TU ELECTRIC

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C. Lance Terry
Group Vice President, Nuclear

November 21, 1995

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 95-008
UNIT 2 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 reload analyses for CPSES Unit 2, Cycle 3 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 an administrative enhancement is included in the footnotes of the RCS flow - low reactor trip function setpoint. The administrative change is applicable to both Units.

Attachment 1 is the required affidavit. Attachment 2 provides a detailed description of the proposed changes, a safety analysis of the proposed changes and TU Electric's determination that the proposed changes do not involve a significant hazard consideration. Attachment 3 provides the affected Technical Specification pages marked-up to reflect the proposed changes.

TU Electric requests approval of this proposed license amendment by February 28, 1996, 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.

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Should you have any questions, please contact Mr. Jimmy Seawright at (214) 812-4375.

Sincerely,

C. L. Terry

C. L. Terry

By: Roger D. Walker
Roger D. Walker
Regulatory Affairs Manager

JDS/grp
Attachments: 1. Affidavit
2. Description and Assessment
3. Affected Technical Specification pages as revised by all approved license amendments

c - Mr. L. J. Callan, Region IV
Mr. T. J. Polich, NRR
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Attachment 1 to TXX-95288
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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

Roger D. Walker being duly sworn, hereby deposes and says that he is Regulatory Affairs Manager for TU Electric, the licensee herein; that he is duly authorized to sign and file with the Nuclear Regulatory Commission this License Amendment Request 95-008; 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 DALLAS)

Subscribed and sworn to before me, on this 21st day of November, 1995.

Gayle R Peck

Notary Public

NOTARY PUBLIC STATE OF TEXAS
Gayle R. Peck
Notary Public, State of Texas
My Comm. Expires 01/06/98

ATTACHMENT 2 to TXX-95288

DESCRIPTION AND ASSESSMENT

DESCRIPTION AND ASSESSMENT

1. BACKGROUND

The fuel supplier for Comanche Peak Steam Electric Station (CPSES) Unit 2, has changed. Beginning with the third operating cycle (Cycle 3), Siemens Power Corporation (SPC) will supply the nuclear fuel assemblies. For Cycle 3, the Siemens fuel assemblies will be co-resident with the existing Westinghouse Fuel Assemblies. The fuel supplier for CPSES Unit 1, for the current and previous fuel cycles, has already changed from Westinghouse to Siemens.

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 2, Cycle 3 to demonstrate that all applicable limits of the safety analysis are met. These methodologies have already been applied for CPSES Unit 1 and incorporated into the Technical Specification administrative section 6.9.1.6b.

In addition, the minimum required flow will be increased. Currently, the actual RCS flow is approximately 6.6% higher than the minimum required flow rate assumed in the CPSES Unit 2, Cycle 2 accident analyses. For CPSES Unit 2, Cycle 3, TU Electric proposes crediting 3.6% of this flow in the accident analyses, resulting in the definition of a higher RCS minimum required flow rate.

Consistent with the use of TU Electric's methodologies, the TUE-1 DNBR correlation will be used to demonstrate compliance with the DNB acceptance limit (versus the WRB1 correlation upon which the current analyses are based). The Reactor Core Safety Limits, shown in Figure 2.1-1b, must be revised to incorporate the effects of both the new DNBR correlation and the increased RCS flow.

With new Unit 2, Cycle 3 reactor core safety limits, the Overtemperature N-16 reactor trip setpoints must be recalculated to provide adequate protection of the new core safety limits.

In addition, TU Electric proposes to remove the footnotes defining the "Loop minimum measured flow" and the "Loop design flow" associated with Functional Units 12.a and 12.b of Table 2.2-1. The footnote associated with Functional Unit 12.b must be revised for Unit 2 Cycle 3. Rather than continue to revise this footnote (and the corresponding Unit 1 footnote) on a cycle-specific basis, TU Electric has elected to adopt the attributes of NUREG-1431, Revision 1 for both units and delete the footnotes for both units. The specification of a "thermal design flow" or a "minimum measured flow," and the exact values of these flows, are not relevant to the manner in which the RCS flow - low trip setpoint is developed and implemented.

These changes to the plant Technical Specifications are specific to CPSES and are needed to support Unit 2, Cycle 3 operation. The revision to

Functional Unit 12.a of Table 2.2-1 is made for consistency with the proposed Unit 2 revision.

II. DESCRIPTION OF TECHNICAL SPECIFICATION CHANGE REQUEST

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

A. The minimum indicated RCS flow will be increased from 395,200 gpm to 408,000 gpm in Technical Specification 3/4.2.5.

B. Using the methodologies specified in Technical Specification 6.9.1.6b and the revised RCS flow from "A" above, calculations and analyses have been performed to identify the new reactor core safety limit curves for Unit 2. Technical Specification Figure 2.1-1b will be revised to replace the old curves with the new reactor core safety limit curves.

C. Using the new reactor core safety limit curves from "B" above, calculations and analyses have been performed to determine new N-16 related setpoint values and parameters for Unit 2 as noted below:

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

- $T_c°$ from 560.3°F to 560.8°F - constant Tavg
- K_2 from 0.016856/°F to 0.0138/°F
- K_3 from 0.000898/psig to 0.00072/psig
- q_t-q_b range from -52% and +5.5% to -65% and +2.5%
- Overtemperature N-16 setpoint reduction from 2.15% to 1.86% for each percent that the magnitude of q_t-q_b exceeds -65% (current value -52%)
- Overtemperature N-16 setpoint reduction from 2.17% to 1.65% for each percent that the magnitude of q_t-q_b exceeds +2.5% (current value +5.5%)

In Technical Specification 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 decreased from 2.85% to 1.88%.

D. The footnotes "**" and "***," used with Functional Units 12.a and 12.b in Table 2.2-1 will be deleted and the Trip Setpoint and Allowable Value will be expressed in % of instrument span.

In summary, the license amendment request includes changes proposed to support CPSES Unit 2, Cycle 3. These changes will increase the minimum value of Unit 2 RCS loop flow rate, provide new Unit 2 reactor core safety limit curves, provide new Unit 2 Overtemperature N-16 related setpoint values, and remove unnecessary detail from the RCS flow - low reactor trip setpoint. The latter item is also applied to Unit 1 for consistency between the two units.

III. ANALYSIS

TU Electric uses NRC approved reload analysis methodologies for CPSES Units 1 and 2 to determine the reactor 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 fuel assemblies and Siemens fuel assemblies which will be co-resident in the core of CPSES Unit 2 during Cycle 3.

The reactor 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 2, Cycle 3 reactor core configuration, new reactor core safety limits have been calculated.

In addition to the analysis of the reactor core safety limits and the DNB related parameters for the Unit 2, Cycle 3 reactor core configuration (including revised Overtemperature N-16 setpoint equation coefficients), TU Electric also intends to increase the minimum required Reactor Coolant System (RCS) flow rate.

Associated with the change to the minimum required RCS flow rate is the deletion of footnotes in the Table of reactor trip setpoints which define RCS flow rate. This information is not necessary to preserve the consistency with the accident analyses.

The safety implications of these changes are described below.

A. Increase in the Unit 2 required RCS flow rate

Using NRC approved methodologies developed by TU Electric for determining reactor core safety limits, a model of the CPSES Unit 2 mixed core configuration was developed to accurately account for the effects of the different co-resident fuel assembly designs. The minimum required RCS flow rate was increased by 3.6% to provide additional margin which may be used to demonstrate compliance with all applicable limits of the safety analysis.

Increasing the minimum required RCS flow rate by 3.6% is acceptable because approximately a 6.6% difference currently exists between the actual measured Reactor Coolant System (RCS) flow rate and the flow rate assumed in CPSES Unit 2, Cycle 2 safety analyses. For CPSES Unit 2, Cycle 3, 3.6% of this difference will be credited in the accident analyses, resulting in the definition of a higher value of minimum required flow rate. The remaining difference 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 RCS flow requirement in Limiting Condition for Operation (LCO) 3.2.5c will ensure that the RCS flow rate assumed in the CPSES Unit 2, Cycle 3 safety analyses is valid. Increasing the assumed RCS flow rate has no impact on the actual measured RCS flow rate.

B. Revision to the Unit 2 Reactor Core Safety Limits

The fuel supplier for Comanche Peak Steam Electric Station (CPSES) Unit 2, Cycle 3, and several subsequent CPSES Unit 2 cycles, is different from the current supplier. Beginning with Cycle 3, Siemens Power Corporation (SPC) will supply the nuclear fuel assemblies for Unit 2. During Cycle 3, the Siemens fuel assemblies will be co-resident with existing Westinghouse fuel assemblies.

TU Electric has used in-house reload analysis methodologies to determine the reactor core safety limits and to meet applicable limits of the safety analyses for CPSES Unit 2, Cycle 3.

In conjunction with the above methodologies, TU Electric will also use the TUE-1 DNB correlation which has been approved by the NRC for performing DNB-related analyses (see Technical Specification 6.9.1.6b). This correlation has also been approved by the NRC for the core configuration of Westinghouse fuel assemblies and Siemens fuel assemblies, including a mixture of these fuels which will be co-resident in the core of CPSES Unit 2 during Cycle 3. 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.

The in-house methodologies used by TU Electric to determine the reactor 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 reactor core safety limits, an increase in the assumed RCS flow rate, and a safety analysis DNBR limit based on the NRC approved TUE-1 DNB correlation, the core safety limits for CPSES Unit 2, Cycle 3 (Technical Specification 2.1, Figure 2.1-1b) 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).

C. Revision to Unit 2 Overtemperature 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 N-16 trip function initiates a reactor trip which helps 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 operations axial power distribution, the Overtemperature N-16 reactor trip limit is always below the reactor core safety limit. If the axial flux difference is greater than a reference distribution, 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 reduction provides protection consistent with the reactor core safety limits.

Because the reactor core safety limits have changed for CPSES Unit 2, Cycle 3, the Overtemperature N-16 reactor trip setpoint must be recalculated to ensure that the reactor core safety limits remain protected by this reactor trip function. This calculation has been performed in accordance with the methods developed by TU Electric (see Technical Specification 6.9.1.6b) 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 $q_t - q_b$ 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 coefficients are determined assuming a fixed reference (normal operations) axial power distribution; then, the compensation terms $f_1(\Delta q)$ are determined to account 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 reactor 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 the minimum required RCS flow rate) 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 the minimum required Reactor Coolant System 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 value of the minimum required flow rate, a new value of T_c^o 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 setpoints are defined by the 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 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 Technical Specification 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.

The CPSES Unit 2, Cycle 3 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.

D. <u>Deletion of the footnotes associated with the RCS flow - low reactor trip function setpoints</u>

In order to eliminate unnecessary information from the Technical Specifications, thereby reducing the potential for cycle-specific changes, the footnotes associated with the RCS flow - low reactor trip function setpoints are deleted and the Trip Setpoint and Allowable Value will be expressed in % of instrument span. This change is consistent with the Westinghouse Improved Standard Technical Specifications (NUREG-1431, Revision 1).

In theory, with the current language of Technical Specification 2.2, Table 2.2-1, Functional Units 12.a and 12.b, the reactor trip setpoint on low RCS flow could be set such that the trip setpoint corresponded to 90% of the minimum RCS flow rate assumed in the accident analyses. Because the minimum RCS flow rate assumed in the accident analyses is less than the actual flow rate, the setpoint could potentially be set at some value less than 90% of instrument span. In practice, the trip setpoint is set at 90% of the instrument span, where the actual RCS loop flow corresponds to 100% (or perhaps slightly less) of the instrument span. The actual RCS flow is determined to be greater than the value assumed in the accident analysis through compliance with Technical Specification 3.2.5. Even though the deletion of the footnotes has no effect on the current practice, in theory, it could result in RCS flow - low setpoints which are more restrictive than allowed with the current specifications. This restriction is conservative relative to the accident analysis assumptions, and has no impact with respect to actual plant operation. Due to the current method used to set the RCS flow - low setpoint, this change is essentially administrative in nature.

This change is proposed for Unit 2 in lieu of a cycle-specific revision to footnote "***," and is proposed for Unit 1 to maintain consistency between the units.

<u>SUMMARY</u>

To summarize, TU Electric proposes using its in-house, NRC approved reload analysis methodologies to determine the reactor core safety limits and perform DNB-related analyses for the mixed core configuration in CPSES Unit 2, Cycle 3. As a result of the new reactor core safety limits, the Overtemperature N-16 trip setpoints are being recalculated. In performing these analyses, the Reactor Coolant System minimum required flow rate is increased. The Unit 2, Cycle 3 analyses have been performed using methodologies which are NRC approved and satisfy the applicable safety analyses limits. Finally, an administrative change is proposed which would delete the footnotes associated with the RCS flow - low reactor trip setpoint.

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:

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   A. Increase in Unit 2 minimum required flow

   This revision increases the Unit 2 minimum required RCS flow rate assumed in the safety analyses by 3.6%. The actual core flow is unchanged and is approximately 6.6% higher than the value assumed in previous accident analyses. The remaining 3.0% 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 the actual core flow, it does not increase the actual probability or consequences of any postulated accident.

   When considered separately, increasing the minimum required RCS 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 assumed RCS flow will not increase the probability or consequences of an accident previously analyzed.

B. Revision to the Unit 2 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 2, Cycle 3 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 similar range of acceptable conditions. This change means that if a transient were to occur with the plant operating at the limits of the new curve, a higher temperature and power level might be attained than if the plant were operating within the bounds of the old curves. 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.

C. Revision to Unit 2 Overtemperature N-16 Reactor Trip Setpoints, Parameters and Coefficients

As a result of changes discussed, the Overtemperature N-16 reactor trip setpoint has been recalculated. 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 changed. This essentially means if a transient were to occur, the actual temperature and power level could be slightly higher. However, the analyses performed show that, using the TU Electric methodologies, all reactor core safety limits are met and all applicable limits of the safety analysis are met. This parameter has a setpoint which allows the mitigation of postulated accidents and has 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 potentially offer safety benefits. The risk of turbine runbacks or reactor trips due to upper plenum flow anomalies will be minimized with a higher overtemperature setpoint, thus reducing potential challenges to the plant safety systems. A final benefit is that the new methods for considering N-16 setpoints and values will be consistent with Unit 1, 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 will result in slight reduction in the probability of an accident and do not significantly increase the consequences of an accident previously evaluated.

D. Deletion of footnotes associated with the RCS flow - low reactor trip setpoint

In lieu of revising the footnotes to support the Unit 2 Cycle 3 operation, the deletion of the footnote is proposed. Further, for consistency with Unit 2, the same change is proposed for Unit 1. This change will not affect current plant practice; however, it will impose a more restrictive RCS flow - low setpoint than is currently required. The RCS flow - low reactor trip setpoint is currently specified in Technical Specification Table 2.2-1, Functional Unit 12.b, to be 90% of the minimum measured RCS flow. The proposed change would require the setpoint to be 90% of the instrument span where 100% of instrument span approximately corresponds to the actual RCS flow. The actual RCS flow is verified to be greater than the RCS flow assumed in the accident analysis through compliance with Technical Specification 3.2.5. Thus, through deletion of the footnotes, the RCS volumetric flow corresponding to the reactor trip setpoint will be greater than or equal to the volumetric flow allowed by the current specifications.

In summary, the proposed deletion of the footnotes will have no impact on current plant operations. A possible relaxation of the RCS flow - low setpoint which is currently allowed by the Technical Specifications will be removed without creating the potential for unnecessary plant trips.

The RCS flow - low reactor trip setpoint can have no effect on the probability of an accident. Because the reactor will be tripped at or prior to the conditions assumed in the accident analyses, there will be no effect on the consequences of an accident previously identified.

SUMMARY

The changes in the amendment request applies new NRC approved 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 operational transient to occur has been reduced and there has been no significant impact on the consequences of any accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes involve the use of revised safety analysis values and the calculation of new reactor 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. Do the proposed changes 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. Increase in the Unit 2 minimum required flow

   In performing the DNB-related analyses, the Reactor Coolant System flow rate assumed in these analyses is increased by 3.6 percent to insure that all applicable limits of the safety analysis are met. The Technical Specification 3/4.2.5 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 RCS 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.

   B. Revision to the Unit 2 Reactor Core Safety Limits

   The TU Electric reload analysis methods have been used to determine new reactor core safety limits. All applicable safety analysis limits have been met. The methods used are wholly consistent with Technical Specification BASES 2.1 which is the bases for the safety limits. In particular, the curves assure that for Unit 2, Cycle 3, 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 reactor 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 2, Cycle 3 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 provided 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.

C. Revision to Unit 2 Overtemperature N-16 Reactor Trip Setpoints, Parameters and Coefficients

Because the reactor core safety limits for CPSES Unit 2, Cycle 3 are recalculated, the Reactor Trip System instrumentation setpoint values for the Overtemperature N-16 reactor trip setpoint which protect the reactor 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 most relevant design basis analysis in Chapter 15 of the CPSES Final Safety Analysis Report (FSAR) which is affected by the change in the safety analysis value for the CPSES Unit 2 Overtemperature N-16 reactor trip setpoint is the Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (FSAR Section 15.4.2). This event has 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 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 2, Cycle 3 Overtemperature N-16 reactor trip setpoint is different from previous cycles which provides more operational flexibility to withstand mild transients without initiating automatic protective actions. Although the setpoint is different, 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.

D. Deletion of footnotes associated with the RCS flow - low reactor trip function

The deletion of the footnotes, and the potential relaxation of the RCS flow - low setpoint which could be used, will provide further assurance that, in the event of a partial loss of forced RCS flow or locked rotor transient, a reactor trip signal would be initiated prior to the conditions assumed in the accident analyses. Thus, the accident analyses are unaffected, and there is no reduction in a margin of safety.

SUMMARY

The proposed changes to the CPSES Technical Specifications involve using NRC-approved licensing analysis methods developed by TU Electric to determine the Technical Specification reactor core safety limits and perform DNB-related analysis for CPSES Unit 2, Cycle 3. 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 2, Cycle 3 core configuration. In performing these analyses, the minimum required Reactor Coolant System flow rate is increased by 3.6 percent. Because the core safety limits for CPSES Unit 2, Cycle 3 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.

Using the NRC approved TU Electric methods, the reactor 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 3/4.2.5 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 determined that the proposed amendment would change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10CFR20, or would change an inspection or surveillance requirement. 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.