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Ref: 10CFR50.90

CPSES-200100773
Log # TXX-01042
File # 236

April 5, 2001

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 01-05
INCREASE IN UNIT 1 AND UNIT 2
REACTOR POWER TO 3458 MWt

REF:
1) TXU Electric Letter logged TXX-98180, from C. L. Terry to the NRC dated July 17, 1998

2) Caldon Engineering Report - 160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM√™ System," Revision 0, May 2000.

3) TXU Electric Letter logged TXX-93339, from William J. Cahill, Jr. to the NRC dated October 4, 1993

Gentlemen:

Pursuant to 10CFR50.90, TXU 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 Unit 1 and Unit 2 Operating License and the CPSES Units 1 and 2 Technical Specifications. These changes apply to both CPSES Unit 1 and Unit 2.

Based on the Caldon topical report submitted per Reference 1 and as supplemented by Reference 2, TXU Electric requests an increase in the licensed power for operation of both CPSES Unit 1 and Unit 2 to 3458 MWt. This power level represents an

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increase of approximately 1.4% of the currently licensed power level for Unit 1 and an increase of approximately 0.4% for Unit 2. The licensed power level for Unit 2 was previously increased by 1% as summarized in Reference 1. The information supporting the current license amendment request is largely a repetition of the material submitted, reviewed and approved in the previous 1% power uprate for Unit 2. In addition, TXU Electric requests that Texas Municipal Power Agency (TMPA) be removed from both Unit 1 and Unit 2 Licenses since transfer of ownership from TMPA to TXU Electric was completed as discussed in Reference 3.

Attachment 1 is the required affidavit. Attachment 2 provides a detailed description of the proposed changes, a safety analysis of the proposed changes and TXU Electric's determination that the proposed changes do not involve a significant hazard consideration. Attachments 3 and 4 provide markups of the affected pages of Operating Licenses NPF-87 and NPF-89 for Units 1 and 2, respectively. Attachment 5 provides the Technical Specification pages marked-up to reflect the proposed changes. Attachment 6 provides the retyped version of the affected Technical Specifications pages with the proposed changes incorporated for Unit 2 implementation. Attachment 7 provides the retyped version of the affected Technical Specifications pages with the proposed changes incorporated for Unit 1 implementation.

The analyses of the effects of the proposed power increase on the Balance of Plant (BOP) systems are presented in Section III.C of Attachment 2. These analyses are complete for Unit 2 and for those systems common to the two units, but not all analyses supporting Unit 1 have been finalized. Due to the similarity between the two units, it is expected that all relevant design criteria will be met. The Unit 1 BOP system analyses will be completed prior to implementation of the power uprate on Unit 1.

TXU Electric requests approval of this proposed license amendment be targeted for August 1, 2001, with implementation of the changes to occur during Refueling Outage 9 for Unit 1 and within 60 days after NRC approval for Unit 2.

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

This communication contains the following revised commitments which will be completed as noted:

CDF Number	Commitment
27185	LEFM√ operation and the associated quantification of power measurement uncertainty will apply whenever the facility operates at the new, higher rated power.
27184	The modification to LOCA Analysis methods to incorporate the power measurement uncertainty based on the use of LEFM√ will be included in the next annual ECCS report prepared in accordance with 10CFR50.46.

This communication contains the following new commitments which will be completed as noted:

27228	Prior to implementation of the proposed uprate in Unit 1, TXU Electric will determine whether additional preventive actions are required in addition to those previously taken and reported in response to NRC Bulletin 88-02.
27229	The detailed evaluation of Unit 1 non-NSSS systems, structures, and components and related programs will be completed prior to implementation of the requested Unit 1 uprate. BOP design attributes, such as certain piping analyses that are unit-specific, have been evaluated for Unit 2 and will be completed for Unit 1 prior to implementation of the requested uprate in Unit 1.

The CDF number is used by TU Electric for the internal tracking of CPSES commitments.

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Should you have any questions, please contact Mr. J. D. Seawright at (254) 897-0140.

Sincerely,

C. Lance Terry

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

JDS/grp
Attachments:
1. Affidavit
2. Description and Assessment
3. Affected pages to the Unit 1 Operating Licenses (NPF-87)
4. Affected pages to the Unit 2 Operating Licenses (NPF-89)
5. Markup of affected pages to the Technical Specification and Bases
6. Retyped affected pages to the Technical Specification and Bases

c - Mr. E. W. Merschoff, Region IV
Mr. D. H. Jaffe, NRR
Mr. J. I. Tapia, Region IV
Resident Inspectors, CPSES

Mr. Arthur C. Tate
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Austin, Texas 78704

Attachment 1 to TXX-01042
Page 1 of 1

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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

<u>AFFIDAVIT</u>

Roger D. Walker being duly sworn, hereby deposes and says that he is Regulatory Affairs Manager of TXU Electric, the licensee herein; that he is duly authorized to sign and file with the Nuclear Regulatory Commission this License Amendment Request 01-05; 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 Somervell)

Subscribed and sworn to before me, on this 4th day of April, 2001.

Gayle R. Peck Jespersen

Notary Public

NOTARY PUBLIC STATE OF TEXAS
Gayle R. Peck Jespersen
Notary Public, State of Texas
My Comm. Expires 03/16/02

ATTACHMENT 2 to TXX-01042

DESCRIPTION AND ASSESSMENT

DESCRIPTION AND ASSESSMENT

1.0 INTRODUCTION

The purpose of this License Amendment Request is to increase the licensed core power levels for both CPSES Unit 1 and Unit 2 to 3458 MWt. Along with the proposal to increase the Rated Thermal Power to 3458 MWt, TXU Electric also proposes continued use of topical reports identified in CPSES Technical Specification 5.6.5b. These topical reports describe the NRC-approved methodologies which support the CPSES safety analysis, including the small break and large break loss of coolant accidents analyses. In many of these topical reports, reference is made to the use of a 2% uncertainty applied to the reactor power, consistent with Appendix K. Through Amendment 72 to the CPSES Technical Specifications, the NRC approved use of these methods with a power uncertainty of 1% RTP. TXU Electric proposes that the approval for the continued use of these topical reports be extended to a power uncertainty of 0.6% RTP and that the NRC acknowledge that the change in the power uncertainty does not constitute a significant change as defined in 10CFR50.46 and Appendix K. In addition, TXU Electric is proposing to remove Texas Municipal Power Agency (TMPA) from both Unit 1 and Unit 2 Licenses since transfer of ownership from TMPA to TXU Electric is complete.

The Technical Specification Bases and FSAR will be updated by TXU Electric based on the approved License Amendment when issued.

2.0 DESCRIPTION OF OPERATING LICENSE CHANGE REQUEST

The Operating License for Unit 1 (NPF-87), section 2.C(1), identifies the maximum core thermal power level for which CPSES Unit 1 is authorized to operate as 3411 MWt. With the use of the Caldon, Inc., Leading Edge Flow Meter (LEFM(check)), TXU Electric proposes changing the maximum core power level to 3458 MWt. In addition, references to Texas Municipal Power Agency (TMPA) are removed and sections 2.B(2) and 2. J. of the license are deleted.

The Operating License for Unit 2 (NPF-89), section 2.C(1), identifies the maximum core thermal power level for which CPSES Unit 2 is authorized to operate as 3445 MWt. With the use of the Caldon, Inc., Leading Edge Flow Meter (LEFM(check)), TXU Electric proposes changing the maximum core power level to 3458 MWt. In addition, references to Texas Municipal Power Agency (TMPA) are removed and sections 2.B(2) and 2. J. of the license are deleted.

TXU Electric proposes changing the definition of RATED THERMAL POWER (definition 1.28) in the Technical Specifications to read:

> 1.28 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3458 MWt.

TXU Electric has performed calculations to demonstrate that with the use of the LEFM(check), the uncertainty associated with the core power measurement may be reduced to less than 0.6% RTP. Many of the safety analyses supporting the design and operation of CPSES retain a power uncertainty of 2% RTP after the previous uprate to 3445 MWt for Unit 2. Where practical, TXU Electric proposes to continue the use of existing analyses to justify operation at the newly uprated power conditions. That is, analyses performed at 3411 MWt with the application of a 2% RTP uncertainty allowance or performed at 3445 MWt with the application of a 1% RTP uncertainty are equivalent to analyses performed at 3458 MWt with the application of a 0.6% RTP uncertainty allowance. Given this philosophy, the safety analysis limits of those Reactor Protection System functions that are expressed as a percentage of RTP will change. The affected power range reactor trip functions are the overtemperature N-16, overpower N-16, and neutron flux .

The overtemperature reactor trip function is explicitly analyzed on a cycle-specific basis. The coefficients of the overtemperature setpoint equation are contained in the cycle-specific Core Operating Limits Report. It is anticipated that continued compliance with the Allowable Value presented in the Technical Specification will be achieved without further changes.

Expressed as a percentage of RTP, the safety analysis limit used for the power range neutron flux and Overpower N-16 reactor trip functions was decreased from 118% of RTP to 116.9% RTP. For the power range high neutron flux function, sufficient margin was available such that the Nominal Trip Setpoint remained valid; however, it was necessary to revise the Allowable Value. For the overpower N-16 reactor trip function, it was necessary to revise both the Nominal Trip Setpoint and the Allowable Value. Therefore, TXU Electric also proposes changing the Allowable Values of the Overpower N-16 reactor trip function to "< 112.9% RTP" and the Power Range Neutron Flux - High reactor trip function to "< 110.8%" RTP. The values for the Nominal Trip Setpoints are located in the Technical Specification Bases. These and other conforming changes to the Technical Specification Bases are provided for information.

To summarize, TXU Electric proposes changing the reactor core licensed power level for both CPSES Unit 1 and Unit 2 to 3458 MWt. Associated with these changes are revisions to the Allowable Values of the Overpower N-16 reactor trip function and the Power Range Neutron Flux - High reactor trip functions. TXU Electric also proposes to change the administrative controls in the Technical Specifications (5.6.5b) regarding the assumed initial power level to be consistent with the power uprate. In addition, references to Texas Municipal Power Agency (TMPA) is removed from the Operating License.

3.0 BACKGROUND

The CPSES Unit 1 and Unit 2 Facility Operating License Nos. NPF-87 and NPF-89 note that the transfer of ownership interest from Texas Municipal Power Agency (TMPA) to TXU Electric was previously authorized by Amendment Nos. 9 and 8, respectively, to Construction Permit Nos. CPPR-126 and CPPR-127, respectively, to take place in 10 installments as set forth in the Agreement. At the completion of the last installment, TMPA would no longer retain any ownership interest.

Comanche Peak Steam Electric Station (CPSES) Unit 1 and Unit 2 were originally licensed for a core Rated Thermal Power (RTP) of 3411 MWt. In 1999, TXU sought and received permission from the NRC to increase the licensed core power for Unit 2 based on the use of high-accuracy feedwater flow measurement systems and an exemption to certain requirements of 10CFR50, Appendix K. Effective July 2000, the NRC revised 10CFR50, Appendix K, to allow licensees to use an initial power level of less than 1.02 times the licensed power level provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error.

Through the use of the LEFM(check) feedwater flow measurement system provided by Caldon, Inc., the power level instrumentation error for the CPSES units can be demonstrated to be less than 0.6% of full power. Therefore, using the existing safety analyses (typically performed at a power level of 1.02 times the licensed power level of 3411 MWt), the licensed power level can be increased by 1.4% of 3411 MWt, to 3458 MWt.

4.0 TECHNICAL ANALYSIS

A transfer of Texas Municipal Power Agency (TMPA) ownership share to TXU Electric was approved by the NRC and appears as a license notation in the CPSES Operating License. TXU Electric made the final installment payment on August 24, 1993, and has received the final deed. TMPA no longer has ownership interest in CPSES. As discussed in the NRC SER issued with the Corporate name change from TU Electric to TXU Electric, TXU Electric is the sole owner of CPSES, Units 1 and 2.

The analysis of the Leading Edge Flow Meter (LEFM) is presented in six sections. An overview of the LEFM instrumentation used to measure the main feedwater mass flow and temperature for input into the plant calorimetric measurement is presented in Section A. The evaluations supporting the Nuclear Steam Supply System (NSSS) and the Balance of Plant (BOP) are presented in Sections B and C, respectively. The evaluation of the effects on the radiological consequences is presented in Section D. Miscellaneous issues, such as the effects of the proposed power uprate on the station blackout evaluations and the Individual Plant Examination conclusions, are addressed in Section E. Finally, the effects of the proposed power uprates on plant operations and procedures are assessed in Section F.

Many of the existing safety analyses, particularly those presented in FSAR Chapter 15, incorporate a core thermal power uncertainty allowance of ±2% RTP referenced to the original rating of 3411 MWt. With the use of the LEFM, it is proposed to reduce the required power uncertainty allowance to ±0.6% RTP. These same analyses with a power uncertainty allowance of ±2% RTP support the revised Rated Thermal Power with a corresponding power uncertainty allowance of ±0.6% RTP. These analyses, which were performed in accordance with the approved methodologies described in Technical Specification 5.6.5, continue to comply with all applicable event acceptance criteria, and thus, will not be discussed further.

In contrast, the remaining safety analyses supporting the design and operation of CPSES were performed at the nominal, 3411 MWt RTP condition. These analyses were evaluated using design conditions representing the proposed power uprate, at a minimum, and more generally corresponding to the original CPSES Engineered Safeguards Design (ESD) rating of 3565 MWt. Recognizing that the evaluation for a larger power increase would encompass the proposed increase and entail essentially the same amount of resources, the 3565 MWt RTP design condition was selected to support the potential pursuit of additional power increases in the future. However, only a core thermal power increase to 3458 MWt is requested at this time.

A. LEADING EDGE FLOW METER (LEFM)

The LEFM is an ultrasonic flow meter consisting of a common control cabinet in the Control Room and spool pieces located in each unit's 30-inch main feedwater line. The LEFM uses acoustic energy pulses to determine the final feedwater mass flow rate. Transducers that transmit and receive the pulses are mounted in the LEFM spool piece at an angle of 45° to the flow stream. The sound will travel faster when the pulse traverses the pipe with the flow and slower when the pulse traverses the pipe against the flow. The LEFM uses these transit times and time differences between pulses to determine the fluid velocity and its temperature. The LEFM computer also uses this calculated temperature and the independently measured fluid pressure to account for changes in the LEFM's spool piece dimensions. The adjusted cross-section area, velocity, and temperature are then used to determine the final feedwater mass flow rate. The mass flow rate is displayed on the local display panel and transmitted to the plant process computer as an input to determine the reactor thermal output based on an energy balance of the secondary system. This technology provides significantly higher accuracy and reliability than flow instruments which use differential pressure measurements and temperature instruments which use conventional thermocouple or resistance thermometers.

The LEFM(check) is an improved system for use in determining and monitoring feedwater flow in nuclear power plants. The LEFM(check) provides increased safety by providing on-line verification of the accuracy of the feedwater flow and temperature measurements upon which NSSS thermal power determinations are based. In addition, the LEFM(check) provides a significant improvement in accuracy and an increase in reliability of flow and temperature measurements.

The improved accuracies achievable with the LEFM(check) are valid while the instruments are performing as designed. The on-line verification features of the LEFM(check) provide the ability to assure on-line that performance is consistent with the design basis.

The LEFM(check) provides measurements of feedwater mass flow and temperature yielding a ±0.6% RTP uncertainty in thermal power, substantially more accurate than the typical ±2% RTP obtained with inputs from the conventional venturi-based instrumentation or the ±1.4% RTP uncertainty obtainable with precision venturi-based instrumentation. Furthermore, the probability of operation above 1.02 times the original licensed power level of 3411 MWt with use of original instrumentation is approximately the same as it is for the proposed licensed power level (3458 MWt) with use of the LEFM(check).

The LEFM(check) indications of feedwater mass flow and feedwater temperature will be directly substituted for the venturi-based mass flow indication and the resistance temperature detector (RTD) indications in the plant calorimetric calculation performed with the plant computer. As an alternative, the calorimetric power can be manually calculated, using the LEFM(check) indications and following prescribed procedures. In either of these cases, the use of the LEFM(check) will be limited to the calorimetric power determination. The venturi-based feedwater flow indication will continue to be used for feedwater control and other functions that it currently fulfills. Further, the venturi-based indication may be adjusted periodically on the basis of the LEFM(check) indication, so that it serves as a backup in the event that the LEFM(check) system is not available.

Use of the enhanced LEFM system supports the proposed reduction in the uncertainty in the core thermal power measurement while maintaining the original licensing basis of CPSES. The increase to the license operating power level of 3458 MWt, plus an allowance of ±0.6% RTP power uncertainty, continues to be bounded by the original analyses of record for the ECCS and the safety analyses based on the ECCS.

The methodology used to calculate the power calorimetric uncertainty is unchanged by the inclusion of the LEFM(check) feedwater flow and temperature indications. This methodology is consistent with the recommendations by ASME PTC 19.1 - 1985, "Measurement Uncertainty." This method is the same as that typically used in Westinghouse power plants with standard venturi-based feedwater flow instrumentation, with the exception that TXU Electric treats the venturi coefficient error from loop to loop as a dependent, or systematic, error in flow. As a result, the feedwater flow uncertainty is not reduced by a factor of the square root of the number of channels to account for the four individual loop measurements, as is believed to be a more typical practice.

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

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

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

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

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

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

B. NUCLEAR STEAM SUPPLY SYSTEM (NSSS)

The CPSES Power Uprate Project was completed consistent with the methodology established in WCAP-10263, "A Review Plan for Uprating the Licensed Power of a PWR Power Plant," dated January, 1983. The methodology establishes the general approach and criteria for uprate projects including the broad categories that must be addressed, such as NSSS performance parameters, design transients, systems, components, accidents and nuclear fuel as well as interfaces between the NSSS and Balance of Plant (BOP) systems. Inherent in this methodology are key points that promote correctness, consistency and licensability. The key points include the use of well-defined analysis input assumptions/parameter values, use of NRC currently approved analytical techniques and use of plant-specific currently applicable licensing criteria and standards. The evaluations and analyses described in this document have been completed consistent with this methodology.

B.1 NSSS PARAMETERS

INPUT PARAMETERS AND ASSUMPTIONS

The major inputs used in the development of the NSSS parameters used by Westinghouse in the re-evaluation of the NSSS design are summarized below. The values denoted as "current" reflect the original design for each unit.

The NSSS power level for the uprating analysis was established as 3582 MWt (3565 MWt core). This is approximately 4.5% higher than the original NSSS power rating of 3425 MWt (3411 MWt core). The 4.5% increase was selected to allow for the planned 1.4% increase relative to the original power ratings of each CPSES unit and provide margin for any future increases, up to 3582 MWt (3565 MWt core), if desired. This core thermal power rating corresponds to the original CPSES Engineered Safeguards Design rating. The NSSS power level for the uprating is based on a net RCS heat input of 17 MWt (3582 - 3565 MWt) for the reactor coolant pumps, which is more representative of the actual net reactor coolant pump heat input addition.

The reactor coolant system flowrate of 97,900 gpm/loop was revised from the original value of 95,700 gpm/loop. This flow was applied for all cases, even those which assumed 0% steam generator tube plugging (SGTP).

Two values of SGTP have been assumed: 0% and 10%. The 10% SGTP level supports a peak SGTP level of 15% in any one steam generator provided that the average level of plugging of all four steam generators is < 10%.

A range of full power normal operating T_{AVG} from 585.7°F to 592.7°F has been analyzed. This range represents a ±3.5°F variation from the original design T_{AVG} of 589.2°F and will allow for greater operating flexibility.

A full power feedwater temperature range of 390°F to 444.6°F was selected for the analyses. This range supports greater operating flexibility by allowing the plant to reduce feedwater temperature to maintain 100% thermal power in the event that the Steam Generator outlet steam pressure is insufficient to satisfy the turbine volumetric flow limit.

DISCUSSION OF PARAMETER CASES

Table IV-1 summarizes the NSSS parameter cases that were developed and used as the basis for the uprating project. The original design parameters are also shown for comparison purposes. A description of the three uprate cases follows.

Case 1 represents the uprated conditions with the lower reactor vessel average temperature of 585.7°F and a 10% SGTP level. It yields the lowest possible initial primary side temperatures for the analyses as well as the minimum secondary side steam generator temperature, pressure and flow.

Case 2 represents the uprated conditions with the higher reactor vessel average temperature of 592.7°F and a 10% SGTP level. It yields the highest possible initial primary side temperatures for the analyses.

Case 3 represents the uprated conditions with the higher reactor vessel average temperature of 592.7°F and a 0% SGTP level. It yields the highest possible initial steam temperature, steam pressure, and steam flow.

Table IV-1
Unit 1 and Unit 2 NSSS Revised Design Parameters

BASIC COMPONENTS			
Reactor Vessel, ID, in.	173	Isolation Valves	No
Core		Number of Loops	4
Number of Assemblies	193	Steam Generator	
Rod Array	17x17	Model	D4 (U1)/D5 (U2)
Rod OD, in.	0.374	Shell Design Pressure, psia	1300
Number of Grids	8	Reactor Coolant Pump	
Active Fuel Length, in.	144	Model/Weir	93A/Yes
Number of Control Rods	53	Pump Motor, hp	7000
		Frequency, Hz	60

UPRATING OPERATING DESIGN PARAMETERS	Original	CASE 1(2)	CASE 2(2)	CASE 3(2)
NSSS Power, %	100	100	100	100
MWt	3425	3582	3582	3582
10^6 BTU/hr	11,687	12,222	12,222	12,222
Reactor Power, MWt	3411	3565	3565	3565
10^6 BTU/hr	11,639	12,164	12,164	12,164
Assumed Flow, Loop gpm	95,700	97,900	97,900	97,900
Reactor 10^6 lb/hr	142.0	146.1	144.6	144.6
Reactor Coolant Pressure, psia	2250	2250	2250	2250
Core Bypass, %	5.8	5.8	5.8	5.8
Reactor Coolant Temperature, °F				
Core Outlet	622.1	619.4	626.0	626.0
Vessel Outlet	618.8	616.1	622.7	622.7
Core Average	592.5	589.1	596.2	596.2
Vessel Average	589.2	585.7(3)	592.7(3)	592.7(3)
Vessel/Core Inlet	559.6	555.3	562.7	562.7
Steam Generator Outlet	559.3	555.0	562.4	562.4
Steam Generator				
Steam Temperature,°F	544.6	533.6/537.2	541.9/545.4	545.7/549.2
Steam Pressure, psia	1000	913(1)/941(1)	978(1)/1007(1)	1009(1)/1039(1)
Steam Flow, 10^6 lb/hr total	15.14	15.89/14.77	15.93/14.81	15.96/14.83
Feed Temperature,°F	440	444.6/390	444.6/390	444.6/390
Moisture, % max.	0.25	0.25	0.25	0.25
Tube Plugging, %	0	10	10	0
Zero Load Temperature,°F	557	557	557	557

HYDRAULIC DESIGN PARAMETERS	
Mechanical Design Flow, gpm	105,000
Minimum Measured Flow, gpm total	394,000

FOOTNOTES:
(1) Steam pressures listed are applicable to Unit 2 generators, add 7 psi to reflect steam pressure for Unit 1 generators.
(2) For system and component analysis only.
(3) T_{AVG} range +/-3.5°F.

B.2 DESIGN TRANSIENTS

NSSS DESIGN TRANSIENTS

The revised design conditions in Table IV-1 and the NSSS design transients applicable to the uprated conditions serve as primary inputs to the evaluation and analysis of the stress and fatigue analyses of the NSSS systems and components. Current primary and secondary design transients were reviewed in order to determine their continued applicability to uprated conditions.

Primary Side Transients

The review of the primary side design conditions listed in Table IV-1 indicates that the full power values of vessel outlet, vessel inlet and vessel average temperatures (hot leg, cold leg, and loop average temperatures) vary by less than 4°F from the previously applicable design values. Given the conservative assumptions used to develop the current design transients (e.g., initial conditions, unavailability of control systems during certain transients), a 4°F change in primary side full power temperatures is considered insignificant during all transient conditions. Therefore, the revised conditions have negligible impact on the primary side design transients, and the previously applicable NSSS design transients for the primary side continue to apply, without modification, at the uprated conditions.

Secondary Side Transients

With regard to secondary design parameters, the revised design conditions in Table IV-1 indicate that the plant may operate with lower full power values for steam temperature, steam pressure and feedwater temperature. Lower nominal steam temperatures (e.g., from 544.6 to 533.6°F) and pressures (e.g., from 1000 psia to 913 psia) result in larger changes from initial conditions than the range reflected in the current NSSS design transients. Similarly, at uprated conditions a 50°F reduction in feedwater temperature (i.e., from 440 to 390°F) results in secondary-side operating conditions that are more limiting than were determined at higher feedwater temperatures. Therefore, some of the existing secondary side design transients were modified to reflect the lower nominal feedwater and steam temperatures. These modified design transients were used in the steam generator analyses.

AUXILIARY EQUIPMENT DESIGN TRANSIENTS

The review of the NSSS auxiliary equipment design transients was based on a comparison between the revised operating conditions in Table IV-1 and the parameters which make up the current auxiliary equipment design transients.

This review determined that only those temperature transients affected by a change in T_{cold} were impacted. These transients are currently based on an assumed full load NSSS worst case T_{cold} of 560°F and have been modified to reflect the new worst case value of 563°F.

CONCLUSIONS

The revised design conditions did not require a change to the primary side design transients. However, the revised conditions did require modification of several secondary side transients which have been considered in the steam generator design discussed later. In addition, the auxiliary transient temperature curves were changed to accommodate the T_{cold} increase from 560°F to 563°F and have been considered in the design of the NSSS auxiliary equipment discussed later.

B.3 NSSS SYSTEMS

This section presents the results of the evaluations and analyses performed in the NSSS systems area to support the revised design conditions in Table IV-1. The systems addressed in this chapter include Fluid Systems and NSSS/BOP interface Systems. The results and conclusions of each analysis are presented within each subsection.

NSSS FLUID SYSTEMS

Reactor Coolant System

The Reactor Coolant System (RCS) consists of four heat transfer loops connected in parallel to the reactor vessel. Each loop contains a reactor coolant pump (RCP), which circulates the water through the loops and reactor vessel, and a steam generator (SG), where heat is transferred to the main steam system (MSS). In addition, the RCS contains a pressurizer which controls the RCS pressure through electrical heaters, water sprays, power operated relief valves (PORVs) and spring loaded safety/relief valves. The steam discharged from the PORVs and safety/relief valves flows through interconnecting piping to the pressurizer relief tank (PRT).

Assessments were performed to demonstrate that the RCS design basis functions could still be met at the revised design conditions.

Pressurizer spray flow capability was calculated considering the T_{avg} range of 585.7 to 592.7°F. The calculations demonstrate that the minimum required spray flow of 900 gpm can be achieved over the entire range of anticipated RCS process conditions.

Also, the maximum expected T_{hot} at uprated conditions is 622.7°F. This temperature is well within the RCS loop design temperature of 650°F.

With respect to the PRT, the revised T_{avg} range will change the nominal full load pressurizer steam volume at uprated conditions. In general, the reference nominal pressurizer level is coordinated with RCS T_{avg} such that an increase in T_{avg} raises the nominal pressurizer reference level condition. With respect to the PRT discharge analysis, a lower RCS T_{avg} condition is more limiting than a higher RCS T_{avg} condition, since pressurizer level is lower and steam volume is larger. It was determined that the current PRT level setpoint was still sufficient to accommodate the expected increase in steam volume from the reduced T_{avg} from 589.2°F to 585.7°F.

Chemical and Volume Control System

The Chemical Volume and Control System (CVCS) provides for boric acid addition, chemical additions for corrosion control, reactor coolant clean-up and degasification, reactor coolant make-up, reprocessing of water letdown from the RCS, and RCP seal water injection. During plant operation, reactor coolant flows through the shell side of the regenerative heat exchanger and then through a letdown orifice. The regenerative heat exchanger reduces the temperature of the reactor coolant and the letdown orifice reduces the pressure. The cooled, low pressure water leaves the reactor containment and enters the auxiliary building. A second temperature reduction occurs in the tube side of the letdown heat exchanger followed by a second pressure reduction due to the low pressure letdown valve. After passing through one of the mixed bed demineralizers, where ionic impurities are removed, coolant flows through the reactor coolant filter and enters the Volume Control Tank (VCT).

In the assessment of CVCS operation at revised RCS operating temperatures, the maximum expected RCS T_{cold} must be less than or equal to the applicable CVCS design temperature and less than or equal to the heat exchanger design inlet operating temperature. The former criterion supports the functional operability of the system and its components. The latter criterion supports the conclusion that the current heat exchanger design operating conditions remain bounding.

With regard to the CVCS thermal performance, the T_{cold} of 562.7°F is slightly higher than the design system inlet temperature of 560°F. This slight increase in heat exchanger inlet temperature can be accommodated by the system. Although the letdown temperature at the outlet of the regenerative heat exchanger will increase slightly, the letdown heat exchanger will reduce the letdown system temperature to its design value. The increase in heat transfer to the component cooling water system will be less than 0.6% due to the increase in regenerative heat exchanger outlet temperature. The excess letdown path is used to process excess effluents associated with fluid expansion during plant heatup and thus, is unaffected by the revised T_{cold} at full power conditions. The excess letdown heat exchanger outlet flow is throttled to maintain the desired outlet temperature and efflux. Therefore, operation of this system is unaffected by the temperature change.

Safety Injection System

The Safety Injection System (SIS) is an Engineered Safeguards System used to mitigate the effects of postulated design basis events. The basic functions of this system include providing short and long term core cooling, and maintaining core shutdown reactivity margin. The SIS is also referred to as the emergency core cooling system (ECCS).

The SIS is comprised of three subsystems. The passive portion of the system is the four accumulator vessels which are connected to each of the RCS cold leg pipes. Each accumulator contains borated water under pressure (nitrogen cover gas). The borated water automatically injects into the RCS when the pressure within the RCS drops below the operating pressure of each of the accumulators.

The "active" part of the SIS injects borated water into the reactor following a break in either the reactor or steam systems in order to cool the core and prevent an uncontrolled return to criticality. Two safety injection (SI) pumps and two residual heat removal (RHR) pumps take suction from the Refueling Water Storage Tank (RWST) and deliver borated water to four cold leg connections via the accumulator discharge lines. In addition, two centrifugal charging pumps take suction from the RWST on SI actuation and provide flow to the RCS via separate SI connections on each cold leg. This arrangement of SI pumps can provide safety injection flow at any RCS pressure up to the set pressure of the pressurizer safety valves.

The revised design conditions have no direct effect on the overall performance capability of the SIS. These systems will continue to deliver flow at the design basis RCS and containment pressures since there are no changes in the RCS operating pressure.

Other NSSS Systems

Other NSSS systems which were reviewed and found to be acceptable for power uprate were the Boron Recycle System, the Boron Thermal Regeneration System, and the Gaseous Waste Processing System.

Cold Overpressure Mitigation System (COMS)

COMS is designed to protect the RCS from overpressure events when the RCS is below a temperature of approximately 350°F. Changes to full power operating parameters, such as NSSS power, T_{avg}, and RCS pressure do not impact COMS. An increase in steam generator tube plugging to 10%, which reduces the RCS volume, can have a small impact on the peak RCS pressure experienced during the design basis mass injection event. An evaluation confirmed that the current mass injection event is still valid for the revised design conditions.

NSSS/ BOP INTERFACE SYSTEMS

The following Balance-of-Plant (BOP) fluid systems were reviewed for compliance with Westinghouse Nuclear Steam Supply Systems (NSSS)/BOP interface guidelines. These guidelines do not necessarily reflect design requirements; some are provided to ensure the assumptions of the NSSS analyses remain valid, and the remainder are intended to facilitate operation of the plant.

A comparison of the revised design conditions with the current design conditions previously evaluated for systems and components indicates differences that could impact the performance of these systems.

Main Steam System

The uprating coupled with the potential reduction in full-load steam pressure to the average minimum value of 920 psia for Unit 1 (913 psia for unit 2) impacts the main steam line pressure drop. At the revised design conditions, the steam line pressure drop could increase by as much as 21.8 percent due to the increased steam flow and lower steam density.

The following summarizes the evaluation of the major steam system components relative to the power uprate conditions. The major components of the Main Steam System (MSS) are the Steam Generator Main Steam Safety Valves (MSSVs), the SG Atmospheric Relief Valves (ARVs), and the Main Steam Isolation Valves (MSIVs).

Steam Generator Main Steam Safety Valves

The MSSVs must have sufficient capacity so that main steam pressure does not exceed 110 percent of the MSS design pressure (the maximum pressure allowed by the ASME B&PV Code). Each operating unit at Comanche Peak has twenty MSSVs with a total capacity of 18.19×10^6 lbm/hr. Therefore, based on the revised design conditions, the capacity of the installed MSSVs meets the required sizing criterion.

Steam Generator Atmospheric Relief Valves (ARVs)

The primary function of the ARVs is to provide a means for decay heat removal and plant cooldown by discharging steam to the atmosphere when either the condenser, the condenser circulating water pumps, or steam dump to the condenser is not available. Under such circumstances, the ARVs in conjunction with the Auxiliary Feedwater System (AFWS) permit the plant to be cooled down from the pressure setpoint of the lowest-set MSSVs to the point where the Residual Heat Removal System (RHRS) can be placed in service. During cooldown, the ARVs are either automatically or manually controlled. Each ARV controller automatically compares steam line pressure to the pressure setpoint, which is manually set by the plant operator.

In the event of a steam generator tube rupture event in conjunction with loss of offsite power, the ARVs are used to cool down the RCS to a temperature that permits equalization of the primary and secondary pressures at a pressure below the lowest-set MSSV. RCS cooldown and depressurization are required to preclude steam generator overfill and to terminate activity release to the atmosphere.

For the revised design conditions, each steam generator ARV is required to have a capacity at least equal to 62,150 lbm/hr at 100 psia inlet pressure. At these conditions, this capacity permits a plant cooldown to RHRS operating conditions in 5 hours (at a cooldown rate of 50°F/hr) assuming a minimum of 2 hours at hot standby. This sizing is compatible with normal cooldown capability and minimizes the water supply required by the AFWS. This is based on one train of auxiliary feedwater (AFW) operating and flow going through three SGs. The design capacity of the installed ARVs meets the required sizing criterion.

Main Steam Isolation Valves and Main Steam Isolation Bypass Valves

The MSIVs are located outside the containment and downstream of the MSSVs. The valves function to prevent the uncontrolled blowdown of more than one steam generator and to minimize the RCS cooldown and containment pressure to within acceptable limits following a main steam line break. To accomplish this function, the original design requirements specified that the MSIVs must be capable of closure within 5 seconds of receipt of a closure signal against steam break flow conditions in either the forward or reverse direction.

Rapid closure of the MSIVs following postulated steam line breaks causes a significant differential pressure across the valve seats and a thrust load on the main steam system piping and piping supports in the area of the MSIVs. The worst cases for differential pressure increase and thrust loads are controlled by the steam line break area (i.e., mass flow rate and moisture content), throat area of the steam generator flow restrictors, valve seat bore, and no-load operating pressure. Since these variables and no-load operating pressure are not impacted by the revised design conditions, the design loads and associated stresses resulting from rapid closure of the MSIVs will not change. Consequently, the revised design conditions have no significant impact on the interface requirements for the MSIVs.

The MSIV bypass valves are used to warm up the main steam lines and equalize pressure across the MSIVs prior to opening the MSIVs. The MSIV bypass valves perform their function at no-load and low power conditions where the revised design conditions have no significant impact on main steam conditions (e.g., steam flow and steam pressure). Consequently, the revised design conditions have no significant impact on the interface requirements for the MSIV bypass valves.

Steam Dump System

The steam dump system creates an artificial steam load by dumping steam from upstream of the turbine valves to the main condenser. The Westinghouse sizing criterion recommends that the steam dump system (valves and pipe) be capable of discharging 40% of the rated steam flow at full-load steam pressure to permit the NSSS to withstand an external load reduction of up to 50% of plant rated electrical load without a reactor trip. To prevent a trip, this transient requires all NSSS control systems to be in automatic, including the Reactor Control System, which accommodates 10% of the load reduction. A steam dump capacity of 40% of rated steam flow at full load steam pressure also prevents MSSV lifting following a reactor trip from full power.

The steam dump system is composed of twelve condenser steam dump valves. The total capacity for all twelve valves exceeds the Westinghouse sizing criterion of 40% of rated steam flow.

NSSS operation within the revised design conditions at lower steam generator pressures and higher steam flows will result in a reduced steam dump capability.

An evaluation for these conditions indicates that the total steam dump capacity still exceeds the required sizing criterion.

In addition, to provide effective control of flow on large step load reductions or plant trip, the steam dump valves are required to go from full-closed to full-open in 3 seconds at any pressure between 50 psi less than full load pressure and steam generator design pressure. The dump valves are also required to modulate to control flow. These requirements remain applicable for the revised design conditions.

Condensate and Feedwater System

The Condensate and Feedwater System (C&FS) must automatically maintain steam generator water levels during steady-state and transient operations. The revised design conditions will impact both feedwater volumetric flow and system pressure drop.

The major components of the C&FS are the Main Feedwater Line Flow Restricting Orifices, Feedwater Isolation Valves, the Feedwater Control Valves, and the Condensate and Feedwater System Pumps.

Main Feedwater Line Flow Restricting Orifices

The main feedwater line to each steam generator incorporates a Feedwater Bypass System (FBS). The function of the FBS is to minimize the potential occurrence of water hammer in the steam generators, and to mitigate flow induced tube vibration in the steam generators. The FBS includes the feedwater split flow bypass line which connects the main feedwater line to the feedwater preheater bypass line inside containment. The feedwater preheater bypass line delivers flow into the steam generator auxiliary feedwater nozzle. The main purpose of the feedwater split flow bypass is to limit flow through the preheater to control flow-induced tube vibration. The required flow split between the auxiliary nozzle and the main feedwater nozzle is facilitated by a flow restricting orifice installed in each main feedwater line just downstream of the feedwater split flow bypass line connection. The Unit 1 orifices are less restrictive than Unit 2 and therefore allow more flow through the main nozzles. An evaluation of the full load main feedwater flow-split for the range of revised design conditions (up to 3582 MWt) indicates that the main nozzle flow rate will be below the maximum permissible flow to the steam generator main nozzles for both units. Although the increase in main feedwater preheater flow in Unit 1 is projected to increase the rate of SG tube wear, any such increase is anticipated to be modest and the current Unit 1 eddy current inspection program is adequate to monitor any increased tube wear.

Feedwater Isolation Valves/Feedwater Control Valves

The feedwater isolation valves (FIVs) are located outside containment and downstream of the feedwater control valves (FCVs). The valves function in conjunction with the primary isolation signals to the FCVs and backup trip signals to the feedwater pumps to provide redundant isolation of feedwater flow to the steam generators following a steam line break or a malfunction in the steam generator level control system. Isolation of feedwater flow is required to prevent containment overpressurization and excessive reactor coolant system cooldowns. To accomplish this function, the FIVs and the backup FCVs must be capable of fast closure, that is within about 5 seconds following receipt of any feedwater isolation signal.

The quick-closure requirements imposed on the FIVs and the backup FCVs causes dynamic pressure changes that may be of large magnitude and must be considered in the design of the valves and associated piping. The worst loads occur following a steam line break from no load conditions with the conservative assumption that all feedwater pumps are in service providing maximum flow following the break. Since these conservative assumptions are not impacted by the revised design conditions, the design loads and associated stresses resulting from rapid closure of these valves will not change.

Condensate and Feedwater System (C&FS) Pumps

The C&FS available head in conjunction with the FCV characteristics must provide sufficient margin for feed control to ensure adequate flow to the steam generators during steady-state and transient operation. A continuous steady feed flow should be maintained at all loads. To assure stable feedwater control, with variable speed feedwater pumps, the pressure drop across the FCVs at rated flow (100 percent power) should be approximately equal to the dynamic losses from the feed pump discharge through the steam generator. In addition, adequate margin should be available in the FCVs at full load conditions to permit a C&FS delivery of 103 percent of rated flow with a 75 psi pressure increase above the full load pressure with the FCVs fully open. Because actual operating conditions may change over time due to tube plugging, etc., the FCV and pump speed controller may be recalibrated as necessary to optimize performance.

To provide effective control of flow during normal operation, the FCVs are required to stroke open or closed in 20 seconds over the anticipated inlet pressure control range (approximately 0-1600 psig). Additionally, rapid closure of the FCVs is required in 5 seconds after receipt of a trip close signal in order to mitigate certain transients and accidents. These requirements are still applicable at the uprated conditions. Other C&FS evaluations, including the feedwater and condensate pumps, are contained in the Balance of Plant section.

Auxiliary Feedwater System (AFWS)

The AFWS supplies feedwater to the secondary side of the steam generators at times when the normal feedwater system is not available, thereby maintaining the heat sink of the steam generators. The system provides feedwater to the SGs during normal unit startup, hot standby, and cooldown operations and also functions as an Engineered Safeguards System. In the latter function, the AFWS is directly relied upon to prevent core damage and system overpressurization in the event of transients and accidents such as a loss of normal feedwater or a secondary system pipe break.

Auxiliary Feedwater Storage Requirements

The AFWS pumps are normally aligned to take suction from the condensate storage tank (CST). To fulfill the Engineered Safety Features (ESF) design functions, sufficient feedwater must be available during transient or accident conditions to enable the plant to be placed in a safe shutdown condition.

Sufficient CST useable inventory must be available to bring the unit from full power to hot standby conditions, maintain the plant at hot standby for 4 hours, and then cooldown the RCS to the residual heat removal system cut-in temperature (350°F) in 5 hours. In addition, the CPSES licensing bases requires that the minimum useable inventory must also be adequate to support 18 hrs of decay heat removal at no-load plant conditions. In light of these requirements, an analysis was performed at the revised design conditions which demonstrated that the existing minimum useable inventory of 249,100 gals is adequate.

Steam Generator Blowdown System

The Steam Generator Blowdown System is used in conjunction with the Chemical Feed and Sampling Systems to control the chemical composition of the steam generator shell water within the specified limits. The blowdown system also controls the buildup of solids in the steam generator water.

Two blowdown locations are provided for each Steam Generator, a tube sheet connection and a shell connection. The tube sheet maximum allowable blowdown flow rate decreases with power. However, the total allowable blowdown flow from each steam generator can be maintained constant since the shell blowdown flow can be increased by the same amount. Therefore, the ability of the system to control the rate of addition of dissolved solids to the secondary systems by condenser leakage or makeup water is not impacted by the revised design conditions. Also, the reduction in tube sheet blowdown capability is not significant in terms of the ability of the system to control the rate of generation of particles by erosion-corrosion effects within the secondary systems, since the required blowdown rate from the tube sheet to control particulate is well within the allowable tube sheet blowdown flow rate.

The actual required blowdown flow rates during plant operation will not be significantly impacted by the revised design conditions, since neither the rate of addition of dissolved solids or the rate of addition of particulates into the steam generators will be significantly impacted.

CONCLUSIONS

The following is a brief summary of the NSSS/BOP interface evaluation conclusions.

Main Steam System

Operation at reduced steam pressures and corresponding higher pressure drops will have a negative but acceptable effect on plant heat rate.

The capacity of the installed MSSVs meets the Westinghouse sizing criterion for the proposed range of NSSS operating conditions.

The capacity of the installed ARVs meets the Westinghouse sizing criterion for the proposed range of NSSS operating conditions.

The MSIVs and MSIV bypass valves are not adversely impacted by the uprating.

Steam Dump System

The capacity of the steam dump system continues to comply with the Westinghouse sizing criterion for the proposed range of NSSS operating conditions.

Condensate and Feedwater System

For the range of uprated NSSS operating conditions, the main feedwater flow restriction orifices provide an acceptable flow split between the steam generator main nozzles and auxiliary nozzles at full load.

Auxiliary Feedwater System

The minimum flow requirements of the AFWS are dictated by accident analyses which are unaffected by the proposed uprate. Therefore, the AFWS performance remains acceptable at the uprated operating condition.

The CST minimum useable inventory of 249,100 gals remains adequate and complies with the plant design bases requirements for the range of uprated NSSS operating conditions.

Steam Generator Blowdown System

The actual required blowdown flow rates during plant operation are not significantly impacted by power uprate, since neither the rate of addition of dissolved solids nor the rate of addition of particulates into the steam generators is significantly impacted by power uprate.

B.4 NSSS COMPONENTS

REACTOR VESSEL

The reactor vessel (RV) was evaluated at the revised design conditions with respect to the structural acceptability of the vessel and reactor vessel integrity in terms of the impact due to neutron fluence.

Reactor Vessel Structural Evaluation

Evaluations were performed to assess the effects of the revised design conditions in Table IV-1 on the most limiting vessel locations with regard to ranges of stress intensity and fatigue usage factors in each of the regions, as identified in the reactor vessel stress reports and addenda. The evaluations considered a limiting worst case set of operating parameters from among the high temperature conditions, the low temperature conditions, and the original design basis. As a result, all of the design conditions in Table IV-1 are fully enveloped by the evaluations, and reactor vessel operation in accordance with the parameters for the remainder of the current operating licenses is justified.

The revised design conditions, in particular T_{hot} and T_{cold}, increase the maximum ranges of stress intensity and fatigue usage for the primary outlet and inlet nozzles. These increased stress intensities remain within the acceptance criterion of $3S_m$, and the maximum cumulative fatigue usage factors continue to remain below the acceptance limit of 1.00.

Reactor Vessel Integrity - Neutron Irradiation

Several analyses are performed to determine the impact that neutron irradiation has on reactor vessel integrity. The most critical area is the beltline region of the reactor vessel since it is predicted to be most susceptible to neutron damage. These analyses include a surveillance capsule withdrawal schedule, heat-up and cooldown pressure-temperature limit curves, pressurized thermal shock calculations and upper shelf energy evaluations. All of these analyses and evaluations can be affected by changes in the neutron fluences and operating temperatures and pressures. A power uprating to a core thermal power of 3565 MWt would be expected to increase the neutron fluences due to the increased power distributions. However, it was determined that based on CPSES operation, the calculated fluences used in the current vessel design, bound the fluences expected to occur with the revised design conditions. In addition, the changes in the RCS temperature are well within the temperature ranges assumed in these analyses. Thus, the revised design conditions do not invalidate these analyses.

Pressurized Thermal Shock

The highest current RT_{PTS} end of license value for the Unit 1 and 2 reactor vessels have been previously docketed as 100°F and 94°F respectively which is nominally 170°F below the screening criteria of the Pressurized Thermal Shock (PTS) Rule (10CFR50.61). Two Unit 1 and one Unit 2 surveillance capsules have been analyzed confirming the similarity between the two vessels in irradiated and nonirradiated material properties. The results of these surveillance capsule evaluations have confirmed that the early projections for CPSES vessel materials were conservative. In addition, the majority of the irradiation-induced shift in vessel material properties occurs early in life. Therefore, with substantial margin to the RT_{PTS} screening criteria and a nominal 1.4% and 0.4% increase in fluence for Units 1 and 2, respectively, the change in the RT_{PTS} value would not be significant and a revised Pressurized Thermal Shock report is unnecessary.

End-of-Life Reactor Vessel Fluence

The existing fast neutron fluence data used in the reactor vessel design remains bounding for the uprated power conditions. This conclusion is based on the most recent fluence evaluation performed in conjunction with the withdrawal of surveillance capsule Y (second capsule) from the CPSES Unit 1 reactor. In this evaluation, the inclusion of the impact of low leakage fuel management reduced the Unit 1 fluence projections by approximately 33% relative to the values used in the Unit 1 reactor vessel design. A similar reduction is anticipated for Unit 2 when the upcoming second surveillance capsule evaluation is performed for the Unit 2 reactor. This 33% margin more than offsets the nominal 1.4% and 0.4% increase in fluence for Units 1 and 2, respectively, that could be caused by the subject uprating. Thus, the fluence values used in the design bound the new best estimate fluence projections including consideration of the uprating.

Cold Leg Temperature

The reference value for T_{avg} will not be changed as part of the power uprate modification. However, because the core power will be increased by 1.4% for Unit 1 and 0.4% for Unit 2, the ΔT will increase by 1.4% for Unit 1 and 0.4% for Unit 2. The current ΔT is approximately 60°F. Thus, with a constant value of T_{avg}, T_{cold} is expected to decrease by approximately 0.4°F for Unit 1 and 0.1°F for Unit 2, and T_{hot} is expected to increase by approximately 0.4°F for Unit 1 and 0.1°F for Unit 2. The expected cold leg temperatures remain within the range assumed in the development of the equations and tables which form the bases for evaluating the neutron irradiation effects on vessel integrity.

REACTOR INTERNALS

The reactor internals support and orient the fuel and control rod assemblies, absorb control rod assembly dynamic loads and transmit these and other loads to the reactor vessel. The internals also direct flow through the fuel assemblies, provide adequate cooling to various internal structures and support the in-core instrumentation. The increase in thermal design flow and changes in the RCS temperatures, reported in Table IV-1, produce changes in the boundary conditions experienced by the reactor internals components. Also, increases in core thermal power may increase nuclear heating rates in the lower core plate, upper core plate and baffle-barrel former region.

Core Bypass Flow Calculation

Bypass flow corresponds to the amount of reactor coolant flow that bypasses the core region and is not considered effective in the core heat transfer process. The principal core bypass flows are the baffle barrel region, vessel head cooling spray nozzles, vessel outlet nozzle gap, baffle plate cavity gap and the thimble tubes. Slight variations in the size of some of the bypass flow paths, such as gaps at the vessel core barrel and outlet nozzles, occur due to unit-specific as-built dimensions or due to different fuel assembly designs and changes in the RCS conditions. Therefore, analyses were performed for each unit to determine core bypass flow values either to demonstrate that the design bypass flow limit is not exceeded or to determine a revised design core bypass flow.

The analysis results indicate that the bypass flow for each unit remains less than the 5.8% assumed in the development of the revised design conditions and, therefore is acceptable.

RCCA Drop Time Analyses

The RCCAs represent the interface between the fuel assemblies and the other internal components. The Technical Specifications require that the RCCA drop time be less than or equal to 2.4 seconds. The revised design conditions, in particular the reduced T_{cold}, can increase the drop time due to the increased fluid density. An evaluation was performed to demonstrate continued compliance with the current technical specification value at the revised design conditions. Further, the RCCA drop times are explicitly confirmed, by measurement after each refueling outage, to meet the times assumed in the accident analyses.

Hydraulic Lift Forces

The reactor internals hold-down spring is essentially a large diameter belleville type spring of rectangular cross section. The purpose of this spring is to maintain a net clamping force between the reactor vessel head flange and upper internals flange and the reactor vessel shell flange and the core barrel flange of the internals. An evaluation was performed to determine hydraulic lift forces on the various reactor internal components to ensure that the reactor internals assembly remains seated and stable for all conditions. Increases in mechanical design flow or changes in operating temperatures can reduce the clamping force. It was determined that, with the revised design conditions, the reactor internals assembly would remain seated and stable.

Mechanical Evaluations

The revised design conditions do not affect the current design bases for seismic and LOCA loads. Thus, it was not necessary to re-evaluate the structural affects from seismic OBE and SSE loads and the LOCA hydraulic and dynamic loads. With regards to flow and pump induced vibration, the current analysis is based on a mechanical design flow which was not impacted by the revised design conditions. The revised design conditions slightly alter the T_{cold} and T_{hot} fluid densities which will slightly change the forces induced by flow. However, these changes are insignificant when compared to the current design temperature ranges. Thus, the impact of the revised design conditions on the mechanical loads is acceptable.

Structural Evaluations

Structural evaluations are required to demonstrate that the structural integrity of the reactor components is not adversely affected by the change in RCS conditions and transients and/or by secondary effects of the change on reactor thermal hydraulic or structural performance. The presence of heat generated in reactor internal components, along with the various fluid temperatures, results in thermal gradients within and between components. These thermal gradients result in thermal stresses and thermal growth which must be accounted for in the design and analysis of various components. The core support structures affected by the revised design conditions (Table IV-1) are discussed in the following sections.

The primary inputs relevant to the evaluations are the revised design conditions in Table IV-1 and the gamma heating rates and the NSSS design transients which remain bounding for the revised design conditions. The gamma heating rates were increased to account for an increase in core thermal power to 3565 MWt.

The reactor internals components subjected to heat generation effects (either directly or indirectly) are the upper and lower core plates, the lower core support, the core baffle plates, the former plates, the core barrel, the neutron panel, the baffle-former bolts and the barrel-former bolts. Note, however, that due to relatively low heat generation rates the upper core plate, the lower core support, and neutron panels experience little, if any, temperature rise over the surrounding reactor coolant.

Baffle-Barrel Region Evaluations

The baffle-barrel regions consist of a core barrel into which baffle plates are installed, supported by bolting interconnecting former plates to the baffle and core barrel.

The baffle-to-former bolts restrain the motion of the baffle plates that surround the core. These bolts are subjected to primary loads consisting of deadweight, hydraulic pressure differentials, seismic loads, as well as secondary loads consisting of preload, and thermal loads resulting from RCS temperatures and gamma heating rates. The baffle-to-former bolt thermal loads are induced by differences in the average metal temperature between the core barrel and baffle plate. In addition to providing structural restraint, the baffles also channel and direct coolant flow such that a "coolable core geometry can be maintained."

The thermal stresses in the core barrel shell in the core active region are primarily due to temperature gradients through the thickness of the core barrel shell. These temperature gradients are caused by the fluid temperatures between the inside and outside surfaces and the contribution of gamma heating.

To demonstrate that the baffle-barrel metal temperatures remain bounded by the generic analysis, a comparison of core power in the fuel assemblies at the periphery was made. The results indicated that the current structural and thermal analysis of record for the baffle-barrel region remain bounding for the revised design conditions. Thus, the baffle-barrel region is structurally adequate for the revised design conditions.

Lower Core Plate Structural Analysis

The lower core plate is a perforated circular plate that supports and positions the fuel assemblies. The plate contains numerous holes provided for fluid flow through the plate to each fuel assembly and the baffle-barrel region. The plate is bolted at the periphery to a ring welded to the inside diameter of the core barrel. The center span of the plate is supported by the lower support columns which are attached at the lower end to the lower support plate.

Temperature differences between components of the lower support assembly induce thermal stresses in the lower core plate. In addition, due to the lower core plate's proximity to the core and thermal expansion of fuel rods at power, the heat generation rates in the lower core plate due to gamma heating result in a significant temperature increase in this component. Thermal expansion of the lower core plate is restricted by the lower support columns, lower support plate and core barrel. These restraining items are exposed to the inlet temperature and have heat generation rates much lower than those found in the lower core plate.

Structural evaluations were performed to demonstrate that the structural integrity of the lower core plate was not adversely affected by the revised design conditions. These evaluations determine that the fatigue usage remains less than 1.0 and the plate is structurally adequate at the revised design conditions.

Upper Core Plate Structural Analysis

The upper core plate positions the upper ends of the fuel assemblies and the lower ends of the control rod guide tubes, thus serving as the transitioning member for the control rods in entry and retraction from the fuel assemblies. It also controls coolant flow exiting from the fuel assemblies and serves as a boundary between the core and the upper plenum. The upper core plate is restrained from vertical movement by the upper support columns which are attached to the upper support plate assembly. Lateral movement is restrained by four equally spaced core plate alignment pins.

The normal and upset stresses in the upper core plate are mainly due to hydraulic, seismic, and thermal loads. The total thermal stresses are due to secondary membrane stress and surface skin stress. Evaluations were performed to determine the impact that the revised design conditions had on the structural integrity of the upper core plate. As a result of the evaluation, it was concluded that the fatigue usage remains less than 1.0 and the plate is structurally adequate for the revised design conditions.

CONTROL ROD DRIVE MECHANISMS (CRDMS)

The pressure boundary portions of the CRDMs are exposed to the vessel/core inlet fluid. The conditions in Table IV-1 indicate that the maximum increase in vessel/core inlet temperature was from 559.6°F to 562.7°F. An analysis was performed to determine the impact that the revised design conditions had on the fatigue usage of the CRDM components. The results indicate that the stress and fatigue usage are still within ASME Code limits.

REACTOR COOLANT LOOP PIPING AND SUPPORTS

The revised design conditions were reviewed for impact on the existing design basis analysis for the reactor coolant loop piping, primary equipment nozzles, primary equipment supports and the pressurizer surge line piping. In particular, the temperature changes associated with the revised conditions could affect the loads in the components, the applicable stresses, and fatigue usage values since these temperature serve as the initial conditions in some of the design transients.

However, the evaluation concluded that the loads remain bounded by the values in the existing analyses and the stresses and fatigue usage values remain below the allowable limit.

Leak-Before-Break (LBB) Analysis

The current LBB evaluation was performed for the primary loops to provide technical justification for eliminating large primary loop pipe rupture as the structural design basis. In order to demonstrate the elimination of RCS primary loop pipe breaks, the following objectives were achieved:

Demonstrate that margin exists between the "critical" crack size and a postulated crack which yields a detectable leak rate.

Demonstrate that there is sufficient margin between the leakage through a postulated crack and the leak detection capability.

Demonstrate margin on applied load.

Demonstrate that fatigue crack growth is negligible.

An evaluation was performed which determined the impact of the revised design conditions on the LBB margins is negligible, and the LBB conclusions remain unchanged.

REACTOR COOLANT PUMPS (RCPs)

Structural Analysis

The pressure boundary portions of the RCPs are exposed to the steam generator outlet fluid. Table IV-1 indicate that the maximum change in steam generator outlet temperature was increased from 559.3 to 562.4°F. An analysis was performed to determine the impact of the revised design conditions on the stresses and fatigue usage of the RCP and associated components. The results indicated that the stress and fatigue usage remain within ASME Code limits.

RCP Motor Analysis

The RCP motors were evaluated for the limiting case loads based on the revised design conditions for continuous operation at revised hot loop rating, operation at revised cold loop rating, starting, and loads on thrust bearings. It was determined that for operation at the revised design conditions, the RCPs continue to comply with their applicable hot and cold loop operating ratings. Thus, the RCPs are able to accelerate at the resultant loads for the limiting case design conditions and the thrust bearings do not exceed their load ratings.

STEAM GENERATORS

Structural Integrity

The bases for the existing structural and fatigue analyses of the steam generators are contained in the Model D4 and Model D5 Steam Generator Stress Reports. An evaluation was performed to demonstrate that continued compliance with the ASME limits is maintained for the revised design conditions. This evaluation considered the most critical components with regard to stress and fatigue usage. The primary inputs for the evaluation are the revised design conditions in Table IV-1 and the secondary side design transients required to accommodate the changes in steam temperature, feedwater temperature and steam pressure. Table IV-1 indicates that the reactor coolant pressure remained unchanged at 2250 psia, while steam pressure could decrease from 1000 psia to a minimum value of 920 psia for Unit 1 (913 psia for Unit 2) or increase to a maximum value of 1046 psia for Unit 1 (1039 psia for Unit 2). Also, the steam temperature could either decrease from a current design value of 544.6°F to a value of 533.6°F or increase to a maximum value of 594.2°F. Finally, the feedwater temperature changed from a design value of 440°F to a design range of 390°F and 445°F.

The evaluation incorporated these inputs by developing scaling factors necessary to calculate the increased stress and fatigue usage. The results indicate that all applicable stresses and fatigue usage values remain within the allowable limits. Thus, the evaluation demonstrated that the critical components of the steam generators continue to comply with the requirements of the ASME Code at the revised design condition.

Thermal-Hydraulic Performance

Secondary side steam generator performance characteristics such as circulation, moisture carryover, hydrodynamic stability, heat flux and others are affected by increases in thermal power, and steam pressure. Steam pressure is, in turn, determined by the power as well as the primary side temperature, tube plugging level and feedwater temperature. The magnitude and importance of changes in the secondary side thermal hydraulic performance characteristics at the uprated power, with increased plugging, reduced primary side temperatures and a feedwater temperature range are assessed below.

Circulation Ratio/Bundle Liquid Flow

The circulation ratio is a measure of tube bundle liquid flow in relation to steam flow and is primarily a function of steam flow. The bundle liquid flow minimizes the accumulation of contaminants on the tube sheet and in the bundle. There is a slight decrease in the bundle flow which also has a minimal affect on its function. Thus, the bundle flows are still adequate.

Hydrodynamic Stability - Damping Factor

The hydrodynamic stability of a steam generator is characterized by the damping factor. A negative value of this parameter indicates a stable unit; i.e., small perturbations of steam pressure or circulation ratio will diminish rather than grow in amplitude causing instability in the control systems.

The Model D4 Steam Generator damping factors remain negative at about the same level for high temperature feedwater conditions (444.6°F) and change from –149 to –35 hr^{-1} for low temperature feedwater conditions. The Unit 2 damping factors remain highly negative, at a level comparable to the current design, for all cases. Thus, the steam generators remain hydro-dynamically stable for all uprated cases.

Heat Flux - Nucleate Boiling Limits

The maximum heat flux transmitted from the primary to the secondary side is evaluated to determine that it remains within the applicable nucleate boiling limits. The peak heat flux increases with power and tube plugging. For uprating, the increased total heat load is passed through the same bundle heat transfer area, increasing the heat flux. For increased tube plugging (10%), the same heat load is passed through a smaller heat transfer area, also increasing the heat flux. However, the heat flux remains well within nucleate boiling limits and is comparable to values for a number of steam generators currently operating.

Secondary Side Pressure Loss

The total secondary side pressure drop for the steam generator increased by about 3 psi for Unit 1, which is considered very small in relation to the total feed system pressure drop. Unit 2 steam pressure losses are equal to or less than the current design pressure loss as a result of the reduced water level and circulation ratio at higher power levels offsetting the increase in steam flow. For the low feed temperature cases (390°F), the lower steam flow causes the total pressure loss to be less than the design condition value. Thus, the revised pressure loss has a negligible affect on feedwater system operation.

Moisture Carryover

Evaluations were performed to predict the amount of moisture carryover from the steam generators at the revised design conditions. Increases in steam flow, and reductions in steam temperature and pressure can increase the moisture carryover. The evaluations concluded that the moisture carryover is predicted to remain less than the current design value of 0.25% with plant operation at the revised design conditions.

U-Bend Fatigue Evaluation

Fluid elastic vibration and fatigue of unsupported, small radius U-bends can occur and lead to significant fatigue usage when "denting" is present at the top tube support plate. The model D5 steam generators installed in CPSES Unit 2 are not susceptible to "denting" and therefore this issue is not applicable to Unit 2. An evaluation was performed and determined that the revised design conditions will increase the susceptibility of several tubes in the Unit 1 steam generators. Prior to implementation of the proposed uprate in Unit 1, TXU Electric will determine whether additional preventive actions are required in addition to those previously taken and reported in response to NRC Bulletin 88-02.

Laser Welded Sleeving

The analytical basis for the use of Laser Welded Sleeving (LWS) at CPSES incorporated operation at up to 3582 MWt. Therefore, since the uprated design conditions have been considered for LWS, the basis for the process at CPSES is unchanged.

Alternate Repair Criteria

Alternate repair criteria provides a technical basis for justification of continued operation of tubes with indications at the tube support plate intersections. The supporting analyses are based on actual operating conditions, rather than design conditions. Therefore, the alternate Repair Criteria is generally not a function of power level. Changes in the primary and secondary temperatures could affect growth rates observed for the previous operating cycle, however, postulated changes in growth rates will be evaluated in accordance with Generic Letter 95-05 and are required to be submitted to the NRC within 90 days following steam generator eddy current inspection.

F* Distance for the Model D4 Steam Generator (Unit 1 only)

The F* distance is the distance of the hardroll expanded portion of a tube which provides a sufficient length of non-degraded tube expansion to resist pullout of the tube from the tubesheet. The F* distance was reevaluated and determined to be unaffected by the changes in normal operation steam pressures. Therefore, the revised design conditions have been considered for F* criteria and the F* value for the Feed Line Break remains valid.

PRESSURIZER

The pressurizer limiting locations from a structural standpoint are the surge nozzle, the spray nozzle, and the upper shell at the point of spray impingement. The pressurizer limiting operating condition occurs when the RCS pressure is high and the RCS hot leg temperature (T_{hot}) and cold leg temperature (T_{cold}) are low. The pressurizer structural evaluation was performed by comparing the key inputs in the current pressurizer stress report with the revised design conditions in Table IV-1. The results indicated that the design conditions used in the original analysis remain bounding for the revised design conditions.

NSSS AUXILIARY EQUIPMENT

The NSSS auxiliary equipment includes the heat exchangers, pumps, valves and tanks. These components were evaluated for the maximum expected cold leg temperature increase from 560°F to 563°F. The results indicated that the components continue to comply with the current design criteria since the fatigue usage values for each component remain below the allowable limit.

B.5 MASS ENERGY RELEASES AND LOCA HYDRAULIC FORCING FUNCTIONS

MASS ENERGY RELEASES

The LOCA (short and long term) and Steam Line Break (inside and outside) Mass and Energy Releases have been reviewed with respect to the 1.4% power uprate for Unit 1 and the 0.4 % uprate for Unit 2. Results of these reviews have determined that a calorimetric uncertainty of ±2% for the original licensed core power was incorporated into the analysis. The LOCA and Steam Line Break Mass and Energy releases originally calculated remain bounding.

LOCA HYDRAULIC FORCING FUNCTIONS

This section discusses the impact that the revised design conditions (Table IV-1) have on the LOCA hydraulic forcing functions. The purpose of a LOCA hydraulic forcing function analysis is to generate the hydraulic forcing functions and hydraulic loads that occur on Reactor Coolant System (RCS) components as a result of a postulated loss-of-coolant accident (LOCA). These forcing functions and loads are considered in the structural design of the NSSS components. In general, LOCA hydraulic forces increase with an increase in RCS coolant density and, consequently, LOCA hydraulic forces increase for lower RCS temperatures. The hydraulic forcing functions and loads that occur as a result of a postulated LOCA are calculated assuming a limiting break location and break area. The most limiting auxiliary line breaks are the 4" pressurizer spray line break on the cold leg and the 6" safety injection line break on the hot leg.

Description of Analyses/Evaluations Performed

An evaluation was performed to demonstrate that the current LOCA hydraulic forcing functions remain bounding for the revised design conditions in Table IV-1. Table IV-1 indicates that the maximum reduction in T_{cold} occurred from 559.6°F to 555.3°F. This temperature change was determined to increase the LOCA forces (due to the density increase) by less than 3% of their current values. The frequency of the LOCA forces pressure transient remained essentially unchanged and the amplitude of the LOCA decompression wave remained smaller when compared to the current values.

The 3% increase in forces was offset by a more accurate model of the loop at the break location. The more accurate model provides for up to a 30% decrease in the current forces. Thus, the current LOCA forces remain bounding for the revised design conditions.

C. BALANCE OF PLANT

A detailed evaluation of Unit 2 non-NSSS systems, structures, components, and related programs was completed which demonstrated continued compliance with all CPSES applicable industry and regulatory requirements at a core thermal power of 3458 MWt. This Unit 2 evaluation also specifically addressed Unit 1 applicability throughout, identifying those unit-specific areas of design documentation that remain to be reviewed to substantiate similar conclusions to support a Unit 1 uprate. Based on the Unit 2 evaluation conclusions, the similarity of the two CPSES units, and awareness of the unit differences that might be sensitive to the revised operating conditions, Unit 1 is expected to also remain in compliance with all CPSES applicable industry and regulatory requirements at a core thermal power of 3458 MWt. The detailed evaluation of Unit 1 non-NSSS systems, structures, and components and related programs will be completed prior to implementation of the requested Unit 1 uprate.

Electrical Load, Voltage, and Short Circuit Values for Auxiliary Electrical Distribution System

As a result of this uprate, no auxiliary load ratings are expected to change, and the loads are not expected to experience demands above their ratings. Therefore, the plant auxiliary electrical load will not change. The main generator electrical parameters remain the same, and the uprate capacity remains within the generator rating. The voltage controls and grid source impedance at the CPSES 345 kV grid will not be affected by this uprate; therefore, the evaluated voltages and short circuit values at different levels of the station auxiliary electrical distribution system will not change as a result of this uprate.

Environmental Qualification for the Safety Related Electrical Equipment

The normal environments for the plant buildings were assessed. The uprate has an insignificant effect on process fluid temperatures in the auxiliary, safeguards and electrical and control buildings. With the exception of the main feedwater system, the increase in the heat loads is caused by the increase in the decay heat load as it is transferred to the Component Cooling Water and Station Service Water Systems. The increase in these system temperatures has been shown to be fractions of a degree. The main feedwater temperature is changing by approximately 1°F. These small changes in fluid temperatures have an insignificant affect on the area temperatures. Similar conclusions were reached following the evaluations of the normal environmental conditions in the containment and fuel building.

The post-accident thermal environmental parameters were generated from computer models of the building structures that calculate the environment created by mass and energy releases during postulated pipe breaks. Evaluations concluded that through the use of the reduced power calorimetric uncertainty to offset the increase in reactor power, the existing mass and energy releases used in the environmental analyses for both inside and outside containment would remain valid. Because the mass and energy releases are not changed, the resulting environments are also unchanged. Therefore, the power uprate has no impact on the CPSES non-radiological equipment qualification program.

Generally, postulated radiation doses impacting equipment qualification depend primarily on post-accident contributions. However, normal-operating dose rate contributions are included in the design basis calculations. These normal-operating contributions are, in all cases, based on Westinghouse source terms which were originally generated for a power level of 104.5% RTP (i.e., 3565 MWt) and assumed 1% fuel defects. The assumption of 1% fuel defects is considered to be very conservative inasmuch as operation with that level of fuel leakage is not anticipated. Therefore, in regard to cases where normal operating equipment qualification dose rate contributions may be significant, it can safely be concluded that a power uprating would not cause dose rates or accumulated doses to exceed design basis values.

The effects of post-accident radiological consequences on equipment qualification were also evaluated. The source term used in the original analyses was generated for operation at a thermal power of 3565 MWt. Revised core fission product inventory calculations were performed; it was concluded that the original source term remains bounding. Based on the revised core fission product inventory, the post-accident gamma source strengths for some energies were found to slightly increase as a result of the power uprate; however, when applied in specific dose rate computations, it was shown that the accumulated doses at all times remain lower than current design-basis values. Therefore, it was concluded that all doses used for equipment qualification remain within existing design basis values.

In summary, the thermal power uprates have a negligible effect on normal environmental conditions and no effect on the environmental conditions currently used for equipment qualification.

Turbine/generator, Isophase Bus, Main Transformers, and Switchyards

The major turbine system components have been evaluated and were determined to be acceptable for continuous operation under the new operating conditions associated with a core rated thermal power of 3458 MWt. The potential missile energy from the high-pressure turbine is less than that from the low pressure turbine because of its much smaller potential missile mass and thicker turbine casing. Thus, the high-pressure turbine potential missiles are bounded by the low pressure turbine potential missiles.

The turbine drain system was evaluated. The required drain system capacity is based on start up and low load conditions, which are not affected by the power uprate. Therefore, it was concluded that the system capacity is adequate and no changes are required.

The moisture separator-reheaters (MSRs) and their sub-systems and components were reviewed and found to be adequate for service under the 1.4% for Unit 1 and 0.4% for Unit 2 uprate operating conditions. The MSR and hot reheat piping design pressures and temperatures are not changed by the uprate modification. Therefore, the lift pressure of the three shell side MSR safety valves is within the design requirements and no changes are required.

The turbine and its auxiliaries have been reviewed for operational impacts of the uprate on the relevant normal and abnormal modes of operation, and it was determined that there is no adverse impact.

The following operational parameters and systems are monitored during start-up and operational changes. Each has been reviewed and the determination was made that the uprate does not require any operational changes:

Main Steam Pressure
Stage Pressures
Stage and Exhaust Temperatures
Casing Temperature
Component Stresses
Expansions
The Bentley Nevada Turbine Supervisory Instrumentation System
The Siemens Berhhrungsloses Schaufel-Schwingungs Information System (BeSSI) LP Blade Vibration Monitoring System

The turbine system instrumentation and controls equipment does not constitute a hardware-imposed operating limit. However, specific setpoints will be revised for impacts on turbine operating parameters that are affected by the uprate. The original turbine design encompassed a “Valves Wide Open” operating condition representing a 5% increase in steam supply to the turbine relative to the 100% (3411 MWt) nominal operating point. This bounding design condition provides assurance that instrument ranges and capabilities retain adequate margin to accommodate the uprate.

The T_{ref} program uses turbine first stage (impulse chamber) pressure as an input to maintain the Reactor Coolant System at the appropriate temperature (T_{avg}). In order to more easily make necessary scaling adjustments and to accommodate making them at power, if necessary, these instrument channels have been re-scaled in units of percent turbine load.

The electrical systems associated with the turbine auxiliary systems are not affected by the uprate.

The steam turbine-driven polyphase generator is a four pole machine rated at 1350 MVA, with an operating point of 1215 MWe at a 0.9 power factor. This rating is based upon 60 psig hydrogen pressure, which is supplemented with water cooling for the stator and rotor.

The peak historical generator output for each unit plus the anticipated net increase due to the associated uprate, lies within the nameplate rating of the generator, Consequently, there will be no generator limitations to prevent operation at a core power of 3458 MWt.

A review of applicable calculations identified no need for any changes to equipment protection relay settings for the generator; although some process alarm setpoints for the generator and the exciter may require adjustment.

To deliver electrical power provided by the generator to the transmission system, each unit is equipped with an isolated phase bus, two main transformers, cabling, and two switchyard breakers. With the exception of the Unit 2 main transformers, which are rated for 650 MVA each, the remaining components are rated to deliver electrical power at or in excess of the main generator nameplate rating of 1350 MVA.

The isophase bus main section is rated at 37,000 amps, with each main transformer branch rated at 18,000 amps. The bus conductor will permit a temperature rise of 55°C, with the enclosure rated at 30°C rise. This will permit a total load (assuming a nominal voltage rating of 22 kV and 36,000 amps) of 1235 MWe at 0.9 pf. These figures are well in excess of the anticipated maximum generator output of 1190 MWe. The Isophase Bus will support the power increase with no modifications.

The Unit 2 transformers are more limiting and have a total capacity of 1300 MVA, just slightly less than the output rating of the generator (1350 MVA.) Since the reactive power of the generator must remain below approximately 400 MVARs due to the voltage rating on the primary windings, most of the MVA capacity can be utilized for real power. With an anticipated increased output of 1190 MWe from the generator, and assuming a maximum reactive output of 400 MVARs, this will result in an apparent output of 1255 MVA. Therefore, the transformers will operate within applicable limits at the power uprate condition.

Standard design practice at TXU Electric requires that switchyard equipment at least meets, but often exceeds, the nameplate rating of the main generator. The switchyard will accept the additional load without the need for any hardware modifications.

In summary, the turbine/generator and major electrical components extending from the isophase bus to the switchyard have adequate design margin to accept the additional power anticipated by the uprate.

Grid Stability and Reliability Analysis

The capacity of a single CPSES unit is less than 3% of the Electric Reliability Council of Texas (ERCOT)-estimated peak load. The uprate of either unit's capacity is negligible and the single-unit capacity will still remain less than 3% of the ERCOT-estimated peak load. Actual disturbances on the ERCOT system have occurred where large amounts of capacity were lost, as high as 10%, with no integrity degradation of the transmission system observed. The power uprates will not impact grid stability and reliability. Finally, availability and reliability of electric power from the transmission network to CPSES will not be affected; therefore, the station will continue to be in conformance with GDC 17.

Other Affected Systems

The following systems were identified as being affected by the uprate and thus required further evaluation: main steam, feedwater, steam generator blowdown, auxiliary feedwater, extraction steam, heater drains, condensate, condensate polishing, circulating water, turbine plant cooling, secondary sampling, spent fuel pool cooling, residual heat removal, component cooling, station service water, and combustible gas control systems.

These affected systems were evaluated based on the revised NSSS parameters (revised RCS temperatures, and revised steam generator temperature and steam flow rate) and the 3458 MWt heat balance. The evaluation process addressed: thermal-hydraulic performance, piping and support qualification, instrumentation and control functionality, equipment performance, and impact on the existing consequences of a pipe break. As noted previously in this section, those design attributes that are common to both units have been evaluated for both units. Design attributes, such as certain piping analyses that are unit-specific, have been evaluated for Unit 2 and will be completed for Unit 1 prior to implementation of the requested uprate in Unit 1.

The programs, components, and structures, identified as affected by the uprate were station blackout, equipment qualification, fire safe shutdown analysis, the concrete temperature at containment penetrations, fuel design, spent fuel storage, HVAC systems, radiological analyses, operating permits, the turbine/generator, isophase bus, main transformers, and switchyard. All of these areas were reviewed for the revised thermal-hydraulics parameters (temperature, pressure, flow rate), decay heat, and radioactivity resulting from the power uprate.

Other than design documentation changes and several instrumentation and control setpoint-related changes, it was determined that no hardware or operational modifications are required for Unit 2 and similar conclusions are anticipated for Unit 1. The only identified impact on plant operation is a slight reduction in the maximum normal (tube sheet) steam generator blowdown mass flow rate (described previously herein). However, total blowdown flow can be maintained by increasing the shell blowdown flow by the equivalent amount. This change will pose no limitation on the power the unit can generate and only marginally limits chemistry control flexibility. This reduction is necessary to preclude excessive erosion of the steam generator blowdown pipe.

D. RADIOLOGICAL CONSEQUENCES

The acceptability of the radiological consequences is based on both the continued validity of amount of fuel failures calculated for each postulated event and the associated radiological source term. Cycle-specific analyses, using the NRC-approved methodologies described in Technical Specification 5.6.5, are performed to ensure that all relevant event acceptance criteria are satisfied. These criteria include the analyzed limits on the extent of calculated fuel failures. A summary of the results of specific transients is provided below. A discussion of the effects of the proposed power uprates on the calculated radiological consequences follows.

Transient and Accident Analysis Methods

A ±2% power calorimetric measurement uncertainty allowance, consistent with 10CFR50, Appendix K requirements, is used in the topical reports identified in CPSES Technical Specification 5.6.5b. These topical reports describe the NRC-approved methodologies that support the CPSES safety analysis, including the small break and large break loss of coolant accident analyses. TXU Electric proposes that these topical reports be approved for use with a 0.6% uncertainty consistent with the reduction of calorimetric uncertainty. The NRC-approved Caldon Report ER-80P (as supplemented) was provided as the basis for this change.

Other conservative assumptions in the accident analyses are unaffected by the change in the uncertainty allowance applied to the initial core power level. For the non-LOCA events presented in FSAR Chapter 15, conservative initial conditions are used as described in FSAR Section 15.0.3.2. Other conservative assumptions considered in the non-LOCA transient analyses are described in FSAR Section 15.0.3. These assumptions include conservative core power distributions and peaking factors, conservative moderator and Doppler fuel temperature reactivity feedbacks, a small value of the control rod trip reactivity worth and conservatively-skewed trip reactivity insertion characteristics. With respect to available equipment and instrumentation, the beneficial effects of control systems are not credited in the analyses, and, in addition, a single failure of equipment or instrumentation required to mitigate the transient is assumed.

The generic models and methods used to analyze the transients are described in the methodology topical reports listed in Improved Technical Specification 5.6.5 and incorporated by reference into the FSAR. Factors that make up the inherent conservatism of the models and methods include the use of the point-kinetics approximation in lieu of a multi-dimensional representation of the reactor core and simplified steam generator models developed to conservatively predict the primary-to-secondary heat transfer rate.

In addition to these somewhat generic assumptions, additional conservative assumptions or models may be applied to specific transients. The more significant of these assumptions are delineated in the "Method of Analysis" discussion provided for each transient analysis presented in FSAR Chapter 15. Examples include maximized main and auxiliary feedwater flows for the main steam line break analysis and minimized auxiliary feedwater flows for the loss of feedwater analysis.

As described in FSAR Section 6.2.1.4, many of these same types of conservative assumptions are applied in the development of the steam line break mass and energy releases used to evaluate the containment response. The steam line break mass and energy release calculation developed for use in the environmental analysis outside of containment is described in FSAR Section 3.6B.2.5.2 Subsection 1. D. This analysis contains the typical, generic, conservative assumptions, as well as additional assumptions designed to increase the severity of the event with respect to the acceptance criteria for this specific application.

In summary, the allowance provided for the power calorimetric uncertainty is but one of several conservative assumptions that are applied to each of the safety analyses. However, through the use of the improved LEFM(check) instrumentation, the use of a smaller value of the power calorimetric uncertainty does not result in a reduction of analytical margin in the safety analyses.

The specific transients discussed below were selected based on questions received during the previous 1% uprate for Unit 2. The described results are specific to Unit 2 Cycle 6, in which the effects of the additional 0.4% RTP power uprate were explicitly considered. The conclusions are typical of all Unit 2 Cycle 6 analyses and of the results expected for the Unit 1 Cycle 10 analyses.

The system analyses for the following events were performed in accordance with the NRC approved methodologies described in Technical Specification (TS) 5.6.5, Item 14. Information contained in TS 5.6.5, Item 13 provides additional information. Where necessary, the comparison against the DNBR limit was performed as described in TS 5.6.5 Items 10, 11, and 12. Using these NRC-approved methods and considering operation at a Rated Thermal Power of up to 3458 MWt, compliance with all relevant event acceptance criteria was demonstrated for the Unit 2 Cycle 6 core configuration.

Uncontrolled RCCA Withdrawal from Power

The relevant acceptance criterion for the uncontrolled rod withdrawal at power event is compliance with the DNBR limit. For the analysis performed to support Unit 2 Cycle 6 operation, the full power cases were analyzed at a power level of 3479 MWt. The assumed initial power level was the licensed core thermal power (3445 MWt) plus an allowance of 1% of the initial power to account for power measurement uncertainties. The initial power assumed for this specific calculation was consistent with the proposed licensed rated power of 3458 MWt with a 0.6% uncertainty allowance. The calculated minimum DNBR for this case is approximately 1.36 (including any effects attributed to mixed cores and the lower plenum flow anomaly) which is greater than the DNBR limit value of 1.16.

Misloaded Fuel Assembly

The relevant acceptance criterion for the misloaded fuel assembly analysis is compliance with the DNBR limit. This event is analyzed by calculating a maximum allowable value of the nuclear enthalpy rise hot channel factor ($F_{\Delta H}$) such that the DNBR acceptance limit is just met. The thermal-hydraulic condition assumed for this evaluation was 3479 MWt, which bounds the current and proposed core power ratings with their associated power measurement uncertainty allowance. A reactor physics calculation is then performed to ensure that for the spectrum of potential misloaded assemblies identified, the resulting nuclear enthalpy rise hot channel factor is less than the maximum allowable value. Although case-specific DNBR calculations are not performed, compliance with the DNBR acceptance criterion is assured through compliance with the maximum allowable value of $F_{\Delta H}$.

Rod Ejection

The rod ejection event is analyzed to ensure compliance with the guidelines of 10CFR100. The source term for this analysis is based on assumptions concerning the integrity of the fuel rods which are confirmed to remain valid on a cycle-specific basis. The source term is based on assumptions of 10% fuel failures and 0.25% fuel melt. Fuel failures are assumed to occur if the DNBR limit is exceeded; fuel melt is assumed to occur if the peak centerline fuel temperature exceeds 4700°F. An additional criterion is that the fuel remains in a coolable geometry. Compliance with an average fuel pellet enthalpy limit of 280 cal/gm is used to ensure that no fuel dispersion occurs and a coolable geometry is maintained. The full power scenarios are analyzed at an assumed initial power of 3458 MWt plus an allowance of 0.6% RTP to account for power measurement uncertainties. For both the beginning of life and end of life full power cases for Unit 2 Cycle 6, the peak average fuel enthalpy is calculated to be 152 cal/gm which is less than the limit of 280 cal/gm. To ensure compliance with the assumptions on fuel failures and fuel melt, maximum allowable values of the nuclear enthalpy rise hot channel factor ($F_{\Delta H}$) and heat flux hot channel factor (F_Q) are calculated such that the respective limits of DNBR and fuel centerline temperature are just met. Reactor physics calculations are then performed to evaluate the distribution of peaking factors in the core for the spectrum of potential ejected RCCAs. A pin census is then performed to calculate the percentage of the core that exceeds the relevant peaking factors.

Dropped RCCA

The dropped rod event is analyzed to demonstrate compliance with the DNBR acceptance limit. For Unit 2 Cycle 6, this analysis was performed using the statistical combination of uncertainties (SCU) method described in Technical Specification 5.6.5, Item 14. Using this method, the system analyses are assumed to be initiated from nominal, full power conditions. The uncertainties in the initial conditions, (in this case, power, pressure, temperature, and $F_{\Delta H}$) are combined statistically and included in the DNBR limit. As such, separate analyses were required to address operation at a Rated Thermal Power of 3411 MWt with an allowance of 2.0% RTP to account for the power calorimetric uncertainty, and operation at a Rated Thermal Power of 3458 MWt with an allowance of 0.6% RTP to account for the smaller power calorimetric uncertainty. Intuitively, one would expect the latter case to be limiting, since more of the initial power is considered in a deterministic, rather than statistical, manner. Such is the case for the Unit 2 Cycle 6 analyses. The minimum DNBR was calculated to be approximately 1.48 which is well above the cycle-specific SCU DNBR limit of 1.34.

Using the approved methods listed in the Technical Specification 5.6.5 and considering operation at a Rated Thermal Power of up to 3458 MWt, compliance with all relevant event acceptance criteria was demonstrated for the Unit 2 Cycle 6 core configuration.

Inadvertent Boron Dilution Event

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

SG tube rupture event

Using the NRC-approved methods described in Technical Specification 5.6.5, Item 16, the SGTR event is analyzed to demonstrate that the calculated dose consequences satisfy the guidelines of 10CFR100. The SGTR event is first analyzed to ensure that the ruptured SG does not completely fill with fluid prior to the time the reactor operators terminate the primary-to-secondary break flow. Assuming success, the single failure scenario that results in the largest radiological dose consequences is the failure to close the atmospheric relief valve on the ruptured steam generator steam line. The source term used for the radiological dose consequence evaluation is based on operation at a power level of 104.5% of 3411 MWt. The mass releases used in the radiological dose consequence evaluation are dominated by the blowdown of the fluid in the ruptured steam generator through the failed-to-close atmospheric relief valve. The primary-to-secondary leak rate during the event is also relatively important. Because of the rapid depressurization of the ruptured SG, the time-dependent mass release is insensitive to small changes in the assumed initial power level. This insensitivity was first identified during the development of the analyses supporting the topical report described in TS 5.6.5, Item 16. While investigating the effects of the proposed 0.4% uprate for Unit 2 and 1.4% for Unit 1, additional calculations were performed at the uprated power. The calculated mass releases for the cases analyzed at the uprated power level are essentially indistinguishable from the original mass releases. Because the mass releases are unchanged and the radiological source term remains bounding, it is concluded that the results of the SGTR event are insensitive to changes related to the proposed power uprate.

Radiological Consequences Calculations

Calculations have been performed to determine potential impact on the radiological consequences from a reactor power level uprating to 3458 MWt based on cycle-specific inputs for Unit 2 Cycle 6. The calculations and principal conclusions obtained are described below. The conclusions also bound the results obtained when the Unit 1 Cycle 9 analyses were extended to include uprating to 3458 MWt.

Calculations address radiation source terms for four general categories of potential radiological consequences:

- post-accident (LOCA) doses affecting equipment qualification (EQ) of safety-related equipment,

- post-accident (LOCA) dose rates affecting vital area accessibility,

- post-accident dose rates in the control room or offsite following a postulated LOCA or a postulated fuel-handling accident, and

- offsite doses from normal-operating effluent releases.

In each case, assessment of potential impact, in comparison to the original design bases, is dependent upon potential changes in fission product inventory at reactor core end-of-life (EOL) and typical use of the inventory data in dose rate or integrated dose computations.

Fission Product Inventories

To assess the impact, additional core fission product inventory calculations were performed assuming full cycle operation at 3458 MWt, with Unit 2 EOL core average burnups of 37,053 MWD/MTU. The calculations indicate that some noble gases (i.e., Kr-85 Xe-131m), halogens (i.e., Br-82, and I-130), and other nuclides (i.e., H-3, Sr-89, Sr-90, Y-90, Y-91, Zr-95, Nb-95, Zr-97, Mo-99, AG-111, SB-125, Ba-137m, Ba-140, Ce-141, Ce-143, Pr-143, Ce-144, Nd-147, Pm-147, and Pm-148) are marginally higher than design basis values.

Post-accident Gamma-Ray Source Strengths

The calculated fission product inventories are appropriately converted to obtain energy-dependent gamma-ray source strengths at various times after shutdown for comparison with the design basis values. Using TID-14844 release fractions, consideration is given to post-accident containment air (inside), reactor coolant (pressurized), sump water (depressurized), and containment air (outside). For the uprate in Unit 2, some calculated reactor coolant source strengths (>4.0 Mev) were found to be higher than design basis values for post-accident times after 30 minutes. Some sump water source strengths (>2.2 Mev) were higher than design basis values for post-accident times after one week. At late post-accident times (i.e., six months to a year), some source strengths (<2.6 Mev) exceed design basis values in all categories. Therefore, application to a typical dose rate computation (discussed below) was necessary to assess the potential impact on radiological consequences.

Post-accident Dose Rates and Doses from Contained Sources

Using the calculated gamma-ray source strengths in typical sample applications indicates that post-accident dose rates from all categories of contained sources during the first few months are lower than design basis values. Therefore, existing postulated dose rates affecting vital area accessibility remain bounding, and increased contributions to EQ doses occur only at later times. Owing primarily to increased burn-up, and the associated larger inventories of long-lived fission products, the post-accident dose rates at later times (6 months to a year) are increased above current dose rates in all source-term categories. However, integration of dose-rates (from t = 0 to t = 1 year) for typical sample cases demonstrates that accumulated doses at all times out to one year remain lower than design basis values.

Post-accident Doses from Released Sources

Investigation of the potential impact on radiological consequences from Containment release (due to a LOCA) or Fuel Building release (due to a fuel handling accident) determined that existing design basis calculations remain bounding. Most important fission product activities remain within design basis values for the proposed power uprating. The LOCA containment source has increased quantities of Kr-85 and Xe-131m. These two nuclides make negligible contributions to postulated control room and offsite whole body doses. For the case of the fuel handling accident, only Xe-131m and Xe-135m have increased activity. However, both of these nuclides make negligible contributions to postulated control room and offsite whole body doses for this accident. In all cases, the important radio-iodine activities (I-131 through I-135) affecting thyroid inhalation doses remain within design basis values; therefore, the existing design basis remains bounding for the proposed uprating.

Doses from Normal Effluent Releases

For the proposed power uprating, calculations demonstrate that offsite doses from normal effluent releases remain significantly below referenced bounding results, which are within 10CFR50 Appendix I limits.

Conclusions

Based on the above, the following is concluded based on cycle-specific inputs for Unit 2 Cycle 6:

1. Calculated fission product activities which increase as a result of uprating to a core power of 3458 MWt do not contribute significantly to radiological consequences.

2. Post-accident gamma-ray source strengths which increase (for some energies and post-shutdown times) due to a proposed uprating do not result in increased accumulated doses.

3. Postulated post-accident vital area accessibility and EQ doses remain within existing design basis values for the proposed uprating.

4. Control Room or offsite postulated doses due to release from a LOCA or fuel handling accident are not impacted by the proposed uprating.

5. Offsite doses due to normal plant effluent releases remain below 10CFR50 Appendix I limits for the proposed uprating.

Similar conclusions are expected for the Unit 1 Cycle 10 analyses.

E. MISCELLANEOUS PROGRAMS

The effects of the proposed power uprates to a Rated Thermal Power of 3458 MWt on several analyses required by regulations are summarized below. These issues are not part of the accident analyses as described in FSAR Chapters 6 and 15, but were performed to address specific issues.

Station Blackout

The existing calculations used to demonstrate the capability to withstand a Station Blackout event of four hours duration without uncovering the core were reviewed for the uprate condition. The later stages of the existing analysis credit operator action to maintain the RCS temperature and pressure below specified limits; the steam generator atmospheric relief valves (ARVs) are used to accomplish this action. The capacity of the ARVs was evaluated and determined to be sufficient to accommodate the uprated condition; therefore, the conclusions of the calculation remain valid, i.e., the time to uncover the core following a Station Blackout event is greater than four hours.

The existing loss of ventilation analyses for the CPSES Station Blackout (SBO) is a four hour transient. The evaluation of the SBO transient is based on emergency operating procedures. Using these procedures, a basic list of the equipment necessary to achieve safe shutdown and restore AC power was developed. The SBO room temperatures identified in the equipment lists were calculated using transient heat-up computer models. The temperatures identified were the peak temperatures calculated for the four hour coping period. Equipment operability was assessed at those peak temperatures, except as noted for individual pieces of equipment.

The areas where the equipment environment was evaluated can be summarized as follows:

UPS and battery rooms
control room
electrical and switchgear rooms
cable spread rooms
diesel generator rooms
pipe tunnel
containment
pipe penetration area rooms
main steam and feedwater penetration areas
turbine driven feedwater pump room
instrument air compressor room
outdoors (turbine building, safeguards roof)

The expected increase in the RCS, main steam, feedwater, and steam generator blowdown operating temperatures associated with the power uprating does not affect the heat loads used to calculate the temperature transients for the first six items. This is because these areas are primarily electrical areas that are not exposed to these process fluids.

The containment environment during a four hour SBO event is significantly less limiting (by greater than 120°F) than the thermal profiles considered for LOCA/MSLB events. A small change in decay heat and initial process temperatures cannot result in a change of such magnitude that the calculated LOCA/MSLB environment will be exceeded. Therefore, it was concluded that a small change in RCS temperature, decay heat, main steam and feedwater temperatures would have no effect on the equipment as evaluated for the SBO event.

The concern for the pipe penetration area rooms is the potential increase in the room heat load resulting from an increase in the steam generator blowdown line temperature. The room temperatures used in the four hour SBO evaluation were obtained from the 30 day loss of HVAC analysis. The piping heat load input used in the loss of ventilation analyses assumed that the unit was also in a LOCA (since the signals obtained from the LOCA tripped the non-safety HVAC). In the 30 day loss of HVAC analyses, the piping heat loads in these rooms included RHR, CVCS, and/or containment spray fluid flowing from the containment sump at post LOCA temperatures (in excess of 200°F), as well as component cooling water (CCW) post accident temperatures. Steam generator blowdown piping is also routed through these rooms. A temperature of 550°F was used in the piping heat load and was held constant for the duration of the transient. The steam generator blowdown is isolated in the SBO scenario and only that portion of the piping up to the blowdown isolation valve would remain hot. The heat source from the remaining piping would decay throughout the transient.

Since the operation of the containment spray, RHR, CCW systems are not postulated in the SBO scenario, it can be concluded that the effects of small changes in steam generator blowdown temperatures are bounded by the significantly larger post-accident piping heat loads. Therefore, small changes in steam generator blowdown temperatures do not impact the environment and the equipment already evaluated for the SBO event.

The primary heat loads in the main steam and feedwater piping penetration areas are obviously from the main steam and feedwater piping. The power uprate results in a lower operating steam temperature and no change to the no-load steam temperature. Therefore, the heat load resulting from the main steam lines will actually decrease during power operation and remain constant at zero power.

The feedwater temperature used in the SBO loss of HVAC analyses was 441°F. The estimated change in feedwater temperature is less that 1°F for the uprate. This increase is expected to have an insignificant affect on the results of these analyses, especially when it is considered that the feedwater is isolated in the transient and the piping is insulated.

Based on the preceding discussions, it is concluded that the small changes in steam temperature and feedwater temperatures do not adversely impact the environment and the equipment already evaluated for the SBO event.

The primary heat load in the turbine-driven auxiliary feedwater pump room is from the main steam piping feeding the turbine. The power uprate results in a lower operating steam temperature and no change to the no-load steam temperature. Therefore, the heat load resulting from the main steam lines will decrease during power operation and remain constant at zero power.

The primary concern in the instrument air compressor room is the potential increase in the room heat load resulting from an increase in the steam generator blowdown line temperature. This room contains the steam generator blowdown heat exchangers and associated piping for both units. The existing calculations modeled the fluid temperature as a constant heat source of 550°F for the duration of the transient. In addition, the instrument air compressor's heat loads were included in the calculations. The affected calculations were reviewed, and it was concluded that the conservative modeling of the heat load sources bound small increases in the blowdown temperature.

The power uprate modification does not change the environment outdoors, therefore, there is no impact to the equipment evaluated for SBO.

To provide for an orderly and safe cooldown of the unit during a station blackout event, the following conditions must be met:

the turbine driven auxiliary feedwater pump must operate to provide feedwater to the SGs,

the SG atmospheric relief valves (ARVs) must cycle open to relieve steam for unit cooldown, and

an adequate supply of water from the condensate storage tank must be available to maintain adequate water level in the steam generators.

To accomplish these tasks, specific air operated valves in the main steam system and the auxiliary feedwater system must be able to be operated from air accumulators that have sufficient capacity to cycle the valves as needed during the controlled unit cooldown. In each case, the required number of valve cycles was established independent of and was determined to be reasonably insensitive to the actual power level. Accordingly, there is no change in the required operation of the AOVs for unit cooldown during a station blackout event as a result of the power uprate and the AOV accumulator sizes are therefore sufficient to provide a safe cooldown during a SBO event.

An evaluation was also performed in which it was concluded that the current minimum available safety grade condensate inventory in the condensate storage tank is sufficient for the uprate condition.

ATWS, Containment Integrity, and IPE

The proposed increase of 1.4% Rated Thermal Power for Unit 1 and 0.4% Rated Thermal Power for Unit 2 is not sufficient to materially affect the progression of any event. Discussions of the effects of the proposed uprate on specific events are provided below:

a) ATWS progression: The current CPSES analytical basis for this "beyond design basis" event is provided in FSAR Section 15.8. Incorporated by reference is a letter from T. M. Anderson of Westinghouse to S. H. Hanauer of the NRC, "ATWS Submittal," NS-TMA-2182, dated December, 1979. The applicable analyses described in that document are based on an NSSS power of 3427 MWt; however, an additional sensitivity study is provided to address a power level 2% higher. Acceptable results were obtained for this case. The proposed power increase falls within the parameter range analyzed in this document. Therefore, there is no significant effect on the ATWS progression.

b) Containment integrity analyses: The mass and energy release calculations used to evaluate the containment integrity were performed at power levels of up to 3479 MWt. For the LOCA mass and energy release calculations, a higher power level of 3565 MWt was used. A spectrum of lower initial power levels was also considered. Analyses initiated from lower power levels were found to be limiting for most containment analyses; thus, these analyses remain unaffected. Those analyses initiated at the current power, or higher power levels, included a 2% power uncertainty. As previously described, through the use of the improved LEFM instrumentation, the 0.6% power uncertainty is used to offset the increase in the operating power level. In all cases, it was determined that the mass and energy release calculations remained valid; therefore, the containment integrity analyses are unaffected by the proposed uprate.

The change in the uncertainty allowance applied to the core power can affect only the initial power used in the analysis; all other conservative assumptions remain unchanged. The mass and energy releases attributed to a secondary system break were calculated for initial power levels of up to 102% RTP, which includes a 2% power uncertainty. A spectrum of lower initial power levels was also considered. Analyses initiated from lower power levels were found to be limiting for secondary system breaks; thus, these containment analyses remain unaffected. For the LOCA mass and energy release calculations, a higher power level of 104.5% RTP (plus a 2% power uncertainty) was used; therefore, these analyses remain valid. As previously described, through the use of the improved LEFM instrumentation, the 0.6% power uncertainty is used to offset the increase in the operating power level. In all cases, it was determined that the mass and energy release calculations remained valid; therefore, the containment integrity analyses are unaffected by the proposed uprate.

In summary, the allowance provided for the power calorimetric uncertainty is but one of several conservative assumptions that are applied to the containment analyses. However, through the use of the improved LEFM instrumentation, the use of a smaller value of the power calorimetric uncertainty does not result in a reduction of analytical margin in the containment analyses.

c) The success criteria used for the CPSES IPE were reviewed and found to not be materially affected by the proposed power uprate. Therefore, it is concluded that the overall IPE results are similarly unaffected, and the proposed power uprate is not risk-significant.

F. PLANT OPERATIONS AND MAINTENANCE

Operations

A review of plant operations has concluded that an increase of this magnitude does not require any material modifications to plant procedures. Further, the responses of the reactor operators to any event will be unaffected by a change of this magnitude.

No changes to control room alarms, controls and displays are required as a direct result of the power uprate for either Unit. The SPDS is also unaffected by the proposed increase in Rated Thermal Power. When the power uprate is put in place, the Nuclear Instrumentation System will simply be adjusted to indicate the new 100% RTP in accordance with Technical Specification requirements and plant administrative controls. Because this power uprate is predicated on the availability of the LEFM(check), procedural guidance (presently in place for Unit 2), supplemented by plant computer displays, will be implemented in Unit 1 to facilitate operation when the LEFM(check) is unavailable. The reactor operators will be trained on the changes in a manner consistent with any other design modification.

Operator training and plant simulator

The CPSES simulator uses Unit 1 as the reference plant. Because the previous 1% power uprate only affected Unit 2, no simulator modifications were required. However, the power uprate for Unit 1 will be reflected in the simulator. Based on the experience gained from the 1% uprate on Unit 2, these changes will be virtually transparent to the reactor operators.

Maintenance and Calibration

The Caldon LEFM(check) system is currently installed at CPSES. The existing maintenance requirements (scope and frequency) and calibration procedures for the LEFM system incorporate the vendor's requirements for the operation of the LEFM(check) system in a manner consistent with the NRC-approved Caldon Engineering Report 80-P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM(check) System," Revision 0.

Current Operations procedures are used to perform a unit calorimetric measurement for the purpose of calibrating the Power Range NIS and N-16 channels. Contingencies and instructions are currently in the procedure in the event that the LEFM system is unavailable. In addition, more formal guidance, including routine surveillance requirements for the LEFM(check) and appropriate contingency actions, has been provided in the Technical Requirements Manual. This guidance directs the operators to operate the plant consistent with the accident analyses and the uncertainty associated with the alternate methods of determining the plant Thermal Power (i.e.,LEFM(check) or venturi-based indications of feedwater flow).

5.0 REGULATORY ANALYSIS

5.1 NO SIGNIFICANT HAZARDS CONSIDERATIONS ANALYSIS

TXU Electric has evaluated whether a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10CFR50.92(c). The following information is provided to address the three questions required for the 10 CFR 50.92 evaluation.

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

   Since the transfer of ownership from Texas Municipal Power Agency (TMPA) to TXU Electric was previously approved by the NRC, removal of TMPA from the operating license is administrative in nature and does not increase in the probability or consequences of any accident.

   The comprehensive analytical efforts performed to support the proposed change in rated thermal power included a review of the NSSS systems and components that could be affected. All systems and components will function as designed and the applicable performance requirements have been evaluated and found to be acceptable.

   The primary loop components (reactor vessel, reactor internals, CRDMs, loop piping and supports, reactor coolant pump, steam generator and pressurizer) continue to comply with their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components. The Leak Before Break analysis conclusions remain valid and thus the limiting break sizes determined in this analysis remain bounding.

   All of the NSSS systems will still perform their intended design functions during normal and accident conditions. The pressurizer spray flow remains above its design value. Thus, the control system design analyses which credit the flow do not need to be modified for changes in this flow. The auxiliary systems and components continue to meet their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components. All of the NSSS/BOP interface systems will continue to perform their intended design functions. The steam generator safety valves will provide adequate relief capacity to maintain the steam generator pressures within design limits. The atmospheric relief valves and steam dump system valves meet design sizing requirements at the uprated power level. The current LOCA hydraulic forcing functions are still bounding for the proposed increase in power.

Because the integrity of the plant will not be affected by operation at the uprated condition and it can be concluded that all structures, systems, and components required to mitigate a transient remain capable of fulfilling their intended function, the effects on the remainder of the safety analyses can be assessed. The reduction in the uncertainty allowance provided for the power calorimetric measurement allows many current safety analyses to be used, without change, to support operation at a core power of 3458 MWt. As such, all FSAR Chapter 15 accident analyses continue to demonstrate compliance with the relevant event acceptance criteria. Those analyses performed to assess the effects of mass and energy releases remain valid. The source terms used to assess radiological consequences have been reviewed and determined to bound operation at the uprated condition based on Unit 2 Cycle 6 inputs. Similar results are expected for cycle-specific Unit 1 analyses at the uprated conditions.

Based on the forgoing, it is concluded that the proposed changes do not result in an increase in the probability or consequences of any accident previously analyzed.

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

   Since the transfer of ownership from Texas Municipal Power Agency (TMPA) to TXU Electric was previously approved by the NRC, removal of TMPA from the operating license is administrative in nature and does not create the possibility of a new or different kind of accident.

   For the proposed change in rated thermal power, no new accident scenarios, failure mechanisms or single failures are introduced. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design function. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety related system. Therefore, it is concluded that 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?

   Since the transfer of ownership from Texas Municipal Power Agency (TMPA) to TXU Electric was previously approved by the NRC, removal of TMPA from the operating license is administrative in nature and does not reduce the margin of safety.

   Operation at the uprated power condition does not involve a significant reduction in a margin of safety. Extensive analyses of the primary fission product barriers have concluded that all relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier and from the standpoint of compliance with the regulated acceptance criteria. As appropriate, all evaluations have been performed using methods that have either been reviewed and approved by the NRC in Section 5.6.5b of the CPSES Technical Specifications, or that are in compliance with all regulatory review guidance and standards. Therefore, it is concluded that the proposed changes do not involve a significant reduction in a margin of safety.

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

5.2 REGULATORY SAFETY ANALYSIS

Applicable Regulatory Requirements/Criteria

Appendix K to 10CFR50, Part I,A., which reads, in part, "For the heat sources...it must be assumed that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level (to allow for instrument error).... An assumed power level lower than the level specified in this paragraph...may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error."

Because of the potentially broad impact of a power uprate, the acceptance limits for many systems, components and analyses and defined by 10CFR50 and the CPSES FSAR need be addressed.

Analysis

Based on the discussions provided in section 4.0 above and using the LEFM(check) system, the power level instrumentation error is less than 0.6%. With the proposed power uprates and the 0.6% instrument error, those safety analyses conducted at 102% of RTP continue to be acceptable.

Other applicable acceptance limits are also addressed in section 4.0, "Technical Analysis."

Conclusion

The License Amendment Request and its references provide sufficient information to conclude that 10CFR50 Appendix K, with respect to power level uncertainty, is satisfied and that the other applicable review topics, and listed in section 4.0, have been adequately addressed. In conclusion, CPSES, with the proposed power upgrades, will continue to be in compliance with NRC Regulations.

6.0 ENVIRONMENTAL EVALUATION

The Comanche Peak Steam Electric Station (CPSES) Final Environmental Statement (FES-OL), NUREG-0775, evaluates the environmental impact of operating Comanche Peak Steam Electric Station Units 1 and 2. The conclusions of the Final Environmental Statement are based on review of information contained in the CPSES Environmental Report Operating Licensee Stage. Deletion of TMPA from the operating license is administrative in nature and therefore does not impact the Final Environmental Statement. The following evaluation provides an assessment of environmental impact associated with a 1.4% power uprate of Unit 1 and a 0.4% power uprate of Unit 2 based on comparisons of the operating parameters established for the power uprate with the parameters and conclusions in the above referenced reports. Power uprate has been widely recognized by the industry as a safe and cost effective method to increase generating capacity.

Section 3.1 of the CPSES Environmental Protection Plan (EPP), Appendix B to the Unit 1 Facility Operating Licenses NPF-87 and Section 3.1 of the CPSES EPP, Appendix B to the Unit 2 Facility Operating Licenses NPF-89 state that "the licensee may make changes in station design or operation or perform tests or experiments affecting the environment provided such activities do not involve an unreviewed environmental question and do not involve a change to the EPP[1]." Section 3.1 requires that an environmental evaluation be prepared and recorded prior to engaging in any activity which may significantly affect the environment. Section 3.1 further states that, "[A] proposed change, test or experiment shall be deemed to involve an unreviewed environmental question if it concerns: (1) a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the FES-OL, environmental impact appraisals, or in any decisions of the Atomic Safety and Licensing Board; or (2) a significant change in effluents or power level; or (3) a matter not previously reviewed and evaluated in the documents specified in (1) of this Subsection, which may have a significant adverse environmental impact."

In accordance with the requirements discussed above, an evaluation assessing the environmental impact of the proposed NSSS power level uprate from 3411 MWt to 3458 MWt has been performed. This evaluation determines that the proposed change in power level is not significant relative to the potential environmental impact. The following environmental evaluation specifically considers thermal effects, radiological effluents and radwaste.

Squaw Creek Reservoir functions as the heat sink for heat rejected by the turbine plant (via the Circulating Water System) primarily through the main condensers and the auxiliary condensers (which serve the feedwater pump turbine drivers). The Circulating Water System also supplies cooling water to the turbine plant cooling water (TPCW) heat exchanger, condenser exhausting vacuum (CEV) pump heat exchangers and the non-safety ventilation chiller condensers.

The temperature limits on the circulating water discharge, as imposed by our Texas Natural Resource Conservation Commission (TNRCC) wastewater discharge permit (permit # 01854) and National Pollutant Discharge Elimination System (NPDES permit #TX0065854), are: "Daily Average Temperature" not to exceed 113°F; and

[1]This provision does not relieve the licensee of the requirements of 10CFR50.59.

"Daily Maximum Temperature" not to exceed 116°F. Circulating water discharge temperature readings are averaged over two-hour periods and the twelve daily two-hour average readings are then averaged to obtain the daily value. For daily temperature compliance monitoring, this daily value is compared to the Daily Maximum Temperature limit. These daily values are then accumulated for each calendar month, averaged, and compared to the Daily Average Temperature limit. Both the average of the daily values and the single highest daily value for each calendar month are then reported to the TNRCC in a routine environmental report.

A review of historical data revealed that the highest daily average and daily maximum discharge temperatures to-date are 111 °F and 113 °F, respectively (2 °F below the daily average limit and 3 °F below the daily maximum limit). This condition occurred in August 1997, while both units were at 100% load. The second highest values are 108 °F (July 1996) and 112 °F (September 1995) for daily average and daily maximum, respectively. This historical data indicates adequate margin between actual operational values and permit limits exists.

The impact of the 1.4% uprating for Unit 1 and a 0.4% uprating for Unit 2 on the Circulating Water System will not result in significant temperature changes. The circulating water for both CPSES Units 1 and 2 flows at a combined rate of 2,200,000 gpm with a maximum temperature increase of approximately 15 °F from the inlet to the outlet of the main condenser as identified in the CPSES Environmental Report. Operation of Unit 1 and Unit 2 at 3458 MWt will result in an overall maximum temperature increase of less than 0.25°F, only a fraction of a degree, when comparing the heat duty on the condenser for the uprated condition and the current 100% design. Existing administrative controls ensure the conduct of adequate monitoring such that appropriate actions can be taken to preclude exceeding NPDES permitted limits. No additional monitoring requirements or other changes relative to the NPDES permit are required as a result of the power uprate.

Therefore, as described in the preceding discussions, the proposed uprate for Unit 1 and Unit 2 does not have a significant environmental impact on Squaw Creek Reservoir.

The Component Cooling Water (CCW) System removes heat from various safety and non-safety related equipment and transfers it in a closed loop to the Station Service Water (SSW) System, from which it is transferred to an ultimate heat sink. The closed loop provides an intermediate barrier to contain radioactive or potentially radioactive sources, thus precluding direct leakage of radioactive fluids into the ultimate heat sink. The Safe Shutdown Impoundment (SSI) serves as the ultimate heat sink to safely operate, shut down, and cool down the unit. Since CPSES is a multiple unit station, the SSI is required to safely dissipate the heat from an accident in one unit, and to permit the concurrent safe shutdown and cooldown of the second unit.

The performance of the SSW System is measured by its ability to remove heat from each SSW-cooled component and transfer that heat to the SSI. The ability of the SSW to remove heat from a component is a function of the SSI (supply) temperature and the SSW flow rate through the component. In order to comply with safety analysis and equipment limits, the SSW must supply water from the SSI at no more than 102°F and no less than 40°F during normal operation. The maximum SSI temperature during post-DBA events must remain at the currently specified limit of 117°F which occurs seven days after the accident.

Conservatively assuming both units have been uprated and operating at 3458 MWt, the current SSI maximum temperature limit of 102°F is maintained. In addition, the current SSI maximum post-LOCA temperature limit of 117°F continues to be met assuming both units have been uprated to 3458 MWt.

Therefore, as described in the preceding discussions, the proposed uprate for Unit 1 and Unit 2 does not have a significant environmental impact on the SSI.

No significant change in groundwater withdrawal required to supply the sanitary water system or fire protection system will result from power uprate.

Calculations have been performed to determine the potential impact on the radiological effluents from a 1.4% uprate for Unit 1 and a 0.4% uprate for Unit 2 (reactor power level uprating to 3458 MWt). The uprating calculations demonstrate that offsite doses from normal effluent releases remain significantly below bounding limits of 10CFR50 Appendix I. The Gaseous Waste Processing System continues to meet its design basis under the uprated conditions, in that the gas storage tanks have sufficient capacity to store, for decay, the gases produced due to normal operation, including anticipated operational transients. The normal annual average gaseous release remains limited to a small fraction of 10CFR20 limits for identified mixtures.

The solid waste management and liquid waste processing systems are designed to control, collect, process, store and dispose of radioactive wastes due to normal operation including anticipated operational transients. Operation of these systems are primarily influenced by the volume of waste processed. Because these systems are typically operated in a batch mode, the only potential effect is a very slight increase in the frequency at which the batches may be processed. Thus, the amount of the solid waste and liquid waste processed are not expected to significantly change as a result of the uprate.

Design Basis Accident doses for the Exclusion Area Boundary, Low Population Zone and Control Room were computed for CPSES assuming a power level of 3565 MWt (104.5% of the original 100% design). Although the Unit 1 and Unit 2 uprate will result in a small increase in the potential doses, the CPSES analyzed accidents remain bounded by the existing postulated doses which are within applicable General Design Criteria (GDC) 19 and 10CFR100 limits.

In summary, the operating parameters associated with the power uprate for Unit 1 and Unit 2 were evaluated for the potential to affect the radiological effluents and doses. These parameters either retain the same values as the original values evaluated in the Final Environmental Statement or are bounded by those values.

Based on the above evaluation, the plant operating parameters impacted by the proposed Unit 1 and Unit 2 power uprate do not result in any significant adverse environmental impact. The Final Environmental Statement concluded that no significant environmental impact would result from operation of Comanche Peak Steam Electric Station. This conclusion remains valid for the proposed power uprate. In accordance with the above evaluation, it can be concluded that no significant environmental impact will result from the proposed NSSS power level increase to 3458 MWt.

TXU 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. TXU 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.

7.0 REFERENCES

1. WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant," January, 1983.

2. Caldon Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM(check)™ System," Revision 0, March 1997.

3. Caldon Engineering Report -160P, "Supplement to Topical Report ER-80P:Basis for a Power Uprate With the LEFM(check)™ System," Revision 0, May 2000.

4. NRR letter to Mr. J. A. Scalice, Chief Nuclear Officer and Executive Vice President, Tennessee Valley Authority dated January 19, 2001, regarding increase in reactor power of 1.4% for Watts Bar.

5. CPSES License Amendment 72 to NPF-87 and NPF-89 (CPSES Units 1 and 2 respectively), dated September 30, 1999.

6. ASME PTC 19.1-1985, Measurement Uncertainty

7. NRC Inspection Procedure 61706 (7/14/86)

8. Generic Letter 95-05

9. TID-14 844

10. NRR letter to Mr. C. L. Terry from David H. Jaffe dated September 7, 1999, regarding corporate name change from Texas Utilities Electric Company to TXU Electric Company.

11. TXU Electric Letter logged TXX-00040, from C. L. Terry to the NRC dated February 15, 2000, regarding CPSES response to NRC Bulletin 88-02

8.0 PRECEDENCE

The NRC staff has previously reviewed similar documents supporting requests for changes to the Technical Specifications at other plants for an increase in power of 1.4% based on use of the LEFM (check)™ system. The bulk of the technical and regulatory issues for the present request are identical to those reviewed in previous Safety Evaluation concerning the use of Caldon LEFM and may be found in the SE for Watts Bar Nuclear Power Stations, Docket No. 50-390.

ATTACHMENT 3 to TXX-01042

AFFECTED PAGES OF NPF-87 (UNIT 1 OPERATING LICENSE)

NPF-87: Pages 1 through 6

TXU ELECTRIC COMPANY~~, ET. AL.*~~
DOCKET NO. 50-445
COMANCHE PEAK STEAM ELECTRIC STATION, UNIT NO. 1
FACILITY OPERATING LICENSE

License No. NPF-87

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

A. The application for a license filed by TXU Electric Company (TXU Electric) ~~acting for itself and as agent for Texas Municipal Power Agency~~ (~~licensees~~ *licensee*), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

B. Construction of the Comanche Peak Steam Electric Station, Unit No. 1 (the facility), has been substantially completed in conformity with Construction Permit No. CPPR-126 and the application, as amended, the provisions of the Act, and the regulations of the Commission;

C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission (except as exempted from compliance in Section 2.D below);

D. There is reasonable assurance: (i) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I, except as exempted from compliance in Section 2.D below;

E. TXU Electric is technically qualified to engage in the activities authorized by this operating license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

F. The ~~licensees~~ *licensee* ~~have~~ *has* satisfied the applicable provisions of 10 CFR 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

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

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

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Facility Operating License No. NPF-87 subject to the conditions for protection of the environment set forth herein, is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession, and use of source, byproduct, and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40, and 70, except that an exemption to the provisions of 70.24 is granted as described in paragraph 2.D below.

2. Based on the foregoing findings regarding this facility, Facility Operating License No. NPF-87 is hereby issued to the ~~licensees~~ *licensee* to read as follows:

A. This license applies to the Comanche Peak Steam Electric Station, Unit No. 1, a pressurized-water nuclear reactor and associated equipment (the facility), owned by the ~~licensees~~ *licensee* The facility is located on Squaw Creek Reservoir in Somervell County, Texas about 5 miles north-northwest of Glen Rose, Texas, and about 40 miles southwest of Fort Worth in north-central Texas and is described in the ~~licensees'~~ *licensee's* Final Safety Analysis Report, as supplemented and amended, and the ~~licensees'~~ *licensee's* Environmental Report, as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing and Production and Utilization Facilities," TXU Electric to possess, use, and operate the facility at the designated location in Somervell County, Texas in accordance with the procedures and limitations set forth in this license;

(2) *NOT USED* ~~Pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing and Production and Utilization Facilities," Texas Municipal Power Agency to possess the facility at the designated location in Somervell County, Texas in accordance with the procedures and limitations set forth in this license;~~

(3) TXU Electric, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time, special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, and described in the Final Safety Analysis Report, as supplemented and amended;

(4) TXU Electric, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use, at any time, any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) TXU Electric, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required, any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) TXU Electric, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

TXU Electric is authorized to operate the facility at reactor core power levels not in excess of ~~3411~~ *3458* megawatts thermal in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. TXU Electric shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Applicants as defined in Appendix C shall comply with the antitrust conditions delineated in Appendix C to this license; Appendix C is hereby incorporated into this license.

D. The following exemptions are authorized by law and will not endanger life or property or the common defense and security. Certain special circumstances are present and these exemptions are otherwise in the public interest. Therefore, these exemptions are hereby granted pursuant to 10 CFR 50.12.

(1) The facility requires a technical exemption from the requirements of 10 CFR 50, Appendix J, Section III.D.2(b)(ii). The justification for this exemption is contained in Section 6.2.5 of Supplement 22 to the Safety Evaluation Report dated January 1990. The staff's environmental assessment was published on November 14, 1989 (54 FR 47430). Therefore, pursuant to 10 CFR 50.12(a)(1), and 10 CFR 50.12(a)(2)(ii) and (iii), the Comanche Peak Steam Electric Station, Unit 1 is hereby granted an exemption from the cited requirement and instead, is required to perform the overall air lock leak test at pressure P_a prior to establishing containment integrity if air lock maintenance has been performed that could affect the air lock sealing capability.

(2) The facility was previously granted an exemption from the criticality monitoring requirements of 10 CFR 70.24 (see Materials License No. SNM-1912 dated December 1, 1988 and Section 9.1.1 of Supplement 22 to the Safety Evaluation Report dated January 1990). The staff's environmental assessment was published on November 14, 1989 (54 FR 47432). The Comanche Peak Steam Electric Station, Unit 1 is hereby exempted from the criticality monitoring provisions of 10 CFR 70.24 as applied to fuel assemblies held under this license.

(3) The facility requires a temporary exemption from the schedular requirements of 10 CFR 50.33(k) and 10 CFR 50.75. The justification for this exemption is contained in Section 20.6 of Supplement 22 to the Safety Evaluation Report dated January 1990. The staff's environmental assessment was published on November 14, 1989 (54 FR 47431). Therefore, pursuant to 10 CFR 50.12(a)(1), 50.12(a)(2)(iii) and 50.12(a)(2)(v), the Comanche Peak Steam Electric Station, Unit 1 is hereby granted a temporary exemption from the schedular requirements of 10 CFR 50.33(k) and 10 CFR 50.75 and is required to submit a decommissioning funding report for Comanche Peak Steam Electric Station, Unit 1 on or before July 26, 1990.

E. With the exception of 2.C(2) and 2.C(3), TXU Electric shall report any violations of the requirements contained in Section 2.C of this license within 24 hours. Initial notification shall be made in accordance with the provisions of 10 CFR 50.72 with written follow-up in accordance with the procedures described in 10 CFR 50.73(b), (c), and (e).

F. In order to ensure that TXU Electric will exercise the authority as the surface landowner in a timely manner and that the requirements of 10 CFR Part 100.3 (a) are satisfied, this license is subject to the additional conditions specified below: (Section 2.1.1, SER)

(1) For that portion of the exclusion area which is within 2250 ft of any seismic Category I building or within 2800 ft of either reactor containment building, TXU Electric must prohibit the exploration and/or exercise of subsurface mineral rights, and if the subsurface mineral rights owners attempt to exercise their rights within this area, TXU Electric must immediately institute immediately effective condemnation proceedings to obtain the mineral rights in this area.

(2) For the unowned subsurface mineral rights within the exclusion area not covered in item (1), TXU Electric will prohibit the exploration and/or exercise of mineral rights until and unless the licensee and the owners of the mineral rights enter into an agreement which gives TXU Electric absolute authority to determine all activities -- including times of arrival and locations of personnel and the authority to remove personnel and equipment -- in event of emergency. If the mineral rights owners attempt to exercise their rights within this area without first entering into such an agreement, TXU Electric must institute immediately effective condemnation proceedings to obtain the mineral rights in this area.

(3) TXU Electric shall promptly notify the NRC of any attempts by subsurface mineral rights owners to exercise mineral rights, including any legal proceeding initiated by mineral rights owners against TXU Electric.

G. TXU Electric shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report through Amendment 78 and as approved in the SER (NUREG-0797) and its supplements through SSER 24, subject to the following provision:

TXU Electric may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

H. TXU Electric shall fully implement and maintain in effect all provisions of the physical security, guard training and qualification, and safeguards contingency plans, previously approved by the Commission, and all amendments made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain safeguards information protected under 10 CFR 73.21, are entitled: “Comanche Peak Steam Electric Station Physical Security Plan” with revisions submitted through November 28, 1988; “Comanche Peak Steam Electric Station Security Training and Qualification Plan” with revisions submitted through November 28, 1988; and “Comanche Peak Steam Electric Station Safeguards Contingency Plan” with revisions submitted through January 9, 1989.

I. The ~~licensees~~ *licensee* shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

J. *NOT USED* ~~Amendment No. 9 to Construction Permit CPPR-126, issued August 25, 1988, authorized the transfer of 6.2% ownership interest in the facility from Texas Municipal Power Agency to TXU Electric, such transfer to take place in 10 installments as set forth in the Agreement attached to the application for amendment dated March 4, 1988. At the completion of such transfer of interest, Texas Municipal Power Agency shall no longer be a licensee under this license and all references to "licensees" shall exclude Texas Municipal Power Agency.~~

K. This license is effective as of the date of issuance and shall expire at Midnight on February 8, 2030.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Attachments/Appendices:
1. Appendix A - Technical Specifications (NUREG-1399)
2. Appendix B - Environmental Protection Plan
3. Appendix C - Antitrust Conditions

Date of Issuance: April 17, 1990

ATTACHMENT 4 to TXX-01042

AFFECTED PAGES OF NPF-89 (UNIT 2 OPERATING LICENSE)

NPF-89: Pags 1 through 6

TXU ELECTRIC COMPANY~~, ET. AL.*~~
DOCKET NO. 50-446
COMANCHE PEAK STEAM ELECTRIC STATION, UNIT NO. 2
FACILITY OPERATING LICENSE

License No. NPF-89

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

A. The application for a license filed by TXU Electric Company (TXU Electric) ~~acting for itself and as agent for Texas Municipal Power Agency~~ (~~licensees~~ *licensee*), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

B. Construction of the Comanche Peak Steam Electric Station, Unit No. 2 (the facility), has been substantially completed in conformity with Construction Permit No. CPPR-127 and the application, as amended, the provisions of the Act, and the regulations of the Commission;

C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission (except as exempted from compliance in Section 2.D below);

D. There is reasonable assurance: (i) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I, except as exempted from compliance in Section 2.D. below;

E. TXU Electric is technically qualified to engage in the activities authorized by this operating license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

~~*The current owners of the Comanche Peak Steam Electric Station are: TXU Electric Company and Texas Municipal Power Agency. Transfer of ownership from Texas Municipal Power Agency to TXU Electric Company was previously authorized by Amendment No. 8 to Construction Permit CPPR-127 on August 25, 1988 to take place in 10 installments as set forth in the Agreement attached to the application for Amendment dated March 4, 1988. At the completion thereof, Texas Municipal Power Agency will no longer retain any ownership interest.~~

F. The ~~licensees~~ *licensee has* ~~have~~ satisfied the applicable provisions of 10 CFR 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

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

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Facility Operating License No. NPF-89 subject to the conditions for protection of the environment set forth herein, is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession, and use of source, byproduct, and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40, and 70, except that an exemption to the provisions of 70.24 is granted as described in paragraph 2.D below.

2. Pursuant to approval by the Nuclear Regulatory Commission at a meeting on April 6, 1993, the License for Fuel Loading and Low Power Testing, License No. NPF-88, issued on February 2, 1993, is superseded by Facility Operating License No. NPF-89 hereby issued to the ~~licensees~~ *licensee*, to read as follows:

A. This license applies to the Comanche Peak Steam Electric Station, Unit No. 2, a pressurized-water nuclear reactor and associated equipment (the facility), owned by the ~~licensees~~ *licensee*. The facility is located on Squaw Creek Reservoir in Somervell County, Texas about 5 miles north-northwest of Glen Rose, Texas, and about 40 miles southwest of Fort Worth in north-central Texas and is described in the licensee's Final Safety Analysis Report, as supplemented and amended, and the licensee's Environmental Report, as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," TXU Electric to possess, use, and operate the facility at the designated location in Somervell County, Texas in accordance with the procedures and limitations set forth in this license;

(2) *NOT USED* ~~Pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Texas Municipal Power Agency to possess the facility at the designated location in Somervell County, Texas in accordance with the procedures and limitations set forth in this license;~~

(3) TXU Electric, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time, special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, and described in the Final Safety Analysis Report, as supplemented and amended;

(4) TXU Electric, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use, at any time, any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) TXU Electric, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required, any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) TXU Electric, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

TXU Electric is authorized to operate the facility at reactor core power levels not in excess of ~~3445~~ *3458* megawatts thermal in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 72, are hereby incorporated into this license. TXU Electric shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Applicants as defined in Appendix C shall comply with the antitrust conditions delineated in Appendix C to this license; Appendix C is hereby incorporated into this license.

D. The following exemptions are authorized by law and will not endanger life or property or the common defense and security. Certain special circumstances are present and these exemptions are otherwise in the public interest. Therefore, these exemptions are hereby granted:

(1) The facility requires a technical exemption from the requirements of 10 CFR 50, Appendix J, Section III.D.2(b)(ii). The justification for this exemption is contained in Section 6.2.5.1 of Supplement 26 to the Safety Evaluation Report dated February 1993. The staff's environmental assessment was published on January 19, 1993 (58 FR 5036). Therefore, pursuant to 10 CFR 50.12(a)(1), 10 CFR 50.12(a)(2)(ii) and (iii), the Comanche Peak Steam Electric Station, Unit 2 is hereby granted an exemption from the cited requirement and instead, is required to perform the overall air lock leak test at pressure P_a prior to establishing containment integrity if air lock maintenance has been performed that could affect the air lock sealing capability.

(2) The facility was previously granted exemption from the criticality monitoring requirements of 10 CFR 70.24 (see Materials License No. SNM-1986 dated April 24, 1989 and Section 9.1.1 of SSER 26 dated February 1993.) The staff's environmental assessment was published on January 19, 1993 (58 FR 5035). The Comanche Peak Steam Electric Station, Unit 2 is hereby exempted from the criticality monitoring provisions of 10 CFR 70.24 as applied to fuel assemblies held under this license.

E. With the exception of 2.C(2) and 2.C(3), TXU Electric shall report any violations of the requirements contained in Section 2.C of this license within 24 hours. Initial notification shall be made in accordance with the provisions of 10 CFR 50.72 with written followup in accordance with the procedures described in 10 CFR 50.73(b), (c), and (e).

F. In order to ensure that TXU Electric will exercise the authority as the surface landowner in a timely manner and that the requirements of 10 CFR 100.3 (a) are satisfied, this license is subject to the additional conditions specified below: (Section 2.1, SER)

(1) For that portion of the exclusion area which is within 2250 ft of any seismic Category I building or within 2800 ft of either reactor containment building, TXU Electric must prohibit the exploration and/or exercise of subsurface mineral rights, and if the subsurface mineral rights owners attempt to exercise their rights within this area, TXU Electric must immediately institute immediately effective condemnation proceedings to obtain the mineral rights in this area.

(2) For the unowned subsurface mineral rights within the exclusion area not covered in item (1), TXU Electric will prohibit the exploration and/or exercise of mineral rights until and unless the licensee and the owners of the mineral rights enter into an agreement which gives TXU Electric absolute authority to determine all activities -- including times of arrival and locations of personnel and the authority to remove personnel and equipment -- in event of emergency. If the mineral rights owners attempt to exercise their rights within this area without first entering into such an agreement, TXU Electric must immediately institute immediately effective condemnation proceedings to obtain the mineral rights in this area.

(3) TXU Electric shall promptly notify the NRC of any attempts by subsurface mineral rights owners to exercise mineral rights, including any legal proceeding initiated by mineral rights owners against TXU Electric.

G. TXU Electric shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report through Amendment 87 and as approved in the SER (NUREG-0797) and its supplements through SSER 27, subject to the following provision:

TXU Electric may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

H. TXU Electric shall fully implement and maintain in effect all provisions of the physical security, guard training and qualification, and safeguards contingency plans, previously approved by the Commission, and all amendments made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain safeguards information protected under 10 CFR 73.21, are entitled: "Comanche Peak Steam Electric Station Physical Security Plan" with revisions submitted through January 14, 1993; "Comanche Peak Steam Electric Station Security Training and Qualification Plan" with revisions submitted through June 10, 1991; and "Comanche Peak Steam Electric Station Safeguards Contingency Plan" with revisions submitted through December 1988.

I. The ~~licensees~~ *licensee* shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

J. *NOT USED* ~~Amendment No. 8 to Construction Permit CPPR-127, issued August 25, 1988, authorized the transfer of 6.2% ownership interest in the facility from Texas Municipal Power Agency to TXU Electric, such transfer to take place in 10 installments as set forth in the Agreement attached to the application for amendment dated March 4, 1988. At the completion of such transfer of interest, Texas Municipal Power Agency shall no longer be a licensee under this license and all references to "licensees" shall exclude Texas Municipal Power Agency.~~

K. This license is effective as of the date of issuance and shall expire at Midnight on February 2, 2033.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Attachments/Appendices:
1. Appendix A - Technical Specifications (NUREG-1468)
2. Appendix B - Environmental Protection Plan
3. Appendix C - Antitrust Conditions

Date of Issuance: April 6, 1993

ATTACHMENT 5 to TXX-01042

MARKUP OF AFFECTED PAGES

OF THE TECHNICAL SPECIFICATIONS

Page 1.1-6

Page 3.3-15 and 16

Page 5.0-32, 33, and 34

Page B 3.3-64 (for information only)

1.1 Definitions (continued)

QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of *3458 Mwt.* ~~3411 Mwt for Unit 1 and 3445 Mwt for Unit 2.~~
REACTOR TRIP SYSTEM (RTS) RESPONSE	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint TIME at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the hot zero power temperatures.

(continued)

Table 3.3.1-1 (page 1 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE(a)
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.14	NA
	3(b), 4(b), 5(b)	2	C	SR 3.3.1.14	NA
2. Power Range Neutron Flux					
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 110.8% RTP ~~≤ 111.7% RTP (Unit 1)~~ ~~≤ 111.1% RTP (Unit 2)~~
b. Low	1(c), 2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 27.7% RTP
3. Power Range Neutron Flux Rate High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11	≤ 6.3 % RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1(c), 2(d)	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 31.5% RTP
5. Source Range Neutron Flux	2(e)	2	I,J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 1.4 E5 cps
	3(b), 4(b), 5(b)	2	J,K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≤ 1.4 E5 cps

(continued)

(a) The Allowable Value defines the limiting safety system setting. See the Bases for the Trip Setpoints.
(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(c) Below the P-10 (Power Range Neutron Flux) interlock.
(d) Above the P-6 (Intermediate Range Neutron Flux) interlock.
(e) Below the P-6 (Intermediate Range Neutron Flux) interlock.

Table 3.3.1-1 (page 2 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE(a)
6. Overtemperature N-16	12	4	E	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	Refer to Note 1
7. Overpower N-16	12	4	E	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	*≤ 112.9% RTP* ~~≤ 114.5% RTP (Unit 1)~~ ~~≤ 113.4% RTP (Unit 2)~~
8. Pressurizer Pressure					
a. Low	1(g)	4	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤ 1863.6 psig (Unit 1) ≤ 1865.2 psig (Unit 2)
b. High	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤ 2400.8 psig (Unit 1) ≤ 2401.4 psig (Unit 2)
9. Pressurizer Water Level - High	1(g)	3	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 93.9% of instrument span

(continued

(a) The Allowable Value defines the limiting safety system setting. See the Bases for the Trip Setpoints.
(b) Below the P-6 (Intermediate Range Neutron Flux) interlock.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1) Moderator temperature coefficient limits for Specification 3.1.3,

2) Shutdown Rod Insertion Limit for Specification 3.1.5,

3) Control Rod Insertion Limits for Specification 3.1.6,

4) AXIAL FLUX DIFFERENCE Limits and target band for Specification 3.2.3,

5) Heat Flux Hot Channel Factor, K(Z), W(Z), F_Q^{RTP}, and the $F_Q^C(Z)$ allowances for Specification 3.2.1,

6) Nuclear Enthalpy Rise Hot Channel Factor Limit and the Power Factor Multiplier for Specification 3.2.2.

7) SHUTDOWN MARGIN values in Specifications 3.1.1, 3.1.4, 3.1.5, 3.1.6 and 3.1.8.

8) Refueling Boron Concentration limits in Specification 3.9.1.

9) Overtemperature N-16 Trip Setpoint in Specification 3.3.1.

10) Reactor Coolant System pressure, temperature, and flow in Specification 3.4.1.

11) Reactor Core Safety Limit figures (Safety Limit 2.1.1)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. When an initial assumed power level of 102 percent of rated power is specified in a previously approved method, ~~101~~ *100.6* percent of rated power may be used only when feedwater flow measurement (used as input for reactor thermal power measurement) is provided by the leading edge flowmeter (LEFM√) as described in document number 20 listed below. When feedwater flow measurements from the LEFM√ are not available, the originally approved initial power level of 102 percent of rated thermal power shall be used.

Future revisions of approved analytical methods listed in this technical specification that currently assume 102 percent of rated power shall include

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (continued)

the condition given above allowing use of ~~101~~ *100.6* percent of rated power in safety analysis methodology when the LEFM√ is used for feedwater flow measurement.

The approved analytical methods are described in the following documents:

1) WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).

2) WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT," September 1974 (W Proprietary).

3) T. M. Anderson To K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980--Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.

4) NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.

5) WCAP-10216-P-A, Revision 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary).

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-A, "Power Distribution Control Analysis and Overtemperature N-16 and Overpower N-16 Trip Setpoint Methodology, " June 1994.

(continued)

5.6.5 CORE OPERATING LIMITS REPORT (continued)

10) RXE-88-102-P-A, "TUE-1 Departure from Nucleate Boiling Correlation", July 1992.

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

12) RXE-89-002-A, "VIPRE-01 Core Thermal-Hydraulic Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications", September 1993.

13) RXE-91-001-A, "Transient Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications", October 1993.

14) RXE-91-002-A, "Reactivity Anomaly Events Methodology", October 1993.

15) RXE-90-007-A, "Large Break Loss of Coolant Accident Analysis Methodology", April 1993.

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

17) RXE-91-005-A, "Methodology for Reactor Core Response to Steamline Break Events," February 1994.

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

19) RXE-95-001-P-A, "Small Break Loss of Coolant Accident Analysis Methodology," September 1996.

20) Caldon, Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√ System," Revision 0, March 1997 *and Caldon Engineering Report - 160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM√™ System," Revision 0, May 2000.*

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

Table B 3.3.1-1
Reactor Trip System Setpoints

	Function	Nominal Trip Setpoint
1	Manual Reactor Trip	N/A
2.a	Power Range Neutron Flux, High	≤ 109% RTP
2.b	Power Range Neutron Flux, Low	≤ 25% RTP
3	Power Range Neutron Flux Rate, High Positive Rate	≤ 5% RTP with a time constant ≥ 2 seconds
4	Intermediate Range Neutron Flux	≤ 25% RTP
5	Source Range Neutron Flux	≤ 10^5 cps
6	Overtemperature N-16	See Note 1, Table 3.3.1-1
7	Overpower N-16	*≤ 110% RTP* ~~≤ 112 RTP (Unit 1)~~ ~~≤ 110 RTP (Unit 2)~~
8.a	Pressurizer Pressure, Low	≥ 1880 psig
8.b	Pressurizer Pressure, High	≤ 2385 psig
9	Pressurizer Water Level - High	≤ 92% span
10	Reactor Coolant Flow - Low	≥ 90% of nominal flow
11	Not Used.	
12	Undervoltage RCPs	≥ 4830 volts
13	Underfrequency RCPs	≥ 57.2 Hz

continued

ATTACHMENT 6 to TXX-01042
RETYPED AFFECTED PAGES
OF THE TECHNICAL SPECIFICATIONS

Page 1.1-6 (Unit 2 Implementation)

Page 3.3-15 and 16 (Unit 2 Implementation)

Page 5.0-32, 33, and 34 (Unit 2 Implementation)

Page 5.0-32, 33, and 34 (Unit 2 Implementation)

Page 1.1-6, 3.3-15 and 3.3-16 (Units 1 and 2 Implementation)

Page B 3.3-64 (for information only)

Attachment 6 pages 1 through 6 to be used for Unit 2 implementation of License Amendment. **Attachment 6 pages 7, 8 and 9 to be used for Unit 1 implementation of License Amendment. (Post implementation of Unit 2)**

1.1 Definitions (continued)

QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER	RTP shall be a total reactor core heat transfer rate to the (RTP) reactor coolant of 3411 Mwt for Unit 1 and 3458 Mwt for Unit 2.
REACTOR TRIP	The RTS RESPONSE TIME shall be that time interval from
SYSTEM (RTS) RESPONSE	when the monitored parameter exceeds its RTS trip setpoint TIMEat the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the hot zero power temperatures.

(continued)

Table 3.3.1-1 (page 1 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE(a)
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.14	NA
	3(b), 4(b), 5(b)	2	C	SR 3.3.1.14	NA
2. Power Range Neutron Flux					
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 111.7% RTP (Unit 1) ≤ 110.8% RTP (Unit 2)
b. Low	1(c), 2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 27.7% RTP
3. Power Range Neutron Flux Rate High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11	≤ 6.3 % RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1(c), 2(d)	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 31.5% RTP
5. Source Range Neutron Flux	2(e)	2	I,J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 1.4 E5 cps
	3(b), 4(b), 5(b)	2	J,K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≤ 1.4 E5 cps

(continued)

(a) The Allowable Value defines the limiting safety system setting. See the Bases for the Trip Setpoints.
(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(c) Below the P-10 (Power Range Neutron Flux) interlock.
(d) Above the P-6 (Intermediate Range Neutron Flux) interlock.
(e) Below the P-6 (Intermediate Range Neutron Flux) interlock.

Table 3.3.1-1 (page 2 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[a]
6. Overtemperature N-16	12	4	E	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	Refer to Note 1
7. Overpower N-16	12	4	E	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤114.5% RTP (Unit 1) ≤ 112.9% RTP (Unit 2)
8. Pressurizer Pressure					
a. Low	1(g)	4	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤ 1863.6 psig (Unit 1) ≤ 1865.2 psig (Unit 2)
b. High	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤ 2400.8 psig (Unit 1) ≤ 2401.4 psig (Unit 2)
9. Pressurizer Water Level - High	1(g)	3	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 93.9% of instrument span

(continued

(a) The Allowable Value defines the limiting safety system setting. See the Bases for the Trip Setpoints.
(b) Below the P-6 (Intermediate Range Neutron Flux) interlock.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1) Moderator temperature coefficient limits for Specification 3.1.3,

2) Shutdown Rod Insertion Limit for Specification 3.1.5,

3) Control Rod Insertion Limits for Specification 3.1.6,

4) AXIAL FLUX DIFFERENCE Limits and target band for Specification 3.2.3,

5) Heat Flux Hot Channel Factor, K(Z), W(Z), F_Q^{RTP}, and the $F_Q^C(Z)$ allowances for Specification 3.2.1,

6) Nuclear Enthalpy Rise Hot Channel Factor Limit and the Power Factor Multiplier for Specification 3.2.2.

7) SHUTDOWN MARGIN values in Specifications 3.1.1, 3.1.4, 3.1.5, 3.1.6 and 3.1.8.

8) Refueling Boron Concentration limits in Specification 3.9.1.

9) Overtemperature N-16 Trip Setpoint in Specification 3.3.1.

10) Reactor Coolant System pressure, temperature, and flow in Specification 3.4.1.

11) Reactor Core Safety Limit figures (Safety Limit 2.1.1)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. When an initial assumed power level of 102 percent of rated power is specified in a previously approved method, 100.6 percent of rated power may be used only when feedwater flow measurement (used as input for reactor thermal power measurement) is provided by the leading edge flowmeter (LEFM√) as described in document number 20 listed below. When feedwater flow measurements from the LEFM√ are not available, the originally approved initial power level of 102 percent of rated thermal power shall be used.

Future revisions of approved analytical methods listed in this technical specification that currently assume 102 percent of rated power shall include

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (continued)

the condition given above allowing use of 100.6 percent of rated power in safety analysis methodology when the LEFM√ is used for feedwater flow measurement.

The approved analytical methods are described in the following documents:

1) WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).

2) WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT," September 1974 (W Proprietary).

3) T. M. Anderson To K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980--Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.

4) NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.

5) WCAP-10216-P-A, Revision 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary).

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-A, "Power Distribution Control Analysis and Overtemperature N-16 and Overpower N-16 Trip Setpoint Methodology, " June 1994.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (continued)

10) RXE-88-102-P-A, "TUE-1 Departure from Nucleate Boiling Correlation", July 1992.

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

12) RXE-89-002-A, "VIPRE-01 Core Thermal-Hydraulic Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications", September 1993.

13) RXE-91-001-A, "Transient Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications", October 1993.

14) RXE-91-002-A, "Reactivity Anomaly Events Methodology", October 1993.

15) RXE-90-007-A, "Large Break Loss of Coolant Accident Analysis Methodology", April 1993.

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

17) RXE-91-005-A, "Methodology for Reactor Core Response to Steamline Break Events," February 1994.

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

19) RXE-95-001-P-A, "Small Break Loss of Coolant Accident Analysis Methodology," September 1996.

20) Caldon, Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√ System," Revision 0, March 1997 and Caldon Engineering Report - 160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM√™ System," Revision 0, May 2000.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

1.1 Definitions (continued)

QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3458 Mwt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the hot zero power temperatures.

(continued)

Table 3.3.1-1 (page 2 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE(a)
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.14	NA
	3(b), 4(b), 5(b)	2	C	SR 3.3.1.14	NA
2. Power Range Neutron Flux					
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 110.8% RTP
b. Low	1(c), 2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 27.7% RTP
3. Power Range Neutron Flux Rate High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11	≤ 6.3 % RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1(c), 2(d)	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 31.5% RTP
5. Source Range Neutron Flux	2(e)	2	I,J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 1.4 E5 cps
	3(b), 4(b), 5(b)	2	J,K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≤ 1.4 E5 cps

(continued)

(a) The Allowable Value defines the limiting safety system setting. See the Bases for the Trip Setpoints.
(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(c) Below the P-10 (Power Range Neutron Flux) interlock.
(d) Above the P-6 (Intermediate Range Neutron Flux) interlock.
(e) Below the P-6 (Intermediate Range Neutron Flux) interlock.

Table 3.3.1-1 (page 2 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[a]
6. Overtemperature N-16	12	4	E	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	Refer to Note 1
7. Overpower N-16	12	4	E	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤ 112.9% RTP
8. Pressurizer Pressure					
a. Low	1(g)	4	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤ 1863.6 psig (Unit 1) ≤ 1865.2 psig (Unit 2)
b. High	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤ 2400.8 psig (Unit 1) ≤ 2401.4 psig (Unit 2)
9. Pressurizer Water Level - High	1(g)	3	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 93.9% of instrument span

(continued

(a) The Allowable Value defines the limiting safety system setting. See the Bases for the Trip Setpoints.
(b) Below the P-6 (Intermediate Range Neutron Flux) interlock.