uncertainty due to the thermal non-repeatability of the pressurizer pressure transmitters is removed. Also, an operational enhancement is added to statistically include the sensor measurement and test equipment (SMTE) allowance associated with the N-16 power indication into the statistical setpoint determination of the Reactor Trip System Instrumentation trip setpoints which will reduce the required frequency of N-16 power readjustment. The Unit 1, Cycle 4 analyses have been performed using methodologies which will be NRC approved and satisfy the applicable safety analyses limits.

IV. SIGNIFICANT HAZARDS CONSIDERATIONS ANALYSIS

TU Electric has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10CFR50.92(c) as discussed below:

Does the proposed change:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated?

A. Revision to incorporate TU Electric's topical reports

The NRC assures that appropriate core operating limits are applied by requiring that the operating limits be determined using NRC approved analytical methods. These approved methods are listed in TS Section 6.9.1.6b. TU Electric has developed the in-house analysis capability to determine and confirm core operating limits. The TU Electric methodology has been documented in a series of TU Electric submittals which get approved by the NRC. This TS revision adds the TU Electric documents (which are NRC approved or will be approved prior to Unit 1, Cycle 4) to the list of acceptable methods.

Because the revision is administrative only, it cannot directly affect the probability or the consequences of any previously evaluated accident. The core operating limits are set to assure that relevant plant parameters are maintained such that potential accidents are within the bounds of the accident analyses. Because the applicable limits of the safety analysis will continue to be met, there is no significant impact on the consequences of an accident previously evaluated. In addition, since the core operating limits do not affect any accident initiators, the change has no impact on the probability of any accident previously analyzed.

B. Increase in Unit 1 Thermal Design Flow

This revision increases the Unit 1 Thermal Design Flow rate assumed in the safety analyses by 3.5%. The actual core flow is unchanged and is approximately 7.9% higher than the value assumed in previous accident analyses. The remaining 4.4% flow is sufficient to account for all uncertainties associated with the core flow measurement. Since this change only involves analysis methodology and does not affect actual core flow, it does not increase the actual probability or consequences of any postulated accident.

When considered separately, increasing the thermal design flow is a conservative change. Although there is no impact on the initiation of any postulated accidents, the potential severity of the affected accidents is typically less when flow is increased. In general, the increased ability to remove heat from the fuel will reduce the peak temperature seen by the fuel and reduce the potential for undesirable boiling conditions. Thus, the increase in the thermal design flow will not increase the probability or consequences of an accident previously analyzed.

C. Increase in Unit 1 Minimum Pressurizer Pressure

The CPSES Unit 1 safety analysis value for pressurizer pressure was assessed a penalty (-12 psi, treated as a bias on pressurizer pressure uncertainty) due to the non-repeatability of the Barton 763 pressure transmitters at high temperatures. This penalty was assessed so that the same control range could be retained for pressurizer pressure. These transmitters were refurbished and now have acceptable repeatability. The penalty can be removed from the safety analysis value without affecting the normal control range. Removing the penalty increases the assumed safety analysis value. Because the same control range is being maintained and the transmitters were previously refurbished, it is expected that actual pressurizer pressure range that is maintained in the unit will not be changed as a result of this change in the safety analysis value. Thus, when considering normal plant operations, this change by itself is not expected to have any impact on the actual probability or consequences of an accident.

In general, increasing the required minimum pressurizer pressure is a conservative change. An increase in pressure delays the onset of the various modes of boiling and allows better heat transfer which can be expected to result in lower peak fuel temperatures. Thus, the increase in minimum indicated pressurizer pressure will not change the probability of an accident but will tend to decrease the severity of the analysis results for accidents previously analyzed.

D. Revision to the Unit 1 Core Safety Limits

Analyses of reactor core safety limits are required as part of reload calculations for each cycle. TU Electric has performed in-house analyses of the Unit 1, Cycle 4 core to determine the reactor core safety limits. The newer methodologies and safety analysis values result in new operating curves which, in general, permit plant operation over a broader range of acceptable conditions. This increase means that if a transient were to occur with the plant operating at the limits of the new curve, the transient might be more severe than if the plant were operating within the bounds of the old curve. However, since the new curves were developed using approved methodologies which are wholly consistent with and do not represent a change in the Technical Specification bases for safety limits, all applicable postulated transients will continue to be properly mitigated. As a result, there will be no significant increase in the consequences, as determined by accident analyses, of any accident previously evaluated.

E. Revision to Unit 1 Overtemperature and Overpower N-16 Reactor Trip Setpoints, Parameters and Coefficients

As a result of changes discussed in paragraphs "A", "B", "C" and "D" above, the Overtemperature N-16 reactor trip setpoint has been recalculated. An additional uncertainty allowance has also been added to the statistical combination of uncertainties used to determine both the Overtemperature and Overpower N-16 reactor trip setpoints and parameters. These trip setpoints help ensure that the core safety limits are maintained and that all applicable limits of the safety analysis are met.

Based on the calculations performed, the safety analysis value for Overtemperature N-16 reactor trip setpoint has increased. This essentially means if a transient were to occur, the actual course of the transient could be slightly more severe. However, the analyses performed show that, using the new methodologies, all core safety limits are met and all applicable limits of the safety analysis are met. The safety analysis value for Overpower N-16 remains unchanged. Both of these parameters have setpoints to allow the mitigation of postulated accidents and have no impact on accident initiation. Therefore, the changes in safety analysis values do not involve an increase in the probability of an accident and, based on satisfying the core safety limits and all applicable safety analysis limits, there is no significant increase in the consequences of any accident previously evaluated.

In addition, the changes result in setpoint values which offer safety benefits. By including an additional allowance to the combination of uncertainties used to determine these setpoints, the required frequency of N-16 power indication readjustments has been reduced. Not only does this reduce the wear on the hardware but it also reduces the potential for personnel error while working on sensitive safety related process equipment. The higher Overtemperature N-16 setpoints offer another operational improvement. The risk of turbine runbacks or reactor trips due to upper plenum flow anomalies will be minimized, thus reducing potential challenges to the plant safety system. A final benefit is that the new methods for considering N-16 setpoints and values will be consistent with Unit 2, which reduces the potential for personnel error due to unit differences.

Considering both the safety analysis impact and the benefits described above, the changes in N-16 setpoints and parameters probably reduce the probability of an accident and do not significantly increase the consequences of an accident previously evaluated.

SUMMARY

The changes in the amendment request provide new methodologies, changes in safety analysis values, new core safety limits and new N-16 setpoint and parameter values to assure that all applicable safety analysis limits have been met. The potential for an accident to occur has been reduced and there has been no significant impact on the consequences of any accident previously evaluated.

2. Create the possibility of a new o[illegible]ent kind of accident from any accident previously evalu[illegible]

The proposed changes involve the use of new analysis methodologies, revised safety analysis values, and the calculation of new core safety limits and reactor trip setpoints. As such, the changes play an important role in the analysis of postulated accidents but none of the changes effect plant hardware or the operation of plant systems in a way that could initiate an accident. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) Involve a significant reduction in a margin of safety?

In reviewing and approving the methods used for safety analyses and calculations, the NRC has approved the safety analysis limits which establish the margin of safety to be maintained. While the actual impact on safety is discussed in response to question 1, the impact on margin of safety is discussed below.

A. Revision to incorporate TU Electric's topical reports

The use of the methodology contained in the TU Electric topical reports does not in itself have any impact on the margin of safety. Satisfaction of event-specific acceptance criteria provide the margin of safety. The methodologies demonstrate, in a conservative manner (through input selection), that the event acceptance criteria are satisfied.

The proposed methods developed by TU Electric have been approved by the NRC or approval is expected soon. When approved, the methods identify the methodologies, correlations, etc. that may be used by TU Electric and establish the applicable safety analysis limits that must be met. Therefore, including these new methods in the TS does not change the margin of safety, [illegible] merely incorporates the previously approved margin of safety in the TS.

B. Increase in the Unit 1 Thermal Design Flow

In performing the DNB-related analyses, the Reactor Coolant System Thermal Design Flow rate assumed in these analyses is increased by 3.5 percent to insure that all applicable limits of the safety analysis are met. The TS 3.2.5 limit for this parameter will be changed to insure that it is maintained within the normal steady-state envelope of operation assumed in the transient and accident safety analyses (i.e., ensuring that the Thermal Design Flow rate assumed in the safety analyses remains valid). The Technical Specification limits are consistent with the initial safety analysis assumption (plus uncertainties) and have been analytically demonstrated to be adequate to maintain a minimum DNBR at or above the safety analysis DNBR limit throughout each analyzed transient. Because the 95/95 DNBR acceptance criteria is met with the proposed change and assumptions of the safety analyses are maintained valid by the Technical Specification limits, there is no change in a margin of safety.

C. Increase in the Unit 1 Minimum Pressurizer Pressure

The removal of the bias on the CPSES Unit 1 pressurizer pressure due to refurbishment of the pressure transmitters by the vendor has allowed TU Electric to increase the minimum pressurizer pressure value used in the safety analysis. The TS 3.2.5 limit for this parameter will be changed to ensure that it is maintained within normal steady state envelope of operation assumed in the transient and accident analyses (i.e., ensuring pressurizer pressure assumed in accident analyses remains valid). The Technical Specification limits are consistent with the safety analysis assumptions (plus uncertainties) and have been analytically demonstrated to be adequate to maintain a minimum DNBR at or above the safety analysis limit throughout each analyzed transient. Because the 95/95 DNBR acceptance criteria is met with the proposed change and assumptions of the safety analyses are maintained valid by the Technical Specification limits, there is no change in the margin of safety.

D. Revision to the Unit 1 Core Safety Limits

The TU Electric reload analysis methods (see A above) have been used to determine new core safety limits. All applicable safety analysis limits have been met. The methods used are wholly consistent with TS BASES 2.1 which is the bases for the safety limits. In particular, the curves assure that for Unit 1, Cycle 4, the calculated DNBR is no less than the safety analysis limit and the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

In conjunction with the core safety limit methodology, the NRC approved TUE-1 DNB correlation is used for performing DNB-related analyses. This correlation will be applied to the core configuration of CPSES Unit 1, Cycle 4 and future core configurations. The TUE-1 correlation DNBR limit is established such that there is a 95 percent probability with 95 percent confidence level that DNB will not occur when the minimum DNBR for the limiting fuel is greater than or equal to the TUE-1 correlation DNBR limit. This 95/95 criteria defines the "margin of safety" for the DNB-related analysis and remains valid even though the DNB correlation and associated correlation limit are changed. Margin is retained in the DNB-related analysis for known and potential effects such as hydraulic differences between the two co-resident fuel assembly designs and the presence of the Reactor Coolant System lower plenum flow anomaly. The TUE-1 correlation DNBR limit plus margin constitutes the safety analysis DNBR limit. The accident analyses are performed to ensure that the safety analysis DNBR limit acceptance criteria are satisfied. Because the 95/95 DNBR acceptance criteria remains valid and continues to be satisfied, no change in a margin of safety occurs.

E. Revision to Unit 1 Overtemperature and Overpower N-16 Reactor Trip Setpoints, Parameters and Coefficients

Because the core safety limits for CPSES Unit 1, Cycle 4 are recalculated, the Reactor Trip System instrumentation setpoint values for the Overtemperature N-16 reactor trip setpoint which protect the core safety limits must also be recalculated. The Overtemperature N-16 reactor trip setpoint helps prevent the core and Reactor Coolant System from exceeding their safety limits during normal operation and design basis anticipated operational occurrences. The design basis analyses in Chapter 15 of the CPSES Final Safety Analysis Report (FSAR) affected by the change in the safety analysis value for the CPSES Unit 1 Overtemperature N-16 reactor trip setpoint are the Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Full Power (FSAR Section 15.4.2), and Inadvertent Opening of a Pressurizer Safety or Relief Valve (FSAR Section 15.6.1). These affected events have been re-analyzed with the revised safety analysis value for the Overtemperature N-16 reactor trip setpoint to demonstrate compliance with event specific acceptance criteria. Because all event acceptance criteria are satisfied, there is no degradation in a margin of safety.

The nominal Reactor Trip System instrumentation setpoints values for the Overtemperature N-16 reactor trip setpoint (Technical Specification Table 2.2-1) are determined based on a statistical combination of all of the uncertainties in the channels to arrive at a total uncertainty. The total uncertainty (which includes the addition of the indicated N-16 power SMTE allowance discussed below) plus additional margin is applied in a conservative direction to the safety analysis trip setpoint value to arrive at the nominal and allowable values presented in Technical Specification Table 2.2-1. Meeting the requirements of Technical Specification Table 2.2-1 assures that the Overtemperature N-16 reactor trip setpoint assumed in the safety analyses remains valid. The CPSES Unit 1, Cycle 4 Overtemperature N-16 reactor trip setpoint is higher than previous cycles which provides more operational flexibility to withstand mild transients without initiating automatic protective actions. Although the setpoint is higher, the Reactor Trip System instrumentation setpoint values for the Overtemperature N-16 reactor trip setpoint are consistent with the safety analysis assumption which has been analytically demonstrated to be adequate to meet the applicable event acceptance criteria. Thus, there is no reduction in a margin of safety.

The inclusion of the additional SMTE uncertainty for indicated N-16 power into the channel statistical allowance will increase the total statistical combination of all uncertainties associated with the channels for the Overtemperature N-16 and Overpower N-16 reactor trip setpoints. The increase in the channel total uncertainty is accounted for in determination of the nominal setpoint presented in Table 2.2-1 for these reactor trip functions. The safety analysis values for the Overtemperature N-16 and Overpower N-16 reactor trip setpoints which use the indicated N-16 power are not affected by this enhancement. The change in the "S" term only affects the determination of channel operability and has no impact on the nominal setpoints presented in the Technical Specification Table 2.2-1. Incorporating the indicated N-16 power SMTE allowance into the statistical treatment of the Overtemperature N-16 and Overpower N-16 reactor trip setpoint does not reduce a margin of safety because the nominal and allowable setpoints continue to be determined in such a way as to assure that the assumptions in the accident analyses are valid.

SUMMARY

The proposed changes to the CPSES Technical Specifications involve using NRC-approved (or soon to be approved) licensing analysis methods developed by TU Electric to determine the Technical Specification core safety limits and perform DNB-related analysis for CPSES Unit 1, Cycle 4. The DNB-related analyses are performed by TU Electric using a qualified, state-of-the-art departure from nucleate boiling (DNB) correlation, TUE-1, which has also been approved by the NRC for the CPSES Unit 1, Cycle 4 core configuration. In performing these analyses, the Reactor Coolant System Thermal Design Flow rate is increased by 3.5 percent and the removal of the bias on the system pressure uncertainty due to the thermal non-repeatability of the pressurizer pressure transmitters is credited. Because the core safety limits for CPSES Unit 1, Cycle 4 are recalculated, the Reactor Trip System instrumentation setpoints values for the Overtemperature N-16 reactor trip setpoint which protect the core safety limits are also recalculated. In conjunction with the Overtemperature N-16 reactor trip setpoint calculation, an operational enhancement is added to statistically include the sensor measurement and test equipment (SMTE) allowance associated with the indicated N-16 power into the statistical setpoint determination of the Reactor Trip System Instrumentation trip setpoints.

Using the NRC approved TU Electric methods, the core safety limits are determined such that all applicable limits of the safety analyses are met, particularly the 95/95 DNBR limit. The Technical Specification 3.2.5 limits for the DNB Parameters insure the assumptions in the safety analyses remain valid. Because the applicable event acceptance criteria continue to be met, there is no significant reduction in the margin of safety.

Based on the above evaluations, TU Electric concludes that the activities associated with the above described changes present no significant hazards consideration under the standards set forth in 10CFR50.92(c) and, accordingly, a finding by the NRC of no significant hazards consideration is justified.

V. ENVIRONMENTAL EVALUATION

TU Electric has evaluated the proposed changes and has determined that the changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10CFR51.22(c)(9). Therefore, pursuant to 10CFR51.22(b), an environmental assessment of proposed change is not required.