TU ELECTRIC

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February 2, 1996

C. Lance Terry
Group Vice President

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NOS. 50-445 AND 50-446
SUPPORTING INFORMATION FOR LICENSE AMENDMENT REQUEST 95-008
UNIT 2 RELOAD ANALYSES AND UNIT 1 REACTOR COOLANT FLOW
(TAC NOS. M94167 AND M94204)

REF: 1) TU Electric letter logged TXX-95288 from C. L. Terry to the NRC dated November 21, 1995

2) NRC letter requesting addition information concerning CPSES License Amendment Request 95-008 from T. J. Polich to C. L. Terry dated January 30, 1996.

Gentlemen:

By Reference 1) above, TU Electric requested an amendment to the CPSES Unit 1 Operating License (NPF-87) and CPSES Unit 2 Operating License (NPF-89). The License Amendment would change the CPSES Units 1 and 2 Technical Specifications by revising the core safety limit curves and the 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.

By this letter TU Electric provides information to facilitate review of the License Amendment Request in response to Reference 2) above.

The attached information is typically included in each Reload Safety Evaluation performed in accordance with 10CFR50.59 prior to the start of a specific operating cycle and, for Unit 2 Cycle 3, is predicated on the approval of the License Amendment Request. The 50.59 evaluation and supporting calculations will be available for audit upon completion of Cycle 2 and finalizing of Cycle 3 design for Unit 2; however, the requested information is summarized and reproduced here to support the request for a timely review.

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Please contact My. J. D. Seawright at 214/812-4375 if further information is needed to complete the review.

Sincerely,

C. L. Terry

By: DRWoodlan
D. R. Woodlan
Docket Licesing Manager

JDR/jr
Attachment

cc: Mr. L. J. Callan, Region IV
Mr. W. D. Johnson, Region IV
Mr. T. J. Polich, NRR w\encls (2)
Resident Inspectors clo

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NRC Question 1:

You have discussed the use of different co-resident fuel assembly designs in reference 1 (page 1 of 13, Attachment 2). Please provide the reference for the method that has been used for the core reload with mixed fuel for CPSES Unit 2, Cycle 3. Have all the provisions from the reference been satisfied such as that required for the analysis for the effect of stress from seismic forces between the different fuel types (Siemens and Westinghouse) and the DNBR penalty factors required for transition cores?

TU Electric Response:

A. The methods used for the calculation of the mixed core DNB penalty are the same as those in the TU Electric report "VIPRE-01 Core Thermal-Hydraulic Analysis Methods for CPSES Licensing Applications," which is identified in TS 6.9.1.6b, Item 12. A Unit 2, Cycle 3 full core model was developed and used to assess the effects of the mixed core on the DNBR.

B. The effects of the mixed core on the large break LOCA analysis were evaluated in accordance with TU Electric report, "Large Break Loss of Coolant Accident Analysis Methodology," TS 6.9.1.6b, Item 15.

C. Both mechanical and thermal-hydraulic compatibility between the co-resident Westinghouse (W) and Siemens Power Corporation (SPC) supplied fuel assemblies are evaluated in the Reload Safety Evaluation. Both SPC and W have performed evaluations which demonstrate that their respective fuel assembly designs meet all applicable design criteria. In addition, SPC has evaluated the interaction between the co-resident W supplied and SPC supplied fuel assemblies and has confirmed that all applicable design criteria are satisfied.

NRC Question 2:

You have discussed meeting the minimum measured flow requirement in Technical Specification (TS) 3.2.5c in Reference 1. Will this reload incorporate low leakage core loading? If so, this type of loading has resulted in increased hot [leg] streaming in many plants that has resulted in reduced indicated Reactor Coolant System (RCS) flow rates. Will this reduced indicated RCS flow be a problem for CPSES Unit 2, Cycle 3? Please provide the total flow rates in gpm measured from the calorimetric heat balance for the current cycles for Units 1 and 2. Also please provide the references that approved the 1.8% uncertainty for the flow measurement and the 0.5% for the effects of the lower plenum flow anomaly.

TU Electric Response:

A. The Unit 2 Cycle 3 core configuration is a "low leakage" core design, as were the Unit 2 Cycle 2 and Unit 1 Cycles 2 through 5 core configurations. The reduction in indicated RCS flow seen with low leakage core designs is caused by hot leg temperature streaming. At CPSES, the N-16 based Transit Time Flow Meter (TTFM) is used to perform the precision flow calorimetric measurement. This measurement technique, and the associated accuracy of the flow measurement, is unaffected by the hot leg temperature streaming phenomenon. The evaluation of the existing flow margin is based on Unit 2 Cycle 2 operation, in which a low leakage core configuration is used. No significant degradation in flow in anticipated for Cycle 3.

B. For CPSES-1, Cycle 5, the "as-measured" RCS flow rate was 410,948 gpm. For CPSES-2, Cycle 2, the "as-measured" RCS flow rate was 421,610 gpm.

C. The 1.8% uncertainty for the RCS flow measurement is incorporated into Technical Specification 3.2.5 and is based on uncertainty calculations originally performed for CPSES-1 by Westinghouse. The calculations have since been updated by TU Electric for both CPSES units using the Westinghouse methodology. The 1.8% allowance remains valid for CPSES-2. The 1.8% uncertainty is approved in the CPSES Technical Specifications (through Amendments 44/30) and in NUREG-0797, SSER 12, Pages 4-1 and 4-2.

D. The allowance for the lower plenum flow anomaly was obtained from WCAP-11528, "RCS Flow Anomaly Investigation Report," April 1988 and confirmed through plant-specific measurements.

NRC Question 3:

Please provide the reference for the approved method used for obtaining the Overtemperature N-16 reactor trip setpoint [and] for obtaining the total uncertainty as discussed in reference 1 (pages 1, 2, and 3 of 13, Attachment 2 and TS Table 2.2-1) and the Overpower N-16 trip setpoint (page 5 of 13, Attachment 2, and TS Table 2.2-1).

TU Electric Response:

The Unit 2 Cycle 3 overtemperature N-16 setpoint was developed in accordance with TS 6.9.1.6b, Item 9, "Power Distribution Control Analysis and Overtemperature N-16 and Overpower N-16 Trip Setpoint Methodology."

The Westinghouse setpoint application to CPSES-1 is summarized in WCAP-12123 and was reviewed by the NRC prior to issuing the Unit 1 operating license. This review is documented in NUREG-0797 SSER 22 (Page 7-7). This setpoint methodology was used by Westinghouse in the calculation of the Reactor Trip System (RTS) and the Engineered Safety Features Actuation System (ESFAS) setpoints for the CPSES-1 Technical Specifications. TU Electric applied this methodology in the calculation of the RTS and ESFAS setpoints which were approved for incorporation into the original CPSES-2 Technical Specifications and also into past revisions to the CPSES-1 Technical Specifications. TU Electric used the same methodology for the revised overtemperature N-16 uncertainty calculations that was used for the original CPSES-2 Technical Specifications which have been approved by the NRC.

NRC Question 4:

Please explain the difference between how the power is calculated using the N-16 power indication and that from the calorimetric power indication as discussed in Reference 1 (page 11 of 21, Attachment 2).

TU Electric Response:

The calorimetric indication of core power is developed from a secondary plant heat balance. The secondary plant thermal power is determined from measurements of the feedwater flow and enthalpy and steam enthalpy. The reactor coolant pump heat addition and charging/letdown losses are then used to determine the core thermal power. The N-16 power indication is then normalized to indicate the calorimetric power. This process is identical to that used to normalize the Nuclear Instrumentation System excore power indication.

NRC Question 5:

Please provide a list of the NRC approved codes, with the titles of the approved reports, used for the Unit 2, Cycle 3 reload analysis.

TU Electric Response:

The NRC approved methods are listed in Technical Specification 6.9.1.6b, and is also detailed in the cycle-specific Core Operating Limits Report. The relevant information (excerpts from Technical Specification 6.9.1.6b) is reproduced below.

6) WCAP-10079-P-A, "NOTRUMP, A NODAL TRANSIENT SMALL BREAK AND GENERAL NETWORK CODE," August 1985, (W Proprietary).

7). WCAP-10054-P-A, "WESTINGHOUSE SMALL BREAK ECCS EVALUATION MODEL USING THE NOTRUMP CODE," August 1985, (W Proprietary).

8). WCAP-11145-P-A, "WESTINGHOUSE SMALL BREAK LOCA ECCS EVALUATION MODEL GENERIC STUDY WITH THE NOTRUMP CODE," October 1986, (W Proprietary).

9). RXE-90-006-P, "Power Distribution Control Analysis and Overtemperature N-16 and Overpower N-16 Trip Setpoint Methodology," February 1991.

10). RXE-88-102-P, "TUE-1 Departure from Nucleate Boiling Correlation," January 1989.

11). RXE-88-102-P, Sup. 1, "TUE-1 DNB Correlation - Supplement 1," December 1990.

12). RXE-89-002, "VIPRE-01 Core Thermal-Hydraulic Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications," June 1989.

13). RXE-91-001, "Transient Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications," February 1991.

14). RXE-91-002, "Reactivity Anomaly Events Methodology," May 1991.

15). RXE-90-007, "Large Break Loss of Coolant Accident Analysis Methodology," December 1990.

16). TXX-88306, "Steam Generator Tube Rupture Analysis," March 15, 1988.

17). RXE-91-005, "Methodology for Reactor Core Response to Steamline Break Events," May, 1991.

19) RXE-94-001-A, "Safety Analysis of Postulated Inadvertent Boron Dilution Event in Modes 3, 4,and 5," February 1994.

<u>NRC Question 6:</u>

You mention on page 11 of 13 of Attachment 2 that 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) and that all acceptance criteria were satisfied. Please provide information on all the

Chapter 15 accident analyses that were performed for CPSES Unit 2 Cycle 3 and indicate what approved codes were used for each accident or transient and why the results were acceptable (i.e., met the DNBR requirement, met the pressure requirement, etc.).

<u>TU Electric Response:</u>

A. When preparing the License Amendment Request, all events were reviewed. Those events for which the Overtemperature N-16 trip function provides a primary protective or mitigative function were identified. With the exception of the Uncontrolled Rod Withdrawal from Power (RWAP) event, none of the events are "limiting" with respect to the DNBR event acceptance criterion. Therefore, the discussion in the License Amendment Request is based only the RWAP event, which is the most limiting of those events for which DNBR protection is provided by the overtemperature reactor trip function. The analyses of this event demonstrated that the relevant event acceptance criteria (DNB and overpressure) are satisfied. Note that the RWAP event was analyzed using deterministic DNB methods.

B. A table of the relevant event acceptance criterion for each non-LOCA FSAR Chapter 15 event considered during the core reload evaluation process is provided in the NRC's Safety Evaluation Report "Comanche Peak Steam Electric Station Units 1 and 2, Topical Report RXE-91-001, 'Transient Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications' (TAC No. M79866)," T. A. Bergman (NRC) to W. J. Cahill (TUEC), July 16, 1993. This table is attached for your convenience. The LOCA evaluations are performed in accordance with Technical Specification 6.9.1.6b, Items 6, 7, 8, and 15. The most relevant event acceptance criterion, the peak clad temperature, is reported to the NRC in accordance with 10CFR50.46.

During the Reload Safety Evaluation performed in accordance with 10CFR50.59 prior to the start of a specific operating cycle, it is confirmed that the methods listed in Technical Specification 6.9.1.6b are used to ensure that each of the relevant event acceptance criteria are satisfied.

Table IV-1
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FSAR SECTION Event Acronym	15.1.1,2 FWM	15.1.3 ELI	15.1.4 MSSV	15.1.5 MSLB	15.2.3 TT	15.2.3 TT	15.2.6 LOAC
Acceptance Criterion	DNBR	DNBR	DNBR	DNBR	DNBR	RCS Press	DHR@
Power#	High/Zero	High	Zero	Zero	High	High	High
Przr Pressure#	Low	Low	Nominal	Nominal	Low	High	High
Przr Level	High	High	Nominal	Nominal	High	High	High
RCS T-avg#	High	High	Nominal	Nominal	High	High	High
RCS Loop Flow#	Low	Low	Low	Low	Low	Low	Low
SG Level	Low	High	High	High	High	High	High
Fuel Temp	Low	Low	-	-	High	High	High
Przr Prs Cntl*	On	Off	Off	Off	On	Off	Off
SG Wtr Lvl Cntl	On	On	-	-	-	-	-
Rod Cntl	On/Off	On/Off	Off	Off	Off	Off	Off
Turbine Cntl	Load	Load	-	-	-	-	Load
Rx Trip Signal	OTN16	None	SI	SI	HPP,Lo SGWL	Hi Pres	Lo SGWL
ECCS Act Signal	None	None	Lo StmP	Lo StmP	None	None	None
Turbine Trip	Hi SGWL	None	SI	SI	Rx Trip	-	Rx Trip
Stm Iso. Signal	None	None	Lo St[illegible]	Lo StmP	None	None	None
MFW Iso. Signal	Hi SGWL	None	MS Isol	MS Isol	SGWLC	SGWLC	-
Moderator Temp Coef	Range	Range	Most Negative	Most Negative	Range	Range	Least Negative
Doppler Fuel Temp Coeff	Least Negative	Least Negative	Least Negative	Least Negative	Most Negative	Most Negative	Most Negative
Eff Delayed Neutron Frac	Minimum	Range	Minimum	Minimum	Range	Range	Maximum

* Pressurizer Sprays and PORVS only.
\# Nominal Values are used if SCU DNB methodology is used.
@ Decay Heat Removal with Auxiliary Feedwater System

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FSAR SECTION Event Acronym	15.2.7 LOFW	15.2.8 MFLB	15.3.1 F·OF	15.3.2 CLOF	15.3.3 LR/SS	15.4.1 RWS	15.4.2 RWP
Acceptance Criterion	DHR[@]	DHR[@]	DNBR	DNBR	RCS Pres/DNBR	DNBR	DNBR
Power[#]	High	High	High	High	High	Zero	Range
Przr Pressure[#]	High	High	Low	Low	High/Low	Low	Low
Przr Level	High	High	High	High	High	High	High
RCS T-avg[#]	High	High	High	High	High	High	High
RCS Loop Flow[#]	Low	Low	Low	Low	Low	Low	Low
SG Level	High	High/Low	-	-	-	-	-
Fuel Temp	High	High	High	High	High	-	High
Przr Prs Cntl[^]	Off	Off	On	On	Off/On	Off	On
SG Wtr Lvl Cntl	-	-	On	On	On	On	On
Rod Cntl	Off	Off	Off	Off	Off	Off	Off
Turbine Cntl	Load	Load	-	-	-	Load	Load
Rx Trip Signal	Lo SGWL	Lo SGWL	Lo Flow	UV/UF	Lo Flow	Hi Flux Lo Stpt	Hi Flux OT/OP N16
ECCS Act Signal	None	Lo StmP	None	None	None	None	None
Turbine Trip	Rx Trip	Rx Trip	Rx Trip	Rx Trip	Rx Trip	None	Rx Trip
Stm Iso. Signal	None	Lo StmP	None	None	None	None	None
MFW Iso. Signal	-	Lo StmP	None	None	None	None	None
Moderator Temp Coef	Least Negative	Most Negative	Least Negative	Least Negative	Least Negative	Least Negative	Range
Doppler Fuel Temp Coef	Most Negative	Most Negative	Most Negative	Most Negative	Most Negative	Least Negative	Range
Eff Delayed Neutron Frac	Maximum	Maximum	Maximum	Maximum	Maximum	Maximum	Maximum

^ Pressurizer Sprays and PORVS only.
\# Nominal Values are used if SCU DNB methodology is used.
@ Decay Heat Removal with Auxiliary Feedwater System

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FSAR SECTION Event Acronym	15.4.3 DropRod	15.4.3a SRWP	15.4.8 CRE	15.5.1 ECCS	15.6.1 RCS DP
Acceptance Criterion	DNBR	DNBR	Pellet enthalpy/PCT	DNBR	DNBR
Power#	High	High	High/Zero	High	High
Przr Pressure#	Low	Low	High	Low	Low
Przr Level	Low	High	High	High	High
RCS T-avg#	High	High	High	High	High
RCS Loop Flow#	Low	Low	Low	Low	Low
SG Level	-	-	-	-	-
Fuel Temp	High	High	High	High	High
Przr Prs Cntl*	On	On	-	Off	Off
SG Wtr Lvl Cntl	On	On	On	On	On
Rod Cntl	On	On	-	Off	Off/On
Turbine Cntl	Load	Load	-	Load	Load
Rx Trip Signal	Hi Flux	Hi Flux; OTN16	Hi Flux	Lo PrzP	Lo PrzP OTN16
ECCS Act Signal	None	None	None	None	None
Turbine Trip	Rx Trip	Rx Trip	Rx Trip	Rx Trip	Rx Trip
Stm Iso. Signal	None	None	None	None	None
MFW Iso. Signal	None	None	None	None	None
Moderator Temp Coef	Range	Range	Least Negative	Least Negative	Range
Doppler Fuel Temp Coef	Least Negative	Range	Least Negative	Least Negative	Range
Eff Delayed Neutron Frac	Maximum	Maximum	Minimum	Maximum	Maximum

* Pressurizer Sprays and PORVS only.
\# Nominal Values are used if SCU DNB methodology is used.

NRC Question 7:

Provide input parameters for (power, pressure, temperature, flow, and power density) used to calculate DNBR and other Chapter 15 analyses for Unit 2 Cycle 3 and the resultant DNBR value.

TU Electric Response:

The initial conditions used in the transient analyses for Unit 2 Cycle 3 are separated into 2 columns. When deterministic DNB methods are to be used, the left-most column is applicable. When statistical DNB methods are used, the right-most column is applicable for Unit 2 Cycle 3.

Parameter	Deterministic	Statistical
Maximum Rated Thermal Power (% of 3411 MW)	102%	100%
Pressurizer Pressure (psia)	2220 (DNB-limited) 2280 (overpressure-limited)	2250 (System analysis) 2280 (DNB analysis)
T-average at 100% RTP (°F)	595.7	589.2
RCS Flow (gpm)	≤ 400,800	408,000
Average Power Density (kw/ft)	5.55	5.445
DNBR Limit Value	1.16	1.429

Experience has shown that, with the use of TU Electric methods and CPSES core designs, the event for which the calculated minimum DNBR most closely approaches the DNBR limit value is typically the dropped rod event; although any event can be made to appear "limiting", depending on how much conservatism is included in the evaluation. For the preliminary, Unit 2 Cycle 3 analysis of the dropped rod event, in which the Statistical Combination of Uncertainties (SCU) DNB methodology is used, a minimum DNBR of approximately 1.50 was calculated. This value is greater than the DNBR limit value and is, therefore, acceptable.

Of more relevance to this License Amendment Request is the analysis of the RWAP event. In order to provide allowances for future uses, these evaluations were performed with peaking factors greater than expected to be required for Unit 2 Cycle 3 operation. For the preliminary, Unit 2 Cycle 3 analysis of the RWAP event, in which the deterministic DNB methodology is used, a minimum DNBR of approximately 1.34 was calculated for the case initiated from 102% RTP. For the case initiated from 12% RTP, the minimum DNBR was calculated, on a preliminary basis, to be approximately 1.17. Both of these values are greater than the deterministic DNBR limit of 1.16 and are, therefore, acceptable.

The "resultant" DNBR for each transient is confirmed to be greater than the appropriate DNBR limit value (1.16 for deterministic methods, 1.429 for statistical methods for Unit 2 Cycle 3). Through the Reload Safety Evaluation evaluation process, it is confirmed, prior to the start of a specific operating cycle, that all analyses are performed in accordance with the methodology approved by the NRC and listed in Technical Specification 6.9.1.6b.

NRC Question 8:

Provide the uncertainty values and bases used in the statistical combination of uncertainties as required by the safety evaluation report that approved RXE-91-002, "Reactivity Anomaly Event Methodology," dated January 19.1993.

TU Electric Response:

TU Electric's topical report RXE-91-002 contained a demonstration application of the Statistical Combination of Uncertainties (SCU) methodology. The values used in the demonstration application were applicable to Unit 1 Cycle 1. As stated in the NRC's Safety Evaluation Report, Technical Evaluation Report, and the Responses to the Request for Additional Information related to the TU Electric's report RXE-91-002, TU Electric will use unit- and cycle-specific values when applying the SCU methodology. This information is provided, as required by the forgoing documents, in the applicable TU Electric calculations.

When the Unit 2 Cycle 3 Reload Safety Evaluation is completed, in accordance with 10CFR50.59 and predicated on the approval of License Amendment Request 95-008, no unreviewed safety questions will exist and it is anticipated that no additional changes to the plant Technical Specifications will be required. All analyses will be performed in accordance with Technical Specification 6.9.1.6. If, as expected, no unreviewed safety questions are identified, no additional licensing submittals will be required.

For Unit 2 Cycle 3, the uncertainty values expected to be used in the SCU applications are reproduced below. The DNBR uncertainty factor is 0.8278. Temperature and flow biases, totaling ~ 0.027 DNB, are treated appropriately. The bases are described in the approved topical report, which includes the additional questions and responses and the NRC's safety evaluation report.

Parameter	Sensitivity (Δ DNB / Δ change in parameter)	Coefficient of Variance (σ/μ)
Pressure	-1.577	0.00811
Temperature	-9.793	0.00508
Power	-2.605	0.01351
Flow	-1.600	0.01160
FdelH	-3.374	0.02432