February 24, 2000 Template = NRR-058

Mr. C. Lance Terry
Senior Vice President
& Principal Nuclear Officer
TXU Electric
Attn: Regulatory Affairs Department
P. O. Box 1002
Glen Rose, Texas 76043

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: INCREASE IN ALLOWABLE SPENT FUEL STORAGE CAPACITY AND CREDIT FOR SOLUBLE BORON IN THE SPENT FUEL POOL (TAC NOS. MA4841 AND MA4842)

Dear Mr. Terry:

The Commission has issued the enclosed Amendment No. 74 to Facility Operating License No. NPF-87 and Amendment No. 74 to Facility Operating License No. NPF-89 for Comanche Peak Steam Electric Station , Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated February 11, 1999, as supplemented by letters dated September 3 and December 20, 1999.

The amendments change the TSs to authorize an increase in the allowable spent fuel storage capacity and the crediting of soluble boron, in the spent fuel pool, for spent fuel reactivity control.

A copy of our related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

David H. Jaffe, Senior Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-445 and 50-446

Enclosures:
1. Amendment No. 74 to NPF-87
2. Amendment No. 74 to NPF-89
3. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TXU ELECTRIC COMPANY

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT NO. 1

DOCKET NO. 50-445

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 74
License No. NPF-87

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

   A. The application for amendment by TXU Electric Company (TXU Electric) dated February 11, 1999, as supplemented by letters dated September 3 and December 20, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment 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;

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-87 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 74 , 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. The license amendment is effective as of its date of issuance and shall be implemented no later than June 30, 2000.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A Gramm

Robert A. Gramm, Chief, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 24, 2000

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TXU ELECTRIC COMPANY

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT NO. 2

DOCKET NO. 50-446

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 74
License No. NPF-89

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

   A. The application for amendment by TXU Electric Company (TXU Electric) dated February 11, 1999, as supplemented by letters dated September 3 and December 20, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment 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;

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-89 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 74 , 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. This license amendment is effective as of its date of issuance and shall be implemented no later than June 30, 2000.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A Gramm

Robert A. Gramm, Chief, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 24, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 74

TO FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 74

FACILITY OPERATING LICENSE NO. NPF-89

DOCKET NOS. 50-445 AND 50-446

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
iii	iii
3.7-35	3.7-35
3.7-36	3.7-36
3.7-37	3.7-37
3.7-38	3.7-38
3.7-39	3.7-39
3.7-40	3.7-40
3.7-41	3.7-41
3.7-42	3.7-42
3.7-43	3.7-43
3.7-44	3.7-44
–	3.7-45
–	3.7-46
–	3.7-47
4.0-2	4.0-2
4.0-3	4.0-3

(continued)

3.7 PLANT SYSTEMS

3.7.16 Fuel Storage Pool Boron Concentration

LCO 3.7.16 The fuel storage pool boron concentration shall be $\geq$ 2000 ppm.

APPLICABILITY: When fuel assemblies are stored in the fuel storage pool .

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool boron concentration not within limit.	----------------NOTE---------------- LCO 3.0.3 is not applicable. --------------------------------------	
	A.1 Suspend movement of fuel assemblies in the fuel storage pool	Immediately
	AND	
	A.2 Initiate action to restore fuel storage pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify the fuel storage pool boron concentration is within limit.	7 days

3.7 PLANT SYSTEMS

3.7.17 Spent Fuel Assembly Storage

LCO 3.7.17 The combination of initial enrichment, burnup and decay time of each spent fuel assembly stored in high density racks shall be within either (1) the "acceptable" domain of Figure 3.7.17-1 in a 4 out of 4 configuration, (2) the "acceptable" domain of Figure 3.7.17-2 in a 3 out of 4 configuration, (3) the "acceptable" domain of Figure 3.7.17-3 in a 2 out of 4 configuration, or (4) shall be stored in a 1 out of 4 configuration. The acceptable storage configurations are shown in Figure 3.7.17-4.

APPLICABILITY: Whenever any fuel assembly is stored in high density racks of the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ------NOTE-------------- LCO 3.0.3 is not applicable. ---------------------------- Initiate action to move the noncomplying fuel assembly to an acceptable storage location .	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.1	Verify by administrative means the initial enrichment, burnup and decay time of the fuel assembly is in accordance with either (1) the "acceptable" domain of Figure 3.7.17-1 in a 4 out of 4 configuration, (2) the "acceptable" domain of Figure 3.7.17-2 in a 3 out of 4 configuration, (3) the "acceptable" domain of Figure 3.7.17-3 in a 2 out of 4 configuration, or (4) a 1 out of 4 configuration. The acceptable storage configurations are shown in Figure 3.7.17-4 .	Prior to storing the fuel assembly in high density racks

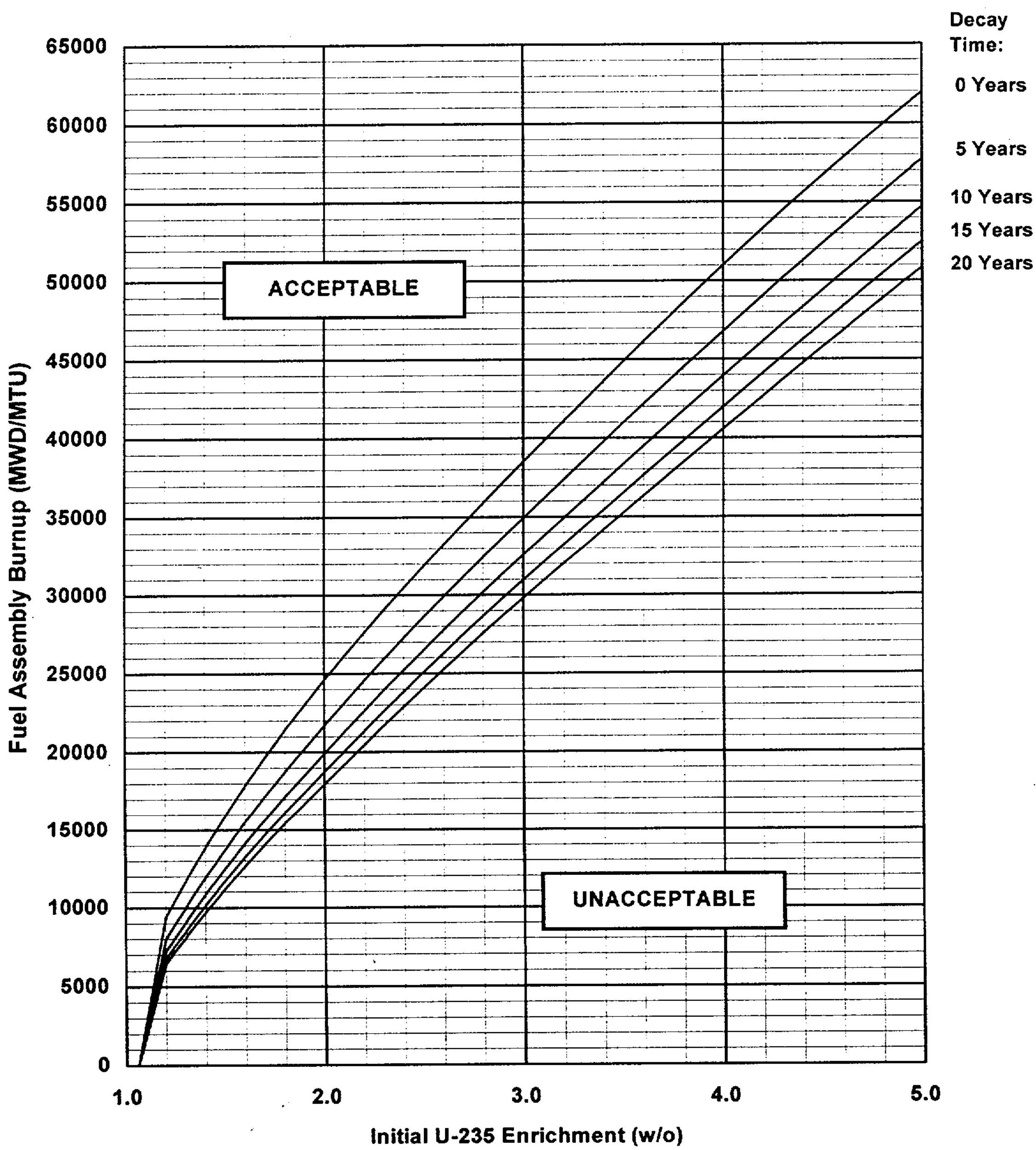

FIGURE 3.7.17-1
Fuel Assembly Burnup vs. U-235 Enrichments vs. Decay Time Limits for All Cell Storage in High Density Spent Fuel Storage Racks

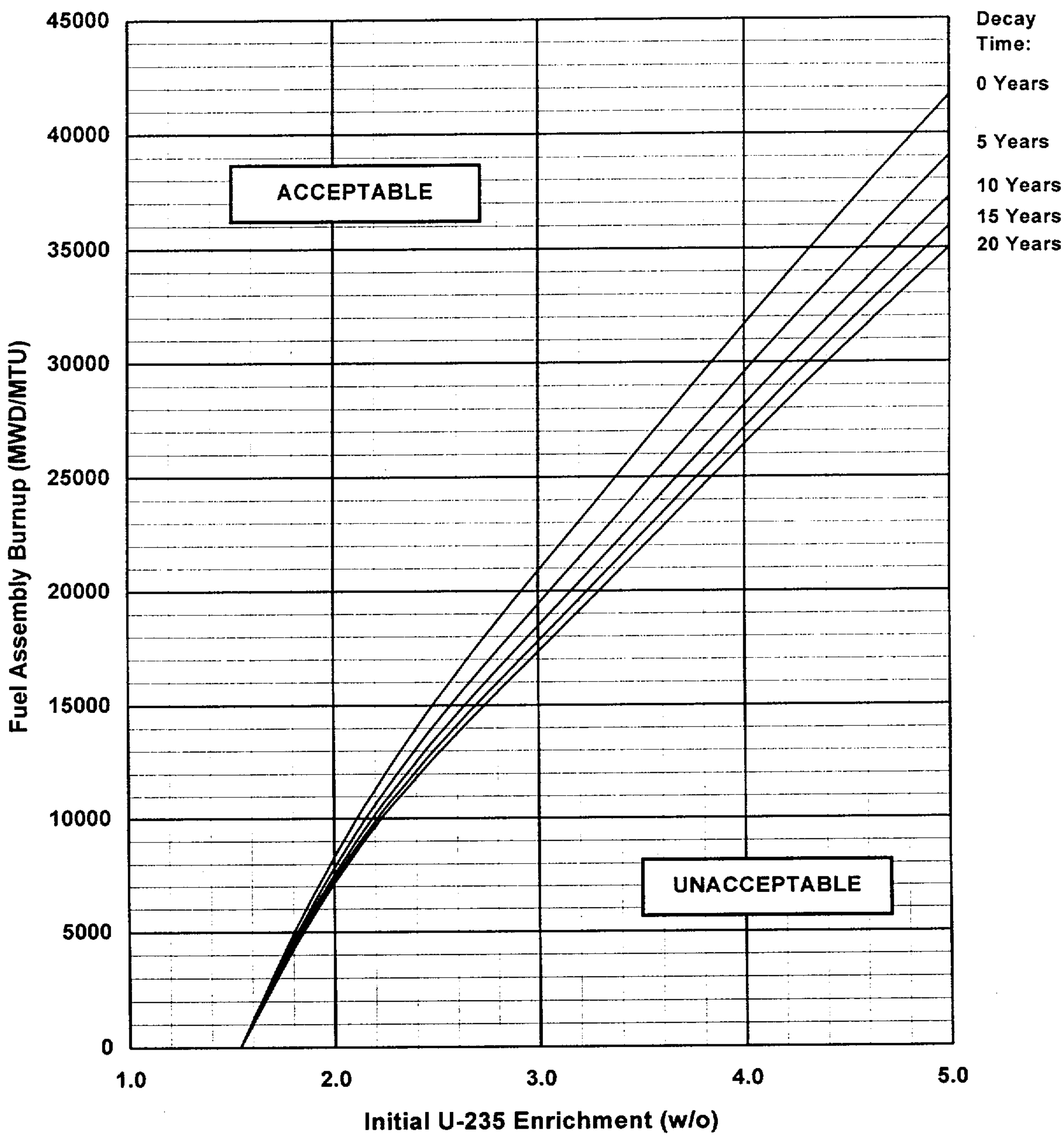

Figure 3.7.17-2
Minimum Burnup vs. Initial U-235 Enrichment vs. Decay Time
For a 3 out of 4 Storage Configuration in High Density Racks

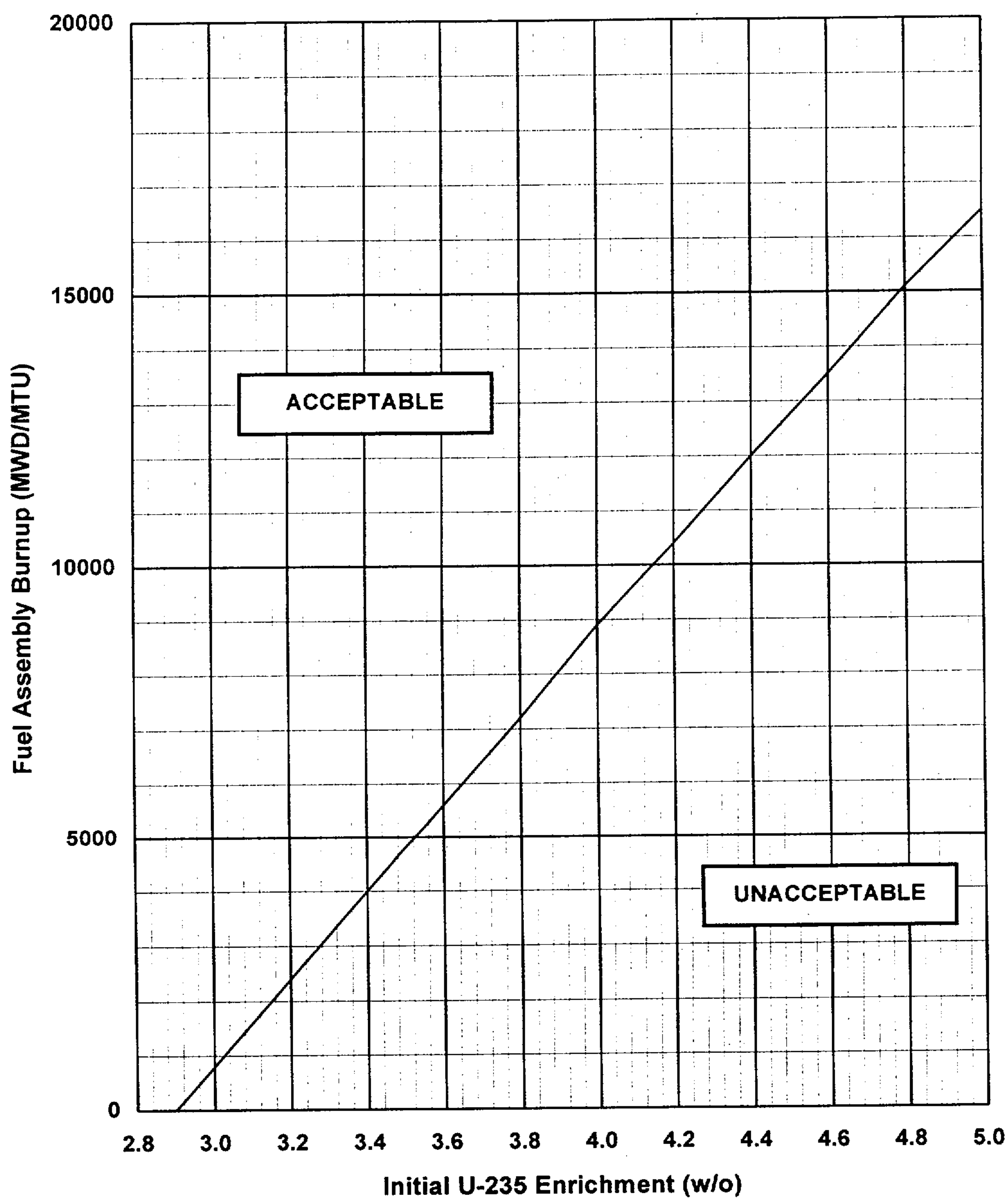

Figure 3.7.17-3
Minimum Burnup vs. Initial U-235 Enrichment
For a 2 out of 4 Storage Configuration in High Density Racks

A	A	A	A	A	A
A	A	A	A	A	A
A	A	A	A	A	A
A	A	A	A	A	A
A	A	A	A	A	A
A	A	A	A	A	A

	B		B		B
B	B	B	B	B	B
	B		B		B
B	B	B	B	B	B
	B		B		B
B	B	B	B	B	B

C		C		C	
	C		C		C
C		C		C	
	C		C		C
C		C		C	
	C		C		C

	D		D		D
	D		D		D
	D		D		D

A High density (all cell), new or partially spent fuel assemblies in the "acceptable" domain of Figure 3.7.17-1.

B High density (3/4), new or partially spent fuel assemblies in the "acceptable" domain of Figure 3.7.17-2.

C High density (2/4), new or partially spent fuel assemblies in the "acceptable" domain of Figure 3.7.17-3.

D High density (1/4), new or partially spent fuel assemblies which are stored in an expanded checkerboard (1 out of 4).

□ - empty

Note: All possible 2 by 2 matrices containing high density rack cells shall comply with at least one of the following: (1) within the "acceptable" domain of Figure 3.7.17-1 in a 4 out of 4 configuration, (2) within the "acceptable" domain of Figure 3.7.17-2 in a 3 out of 4 configuration, (3) within the "acceptable" domain of Figure 3.7.17-3 in a 2 out of 4 configuration, or (4) a 1 out of 4 configuration.

Figure 3.7.17-4
Storage Configurations (all cell, 3/4, 2/4, 1/4) in High Density Racks

3.7 PLANT SYSTEMS

3.7.18 Secondary Specific Activity

LCO 3.7.18 The specific activity of the secondary coolant shall be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3. <u>AND</u> A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.18.1 Verify the specific activity of the secondary coolant is ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131.	31 days

3.7 PLANT SYSTEMS

3.7.19 Safety Chilled Water

LCO 3.7.19 Two safety chilled water trains shall be OPERABLE

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One safety chilled water train inoperable.	A.1	Restore safety chilled water train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.19.1	-----NOTE----- Isolation of safety chilled water flow to individual components does not render the safety chilled water system inoperable. ---------- Verify each safety chilled water manual, power operated, and automatic valve servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.19.2	Verify each safety chilled water pump and chiller starts on an actual or simulated actuation signal.	18 months

3.7 PLANT SYSTEMS

3.7.20 UPS HVAC System

LCO 3.7.20 Two UPS HVAC System Trains shall be OPERABLE

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One UPS HVAC System train inoperable.	A.1	Verify the affected UPS & Distribution Room is supported by an OPERABLE UPS A/C Train.	Immediately
	<u>AND</u>		
	A.2	Restore the inoperable UPS HVAC train to OPERABLE status.	30 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Two UPS HVAC System trains inoperable. <u>OR</u> Required Action A.1 and associated Completion Time not met.	B.1	Verify air circulation is maintained by at least one UPS A/C Train.	Immediately
	<u>AND</u>		
	B.2	Verify the air temperature in the affected UPS & Distribution Room(s) does not exceed the maximum temperature limit for the room(s).	12 hours AND Once per 12 hours thereafter
	<u>AND</u>		
	B.3	Restore UPS HVAC System train to OPERABLE status.	72 hours
C. Required Action B.1 and associated Completion Time not met.	C.1	Restore the required support.	1 hour
D. Required Action and associated Completion Time of Required Action A.2, B.2, B.3 or C.1 not met.	D.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	D.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.20.1	Verify each required UPS & Distribution Room Fan Coil Unit operates ≥ 1 continuous hour.	31 days
SR 3.7.20.2	Verify each required UPS A/C train operates for ≥ 1 continuous hour.	31 days
SR 3.7.20.3	Verify each required UPS A/C train actuates on an actual or simulated actuation signal.	18 months

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} < 1.0$ when fully flooded with unborated water which includes an allowance for uncertainties as described in Section 4.3 of the FSAR.

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 750 ppm, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR;

d. A nominal 9 inch center to center distance between fuel storage locations in high density fuel storage racks;

e. A nominal 16 inch center to center distance between fuel assemblies placed in low density fuel storage racks;

f. New or partially spent fuel assemblies may be allowed restricted storage in a 1 out of 4 configuration in high density fuel storage racks (as shown in Figure 3.7.17-4) or unrestricted storage in low density fuel storage racks.

g. New or partially spent fuel assemblies with a discharge burnup in the "acceptable" domain of Figure 3.7.17-1 may be allowed unrestricted (all cell) storage in high density fuel storage racks as shown in Figure 3.7.17-4.

h. New or partially spent fuel assemblies with a discharge burnup in the "acceptable" domain of Figure 3.7.17-2 may be allowed restricted storage in a 3 out of 4 configuration in high density fuel storage racks as shown in Figure 3.7.17-4.

(continued)

4.0 DESIGN FEATURES

4.3.1.1 (continued)

i. New or partially spent fuel assemblies with a discharge burnup in the "acceptable" domain of Figure 3.7.17-3 may be allowed restricted storage in a 2 out of 4 configuration in high density fuel storage racks as shown in Figure 3.7.17-4.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR; and

d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pools are designed and shall be maintained to prevent inadvertent draining of the pool below elevation 854 ft.

4.3.3 Capacity

The spent fuel storage pools are designed and shall be maintained with a storage capacity limited to no more than 2026 fuel assemblies.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 74 TO

FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 74 TO

FACILITY OPERATING LICENSE NO. NPF-89

TXU ELECTRIC COMPANY

COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-445 AND 50-446

1.0 INTRODUCTION

By application dated February 11, 1999 (Reference 1), as supplemented by letters dated September 3 (Reference 2) and December 20, 1999 (Reference 3), TXU Electric Company (the licensee) requested changes to the Technical Specifications (TSs) for the Comanche Peak Steam Electric Station (CPSES), Units 1 and 2. The proposed changes would revise the TSs to allow an increase in the allowable spent fuel storage capacity and the crediting of soluble boron, in the spent fuel pool (SFP), for spent fuel reactivity control. The September 3 and December 20, 1999, supplements provided clarifying information that did not change the Nuclear Regulatory Commission (NRC) staff's proposed no significant hazards consideration determination or the scope of the February 11, 1999, application.

2.0 BACKGROUND

The CPSES spent fuel storage facilities are described in the CPSES Updated Final Safety Analysis Report (UFSAR), Section 9.1.2. CPSES is serviced by a common Fuel Building which contains two SFPs, SFP Number 1 (SFP1) and SFP Number 2 (SFP2) . The SFP1 is filled to capacity with low density spent fuel storage racks containing a total installed capacity of 556 fuel assemblies. On February 9, 1996, the NRC staff issued License Amendment Nos. 46 and 32 to the Facility Operating Licenses for CPSES Units 1 and 2, respectively (Reference 4). These license amendments authorized the use of high density spent fuel storage racks, fabricated by Westinghouse, in SFP2. Although the high density spent fuel storage racks had a capacity of 1,470 fuel assemblies, only 735 spent fuel assemblies were to be stored due to SFP reactivity considerations; however, the December 30, 1994, application (Reference 5) provided a structural evaluation assuming a full 1,470 fuel assembly storage capacity. In addition, Reference 5 provided thermal analyses to determine bulk SFP temperature based upon 3,386 assemblies in the SFPs. Although the high density spent fuel storage racks were designed to utilize Boraflex as a neutron absorbing material, the Boraflex was removed by the licensee due to questions concerning degradation of this material in other spent fuel storage applications.

The February 11, 1999, application, as supplemented by letters dated September 3 and December 20, 1999, requests an increase in the maximum number of spent fuel assemblies that may be stored in the CPSES, SPF2, from 735 to 1,470 spent fuel assemblies (for a total CPSES SFP fuel storage increase from 1,291 to 2,026 fuel assemblies). The increase in spent fuel storage capacity would be achieved by crediting the soluble boron in the SFP for reactivity control in SFP2. The SFP1 low density spent fuel racks are not affected by the application for license amendment. No new spent fuel storage racks would be installed in SFP1 or SFP2 as proposed in the application for license amendment.

3.0 EVALUATION

The NRC staff reviewed the ability of the existing high density spent fuel storage facility to accommodate the additional spent fuel assemblies in the existing spent fuel storage racks. The NRC staff reviewed the ability of the spent fuel storage racks and SFPs to support the additional physical loads and the SFPs and auxiliary systems to cope with the thermal loads. In addition, the NRC staff reviewed the potential radiological consequences of additional spent fuel storage. Finally, the NRC staff performed a detailed review of the use of soluble boron in the SFP to control the reactivity associated with the proposed spent fuel storage configurations; this aspect was considered to be the only significant change in the previous CPSES licensing basis for spent fuel storage (References 4 and 5, and the CPSES UFSAR).

3.1 Structural Evaluation

The purpose of the NRC staff's structural evaluation was to assure the structural integrity and functionality of the racks, the stored fuel assemblies and the SFP structure subject to the effects of the postulated loads (Appendix D of Standard Review Plan (SRP, Reference 6), Section 3.8.4) and fuel handling accidents with regard to the proposed increase in the number of spent fuel assemblies to be stored in the SFP2.

The high density spent fuel storage racks are seismic Category I equipment, and are required to remain functional during and after a safe shutdown earthquake (SSE). The licensee used a computer program, WECAN, for dynamic analyses to demonstrate the structural adequacy of the CPSES spent fuel rack design under the combined effects of earthquake and other applicable loading conditions. The high density spent fuel storage racks are free-standing and self-supporting equipment, and they are not attached to the floor or walls of the SFP. Nonlinear dynamic models consisting of inertial mass elements, spring elements, gap elements, and friction elements, as defined in the program, were used to simulate the three dimensional (3-D) dynamic behavior of the rack and the stored fuel assemblies, including frictional and hydrodynamic effects. The program calculated nodal forces and displacements at the nodes, and then obtained the detailed stress field in the rack elements from the calculated nodal forces.

Analyses of two models were performed: a 3-D single rack (SR) model and a 3-D whole pool multi-rack (MR) model. Several configurations were used for the 3-D SR analyses. The rack was considered to be fully loaded and partially loaded with two different coefficients of friction (μ=0.2 and 0.8) between the rack pedestal and the pool floor to investigate the stability of the rack with respect to overturning. For the 3-D MR analyses, all racks were considered fully loaded to investigate the fluid-structure interaction effects between the racks and the pool walls, as well as those among the racks, and to identify the worst case response for rack movement and for rack member stresses.

The seismic analyses were performed utilizing the direct integration time-history method. One set of three artificial time histories (two horizontal and one vertical acceleration components) were generated from the design response spectra defined in the CPSES UFSAR, Section 3.7B, "Seismic Design." The licensee demonstrated the adequacy of the single artificial time history set used for the seismic analyses by satisfying requirements of both enveloping design response spectra, as well as matching a target power spectral density (PSD) function compatible with the design response spectra as discussed in SRP, Section 3.7.1.

In the 3-D SR and MR analyses, the racks were subjected to the service, upset and faulted loading conditions (American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) Level A, B, and D service limits). The results of the analyses show that the maximum displacement of the racks at the top is about 0.3 inch indicating that there is adequate safety margin against overturning of the racks. The results of the analyses also show that there is no impact potential between the racks and between the rack and the pool wall. The calculated stresses in tension, compression, bending, combined flexure and compression, and combined flexure and tension were compared with corresponding allowable stresses specified in ASME Code, Section III, Subsection NF. The results show that all induced stresses under the SSE loading condition are smaller than the corresponding allowable stresses specified in the ASME Code, indicating that the rack design is adequate.

The licensee also calculated the rack weld stresses at the connections (e.g., baseplate-to-rack, baseplate-to-pedestal, and cell-to-cell connections) under the dynamic loading conditions. The licensee demonstrated that all of the calculated weld stresses are smaller than the corresponding allowable stresses specified in the ASME Code, indicating that the weld connection design of the rack is adequate.

Based on the licensee's comprehensive parametric study (e.g., varying coefficients of friction and fuel loading conditions of the rack), the adequate factor of safety of the induced stresses in the rack when they are compared to the corresponding allowable values provided in the ASME Code, and the licensee's overall structural integrity conclusions supported by both SR and MR analyses, the NRC staff concludes that the spent fuel storage rack modules will perform their safety function and maintain their structural integrity under postulated loading conditions and, therefore, are acceptable when loaded with the maximum proposed allowable number of fuel assemblies.

The licensee analyzed the SFP to demonstrate the adequacy of the structures under fully loaded fuel racks with all storage locations occupied by fuel assemblies. The fully-loaded structures were subjected to the load combinations specified in the CPSES UFSAR.

The CPSES UFSAR shows the predicted factors of safety varying from 1.23 to 3.44 for shear force and bending moment of the concrete walls and slab. In view of the calculated factors of safety, the NRC staff concludes that the licensee's structural analyses demonstrate the adequacy and integrity of the structures under full fuel loading, thermal loading, and SSE loading conditions. Thus, the SFP design is acceptable when loaded with the maximum allowable number of fuel assemblies.

The following three refueling accident cases were evaluated by the licensee: (1) drop of a fuel assembly through an empty cell onto the baseplate of the rack structure, (2) drop of a fuel assembly and control rod assembly onto the top of the rack structure from a drop height of 3.5 feet in a straight attitude, and (3) drop of a fuel assembly and control rod assembly onto the top of the rack structure from a drop height of 3.5 feet in an inclined attitude.

The analyses results show that the load transmitted to the liner through the rack structure is properly distributed through the bearing pads located near the fuel handling area; therefore, the liner would not be damaged by the impact. The NRC staff has reviewed the licensee's analyses results submitted in Reference 5, and concludes that they are acceptable in that they are supported by the parametric studies.

In conclusion, based on the review and evaluation of the licensee's submittals (References 1, 2, and 5), the NRC staff concludes that the licensee's structural analysis and design of the spent fuel rack modules and the SFP structures are adequate to withstand the effects of the applicable loads including that of the SSE. The analysis and design are in compliance with the current licensing basis set forth in the UFSAR and applicable provisions of the SRP, and are therefore acceptable with regard to the proposed increased number of spent fuel assemblies to be stored in the SFP.

3.2 Spent Fuel Pool Auxiliary Systems

The fuel storage building and associated SFPs are serviced by two important auxiliary systems: (1) the SFP cooling and cleanup system, described in CPSES UFSAR, Section 9.1.3, which is designed to remove decay heat from the spent fuel and maintain water clarity and purity, and (2) the SFP area ventilation system, described in CPSES UFSAR, Section 9.4.2, which is designed to maintain suitable environmental conditions (temperature, humidity, and radiation level) for plant personnel and equipment. The impact of the proposed increase in spent fuel storage on these auxiliary systems was evaluated.

3.2.1 Spent Fuel Pool Cooling and Cleanup System

In Reference 4, the NRC staff reviewed the capability of the CPSES SFP Cooling and Cleanup System (SFPCCS) to provide adequate cooling of spent fuel under several postulated conditions assuming an initial combined SFP inventory of 2,820 fuel assemblies. The licensee determined (Reference 5) that the most limiting case, with regard to bulk SFP temperature, involved a normal full core off-load (maximum design condition) of 193 fuel assemblies coincident, plus 94 fuel assemblies from the other unit, for a total of 3,107 fuel assemblies with a single failure of one cooling train. The licensee assumed a decay heat based upon 3,386 fuel assemblies. Under these conditions, the SFP bulk water was calculated to be 191 °F compared to a CPSES UFSAR design value of 200 °F. Even though long-term exposure of concrete structures to temperatures above 150 °F may result in damage to these structures, the NRC staff concluded that these temperatures are not of concern due to their transient nature. With both cooling trains available, the licensee calculated the maximum bulk SFP temperature to be 139 °F. In Reference 4, the NRC staff concluded that the design of the SFPCCS complies with the guidance of SRP 9.1.3 with regard to providing adequate cooling for the postulated spent fuel inventory under normal full core offload conditions, and the maximum SFP bulk temperature for the abnormal full core offload, assuming both trains of the SFPCCS are in operation, was calculated to be below the temperature associated with the onset of bulk boiling and, therefore, the CPSES SFPCCS meets the guidance of SRP, Section 9.1.3 for adequate SFP cooling under abnormal conditions.

In Reference 7, the NRC staff approved a 1% increase in rated thermal power (RTP) for CPSES, Unit 2. The licensee indicated, in Reference 1, that the power increase would result in a small increase in the decay heat of fuel discharged from CPSES, Unit 2. The NRC staff concludes that the increase in decay heat is small and that, for the proposed increase in spent fuel storage capacity from 1,291 to 2,026 spent fuel assemblies, the resulting decay heat load will be within the design capability of the CPSES SFPCCS for normal and abnormal core offload conditions. Accordingly, the conclusions with regard to conformance with SRP, Section 9.1.3 are still applicable with regard to the proposed increase in SFP storage capacity. The NRC staff conclusion is based upon the conservative nature of the existing calculations (decay heat based upon 3,386 fuels assemblies) when compared with the actual proposed spent fuel storage capacity of 2,026 spent fuel assemblies, even assuming a small increase in decay heat associated with approved increase in RTP for CPSES, Unit 2.

3.2.2 Spent Fuel Pool Area Ventilation System

As discussed in Section 3.2.1, herein, the proposed increase in spent fuel storage capacity will not result in a significant increase in SFP temperature and, thus, the SFP evaporation rate should not significantly increase. Since neither SFP area temperature nor humidity will significantly change, it is expected that the ability of the SFP Area Ventilation System to maintain air quality and temperature will not be affected by the proposed increase in spent fuel storage capacity.

3.3 Spent Fuel Pool Thermal-Hydraulic Analyses and Fuel Cladding Integrity

In Reference 4, the NRC staff reviewed the licensee's evaluation (Reference 5) of potential boiling in the SFPs caused by loss of all SFP cooling and concluded that, under the most severe conditions, the minimum time to boiling would be in excess of three hours. In addition, a sufficient number of SFP make-up water sources exist to conform to the provisions of Section 9.1.3 of the SRP with regard to available make-up water sources. In addition, in Reference 4, the NRC staff concluded that fuel cladding integrity would be maintained in that the SFP thermal-hydraulic conditions provide sufficient cooling as follows: (1) for a flow blockage up to 80%, no boiling occurs, and (2) for complete loss of SFP cooling, natural circulation provides sufficient cooling to prevent fuel cladding failure.

In Reference 1, the licensee indicated that previous SFP thermal-hydraulic analyses presented in Reference 5 are still applicable to the proposed increase in fuel storage capacity. The NRC staff finds that the conclusions in Reference 4, with regard to SFP thermal-hydraulics and fuel cladding integrity, remain valid with regard to the proposed increase in fuel storage capacity. This conclusion is based upon the conservative nature of the existing calculations (decay heat based upon 3,386 fuels assemblies) when compared with the actual proposed spent fuel storage capacity of 2,026 spent fuel assemblies, even assuming a small increase in decay heat associated with approved increase in RTP for CPSES Unit 2.

3.4 Accident Evaluation

The NRC staff evaluated the potential radiological consequences of the refueling accident and the dilution of soluble boron in the SFP water.

3.4.1 Boron Dilution

Reference 1 makes use of NRC-approved Westinghouse Owners Group generic methodology for crediting soluble boron as described in Topical Report WCAP-14416-NP-A, Revision 1 (Reference 8). The licensee performed a boron dilution analysis to ensure that sufficient time is available to detect and mitigate the dilution prior to exceeding the 0.95 k_{eff} design basis. The analysis applies to both SFPs since they are essentially identical. Potential events were quantified to show that sufficient time will be available to enable adequate detection and suppression of any dilution event.

Deterministic dilution event calculations were performed for the CPSES SFPs to define the dilution times and volumes necessary to dilute a SFP from an initial boron concentration of 1,800 parts per million (ppm) to a soluble boron concentration of 750 ppm. Proposed TS 3.7.16, "Fuel Pool Boron Concentration," in Reference 1, would require that the SFP boron concentration be $\geq$ 2,000 ppm. Currently, CPSES maintains a boron concentration of 2,400 ppm in the SFPs. CPSES conservatively chose a boron concentration of 1,800 ppm as the initial concentration for the boron dilution event. Based on the criticality analysis, the soluble boron concentration required to maintain the SFP at k_{eff} < 0.95 is 750 ppm.

Each SFP has a water inventory of 300,000 gallons. Assuming a well-mixed SFP, the volume required to dilute the SFP from 1,800 ppm to 750 ppm is approximately 262,600 gallons of non-borated water. The various events that were considered included dilution from the reactor makeup water system, demineralized water system, component cooling water system, fire protection system, and chemical and volume control system letdown. Other events that may affect the boron concentration of the SFP such as pipe cracks and loss of offsite power were also evaluated. The licensee stated that all pipes in the vicinity of the SFP are seismically qualified and supported. As such, a random pipe break was not considered in their analyses. The licensee did follow the guidance of Reference 6, Branch Technical Position, Mechanical Engineering Branch 3-1 (MEB 3-1), for selection of their potential pipe cracks.

Both the demineralized water system and the fire protection system have tanks large enough to dilute the SFP without replenishment. The usable volume of the demineralized water tank is approximately 316,000 gallons. The fire protection system has two 529,500 gallon tanks. The demineralized water system is directly connected to the SFP through a three inch line that is isolated by a closed manual valve. However, the largest dilution rate would be 260 gallons per minute (gpm), which would take over 16 hours to dilute the SFP to 750 ppm. The fire protection system has piping in the vicinity of the SFP. In accordance with MEB 3-1 of the SRP, the licensee postulated a crack developing in the two inch fire protection system line. The estimated dilution rate would be 21 gpm and would take over eight days to dilute the SFP to 750 ppm.

Other evaluated dilution events take longer than 12 hours to reach the minimum boron concentration. These events would be detected by plant personnel during required rounds every 12 hours. Additionally, SRP level instrumentation alarms in the control room. If a dilution event caused the level in one SFP to increase from the low level setpoint to the high level alarm point (one foot span), a dilution of 9,295 gallons would occur before an alarm is received in the control room; this is equivalent to a reduction of boron concentration of 55 ppm from 1,800 ppm. To detect low flow, long term dilution events, the proposed TS would require that the SFP be sampled every seven days. This frequency is consistent with proposed TS 3.7.16.1, "Fuel Storage Pool Boron Concentration," and is acceptable.

The licensee concluded that an unplanned or inadvertent event that would dilute the SFP boron concentration from 1,800 ppm to 750 ppm would be readily detectable by plant personnel via alarms, flooding in the fuel and auxiliary buildings, or by normal operator rounds through the SFP area. The NRC staff finds that the combination of the large volume of water required for a dilution event, TS-controlled SFP concentration and seven-day sampling requirement, and plant personnel rounds would adequately detect a dilution event prior to k_{eff} reaching 0.95 (750 ppm). Therefore, the analysis and proposed TS controls are acceptable for the boron dilution aspects of Reference 1.

Additionally, the criticality analysis for the SFP demonstrated that k_{eff} remains < 1.0 at a 95/95 probability/confidence level even with non-borated water in the SFP. Therefore, even if the SFP was diluted to approximately zero ppm, the spent fuel in the SFP storage racks would remain subcritical.

In conclusion, the NRC staff has reviewed the licensee's submittals (References 1, 2, and 3) and concluded that the boron dilution aspects of the proposed CPSES license amendment request have been acceptably addressed by the licensee. The TS boron concentration of 2,000 ppm or greater, and a seven-day surveillance requirement, both specified in proposed TS 3.7.16, are acceptable to ensure that sufficient time is available to detect and mitigate a dilution event prior to exceeding the design basis k_{eff} of 0.95.

3.4.2 Dropped Fuel Assembly

In Reference 1, the licensee stated that the radiological consequences of a dropped fuel assembly had not changed for the proposed increase in spent fuel storage capacity and that the analysis in the CPSES UFSAR was still applicable. Section 15.7.4 of the CPSES UFSAR defines the dropped fuel assembly event as "...dropping of a spent fuel assembly in the Containment Building or spent fuel storage area floor resulting in the rupture of the cladding of all the fuel rods in the assembly...." The CPSES UFSAR considers the dropped fuel assembly case, in the SFP area (outside the containment) to be the limiting case with a consequential dose at the exclusion area boundary of 53.0 roentgen equivalent man (rem) to the thyroid and .44 rem to the whole body. The corresponding doses at the low population zone are 7.7 rem to the thyroid and 6.29E-02 rem to the whole body. These consequential doses, due to the dropped fuel assembly, are a small fraction of the limits (300 rem to the thyroid/25 rem full body) specified in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 100, Section 100.11.

In Reference 7, the NRC staff approved a 1% increase in RTP for CPSES Unit 2 (Amendment 72 to the Facility Operating Licenses for CPSES, Units 1 and 2.) In the Safety Evaluation associated with Amendment, the NRC staff concluded that there is reasonable assurance that the radiological consequences of accidents, considered in the CPSES UFSAR will remain the same, or bounded by current values, as a result of the 1% increase in RTP.

Accordingly, the NRC staff concludes that the potential radiological consequences of a dropped fuel assembly, associated with the proposed increase in fuel storage capacity, are acceptable.

3.5 Criticality Evaluation

The CPSES spent fuel storage racks were analyzed using the Westinghouse methodology which has been reviewed and approved by the NRC (Reference 8). This methodology takes partial credit for soluble boron in the fuel storage pool criticality analyses and requires conformance with the following NRC acceptance criteria for preventing criticality outside the reactor:

1) k_{eff} shall be < 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties at a 95% probability, 95% confidence (95/95) level as described in WCAP-14416-NP-A; and

2) k_{eff} shall be $\leq$ 0.95 if fully flooded with borated water, which includes an allowance for uncertainties at a 95/95 level as described in WCAP-14416-NP-A.

The analysis of the reactivity effects of fuel storage in the CPSES spent fuel racks was performed with the three-dimensional Monte Carlo code, KENO-Va, with neutron cross sections generated with the NITAWL-II and XSDRNPM-S codes using the 227 group ENDF/B-V cross-section data. Since the KENO-Va code package does not have burnup capability, depletion analyses and the determination of small reactivity increments due to manufacturing tolerances were made with the two-dimensional transport theory code, PHOENIX-P, which uses a 42 energy group nuclear data library from ENDF/B-V data. The analytical methods and models used in the reactivity analysis have been benchmarked against experimental data for fuel assemblies similar to those for which the CPSES racks are designed and have been found to adequately reproduce the critical values. This experimental data is sufficiently diverse to establish that the method bias and uncertainty will apply to rack conditions which include close proximity storage and strong neutron absorbers. The NRC staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the CPSES storage racks with a high degree of confidence.

The cell spacing in the high density racks is a nominal 9-inch center-to-center, and the racks contain no special neutron absorbing material. The spent fuel assemblies in the high density racks are currently stored in either a 1 out of 4 storage configuration (given a 4 cell box-type array, one cell contains a fuel assembly and 3 cells are empty) or a two-out-of-four storage configuration. The licensee is proposing to add two additional storage configurations in the high density racks, a 3 out of 4 (3/4) storage configuration and a four-out-of-four (all cell) storage configuration. The high density CPSES spent fuel storage racks have previously been qualified for storage of various Westinghouse 17x17 fuel assembly types with maximum enrichments up to 5.0 weight percent (w/o) uranium (U)-235. The maximum enrichment includes a manufacturing tolerance of 0.05.

For calculational purposes, the SFP moderator was assumed to be pure water at a temperature of 68 °F and a density of 1.0 grams per cubic centimeter (gm/cc) and the array was assumed to be infinite in lateral (x and y) extent. Uncertainties due to tolerances in fuel enrichment and density, fuel pellet dishing, storage cell inner diameter, storage cell pitch, stainless steel thickness, wrapper plate thickness, assembly position, calculational uncertainty, and methodology bias uncertainty were accounted for. These uncertainties were appropriately determined at the 95/95 probability/confidence level. A methodology bias (determined from benchmark calculations) as well as a reactivity bias to account for the effect of the normal range of SFP water temperatures (50 °F to 150 °F) were included. These biases and uncertainties meet the previously stated NRC staff requirements and are, therefore, acceptable.

For the high density racks, the nominal enrichment required to maintain $k_{eff} < 1.0$ with all cells filled with Westinghouse 17x17 fuel assemblies and no soluble boron in the pool water was found to be 1.06 w/o U-235. This resulted in a nominal k_{eff} of 0.96653. The 95/95 k_{eff} was then determined by adding the temperature and methodology biases and the statistical sum of independent tolerances and uncertainties to the nominal k_{eff} values, as described in Reference 8. This resulted in a 95/95 k_{eff} of 0.99451. Since this value is < 1.0 and was determined at a 95/95 probability/confidence level, it meets the NRC staff's criterion for precluding criticality with no credit for soluble boron and is acceptable.

A similar calculation was performed assuming a 3 out of 4 assembly checkerboard configuration with 1 empty cell and 3 fresh assemblies. The nominal enrichment required to maintain $k_{eff} < 1.0$ for this configuration and no soluble boron was found to be 1.54 w/o U-235. The resulting nominal k_{eff} was 0.97555. The 95/95 k_{eff} was determined to be 0.99576, also meeting the NRC staff's criterion for precluding criticality with no soluble boron, and is, therefore, acceptable.

Soluble boron credit is used to provide safety margin by maintaining $k_{eff} \leq 0.95$ including 95/95 uncertainties. The soluble boron credit calculations assumed the all cell storage configuration moderated by water borated to 200 ppm. As previously described, the individual tolerances and uncertainties, and the temperature and methodology biases, were added to the calculated nominal k_{eff} to obtain a 95/95 value. The resulting 95/95 k_{eff} was 0.93492 for fuel enriched to 1.06 w/o U-235. Since k_{eff} is < 0.95 with 200 ppm of boron and uncertainties at a 95/95 probability/confidence level, the NRC staff's acceptance criterion for precluding criticality with credit for soluble boron is satisfied. The concentration of soluble boron required to maintained $k_{eff} < 0.95$ is well below the minimum SFP boron concentration value of 2,000 ppm required by proposed TS 3.7.16 and is, therefore, acceptable.

A similar calculation for the 3 out of 4 checkerboard configuration under nominal conditions with 200 ppm of soluble boron in the moderator resulted in a k_{eff} of 0.91968. The resulting 95/95 k_{eff} was 0.94024 for fuel enriched to 1.54 w/o U-235, also meeting the NRC staff's acceptance criterion for precluding criticality with credit for soluble boron, and is therefore, acceptable.

The concept of reactivity equivalencing due to fuel burnup was used to achieve the storage of fuel assemblies with enrichments higher than 1.05 w/o U-235 for the all cell storage configuration, and > 1.54 w/o U-235 for the 3 out of 4 checkerboard configuration. The NRC staff has previously accepted the use of reactivity equivalencing predicated upon the reactivity decrease associated with fuel depletion. To determine the amount of soluble boron required to maintain $k_{eff} \leq 0.95$ for storage of fuel assemblies with maximum enrichments up to 5.0 w/o U-235 for both the all cell configuration and the 3 out of 4 configuration, a series of reactivity calculations were performed to generate a set of enrichment versus fuel assembly discharge burnup ordered pairs which all yield an equivalent k_{eff} when stored in the CPSES spent fuel storage racks. These are shown in the licensee's proposed TS Figures 3.7.17-1 and 3.7.17-2 for the all cell configuration and the 3 out of 4 configuration, respectively. These curves represent combinations of fuel enrichment and discharge burnup which yield the same rack k_{eff} as the rack loaded with fresh (zero burnup) fuel with the maximum allowed enrichments derived previously. Uncertainties associated with burnup credit include a reactivity uncertainty of 0.01 Δk at 30,000 Mega-watt Days per Metric-ton of Uranium (MWD/MTU) applied linearly to the burnup credit requirement to account for calculational and depletion uncertainties and 5% on the calculated burnup to account for burnup measurement uncertainty. The NRC staff concludes that these uncertainties conservatively reflect the uncertainties associated with burnup calculations and are acceptable. The amount of additional soluble boron, above the 200 ppm value required above, that is needed to account for these uncertainties is 550 ppm for the all cells configuration and 450 ppm for the 3 out of 4 configuration. This results in a total soluble boron credit of 750 ppm for the all cell configuration and 650 ppm for the 3 out of 4 checkerboard configuration. These values are well below the minimum SFP boron concentration value of 2,000 ppm required by proposed TS 3.7.16 and are, therefore, acceptable.

Proposed TS Figures 3.7.17-1 and 3.7.17-2 also credit the time an assembly has been discharged from the core. Decay time credit is an extension of the burnup credit process and results from the radioactive decay of isotopes in the spent fuel to daughter isotopes, which results in reduced reactivity. Although decay of the fission products has the effect of further reducing the reactivity of the spent fuel, in this amendment request, credit is taken only for the decay of actinides. Decay time credit has been previously approved by the NRC staff (Reference 9).

Two accidents can be postulated for each storage configuration which would increase reactivity beyond the analyzed conditions. The first would be an extension in pool water temperature from the normal range (50 °F to 150 °F) to a range of 32 °F to 212 °F. The second would be the loading of an assembly into a cell for which the restrictions on location, enrichment, or burnup are not satisfied (a misloaded assembly).

Calculations have shown that the misloaded assembly accident for a 3 out of 4 storage configuration results in the highest reactivity increase. The reactivity increase requires an additional 1,150 ppm of soluble boron to maintain $k_{eff} \leq 0.95$. However, for such events, the double contingency principle can be applied. This states that the assumption of two unlikely, independent, concurrent events is not required to ensure protection against a criticality accident. Therefore, the minimum amount of boron required by proposed TS 3.7.16 (2,000 ppm) is more than sufficient to cover any accident and the presence of the additional boron above the concentration required for normal conditions and reactivity equivalencing can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

In order to prevent an undesirable increase in reactivity, the boundaries between the different storage configurations were analyzed. The boundary between storage configuration zones and the boundary between a 1 out of 4, 2 out of 4, or 3 out of 4 storage configuration zone and an all cell storage configuration zone must be controlled to prevent an undesirable increase in reactivity. The fuel storage patterns must comply with the interface requirements shown in Figures 5 and 6 of Westinghouse Report CAC-98-274, "Comanche Peak High Density Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit" submitted as Enclosure 2 to Reference 1.

Based on the review described above, the NRC staff concludes that the criticality aspects of the proposed CPSES license amendment request are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling. The analysis assumed credit for soluble boron, as permitted by WCAP-14416-NP-A, but no credit for the Boraflex neutron absorber panels. The required amount of soluble boron for each analyzed storage configuration is shown in Table 1 (see next page). The criticality analysis conformed to the NRC staff guidance on the regulatory requirements for criticality analysis of fuel storage at light water reactor power plants (Reference 10).

The following storage configurations and U-235 enrichment limits for Westinghouse and Siemens 17x17 fuel assemblies were determined to be acceptable.

All Cell Storage Configuration

Assemblies with initial nominal enrichments no greater than 1.06 w/o U-235 can be stored in any cell location. Fuel assemblies with initial nominal enrichments greater than 1.06 and up to 5.00 w/o U-235 must satisfy the minimum burnup requirements shown in proposed TS Figure 3.7.17-1.

3 out of 4 Storage Configuration

Assemblies with initial nominal enrichments no greater than 1.54 w/o U-235 can be stored in a 3-out-of-4 storage configuration arrangement with empty cells. Fuel assemblies with initial nominal enrichments greater than 1.54 and up to 5.00 w/o U-235 must satisfy the minimum burnup requirements shown in proposed TS Figure 3.7.17-2.

1 out of 4 and 2 out of 4 Checkerboard Storage

The fuel assemblies in the high density fuel storage racks, currently stored in either a 1 out of 4 storage configuration or a 2 out of 4 storage configuration, remain acceptable.

TABLE 1

Summary of Soluble Boron Credit Requirements for CPSES Units 1 and 2

Storage Configuration	Soluble Boron Required for $k_{eff} \leq 0.95$ (ppm)	Soluble Boron Required for Reactivity Equivalencing (ppm)	Total Soluble Boron Credit Required Without Accidents (ppm)
All Cell storage configuration	200	550	750
3 out of 4 storage configuration	200	450	650

3.6 Proposed Technical Specifications

The licensee has proposed changes to the TS to implement the proposed increase in fuel storage capacity; these changes are as follows:

- The licensee has proposed a new Limiting Condition for Operation (LCO), 3.7.16, and a new Surveillance Requirement (SR), 3.7.16, to require that the SFP water boron concentration be maintained at $\geq$2,000 ppm, when fuel assemblies are stored in the SFP. Should the SFP water boron concentration be < 2,000 ppm, the following actions would be required to be completed "immediately:" (1) suspend movement of fuel assemblies in the SFP, and (2) initiate action to restore the boron concentration. A third alternative in the generic Westinghouse TS, immediately verify by administrative means that a verification of fuel location in the SFP (SFP verification) has been performed since the last fuel movement in the SFP, was not proposed by the licensee. As indicated in Section 3.5, herein, maintenance of 2,000 ppm of boron concentration in the SFP water is acceptable to prevent criticality. The absence of the requirement for SFP verification is acceptable in that it would appear in the TS as an alternative "required action." Based upon the above, the NRC staff concludes that the proposed LCO substantially conforms to the generic Westinghouse TS and is acceptable.

- The proposed SR 3.7.16.1, which would require the SFP boron concentration verification every seven days, conforms to the generic Westinghouse TS and is acceptable as described in Section 3.4.1, herein, with regard to the detection of SFP boron dilution.

- The licensee has proposed a revision to CPSES TS 3/4.7.17, "Spent Fuel Assembly Storage." The proposed changes would incorporate limitations for "All Cell Storage" and "3 out of 4 Storage" configurations in TS Figures 3.7.17-1 and 3.7.17-2, respectively. These TS figures are acceptable as described in Section 3.5, herein. A revised TS Figure 3.7.17-1, for the "2 out of 4 Storage Configuration" would become TS Figure 3.7.17-3. The existing TS Figure 3.7.17-1 was found to contain a minor error which resulted in an overly restrictive requirement for storing fuel in a "2 out of 4 Storage Configuration." The NRC staff concludes that the revised TS Figure 3.7.17-1, which becomes TS Figure 3.7.17-3, is acceptable. A new TS Figure 3.7.17-4 would be added to provide a pictorial definition of the various acceptable fuel storage configurations. These pictorial definitions are consistent with the NRC staff's understanding of the definitions of the various acceptable storage arrays. Accordingly, proposed TS Figure 3.7.17-4 is acceptable.

- The licensee has proposed changes to CPSES TS 4.3.1, "Criticality," in order to incorporate the boron concentration for the $k_{eff} < 1.0$ (0 ppm) and $k_{eff} \leq 0.95$ (750 ppm) cases. In addition, the changes to TS 4.3.1 describe the allowable storage configurations. These proposed changes to TS 4.3.1 are acceptable as described in Section 3.5, herein.

- The licensee has proposed a change to CPSES TS 4.3.3, "Capacity," to increase the fuel storage capacity from 1,291 to 2,026 fuel assemblies. This proposed changed is supported by the NRC staff's conclusions, herein, and is acceptable.

In conclusion, the TS changes proposed as a result of the revised criticality analysis evaluated in Section 3.5, herein, are consistent with the NRC-approved methodology described in Westinghouse Topical Report, WCAP-14416-NP-A, Revision 1, (Reference 8). Based on this consistency with the approved methodology and on the above evaluation, the NRC staff finds these TS changes acceptable. The proposed associated Bases changes adequately describe these TS changes and are also acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Texas State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (64 FR 25522, dated May 12, 1999). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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