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Ref: 10CFR50.90
(Proprietary Information Enclosed)

CPSES-200002403
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File # 00236, 10010 (clo)

October 4, 2000

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
LICENSE AMENDMENT REQUEST (LAR) 00-05
REVISION TO TECHNICAL SPECIFICATION
SPENT FUEL ASSEMBLY STORAGE RACKS AND
FUEL STORAGE CAPACITY

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 change into the CPSES Unit 1 and 2 Technical Specifications. This change request applies to both units.

The proposed changes will revise Technical Specification (TS) 3.7.17, "Spent Fuel Assembly Storage," and TS 4.3, "Fuel Storage." These changes revise the specifications for fuel storage to increase the spent fuel storage capacity by (1) replacing the existing twenty low density racks of spent fuel pool one with three Holtec racks and nine Westinghouse racks, (2) adding three Holtec racks to the nine existing Westinghouse racks in spent fuel pool two, (3) revising the spent fuel storage curves in TS 3.7.17, (4) updating the criticality discussion in TS 4.3.1, and (5) increasing the spent fuel storage capacity from "2,026" to "3,373" fuel assemblies in TS 4.3.3.

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Attachment 1 is the required affidavit. Attachment 2 provides a detailed description of the proposed changes, a safety analysis of the proposed changes, TXU Electric's determination that the proposed changes do not involve a significant hazard consideration, a regulatory analysis of the proposed changes and an environmental evaluation. Attachment 3 provides the affected Technical Specification pages marked-up to reflect the proposed changes. Attachment 4 provides proposed changes to the Technical Specification Bases for information only. These changes will be processed per CPSES site procedures. Attachment 5 provides retyped Technical Specification pages which incorporate the requested changes. Attachment 6 provides retyped Technical Specification Bases pages which incorporate the proposed changes.

Enclosure 1 provides the "Licensing Report for Spent Fuel Rack Installation at Comanche Peak Steam Electric Station" that supports this license amendment request. Holtec International considers information contained in Enclosure 1 "Licensing Report for Spent Fuel Rack Installation at Comanche Peak Steam Electric Station," to be proprietary. In accordance with the requirements of 10CFR2.790(b) for withholding of proprietary information from public disclosure, an Affidavit is enclosed (Enclosure 4). Correspondence with respect to the proprietary aspects of the supporting Holtec Affidavit should be addressed to Holtec International, Attention Charles W. Bullard II, Holtec Center, 555 Lincoln Drive West, Marlton, NJ, 08053. Enclosure 5 provides a non-proprietary version of Enclosure 1.

Enclosure 2 provides the Comanche Peak High Density Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit and No Outer Wrapper Plates to support this license amendment request. Enclosure 3 provides the CPSES Spent Fuel Pool Boron Dilution Analysis to support this license amendment request.

TXU Electric requests approval of the proposed License Amendment Request by August 31, 2001, to be implemented no later than January 31, 2002. The extended implementation period is requested due to the significant field modifications required. The approval date was selected to allow for NRC review and to minimize the impact on the sixth refueling outage for Unit 2 (2RF06) (currently scheduled for the spring of 2002).

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

This communication contains new commitments which will be completed as noted in Attachment 7.

Should you have any questions, please contact Mr. Carl B. Corbin at (254) 897-0121.

Sincerely,

C. L. Terry

By: _____
James J. Kelley, Jr.
Vice President of Nuclear Engineering and Support

CBC/cbc

Attachments
1. Affidavit
2. Description and Assessment
3. Markup of Technical Specifications pages
4. Markup of Technical Specifications Bases pages (for information)
5. Retyped Technical Specification Pages
6. Retyped Technical Specification Bases Pages (for information)
7. Commitments

Enclosures
1. Licensing Report for Spent Fuel Rack Installation at Comanche Peak Steam Electric Station (proprietary)
2. Comanche Peak High Density Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit and No Outer Wrapper Plates
3. CPSES Spent Fuel Pool Boron Dilution Analysis.
4. Affidavit (request to withhold proprietary information in Enclosure 1)
5. Non-Proprietary version of Enclosure 1

c - E. W. Merschoff, Region IV (w/o Enclosure 5)
J. I. Tapia, Region IV (w/o Enclosure 5)
D. H. Jaffe, NRR (w/o Enclosure 5)
Resident Inspectors, CPSES (w/o Enclosure 5)
Charles W. Bullard II, Holtec International (w/o Enclosure 5)

Mr. Authur C. Tate
Bureau of Radiation Control
Texas Department of Public Health
1100 West 49th Street
Austin, Texas 78704

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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

AFFIDAVIT

James J. Kelley, Jr., being duly sworn, hereby deposes and says that he is the Vice President of Nuclear Engineering and Support of TXU Electric, the licensee herein; that he is duly authorized to sign and file with the Nuclear Regulatory Commission this License Amendment Request 00-05; 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.

James J. Kelley, Jr.
Vice President of Nuclear Engineering
and Support

STATE OF TEXAS)
)
COUNTY OF)
Somervell

Subscribed and sworn to before me, on this 4th day of October, 2000.

SUSAN C. GRAVATT
NOTARY PUBLIC
STATE OF TEXAS
My Commission Expires 3-23-2001

Susan C Gravatt
Notary Public

ATTACHMENT 2 to TXX-00144

DESCRIPTION AND ASSESSMENT

Description and Assessment

1.0 INTRODUCTION

1.1 Proposed change LAR-00-05 is a request to revise Technical Specifications (TS) 3.7.17, "Spent Fuel Assembly Storage," and TS 4.3, "Fuel Storage," for Comanche Peak Steam Electric Station (CPSES) Units 1 and 2."

1.2 FINAL SAFETY ANALYSIS REPORT (FSAR) SECTION

The CPSES Final Safety Analysis Report (Sections 3.1, 4.3 and 9.1) will be updated as required to reflect this License Amendment Request. The FSAR will be updated after the License Amendment Request has been approved and implemented.

2.0 DESCRIPTION

These changes revise the specifications for fuel storage to increase the spent fuel storage capacity by (1) replacing the existing twenty low density racks of spent fuel pool one with three Holtec racks and nine Westinghouse racks, (2) adding three Holtec racks to the nine existing Westinghouse racks in spent fuel pool two, (3) revising the spent fuel storage curves in TS 3.7.17, (4) updating the criticality discussion in TS 4.3.1.1, and (5) increasing the spent fuel storage capacity from "2,026" to "3,373" fuel assemblies in TS 4.3.3.

The changes to TS 3.7.17, TS 4.3.1.1, and TS 4.3.3 are marked-up (Attachment 3) on the TS pages as issued by License Amendment 74 (Reference 3(d)).

TXU Electric's requested changes to TS Sections 3.7.17 and 4.3 are summarized below.

<u>TS 3.7.17, Spent Fuel Assembly Storage</u>

Specification 3.7.17: Revises section "Spent Fuel Assembly Storage" to change the nomenclature from "high density" to "Region II" racks. Prior to this modification both low density and high density racks were used. After this modification is complete there will be two types of high density racks installed, Region I and Region II. Only those racks designated as Region II will have restrictions regarding storage configurations. The storage configurations consider the enrichment, burnup, and decay time for the fuel assemblies. This generic change in nomenclature is made in several locations as noted below.

Figure 3.7.17-1: Revises the figure which depicts the requirements for a 4 out of 4 storage configuration to reflect the updated criticality analysis. Revises Figure title to change the nomenclature from "All Cell Storage" to "a 4 out of 4 Storage Configuration" and from "High Density Spent Fuel Storage" to

"Region II" racks.

Figure 3.7.17-2: Revises the figure which depicts the requirements for a 3 out of 4 storage configuration to reflect the updated criticality analysis. Revises Figure title to change the nomenclature from "high density" to "Region II" racks.

Figure 3.7.17-3: Revises Figure title to change the nomenclature from "high density" to "Region II" racks.

Figure 3.7.17-4: Revises Figure text and title to change the nomenclature from "all cell" to "4/4" and from "high density" to "Region II" racks. Revises "Note" by adding second paragraph, "Region I and Region II interface restrictions: The Region II 1 out of 4 configuration shall be oriented such that the single fuel assembly resides in the internal row with the empty cells facing Region I. There are no restrictions between the Region II (2/4, 3/4 , and 4/4) and Region I configurations."

TS 4.3.1, Criticality

Specification 4.3.1.1(b): Replaces the period at the end of the sentence with a semicolon.

Specification 4.3.1.1(c): Revises the water boration requirement for a k_{eff} limit $\leq$ 0.95 from "750 ppm" to "800 ppm." This reflects the updated criticality analysis.

Specification 4.3.1.1(d): Revises the nomenclature from "high density" to "Region II."

Specification 4.3.1.1(e): Revises the cell spacing from "nominal 16 inch center to center" to "nominal 10.6 inch by nominal 11 inch center to center." Revises the nomenclature from "low density" to "Region I."

Specification 4.3.1.1(f): Revises the nomenclature from "high density" to "Region II" and from "low density" to "Region I." Replaces the period at the end of the sentence with a semicolon.

Specification 4.3.1.1(g): Revises nomenclature from "(all cell) storage" to "storage in a 4 out of 4 configuration." Revises the nomenclature from "high density" to "Region II." Replaces the period at the end of the sentence with a semicolon.

Specification 4.3.1.1(h): Revises the nomenclature from "high density" to "Region II." Replaces the period at the end of the sentence with a semicolon.

Adds the word "and" to the end of the sentence.

Specification 4.3.1.1(i): Revises the nomenclature from "high density" to "Region II."

TS 4.3.3, Capacity

Specification 4.3.3: The storage capacity of the two spent fuel pools is changed from 2026 to 3373 fuel assemblies.

In summary, the specifications for fuel storage have been revised (1) to identify the k_{eff} limit ($\leq$ 0.95) when fully flooded with water borated to 800 ppm, (2) to update figures to identify storage patterns for the existing and new racks, and (3) to increase the storage capacity from 2026 to 3373 fuel assemblies. The use of the new and existing Region II racks complies with a limiting storage configuration of one out of four (1/4), two out of four (2/4), three out of four (3/4) or four out of four (4/4) based on fuel enrichment, burnup, and decay time. All cells in the Region I racks may be used with no restrictions on fuel enrichment (except as noted by TS 4.3.1.1.a), burnup, or decay time.

These changes revise the specifications to increase the spent fuel storage capacity and update the storage configurations and figures.

For information only, this LAR includes proposed associated changes to the TS Bases 3.7.16, "Fuel Storage Pool Boron Concentration," and TS Bases 3.7.17, "Spent Fuel Assembly Storage." The boron concentration of TS 3.7.16, "Fuel Storage Pool Boron Concentration," is not changed, but the related Bases are updated to reflect the updated criticality analyses.

The changes to TS Bases 3.7.16 and TS Bases 3.7.17 are marked-up (Attachment 4) on the TS Bases pages as submitted by Attachment 4 to Reference 3(a).

TS Bases 3.7.16 / Fuel Storage Pool Boron Concentration

Revises Bases information to reflect the updated criticality analyses, removal of low density racks, and installation of Region I / Region II racks.

TS Bases 3.7.17 / Spent Fuel Assembly Storage

Revises Bases information to reflect the updated criticality analyses, removal of low density racks, and installation of Region I / Region II racks.

Retyped Technical Specification pages and Technical Specification Bases pages which incorporate the proposed changes, are provided in Attachments 5 and 6 respectively.

3.0 BACKGROUND

On February 24, 2000, the NRC staff issued License Amendment No. 74 to the Facility Operating Licenses for CPSES Units 1 and 2, respectively (Reference 3(d)). This submittal reflects the approved but not yet implemented NRC License Amendment 74. License Amendment 74 authorized the credit for soluble boron in the Spent Fuel Pool (SFP) to control reactivity associated with the new storage configurations and an increase in storage capacity from 1291 to 2026 fuel assemblies. No additional racks were authorized by License Amendment 74. On June 12, 2000, the NRC staff issued License Amendment No. 78 to the Facility Operating Licenses for CPSES Units 1 and 2 (Reference 3(f)). License Amendment 78, in part, extended the implementation date for License Amendment 74 from June 30, 2000, to December 31, 2000.

The CPSES spent fuel storage facilities are described in the CPSES Final Safety Analysis Report (FSAR), Section 9.1.2 (Reference 10). CPSES is serviced by a common Fuel Building which contains two spent fuel pools, spent fuel pool number one (SFP1) and spent fuel pool number two (SFP2). Each pool may be used to store fuel from either or both of the CPSES units. SFP1 is filled to capacity with twenty low density spent fuel storage racks (nominal 16 inch center to center spacing, bolted to the liner) containing a total installed capacity of 556 fuel assemblies. SFP2 has nine high density spent fuel storage racks (nominal 9 inch center to center spacing, free standing) containing a total installed capacity of 1,470 fuel assemblies. Although the existing nine high density spent fuel storage racks were designed to utilize Boraflex as a neutron absorbing material, the Boraflex was removed by TXU Electric due to questions concerning degradation of this material in other spent fuel storage applications.

This application requests an increase in the maximum number of spent fuel assemblies that may be stored in SFP1 from 556 to 1,684 fuel assemblies and in SFP2 from 1,470 to 1,689 fuel assemblies (for a total SFP fuel storage capacity increase from 2,026 to 3,373 fuel assemblies). The increase in spent fuel storage capacity is achieved by (1) extending the credit of soluble boron in SFP2 for reactivity control to include SFP1 (crediting of the soluble boron in SFP for reactivity control is currently authorized for SFP2) and (2) installing fifteen racks (twelve in SFP1 and three in SFP2). The existing twenty low density racks in SFP1 will be removed and disposed of by this modification.

At the completion of the proposed storage expansion, each pool will contain three (3) new Holtec racks and nine (9) Westinghouse racks. The Holtec high density racks have a nominal 10.6 inch by nominal 11.0 inch center to center spacing between cell locations, and are referred to in this submittal as "Region I." The Holtec racks use the neutron absorbing material Boral for reactivity control. All cells in the Region I may be used with no restrictions on burnup or decay times. The Westinghouse high density racks (nine currently installed in SFP2 and nine to be installed in SFP1) have a nominal 9 inch center to center spacing between cell locations, and are referred to in this report as "Region II." In the Region II racks, the spent fuel assemblies will be stored in either a one (1) out of four (4)

configuration, a two (2) out of four (4) configuration, a three (3) out of four (4) configuration, or a four (4) out of four (4) configuration, referred to in this submittal as "Region II (1/4)", "Region II (2/4)", "Region II (3/4)", or "Region II (4/4)" respectively. "Region II" collectively refers to all possible configurations (1/4), (2/4), (3/4), and (4/4). The criticality analysis for Region II (3/4 and 4/4) racks credit soluble boron for reactivity control.

The proposed rack layouts for both SFP1 and SFP2 are almost identical and all the racks will be free standing. Similar to the nine Region II racks currently in SFP2, TXU Electric will modify the nine Region II racks for SFP1 to remove the Boraflex material.

The seventh refueling outage for Unit 1 (Fall 1999) is complete. SFP1 and SFP2 currently contain 913 spent fuel assemblies. After the eighth refueling outage for Unit 1 (spring 2001) TXU Electric anticipates that approximately 1101 spent fuel assemblies will be stored in SFP1 and SFP2. To ensure that refueling activities for the sixth refueling outage for Unit 2 (spring 2002) are not impacted, TXU Electric is requesting approval to use three SFP1 Region I racks, three SFP2 Region I racks, and nine SFP1 Region II racks.

To allow use of the Region I / Region II racks for the storage of fuel, TXU Electric is submitting the proposed Technical Specification changes described in Section 2.

4.0 TECHNICAL ANALYSIS

TXU Electric has evaluated the changes to the Technical Specifications and the proposed storage of fuel in the Region I / Region II racks according to the guidance of the NRC Letter to All Power Reactor Licensees, from B. K. Grimes, April 14, 1978, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," as amended by the NRC Letter dated January 18, 1979 (Reference 5), appropriate NRC Regulatory Guides, appropriate NRC Standard Review Plans, and appropriate Industry Codes and Standards as listed or referenced in the enclosed Licensing Report (Reference 6). In addition, TXU Electric has reviewed several previous applications and NRC Safety Evaluation Reports similar to this proposal.

The analysis below evaluates the use of the Region I / Region II racks (including the limitations on their use) and the increased total storage capacity. The analysis evaluates the following areas: (1) Nuclear Criticality, (2) Thermal-Hydraulic issues, and (3) Mechanical, Material and Structural considerations. In addition, a summary is provided of the impact of the changes on design basis analyses.

Nuclear Criticality (Summary)

The current Technical Specifications (1) allow the use of twenty low density racks in SFP1 with a nominal 16 inch center to center spacing between the fuel assemblies, and (2) allow fuel assemblies to be stored in nine SFP2 Region II racks in either a (1/4) or (2/4) storage configuration. CPSES has received approval (not yet implemented) to credit soluble boron

to allow the storage of additional fuel assemblies in the SFP2 Region II racks in either the (3/4) or (4/4) storage configurations.

The nine SFP1 Region II racks to be installed are almost the same as the existing nine SFP2 Region II racks. The wrapper used to hold the Boraflex was reattached in the SFP2 Region II racks after the Boraflex was removed. It was determined that reattachment of the wrapper was unnecessary for the SFP1 Region II racks. The wrapper material is comprised of stainless steel and acts as a neutron absorber. To account for the removal of the wrapper, the SFP2 Region II (3/4 and 4/4) criticality analysis (References 1(a) and 9) is updated to address the SFP1 Region II (3/4) and SFP1 Region II (4/4) storage configurations. CPSES intends to conservatively apply slightly more restrictive SFP1 Region II (3/4 and 4/4) storage configuration requirements to the SFP2 Region II (3/4 and 4/4). The previously approved criticality analysis for SFP2 Region II (1/4 and 2/4) storage configuration remains valid for the SFP1 Region II (1/4 and 2/4) storage configurations (Reference 1), because the analysis does not credit soluble boron nor the presence of the wrapper.

The criticality analysis for the SPF1 Region II (3/4 and 4/4) storage configurations, taking credit for soluble boron contained in the fuel storage pool water, has been performed using the approved methodology found in Reference 9. In the criticality analysis, storage configurations have been defined to ensure that the fuel storage rack k_{eff} will be less than 1.0, including uncertainties, without the presence of any soluble boron in the fuel storage pool. Soluble boron credit is used to provide safety margin by maintaining k_{eff} less than or equal to 0.95, including uncertainties, in the presence of fuel storage pool soluble boron.

The allowed storage of spent fuel in the SFP1 Region II racks depends on the initial enrichment, discharge burnup, and decay time (decay time is applicable only for the (3/4) and (4/4) storage configurations) of the spent fuel assembly.

Six Region I racks are being installed (three in each SFP) and may be used with no restrictions on burnup, decay times, or storage configurations. A fuel storage pool criticality analysis was performed for the Region I racks taking credit for the neutron absorber Boral (Section 4 of Reference 6). The criticality analysis ensures that the fuel storage rack k_{eff} will be less than or equal to 0.95, including uncertainties. Soluble boron in the fuel storage pool is not credited for the Region I racks.

The Region I (SFP1 and SFP2) and Region II (SFP1) criticality analyses have considered a dropped spent fuel assembly, an inadvertent placement of a fresh fuel assembly of the highest permissible enrichment in a location other than a prescribed location, and pool temperature variations outside of the normal operating range. The criticality acceptance criterion remains satisfied for these events.

The Westinghouse criticality analysis methodology used for the Region II (3/4 and 4/4) storage configurations include a generically determined reactivity bias to account for axial, or three-dimensional, burnup effects. In March, 2000, Duke Power notified the NRC of a potential non-conservatism in the Westinghouse generic axial burnup reactivity bias. A

description of the NRC approved generic axial burnup reactivity bias is described in Reference 9. This generic issue is currently under evaluation by Westinghouse, and should it be determined that there is an impact on the analyses presented in this submittal, TXU Electric will submit revised information as appropriate.

The design of the racks provides a minimum water-gap spacing in the North South direction of 1.63" between Region I and Region II rack modules. Calculations have been made across this interface to determine the reactivity effects due to this inter-module spacing.

Technical Specification 3.7.16 previously established the boron concentration requirements ($\geq$ 2000 ppm) for the fuel storage pool water and remains unchanged.

A deterministic boron dilution analysis was performed to define the dilution times, volumes, and sources for CPSES. It was concluded that an event which would result in the dilution of the fuel storage pool from the proposed Technical Specification Limit of 2000 ppm to the design k_{eff} of 0.95 is not a credible event.

Nuclear Criticality Evaluation

Region I

The Region I racks utilize Boral as a neutron absorber for supplementary reactivity control and are designed to safely accommodate fuel of a maximum 5.0 wt% enrichment (4.95±0.05 wt%).

For normal operation, the Region I racks are designed to assure that the effective neutron multiplication factor (k_{eff}) is equal to or less than 0.95 with the racks fully loaded with fuel of the highest anticipated reactivity and flooded with unborated water at the temperature within the operating range corresponding to the highest reactivity. The maximum calculated reactivity includes margin for uncertainties in reactivity calculations including mechanical tolerances. The uncertainties are statistically combined such that the final k_{eff} will be equal to or less than 0.95 with a 95% probability at a 95% confidence level.

The design basis fuel assembly is the 17x17 Westinghouse OFA assembly containing UO_2 at a maximum initial enrichment of 5.0% U-235 by weight (4.95±0.05 wt%). Other fuel assembly designs were evaluated to assure that fuel of the highest reactivity (bounding case) was used in the analyses.

The criticality safety analysis includes consideration of postulated abnormal and accident conditions to assure that the racks will be safe under all credible conditions. The water in the spent fuel storage pool normally contains soluble boron, which would result in very large subcriticality margins under normal operating conditions. However, the criticality safety evaluation is based upon the assumed accident condition in which all soluble boron is lost. The double contingency principle of ANSI N-16.1-1975 and of the April 1978 NRC position statement allows credit for soluble boron under other abnormal or accident

conditions since only a single accident need be considered at one time. Consequences of abnormal and accident conditions have also been evaluated, primarily the effect of increased moderator temperature (including boiling) and the consequence of fuel mis-placement accidents.

In the fuel Region I rack analyses, the primary criticality analyses of the spent fuel storage Region I racks were performed with the three-dimensional KENO5a multi-group transport theory code, using the 238 group SCALE cross section library and the NITAWL subroutine for U-238 resonance shielding effects (Nordheim integral treatment). Independent verification calculations of the reference case were made with the MCNP code, developed by Los Alamos National Laboratory. Benchmark calculations indicate a bias of 0.0030 with an uncertainty of ±0.0012 for KENO5a (95%/95%) and 0.0009±0.0011 (95%/95%) for MCNP.

The CASMO-4 code was used for evaluating small reactivity increments associated with manufacturing tolerances.

In the geometric model used in the calculations, each fuel rod and its cladding were described explicitly and reflecting boundary conditions (zero neutron current) were used at the centerline of the water gap. These boundary conditions have the effect of creating an infinite radial array of storage cells. Water reflectors were used at the ends of the fuel assemblies to define the actual length of the active fuel.

KENO5a Monte Carlo calculations inherently include a statistical uncertainty due to the random nature of neutron tracking. To assure convergence of the KENO5a calculated reactivity, a minimum of 2,000,000 neutron histories in 1000 generations of 2000 neutrons each are accumulated in each calculation.

For the Region I racks, a reference k_{eff} value of 0.9134 was calculated. 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 reference k_{eff} value. This resulted in a 95/95 k_{eff} of 0.9258. Since this value is < 1.0 and was determined at a 95/95 probability/confidence level, it meets the criterion for precluding criticality with no credit for soluble boron. For the accident scenario of the placement of a fresh fuel assembly outside and adjacent to the Region I rack module, it was determined that 150 ppm soluble boron is required to meet the regulatory guidelines.

Region II

The Region II racks were analyzed using the Westinghouse methodology which has been reviewed and approved by the NRC (Reference 9). This methodology takes partial credit for soluble boron in the fuel storage pool criticality analyses and requires conformance with the following acceptance criteria:

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 Reference 9 (WCAP-14416-NP-A); and

2) k_{eff} shall be ≤ 0.95 if fully flooded with borated water, which includes an allowance for uncertainties at a 95/95 level as described in Reference 9.

The analysis of the reactivity effects of fuel storage in the CPSES Region II 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 Westinghouse 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 nine SFP1 Region II racks to be used are the same as the existing nine SFP2 Region II racks with the exception that the wrapper used to hold the Boraflex was not reattached after the Boraflex was removed. The cell spacing in the Region II racks is a nominal 9-inch center-to-center, and the racks contain no special neutron absorbing material. The SFP1 Region II racks have been qualified for storage of various Westinghouse and Siemens 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 w/o.

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, assembly position, calculational uncertainty, and methodology bias uncertainty were accounted for. These uncertainties were appropriately determined at the 95/95 probability/confidence level. A KENO-Va 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.

For the SFP1 Region II racks, the nominal fresh fuel enrichment required to maintain $k_{eff} <$ 1.0 for an (4/4) storage configuration filled with Westinghouse and Siemens 17x17 fuel assemblies and no soluble boron in the pool water was found to be 1.04 w/o U-235. This resulted in a nominal k_{eff} of 0.96756. 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 0.99574. Since this value is < 1.0 and was determined at a 95/95

probability/confidence level, it meets the criterion for precluding criticality with no credit for soluble boron.

A similar calculation was performed assuming a SFP1 Region II (3/4) storage configuration. The nominal fresh fuel enrichment required to maintain $k_{eff} < 1.0$ for this configuration and no soluble boron was found to be 1.51 w/o U-235. The resulting nominal k_{eff} was 0.97785. The 95/95 k_{eff} was determined to be 0.99811, also meeting the criterion for precluding criticality with no soluble boron.

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 SFP1 Region II (4/4) storage configuration to be 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 0.93531 for fuel enriched to 1.04 w/o U-235 in the SFP1 Region II (4/4) storage configuration. Since k_{eff} is ≤ 0.95 with 200 ppm of boron and uncertainties at a 95/95 probability/confidence level, the acceptance criterion for precluding criticality with credit for soluble boron is satisfied. The concentration of soluble boron required to maintained $k_{eff} \leq 0.95$ is well below the minimum SFP boron concentration value of 2,000 ppm required by proposed TS 3.7.16.

A similar calculation for the SFP1 Region II (3/4) storage configuration under nominal conditions with 200 ppm of soluble boron in the moderator resulted in a k_{eff} of 0.91997. The resulting 95/95 k_{eff} was 0.94061 for fuel enriched to 1.51 w/o U-235, also meeting the acceptance criterion for precluding criticality with credit for soluble boron.

The concept of reactivity equivalencing due to fuel burnup was used to allow storage of fuel assemblies with enrichments higher than 1.04 w/o U-235 for the SFP1 Region II (4/4) storage configuration, and greater than 1.51 w/o U-235 for the SFP1 Region II (3/4) storage configuration. 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 SFP1 Region II (4/4) and (3/4) storage configurations, 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 SFP1 Region II spent fuel storage racks. These are shown in the proposed TS Figures 3.7.17-1 and 3.7.17-2 for the SFP1 Region II (4/4) and the (3/4) storage configurations, 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 delta-k_{eff} 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 amount of additional soluble boron, above the 200 ppm value required above, that is needed to account for these uncertainties is 600 ppm for the SFP1 Region II (4/4) storage configuration and 500 ppm for the SFP1 Region II (3/4)

storage configuration. This results in a total soluble boron credit of 800 ppm for the SFP1 Region II (4/4) storage configuration and 700 ppm for the SFP1 Region II (3/4) storage configuration. These values are well below the minimum SFP boron concentration value of 2,000 ppm required by TS 3.7.16.

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.

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 SFP1 Region II (3/4) storage configuration results in the highest reactivity increase. The reactivity increase requires an additional 1,200 ppm of soluble boron to maintain $k_{eff} \leq 0.95$. However, for such events, the double contingency principle of Reference 8 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 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 the SFP1 Region II storage configuration zones (i.e., boundary between: (4/4) to (3/4) and (2/4) storage configurations, (2/4) to (3/4) storage configuration, (1/4) to (4/4) and (3/4) storage configurations, and (2/4) to (1/4) storage configuration) must be controlled to prevent an undesirable increase in reactivity. The fuel storage patterns must comply with the interface requirements shown in Figures 5, 6, 7 and 8 of Westinghouse Report CAB-00-163, "Comanche Peak High Density Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit and No Outer Wrapper Plates," submitted as Enclosure 2 to this letter.

The SFP1 Region II analyses assumed credit for soluble boron, as permitted by Reference 9, 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.

The following summarizes the storage configurations and U-235 enrichment limits from the SFP1 Region II criticality analysis for Westinghouse and Siemens 17x17 fuel assemblies.

SFP1 Region II (4/4) Storage Configuration

Fresh assemblies with initial nominal enrichments no greater than 1.04 w/o U-235 can be stored in any cell location. Fuel assemblies with initial nominal enrichments greater than 1.04 and up to 5.00 w/o U-235 must satisfy the minimum burnup and decay time requirements shown in proposed TS Figure 3.7.17-1. The SFP1 Region II (4/4) analysis bounds the SFP2 Region II (4/4) configuration, therefore the proposed Figure TS 3.7.17-1 is applicable for both spent fuel pools.

SFP1 Region II (3/4) Storage Configuration

Assemblies with initial nominal enrichments no greater than 1.51 w/o U-235 can be stored in a (3/4) storage configuration arrangement with empty cells. Fuel assemblies with initial nominal enrichments greater than 1.51 and up to 5.00 w/o U-235 must satisfy the minimum burnup and decay time requirements shown in proposed TS Figure 3.7.17-2. The SFP1 Region II (3/4) analysis bounds the SFP2 Region II (3/4) configuration, therefore the proposed TS Figure 3.7.17-2 is applicable for both spent fuel pools.

TABLE 1

Summary of Soluble Boron Credit Requirements for SFP1 Region II

SFP1 Region II 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)
(4/4) storage configuration	200	600	800
(3/4) storage configuration	200	500	700

Region I and Region II Interface

The calculational model assumed a 7 x 10 array of each region separated by the 1.63" watergap. Region I and Region II have different design criteria (k-effective values). Consequently the interface calculational model should not result in a k-effective value greater than that of either region alone.

The data indicates that there is no significant interaction between Region I and Region II except for minor coupling between Region I and the Region II (1/4) storage configuration if the fresh fuel assembly in Region 2A is in the row facing Region I. Therefore, it is required that the Region II 1 out of 4 configuration be oriented such that the single fuel assembly resides in the internal row with the empty cells facing Region I. There are no interface restrictions between the Region II (2/4, 3/4, 4/4) and Region I configurations.

Thermal-Hydraulic (Summary)

The additional usage of the Region I / Region II racks and the added storage capacity increase the heat load in the spent fuel pools during normal operating and abnormal conditions. Evaluation has shown that the Spent Fuel Cooling system will keep the bulk water temperature within the normal maximum of 150°F. Also, the HVAC system will maintain the ability to remove the heat generated by the added storage capacity. The maximum pool temperature will be within the existing design basis for the spent fuel racks, spent fuel pool liner, cooling system, spent fuel pool demineralizers, and the fuel cladding. An analysis has been performed to assure compliance with the ACI 318-71 Building Code Requirements for Reinforced Concrete (Reference 7).

Thermal-Hydraulic: Spent Fuel Pool Cooling and Cleanup System

The SFP cooling and cleanup system, described in CPSES FSAR (Reference 10), Section 9.1.3, is designed to remove decay heat from the spent fuel and maintain water clarity and purity. The SFP cooling system's ability to cool the SFPs was evaluated to verify that decay heat can be removed during operating conditions and that acceptable SFP temperatures are maintained. Spent fuel decay heat calculations were performed in accordance with NRC Branch Technical Position ASB 9-2, Residual Decay Energy for Light Water Reactors for Long Term Cooling (Reference 2).

In accordance with the FSAR for CPSES, the normal design temperature of the bulk SFP water is less than 150°F for normal operation (and less than 200°F in the event of a single failure in the spent fuel pool cooling system). The normal design temperature of the SFP heat exchanger is 140°F to protect the resins in the cleanup system.

In accordance with the FSAR for CPSES, the abnormal maximum design condition, assuming both trains of the SFP cooling system are in operation, prevents the onset of bulk boiling and is therefore acceptable.

Because CPSES has two shared spent fuel pools, the assumed design conditions of Section III.1.h of SRP 9.1.3 (Reference 4) do not apply. The CPSES design allows for back-to-back refuelings with sharing of the spent fuel storage which is not accounted for in SRP 9.1.3.

The decay heat bounding analysis performed to support License Amendment Request 94-22 (Reference 1(a)] was based on a core thermal power of 3411 megawatt thermal power (Mwt) for Unit 1 and Unit 2. The decay heat bounding analysis has been updated to consider the

effect of increasing the core thermal power of 3411 MWt by 4.5% (i.e., 3565 MWt) for Unit 1 and Unit 2. Therefore the decay heat of the previous bounding analyses (License Amendment 46/32, based on an assumed total capacity of 3386 spent fuel assemblies) is increased slightly. The conclusions of the previous bounding analyses (e.g., criticality, decay heat, thermal-hydraulic, structural [concrete temperature], total heat rejected to the environment) has been evaluated and determined to remain acceptable. References in this license amendment request to the bounding decay heat analysis refer to this updated analysis (i.e., maximum calculated core thermal power of 3565 MWt and an ultimate storage capacity of 3386 spent fuel assemblies). The current licensed core thermal power is 3411 MWt for Unit 1 and 3445 Mwt (an increase of 1.0% above 3411 MWt) for Unit 2.

The resulting increase in decay heat for the proposed increase in spent fuel storage capacity from 2,026 to 3,373 spent fuel assemblies will be within the design capability of the CPSES Spent Fuel Pool Cooling and Cleanup System for normal and abnormal core offload conditions. Accordingly, the conclusions with regard to conformance with SRP, Section 9.1.3 (Reference 4) are still applicable with regard to the proposed increase in SFP storage capacity.

Thermal Hydraulic: Impact on Support Systems

The impact on support systems (SFP Makeup Water System, Component Cooling Water System, Service Water System, SFP Cleanup System, and HVAC System) remain acceptable.

The minimum time to boiling if cooling to one or both SFPs were lost was evaluated. In the unlikely event of a failure in the non-seismic portion of the cooling system, there is sufficient time to detect the loss of cooling and to restore at least one train of the two trains of safety related cooling prior to boiling. Makeup water sources for loss of cooling is provided from the seismic Category I Reactor Makeup Water Storage Tank for seismic events and from the plant fire suppression hose stations for events which could result in extended pool boiling. The fire suppression system makeup is provided from the Safe Shutdown Impoundment. Therefore, the SFP makeup water sources are sufficient to restore and maintain the water levels in the SFPs for the duration of a loss of cooling to the pools.

The SFP heat loads for various plant operating modes under the previously discussed conditions will be revised along with total heat loads of Component Cooling Water (CCW) heat exchanger and associated CCW temperatures from previous modifications. Comanche Peak's corrective action program has identified that the impact of spent fuel pool heat loads and heat loads of previous modifications were not properly maintained current in the CCW thermal loads analysis. The analyses will be updated prior to implementation of this License Amendment Request (LAR). The CCW System will continue to perform its intended safety functions. This LAR will be supplemented should these conclusions change.

The Station Service Water (SSW) System removes the additional SFP heat loads rejected by the CCW heat exchangers to the Safe Shutdown Impoundment (SSI), which is the

plant's ultimate heat sink. The increase in heat loads, due to the bounding spent fuel pool decay heat analysis over those used in the previous ultimate heat sink hydrothermal analyses, represents less than 1% of the total station heat input which affects the peak SSI temperature. This is smaller than the conservatism's and uncertainties in the analyses. Even with this additional heat load, the calculated peak SSI temperature increase would be insignificant. Therefore, the SSW System and the ultimate heat sink will continue to perform their intended safety functions. Comanche Peak's corrective action program has identified that the change in various plant heat loads were not properly considered (collectively) in the ultimate heat sink analysis. This issue does not involve the Spent Fuel Pool Cooling System's ability to reject heat to the SSI. The analysis will be updated prior to implementation of License Amendment 74 (Credit for Boron). The ultimate heat sink will continue to perform its intended safety functions. This LAR will be supplemented should these conclusions change.

The impact of the use of the Region I and Region II racks on the CPSES SFP cleanup system was evaluated. The system is designed to respond to varying water purity levels ranging from equilibrium conditions of undisturbed fuel storage to transient, relatively high impurity conditions due to refueling activities. In order to protect the resins in the demineralizers, the maximum normal temperature of the water to the purification loops is 140°F for either one or two pump operation. The increased spent fuel storage capacity does not affect the design basis or functional requirements of the cleanup system.

Short term increases in the SFP radiological source term are dependent on crud transport, fission product release, or system cross-flow resulting from refueling operations. The frequency of refueling is determined by operational considerations and fuel design. Since these circumstances are essentially independent of the quantity of spent fuel stored, the cleanup system is negligibly impacted by the increased spent fuel storage capability.

The SFP area ventilation system, described in CPSES FSAR (Reference 10), Section 9.4.2, is designed to maintain suitable environmental conditions (temperature, humidity, and radiation level) for plant personnel and equipment under normal conditions. An evaluation of the ability of the Fuel Building HVAC to maintain air temperatures within the FSAR limits determined that the increase in SFP heat loads due to the proposed modification will not affect the Fuel Building HVAC system during normal operation.

Thermal-Hydraulic: Local Water and Fuel Cladding Temperatures

Under normal conditions, there will be no local boiling in the Region I / Region II rack cells nor is the fuel clad temperature hot enough to allow nucleate boiling on the fuel clad surface.

For the abnormal condition, the bulk SFP water is boiling. The maximum local water temperature exceeds the local boiling temperature at the top of the spent fuel storage racks, so there is boiling in the Region I / Region II rack cells. Evaluations have concluded that the cladding will be not be subjected to departure from nucleate boiling (DNB) and the cladding integrity will be maintained.

Mechanical, Material and Structural

The main safety function of the spent fuel pool and the Region I / Region II racks is to maintain the fuel assemblies in a safe configuration (e.g., decay heat removal, shielding) through normal and abnormal operating conditions. The mechanical, material and structural considerations of the proposed Region I / Region II racks are described or referenced in the CPSES Fuel Storage Licensing Report (Reference 6). The Region I / Region II rack materials are compatible with the spent fuel pool and the fuel assemblies. The Region I / Region II racks have a sufficient margin of safety against tilting and deflection or movement during a seismic event. The Region I / Region II racks do not impact each other or the pool walls, damage fuel assemblies, or cause criticality concerns during a postulated seismic event.

The structural evaluation assured the acceptability of the structural integrity of the racks, the stored fuel assemblies and the SFP liner and structure subject to the effects of the postulated loads (Section 2, 6, 7, and 8 of Reference 6) and fuel handling accidents with regard to the proposed increase in the number of fuel assemblies to be stored in the SFP1 and SFP2.

The spent fuel storage Region I / Region II racks are seismic Category I equipment, and are required to retain structural integrity during and after a safe shutdown earthquake (SSE). The computer program, DYNARACK, was used 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 spent fuel storage Region I / Region II racks are free-standing and self-supporting equipment, and they are not attached to the floor or walls of SFP1 or SFP2. Nonlinear dynamic models consisting of inertial masses, flexural elements, linear and non-linear 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 various nodes, and then computed the stresses in the rack elements from the nodal displacements.

Analyses of two models were performed: a 3-D single rack (SR) model and a 3-D whole pool multi-rack (WPMR) model. Several configurations were used for the 3-D SR analyses. The Region I / Region II rack was considered to be fully loaded and partially loaded with two bounding coefficients of friction between the rack pedestal and the pool floor to investigate the stability of the rack with respect to overturning. For the 3-D WPMR analyses, all Region I / Region II 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 FSAR, Section 3.7B, "Seismic Design" (Reference 10). The adequacy of the single artificial time history set used for the seismic analyses was demonstrated 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 WPMR analyses, the Region I / Region II racks were evaluated 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 there is an adequate factor of safety 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 nor do the racks slide enough to invalidate the gamma heating calculations performed for the pool concrete 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 are less than the corresponding allowable stresses specified in the ASME Code, indicating that the rack design is adequate.

The Region I / Region II rack weld stresses at the connections (e.g., baseplate-to-rack, baseplate-to-pedestal, and cell-to-cell connections) were calculated under the dynamic loading conditions. All of the calculated weld stresses are less than the corresponding allowable stresses specified in the ASME Code, indicating that the weld connection design of the rack is adequate.

Based on the comprehensive parametric study (e.g., varying coefficients of friction and fuel loading conditions of the rack) and adequate factors of safety, it is concluded that the spent fuel storage Region I / Region II rack modules will perform their safety function and maintain their structural integrity under postulated loading conditions. Therefore, the Region I / Region II racks are acceptable when loaded with the maximum proposed allowable number of fuel assemblies.

The spent fuel pool structure was analyzed to demonstrate its adequacy under conditions when fully loaded with Region I / Region II racks with all storage locations occupied by fuel assemblies. The fully-loaded structures were subjected to the load combinations specified in the CPSES FSAR.

The CPSES FSAR shows the predicted factors of safety varying from 1.30 to 2.28 for shear force and bending moment of the concrete walls and slab. In view of the calculated factors of safety, the structural analyses demonstrate the adequacy and integrity of the structures under full fuel loading, thermal loading, and SSE loading conditions. Thus, the spent fuel pool design is acceptable when loaded with the maximum allowable number of fuel assemblies.

The following three refueling accident cases were evaluated: (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 analyses results submitted in Reference 6 are acceptable in that they are supported by the parametric studies.

In conclusion, the structural analysis and design of the spent fuel rack modules and the spent fuel pool 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 FSAR 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 SFP1 and SFP2.

Design Basis Events

This change does not affect the existing design basis events regarding the drop of a fuel cask and a tornado. Impacts of this change on the following potential design basis events were evaluated:

1) a dropped spent fuel assembly;
2) a loss of spent fuel pool cooling;
3) a seismic event;
4) a fuel assembly inadvertently placed in a location other than a prescribed location; and
5) a stuck fuel assembly and the associated uplift force.

(1) a dropped spent fuel assembly:

The increase in spent fuel storage capacity will not increase the probability of a dropped spent fuel assembly. The Region I / Region II racks are designed such that a dropped fuel assembly cannot occupy a position in the Region I / Region II racks other than a normal fuel storage location. The drop of an assembly on top or along side of a rack or in a location other than a prescribed location is addressed below in paragraph number (4).

The radiological consequences of a fuel handling accident are not changed from that described in Final Safety Analysis Report Section 15.7.4. Only the dropped assembly is damaged; the Region I / Region II racks can accommodate the drop without any significant deformation that would affect other stored assemblies. Thus, the consequences of this type of accident will not be increased from previously

evaluated spent fuel assembly drops.

(2) a loss of spent fuel pool cooling:

For the Region I / Region II racks, the criticality acceptance criterion is not violated. Two redundant safety related cooling loops are provided, each capable of simultaneously servicing both of the station spent fuel pools.

In the event of a postulated complete loss of spent fuel pool cooling system capability, a number of available sources of spent fuel pool makeup water could be successfully used to replace losses due to boiling in order to maintain the spent fuel pool (SFP) water level and thereby ensure adequate spent fuel pool cooling.

(3) a seismic event:

The Region I / Region II racks have no connection with the pool walls or with each other. Time histories whose response spectra envelope the design floor response spectra at the pool floor level were used as input to the dynamic analysis of the racks. Fluid coupling was also considered as described in the enclosed CPSES Fuel Storage Licensing Report (Reference 6).

The Region I / Region II racks are designed and fabricated to meet the requirements of applicable portions of NRC Regulatory Guides and Industry Standards. During a seismic event, the Region I / Region II racks will not tip over, or collide against each other or against the pool walls.

The interaction between the fuel assemblies and the Region I / Region II racks has been considered, including variations in the gaps between both fuel assemblies and cell walls and between racks. The resulting impact loads are not capable of causing structural damage to the fuel assemblies.

Seismic analyses of the Fuel Building were revised to incorporate the increase in mass associated with the Region I / Region II racks (filled with spent fuel assemblies). The revised analyses are in accordance with the criteria and methodology described in Section 3.7 of the FSAR. The seismic analyses resulted in revised responses for the Fuel Building. These revised building responses were used to design the Region I / Region II racks. Comanche Peak's corrective action program has identified that the effect of the revised building responses on the balance of Fuel Building structures, systems, and components was not properly considered. The analyses will be updated prior to implementation of this modification. It is anticipated that the design of the Fuel Building structures, systems, and components will remain adequate and that any necessary modifications to the balance of the Fuel Building (none are expected) would be completed prior to implementation of the proposed reracking. This LAR will be supplemented should these conclusions change.

Seismic validation analyses of the Fuel Building, incorporating the increase in mass associated with the Region I / Region II racks, were performed to verify that the revised responses of the Fuel Building were adequate and conservative. The validation analyses were performed using the similar criteria and methodology as the previous reviewed validation analyses used to validate the original plant building responses as described in Reference 11.

Therefore, the consequences of a postulated seismic event will not increase from previously evaluated events.

(4) a fuel assembly (either a fresh unirradiated or a partially depleted fuel assembly) inadvertently placed in a location other than a prescribed location:

The design of the Region I / Region II racks is such that it is possible to insert fuel assemblies in locations other than prescribed storage locations. A fuel assembly could be (1) inadvertently dropped and come to rest horizontally on top of the racks, (2) inadvertently dropped and come to rest vertically on top of the racks, or (3) misloaded within the racks or adjacent to but outside the racks. Credit for borated water in accident situations is justified by the double contingency principle of ANSI/ANS-8.1-1983 (Reference 8). The amount of soluble boron required to offset each of these postulated accidents was evaluated for all possible storage configurations and the limiting value was determined to be 1900 ppm to maintain a k_{eff} less than or equal to 0.95. Thus, in all cases, the design of the Region I / Region II racks ensures that the multiplication factor is less than or equal to the 0.95 limit, including uncertainties.

(5) The uplift force evaluation shows that the racks are able to withstand the vertical uplift force equal to the maximum lifting capacity of the fuel handling bridge crane. The evaluation determined that the damaged region caused by these forces is well above the top edge of the neutron absorber material (active fuel region). For the load applied vertically anywhere along a cell wall, the resultant stress is well below the yield stress of the material.

Boron Dilution Analysis

A boron dilution analysis has been performed for crediting boron in CPSES SFP1 and SFP2 criticality analysis for Region II. The boron dilution analysis is in accordance with NRC-approved Westinghouse Owners Group generic methodology for crediting soluble boron as described in Reference 9. 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. The boron dilution analysis included an evaluation of the following plant specific features:

- Dilution Sources and Flowrates including Piping Failures

- Boration Sources
- Instrumentation
- Administrative Procedures
- Loss of Offsite Power Impact
- Boron Dilution Initiating Events
- Boron Dilution Times and Volumes

The boron dilution analysis was performed to ensure that sufficient time is available to detect and mitigate the dilution before the spent fuel criticality analysis 0.95 k_{eff} design basis is exceeded. The minimum boron concentration requirement to maintain a k_{eff} less than or equal to 0.95 determined from the various analyses is 800 ppm when no fuel handling accident is considered. The most conservative of the minimum soluble boron concentrations from the analyses was chosen as the basis for the minimum boron concentration of the Spent Fuel Pool.

As a result of this SFP boron dilution analysis, it is concluded that an event which would result in the dilution of the SFP boron concentration from the Technical Specification limit of 2000 ppm to 800 ppm is not a credible event. This conclusion is based on the following:

1. In order to dilute the SFP to the design k_{eff} of 0.95, a substantial amount of water (greater than 259,500 gallons) is needed.

2. Since such a large water volume turnover is required, a SFP dilution event would be readily detected by plant personnel via alarms, flooding in the fuel and auxiliary buildings or by normal operator rounds through the SFP area.

3. Evaluations indicate that, based on the flow rates of non-borated water normally available to the SFP, taken in conjunction with significant operator errors and equipment failures, sufficient time is available to detect and respond to a dilution event.

In addition, there is significant conservatism built into this evaluation such as the cooling of the spent fuel pools can be performed by one train supplying common water to both pools. This cooling configuration would allow credit to be taken for the volume of both pools and therefore substantially increase the dilution time estimates. However, because the flexibility exists for the cooling system to be totally dedicated to one pool, only one pool volume is considered in this evaluation.

Deterministic dilution event calculations were performed for the SFPs to define the dilution times and volumes necessary to dilute a SFP from an initial boron concentration of 1900 parts per million (ppm) to a soluble boron concentration of 800 ppm. TS 3.7.16, "Fuel Pool Boron Concentration," requires that the SFP boron concentration be greater than or equal to 2,000 ppm. Currently, CPSES maintains a boron concentration of 2,400 ppm in the SFPs. CPSES conservatively chose a boron concentration of 1900 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 800 ppm. As part of the CPSES criticality analysis, a calculation has been performed on a 95/95 basis to show that the spent fuel rack k_{eff} remains less than 1.0 with nonborated water in the pool. Thus, even if the SFP were diluted to concentrations approaching zero ppm, the fuel in the racks would remain subcritical and the health and safety of the public would be protected.

Each SFP has a water inventory of 300,000 gallons. Assuming a well-mixed SFP, the volume required to dilute the SFP from 1900 ppm to 800 ppm is approximately 259,500 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. All pipes in the vicinity of the SFP are seismically qualified and supported. As such, a random pipe break was not considered in the analyses. The guidance of Reference 4, Branch Technical Position, Mechanical Engineering Branch 3-1 (MEB 3-1), for selection of the potential pipe cracks, was followed.

The analysis concludes that an unplanned or inadvertent event that would dilute the SFP boron concentration from 1,900 ppm to 800 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 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 (800 ppm). Therefore, the analysis and TS controls are acceptable for the boron dilution aspects of this submittal.

The TS boron concentration of 2,000 ppm or greater, and a seven-day surveillance requirement, both specified in TS 3.7.16, ensure that sufficient time is available to detect and mitigate a dilution event prior to exceeding the design basis k_{eff} of 0.95.

Summary

TXU Electric 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.

5.0 REGULATORY ANALYSIS

5.1 No significant Hazards Determination

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

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

   Response: No

   This proposed license amendment includes changes which provide the criteria for acceptable fuel storage in Region I / Region II racks. The revised criteria for acceptable fuel storage in the Region I / Region II racks are discussed below.

   The Region I / Region II racks proposed for Spent Fuel Pool One (SFP1) and Spent Fuel Pool Two (SFP2), are a nominal 10.6 x 11 inch and nominal 9 x 9 inch center to center spacing respectively. The SFP1 Region II racks are similar to the existing Region II racks in SFP2 (nominal 9 x 9 inch center to center). The proposed Region I / Region II racks and the existing Region II racks in SFP2 are free standing whereas the low density racks being removed from SFP1 are bolted to the pool. Administrative controls are used to maintain the specified storage patterns and to assure storage of a fuel assembly in a proper location based on initial U-235 enrichment, burnup, and decay time. The increased storage capacity results in added weight in the pools and additional heat loads.

   There is no significant increase in the probability of an accident concerning the potential insertion of a fuel assembly in an incorrect location in the Region I / Region II racks. TXU Electric has used administrative controls to move fuel assemblies from location to location since the initial receipt of fuel on site. Fuel assembly placement will continue to be controlled pursuant to approved fuel handling procedures and will be in accordance with the Technical Specification spent fuel rack storage configuration limitations.

   There is no increase in the probability of the loss of normal cooling to the fuel storage pool water due to the presence of soluble boron in the pool water for subcriticality control. A concentration of soluble boron similar to that currently approved (Technical Specification 3.7.16) has always been maintained in the fuel storage pool water. The amount of soluble boron required to offset the reactivity increase associated with water temperature outside the normal range was established for the proposed storage

configurations.

The consequences of all of these changes have been assessed and the current acceptance criteria in the licensing basis of CPSES will continue to be met. The nuclear criticality, thermal-hydraulic, mechanical, material and structural designs will accommodate these changes. Potentially affected analyses, including a dropped spent fuel assembly, a loss of spent fuel pool cooling, a seismic event, a fuel assembly placed in a location other than a prescribed location, and a stuck fuel assembly and the associated uplift force continue to satisfy the CPSES licensing basis acceptance criteria. The analysis methods used by TXU Electric are consistent with methods used by TXU Electric in the past or methods used elsewhere in the industry and accepted by the NRC.

Based on the acceptability of the methodology used and compliance with the current CPSES licensing basis, use of the Region I / Region II racks and the increase in storage capacity do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No

The potential for criticality in the spent fuel pool is not a new or different type of accident. The potential criticality accidents have been reanalyzed to demonstrate that the pool remains subcritical.

Soluble boron has been maintained in the fuel storage pool water since its initial operation. The possibility of a fuel storage pool dilution is not affected by the proposed change to the Technical specifications. Therefore, extending the Technical Specification controls for the soluble boron to include the Region II racks in SFP1 will not create the possibility of a new or different kind of accidental pool dilution.

With credit for soluble boron now a major factor in controlling subcriticality for the Region II racks in SFP1 (with no neutron absorber installed), the evaluation of fuel storage pool dilution events previously performed was updated. The results of the updated evaluation concluded that an event which would result in a reduction of the criticality margin below the 5% margin recommended by the NRC is not credible. In addition, the no soluble boron 95/95 criticality analysis assures that a boron concentration of zero ppm will not result in criticality.

The proposed changes which ensure the maintenance of the fuel storage pool boron concentration and storage configuration, do not represent new concepts. The actual boron concentration in the fuel storage pool is currently maintained at 2,400 ppm for SFP1 and SFP2 for refueling purposes. The criticality analysis determined that a boron concentration of 800 ppm (non-accident) and 1,900 ppm (accident) results in a $k_{eff} \leq 0.95$.

For the Region I racks, credit is taken in the reactivity control analysis for the neutron absorber Boral (soluble boron is not credited). The criticality evaluation concluded that the requirement of $k_{eff} \leq 0.95$ when fully flooded with unborated water, including uncertainties, remain satisfied.

There is no significant change in plant configuration, equipment design, or usage of plant equipment. The safety analysis for boron dilution has been performed; however, the criticality analyses assure that the pool will remain subcritical with no credit for soluble boron. Therefore, the proposed changes will not create the possibility of a new or different kind of accident.

The installation and removal of racks meet the requirements of NUREG 0612, "Control of Heavy Loads at Nuclear Power Plants," and current CPSES Technical Requirement 13.9.34, "Refueling – Crane Travel – Spent Fuel Storage Areas."

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

Response: No

The NRC guidance has established that an evaluation of margin of safety should address the following areas:

1) Nuclear criticality considerations
2) Thermal-Hydraulic considerations
3) Mechanical, material and structural consideration

Proposed Technical Specifications 3.7.17 and 4.3 and the associated fuel storage requirements will provide adequate margin to assure that the fuel storage array (Region I and Region II) will always remain subcritical by the 5% margin recommended by the NRC.

While the criticality analysis for Region II utilized credit for soluble boron,

the storage configurations have been defined using k_{eff} calculations to ensure that the spent fuel rack k_{eff} will be less than 1.0 with no soluble boron. The criticality analysis for Region I utilized credit for the neutron absorber material Boral, the storage configurations have been defined using k_{eff} calculations to ensure that the spent fuel rack k_{eff} will be less than or equal to 0.95 with no soluble boron.

Soluble boron credit is used to offset off-normal conditions (such as a misplaced assembly) and to provide subcritical margin such that the fuel storage pool k_{eff} is maintained less than or equal to 0.95.

The loss of substantial amount of soluble boron from the spent fuel pools which could lead to exceeding a k_{eff} of 0.95 has been evaluated and shown not to be credible. These evaluations show that the dilution of the spent fuel pools boron concentration from 1,900 ppm to 800 ppm is not credible and that the Region II spent fuel rack k_{eff} will remain less than 1.0 when flooded with unborated water.

The thermal-hydraulic evaluation of spent fuel pool cooling demonstrates that the temperature margin of safety will be maintained. Evaluation of the spent fuel pool cooling system for the increased heat loads shows that the spent fuel cooling system will maintain the temperature of the bulk spent fuel pool water within the limits of the existing licensing basis. Additionally, it shows that the maximum temperature will be within the existing design temperatures for the Region I / Region II racks, liner, structure, and cooling system and will not have any significant impact on the spent fuel pool demineralizers. Thus, the existing licensing basis remains valid, and there is no significant reduction in the margin of safety for the thermal-hydraulic design or spent fuel cooling.

The main safety function of the spent fuel pool and the Region I / Region II racks is to maintain the spent fuel assemblies in a safe configuration through normal and abnormal operating conditions. The design basis floor responses of the Fuel Building were confirmed to be adequate and conservative and the floor loading will not exceed the capacity of the Fuel Building. The structural considerations of the Region I / Region II racks maintain margin of safety against tilting and deflection or movement, such that the Region I / Region II racks do not impact each other or the pool walls, damage spent fuel assemblies, or cause criticality concerns. Thus, the margin of safety with respect to mechanical, material or structural considerations is not significantly reduced by the use of the Region I / Region II racks.

Therefore the proposed change does not involve a 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

Criterion 2 of 10 CFR 50.36(c)(2)(ii), "A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria: ..... A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier."

10 CFR 50, Appendix A, General Design Criterion (GDC) 2 - Design Bases for Protection Against Natural Phenomena, "Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions to be performed."

GDC 4 - Environmental and Dynamic Effects Design Bases, "Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of- coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with design basis for the piping."

GDC 5 - Sharing of Structures, Systems, and Components, "Structures, systems, and components important to safety shall not be shared between nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions including, in the event of an accident in one unit, an

orderly shutdown and cooldown of the remaining unit."

GDC 61 - Fuel Storage and Handling and Radioactivity Control, "The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions."

GDC 62 - Prevention of Criticality in Fuel Storage and Handling, "Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations."

RG 1.13 Spent Fuel Storage Facility Design Basis, Rev.1, December 1975

RG 1.29 Seismic Design Classifications, Rev.2, February 1976

RG 1.44 Control of the Use of Sensitized Stainless Steel, Rev.0, May 1973

RG 1.60 Design Response Spectra for Seismic Design of Nuclear Power Plants, Rev. 1, December 1973

RG 1.61 Damping Values for Seismic Design of Nuclear Power Plants, Rev.0, October 1973

RG 1.92 Combining Modal Responses and Spatial Components in Seismic Response Analysis, Rev. 1, February 1976

RG 1.124 Service Limits and Loading Combinations for Class I Linear-Type Component Supports, Rev.1, January 1978

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," July 1980

NUREG-0800, Standard Review Plan 3.7, "Seismic Design"

NUREG-0800, Standard Review Plan 3.8.4, "Other Category I Structures"

NUREG-0800, Standard Review Plan 9.1.2, "Spent Fuel Storage": "Nuclear reactor plants include storage facilities for the wet storage of spent fuel assemblies. The safety function of the spent fuel pool and storage racks is to maintain the spent fuel assemblies in a safe and subcritical array during all credible storage conditions and to

provide a safe means of loading the assemblies into shipping casks."

NUREG-0800, Standard Review Plan 9.1.3, "Spent Fuel Pool Cooling and Cleanup System": "... the spent fuel assemblies must be cooled and must remain covered with water during all storage conditions. Other functions performed by the system, not related to safety, include water cleanup for the spent fuel pool, refueling canal, refueling water storage tank and other equipment storage pools; means for filling and draining the refueling canal and other storage pools; and surface skimming to provide clear water in the storage pool." Because CPSES has two shared spent fuel pools, the assumed design conditions of Section III.1.h of SRP 9.1.3 do not apply. The CPSES design allows for back-to-back refuelings with sharing of the spent fuel storage, which is not accounted for in SRP 9.1.3.

Branch Technical Position ABS 9-2, "Residual Decay Energy for Light-Water Reactors for Long Term Cooling," Rev. 2

Analysis

The proposed increase in the capacity of the spent fuel storage pools does not change the compliance with the above general design criteria. The changes described in the License Amendment Request are also consistent with the above Standard Review Plans and Regulatory Guides.

Conclusion

The technical analysis performed by TXU Electric in Section 4, "Technical Analysis," demonstrates the ability of the spent fuel pools and associated support systems to perform their safety function. The changes to increase the spent fuel storage capacity continued compliance with the above regulatory requirements.

6.0 ENVIRONMENTAL EVALUATION

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 (1) a significant hazards consideration, (2) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (3) 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 the proposed change is not required.

7.0. REFERENCES

1. Related TXU Electric TXX Letters and NRC Safety Evaluation for previous License Amendment 46 / 32.

   a) TXU Electric letter logged TXX-94325, from C. L. Terry to the NRC, dated December 30, 1994.
   b) TXU Electric letter logged TXX-95187, from C. L. Terry to the NRC, dated July 28, 1995.
   c) TXU Electric letter logged TXX-95235, from C. L. Terry to the NRC, dated September 14, 1995.
   d) TXU Electric letter logged TXX-95299, from C. L. Terry to the NRC, dated November 29, 1995.
   e) TXU Electric letter logged TXX-96003, from C. L. Terry to the NRC, dated January 2, 1996.
   f) NRC letter from Timothy J. Polich to C. Lance Terry, dated February 9, 1996 (Comanche Peak Steam Electric Station, Units 1 and 2 - License Amendment Nos. 46 and 32 to Facility Operating License Nos. NPF-87 and NPF-89, TAC NOS. M91244 and M91245).
   g) TXU Electric letter logged TXX-96053, from C. L. Terry to the NRC, dated February 23, 1996.

2. NRC Branch Technical Position ASB 9-2, Residual Decay Energy for Light Water Reactors for Long Term Cooling. (Rev.2).

3. Related TXU Electric TXX Letters and NRC Safety Evaluation for previous License Amendments 74 and 78.

   a) TXU Electric letter logged TXX-99001, from C. L. Terry to the NRC, dated February 11, 1999.
   b) TXU Electric letter logged TXX-99158, from C. L. Terry to the NRC, dated September 3, 1999.
   c) TXU Electric letter logged TXX-99247, from C. L. Terry to the NRC, dated December 20, 1999.
   d) NRC letter from David H. Jaffe to C. Lance Terry, dated February 24, 2000, (Comanche Peak Steam Electric Station, Units 1 and 2 - License Amendment No. 74 to Facility Operating License Nos. NPF-87 and NPF-89, TAC NOS. MA4841 and MA4842).
   e) TXU Electric letter logged TXX-00114, from C. L. Terry to the NRC, dated May 30, 1999.

   f) NRC letter from David H. Jaffe to C. Lance Terry, dated June 12, 2000, (Comanche Peak Steam Electric Station, Units 1 and 2 - License Amendment No. 78 to Facility Operating License Nos. NPF-87 and NPF-89, TAC NOS. MA7188 and MA7189).

4. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plant," Branch Technical Position, Mechanical Engineering 3-1 (MEB 3-1)

5. Nuclear Regulatory Commission, Letter to All Power Reactor Licensees, from B. K. Grimes, April 14, 1978, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," as amended by the NRC Letter dated January 18, 1979.

6. Licensing Report for Spent Fuel Rack Installation at Comanche Peak Steam Electric Station (Enclosure 1 to TXX-00144).

7. ACI 318 Building Code Requirements for Reinforced Concrete, 1971.

8. American Nuclear Society, "American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," ANSI/ANS-8.1-1983, October 7, 1983.

9. Westinghouse WCAP 14416-NP-A, Revision 1, November 1996, Westinghouse Spent Fuel Rack Criticality Analysis Methodology.

10. Comanche Peak Steam Electric Station Final Safety Analysis Report, Docket Nos. 50-445 and 50-446.

11. NUREG-0797, Supplement No. 22, Safety Evaluation Report related to the operation of Comanche Peak Steam Electric Station, Units 1 and 2, Docket Nos. 50-445 and 50-446, January 1990.

8.0. PRECEDENTS

The overall Holtec design of the CPSES Region I racks is similar to those presently in service at many other Nuclear Power Plants. Among these plants are Byron Nuclear Station of Commonwealth Edison, Millstone Unit 3 of Northeast Utilities, and Waterford Unit 3 of Entergy Operations. Altogether, there are thousands of storage cells of this design that have been provided by Holtec International to various nuclear plants around the world. The overall design of the CPSES SFP1 Region II racks is the same as those presently in service in the CPSES SFP2 Region II.

ATTACHMENT 3 to TXX-00144

MARKUP OF TECHNICAL SPECIFICATION PAGE

Pages	**3.7-36**
	3.7-37
	3.7-38
	3.7-39
	3.7-40
	3.7-41
	4.0-2
	4.0-3

3.7 PLANT SYSTEMS

3.7.17 Spent Fuel Assembly Storage

Region II

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.

Region II

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 Region II

Replace with new figure

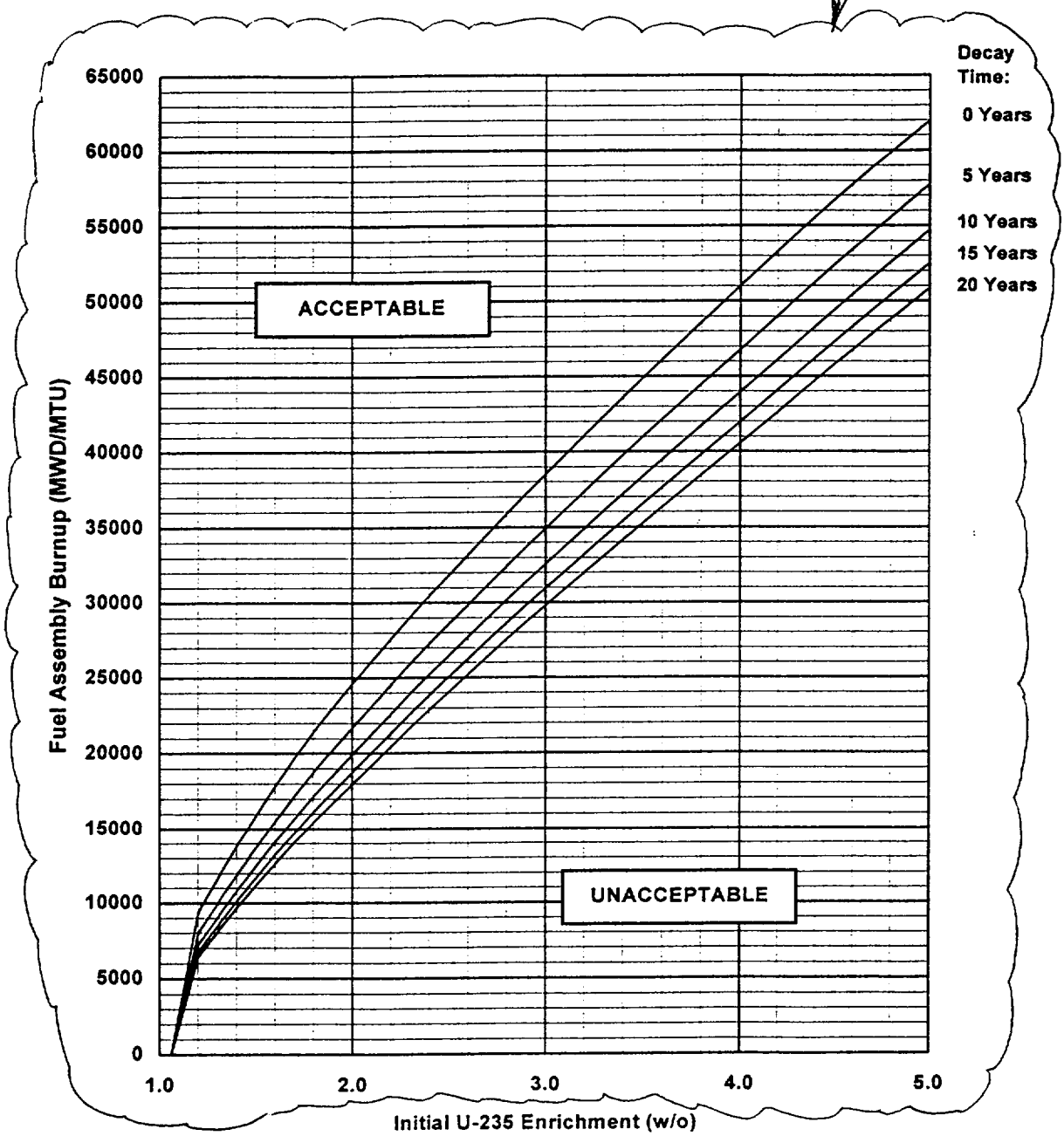

FIGURE 3.7.17-1
Fuel Assembly Burnup vs. U-235 Enrichments vs. Decay Time Limits for
All Cell Storage in High Density Spent Fuel Storage Racks

a 4 out of 4 Storage Configuration

Region II

Replace with new figure

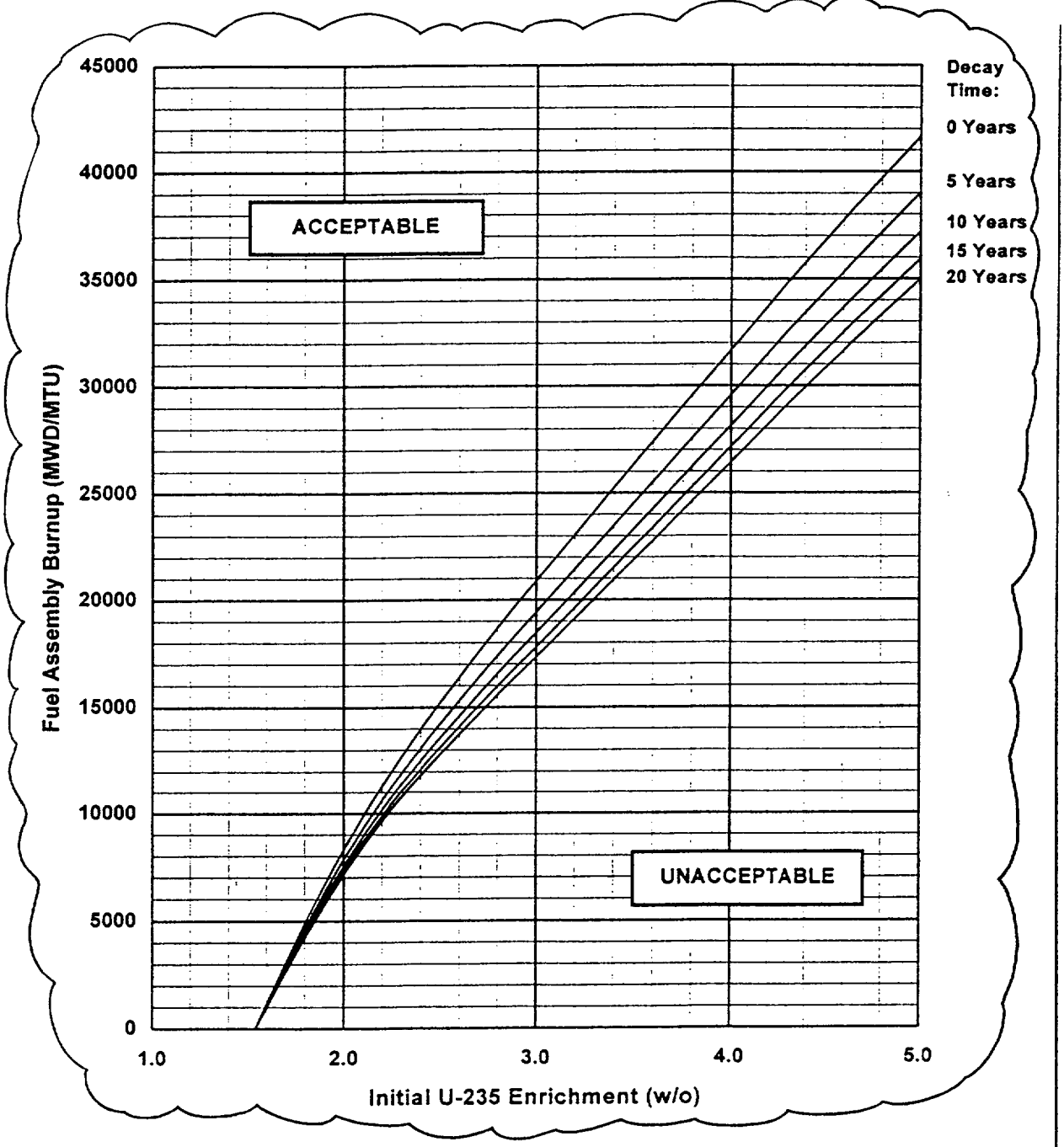

Figure 3.7.17-2
Minimum Burnup vs. Initial U-235 Enrichment vs. Decay Time
For a 3 out of 4 Storage Configuration in High Density Racks

Region II

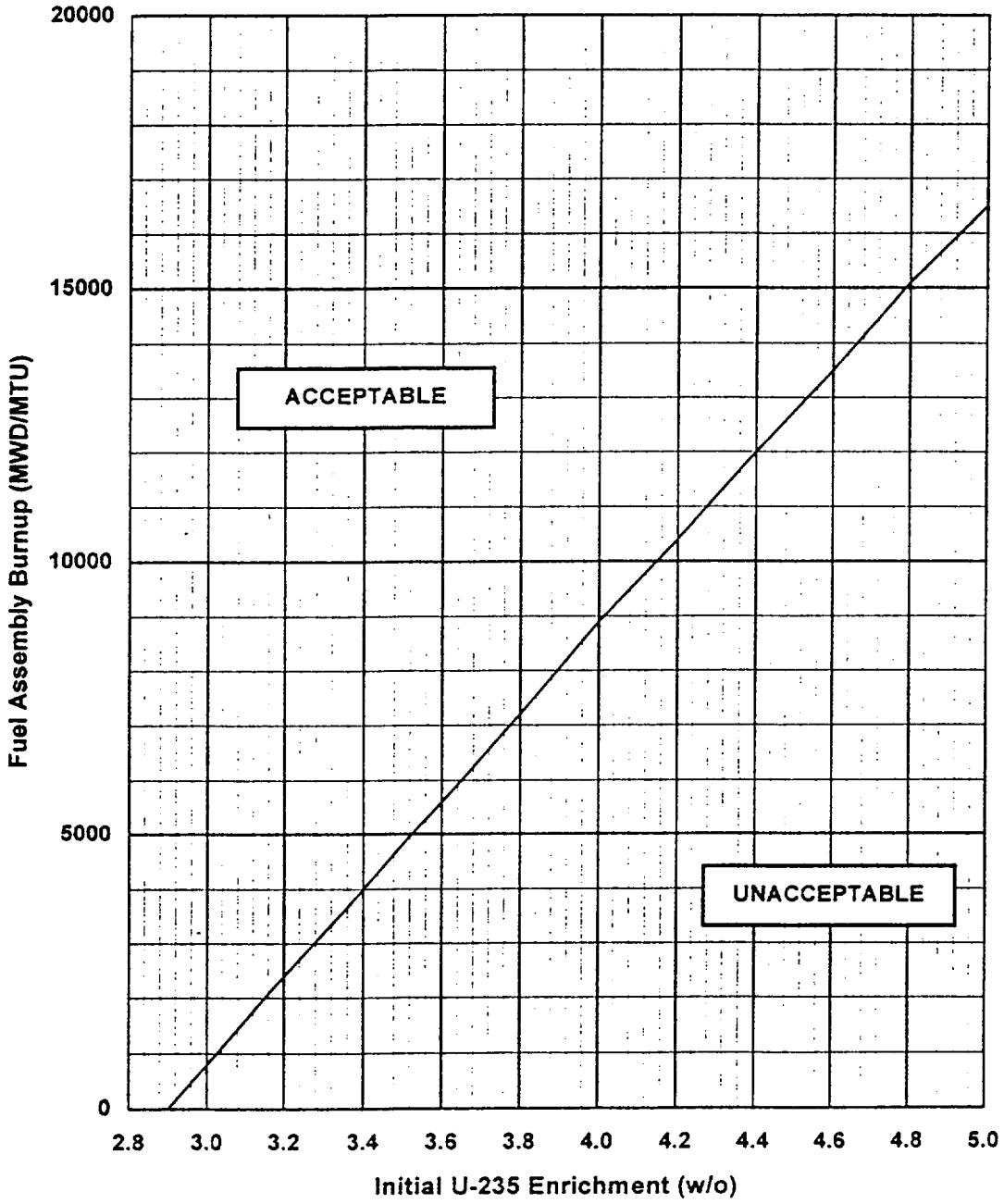

Figure 3.7.17-3
Minimum Burnup vs. Initial U-235 Enrichment
For a 2 out of 4 Storage Configuration in High Density Racks

Region II

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

Region II

A High density (all cell) (4/4), 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

Region II

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

Region II

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

4/4

Region I and Region II interface restrictions: The Region II 1 out of 4 configuration shall be oriented such that the single fuel assembly resides in the internal row with the empty cells facing Region I. There are no interface restrictions between the Region II (2/4, 3/4, 4/4) and Region I configurations

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;

10.6 inch by nominal 11 inch

Region II

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

Region I

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;

Region II

Region I

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;

storage in a 4 out of 4 configuration

Region II

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;

Region II

; and

(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 Region II 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 3373 fuel assemblies.

ATTACHMENT 4 to TXX-00144

MARKUP OF TECHNICAL SPECIFICATION BASES PAGES
(For Information Only)

Pages	**B 3.7-76**
	B 3.7-76a
	B 3.7-76b
	B 3.7-76c
	B 3.7-76d
	B 3.7-76e
	B 3.7-77
	B 3.7-78
	B 3.7.79

B 3.7 PLANT SYSTEMS

B 3.7.16 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND

A common Fuel Building houses facilities for storage and transfer of new and spent fuel. Two pools are provided for CPSES spent fuel storage. Each pool may be used to store fuel from either or both of the CPSES units.

Region II

1 and 2 (SFP1 and SFP2)

In the High Density Rack (HDR) (References 1 and 2) design, the spent fuel storage pool number 2 permits four different configurations (as shown in Figure 3.7.17-4) which, for the purpose of criticality considerations, are considered as separate pools. High Density Racks, with 1470 storage positions, are designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups and decay times 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) a 1 out of 4 configuration.

Region II

1462 and

in SFP1 and SFP2 respectively (2932 total)

as shown in Figure 3.7.17-4

Region I

222 and 219

located in SFP1 and SFP2 respectively (441 total)

Low Density Racks (LDR) with 556 storage positions are located in the spent fuel storage pool number 1 and constitute a fifth configuration within the pools. These LDR are designed to accommodate new fuel with a maximum enrichment of 5.0 w/t % U-235 or spent fuel regardless of the discharge fuel burnup or decay time. Soluble boron is not credited for the storage of spent fuel assemblies within the LDR and there are no storage pattern restrictions associated with the LDR.

Region I racks

Region I racks

The neutron absorber material Boral is credited for the storage of spent fuel assemblies within the Region I racks to maintain Keff less than or equal to 0.95 (Reference 2).

Criticality analyses have been performed (References 2 and 3) which demonstrate that the multiplication factor, k_{eff}, of the fuel and spent fuel storage racks is less than or equal to 0.95. In order to maintain k_{eff} less than or equal to 0.95, the presence of fuel pool soluble boron is credited.

A description of how credit for fuel storage pool soluble boron is used under normal storage configuration conditions is found in Reference 4. The storage configuration is defined using calculations to ensure that k_{eff} will be less than 1.0 with no soluble boron under normal storage conditions including tolerances and uncertainties. Soluble boron credit is then used to maintain k_{eff} less than or equal to 0.95. The pools require 750 ppm of soluble boron to maintain k_{eff} less than or equal to 0.95 for all allowed combinations of storage configurations, enrichments, burnups, and decay

800

(continued)

BACKGROUND (continued)

time limits. The effect of B-10 depletion on the boron concentration for maintaining k_{eff} less than or equal to 0.95 is negligible.

Criticality analyses considering accident conditions have also been performed (References 2 and 3). These analyses establish the amount of soluble boron necessary to ensure that k_{eff} will be maintained less than or equal to 0.95 should pool temperatures fall outside the assumed range or a fuel assembly misload occur. The amount of soluble boron required to mitigate these events is 1800 ppm.

1900

For an occurrence of the above postulated accident condition, the double contingency principle of ANSI/ANS 8.1-1983 (Reference 6) can be applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for these postulated accident conditions, the presence of additional soluble boron in the storage pool water (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.

A boron concentration equal to or greater than 2000 ppm assures that a dilution event which will result in a k_{eff} greater than 0.95 is not credible. This is demonstrated by a boron dilution analysis performed for the CPSES Spent Fuel pools. This conclusion is based on the following: (1) a substantial amount of water is needed in order to dilute the SFP to the design k_{eff} of 0.95, (2) since such a large water volume turnover is required, a SFP dilution event would be readily detected by plant personnel via alarms, flooding in the fuel and auxiliary buildings or by normal operator rounds through the SFP area, and (3) evaluations indicate that, based on the flow rates of non-borated water normally available to the SFP, taken in conjunction with significant operator errors, and equipment failures, sufficient time is available to detect and respond to a dilution event. In addition, there is significant conservatism built into this evaluation; for example, the cooling of the spent fuel pools can be performed by one train supplying common water to both pools. This cooling configuration would allow credit of the volume of both pools and substantially increase the dilution time estimates presented. However, because the flexibility exists for the cooling system to be totally dedicated to one pool, only one pool volume is considered in this evaluation. It should be noted that this boron dilution evaluation considered the boron dilution volumes required to dilute the SFP from 1800 ppm to 750 ppm. The 750 ppm end point was utilized to

1900

800

(continued)

BACKGROUND (continued)

ensure that k_{eff} for the spent fuel racks would remain less than or equal to 0.95. However, as discussed above, calculations for Region II have been performed on a 95/95 basis to show that the spent fuel rack k_{eff} remains less than 1.0 with non-borated water in the pool. Thus, even if the SFP were diluted to concentrations approaching zero ppm, the fuel in the Region II racks would remain subcritical and the health and safety of the public would be protected.

The storage of fuel with initial enrichments up to and including 5.0 weight percent U-235 in the Comanche Peak fuel storage pools has been evaluated. The resulting enrichment, burnup, and decay time limits for the pool are shown in Figures 3.7.17-1 through 3.7.17-4.

APPLICABLE SAFETY ANALYSES

Most fuel storage pool accident conditions will not result in a significant increase in k_{eff}. Examples of such accidents are the drop of a fuel assembly on top of a rack, and the drop of a fuel assembly outside but adjacent to the rack modules.

A dropped assembly accident occurs when a fuel assembly is dropped onto the storage racks. The rack structure is not excessively deformed. An assembly, in its most reactive condition, is considered in the criticality evaluation. Accident analyses have been performed which demonstrate that the dropped assembly which comes to rest horizontally on top of the rack has sufficient water separating it from the active fuel height of stored assemblies to preclude neutronic interaction. This is true even with unborated water. For the borated water condition, the potential for interaction is even less since the water contains boron which is an additional thermal neutron absorber.

However, three accidents can be postulated for each storage configuration that could increase reactivity beyond the analyzed condition. The first postulated accident would be a change in pool temperature to outside the range of normal operating temperatures assumed in the criticality analyses (50°F to 150°F). The second accident would be dropping a fuel assembly into an already loaded cell. The third would be the misloading of a fuel assembly within the racks into a cell for which the restrictions on location, enrichment, burnup, or decay time are not satisfied or adjacent to but outside the racks.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Variations in the temperature of the water passing through the stored fuel assemblies outside the normal operating range were considered in the criticality analysis. The reactivity effects of a temperature range from 32°F to 212°F were evaluated. The increase in reactivity due to the change in temperature is bounded by the misloading accident.

For the accident of dropping a fuel assembly into an already loaded cell, the upward axial leakage of that cell will be reduced; however, the overall effect on the rack reactivity will be insignificant. This is because minimizing the upward-only leakage of just a single cell will not cause any significant increase in reactivity. Furthermore, the neutronic coupling between the dropped assembly and the already loaded assembly will be low due to several inches of assembly nozzle structure which would separate the active fuel regions. Therefore, this accident would clearly be bounded by the misloading accident.

The fuel assembly misloading accident involves placement of a fuel assembly in a location for which it does not meet the requirements for enrichment, burnup, or decay time including the placement of an assembly in a location that is required to be left empty. The result of the misloading is to add positive reactivity, increasing k_{eff} toward 0.95. The maximum required boron to compensate for this event is 1800 ppm, which is below the LCO limit of 2000 ppm.

1900

The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of the 10CFR50.36(c)(2)(ii) .

LCO

The fuel storage pool boron concentration is required to be ≥ 2000 ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential criticality accident scenarios as described in Reference 5. The amount of soluble boron required to offset each of the above postulated accidents was evaluated for all of the proposed storage configurations. The specified minimum boron concentration of 2000 ppm assures that the concentration will remain above these values.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel storage pool

(continued)

No Changes this page

BASES (continued)

ACTIONS

A.1, A.2

When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This action is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. Prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This requirement does not preclude movement of a fuel assembly to a safe position.

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply. If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 9.1.

2. License Amendment Requests 94-22, and 98-08, and 00-05 Spent Fuel Storage Capacity Increase, Docket NOS 50-445 and 50-446, CPSES.

3. Comanche Peak High Density Spent Fuel Rack Criticality Analysis using Soluble Boron Credit, dated April 2000 November, 1998 (Enclosure 2 to TXX-99001 00144).

4. WCAP-14416 NP-A, Rev. 1, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," November 1996.

5. FSAR, Section 15.7.4.

6. American Nuclear Society, "American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," ANSI/ANS-8.1-1983, October 7, 1983.

and No Outer Wrapper Plate

B 3.7 PLANT SYSTEMS

B 3.7.17 Spent Fuel Assembly Storage

BASES

BACKGROUND

A common Fuel Building houses facilities for storage and transfer of new and spent fuel. Two pools are provided for CPSES spent fuel storage. Each pool may be used to store fuel from either or both of the CPSES units.

Region II

1 and 2 (SFP1 and SFP2)

In the High Density Rack (HDR) (References 1 and 2) design, the spent fuel storage pool number 2 permits four different configurations (as shown in Figure 3.7.17-4) which, for the purpose of criticality considerations, are considered as separate pools. High Density Racks, with 1470 storage positions, are designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups and decay times 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) a 1 out of 4 configuration.

Region II

1462 and

in SFP1 and SFP2 respectively (2932 total)

as shown in Figure 3.7.17-4

222 and 219

located in SFP1 and SFP2 repectively (441 total)

Region I

Low Density Racks (LDR) with 556 storage positions are located in the spent fuel storage pool number 1 and constitute a fifth configuration within the pools. These LDR are designed to accommodate new fuel with a maximum enrichment of 5.0 w/t % U-235 or spent fuel regardless of the discharge fuel burnup. Soluble boron is not credited for the storage of spent fuel assemblies within the LDR and there are no storage pattern restrictions associated with the LDR.

Region I racks

Region I racks

Region I racks

A discussion of how soluble boron is credited for the storage of spent fuel assemblies is contained in the BACKGROUND for B 3.7.16.

APPLICABLE SAFETY ANALYSES

Most fuel storage pool accident conditions will not result in a significant increase in k_{eff}. Examples of such accidents are the drop of a fuel assembly on top of a rack, and the drop of a fuel assembly outside but adjacent to the rack modules. However, accidents can be postulated for each HDR storage configuration which could increase reactivity beyond the analyzed condition. A discussion of these accidents is contained in B 3.7.16.

rack

The neutron absorber material Boral is credited for the storage of spent fuel assemblies within the Region I racks to maintain keff less than or equal to 0.95 (Reference 2).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents may be limited to a small fraction of the total operating time.

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, in accordance with Figures 3.7.17-1 through 3.7.17-4, in the accompanying LCO, ensures the k_{eff} of the spent fuel storage pool will always remain $\leq$ 0.95, assuming the pool to be flooded with borated water.

APPLICABILITY

Region II

This LCO applies whenever any fuel assembly is stored in high density racks of the fuel storage pool.

ACTIONS

Region II

A.1

When the configuration of fuel assemblies stored in high density racks of the spent fuel storage pool is not in accordance with Figures 3.7.17-1 through 3.7.17-4, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figures 3.7.17-1 through 3.7.17-4 .

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR verifies, by administrative means, that the initial enrichment, burnup and decay time of the fuel assembly is in accordance with Figures 3.7.17-1 through 3.7.17-4 in the accompanying LCO.

REFERENCES

1. FSAR Section 9.1.
2. License Amendment Requests 94-22, and 98-08, and 00-05 Spent Fuel Storage Capacity Increase, Docket NOS 50-445 and 50-446, CPSES.

ATTACHMENT 5 to TXX-00144

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	4.0-3

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 Region II 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 Region II 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 Region II racks

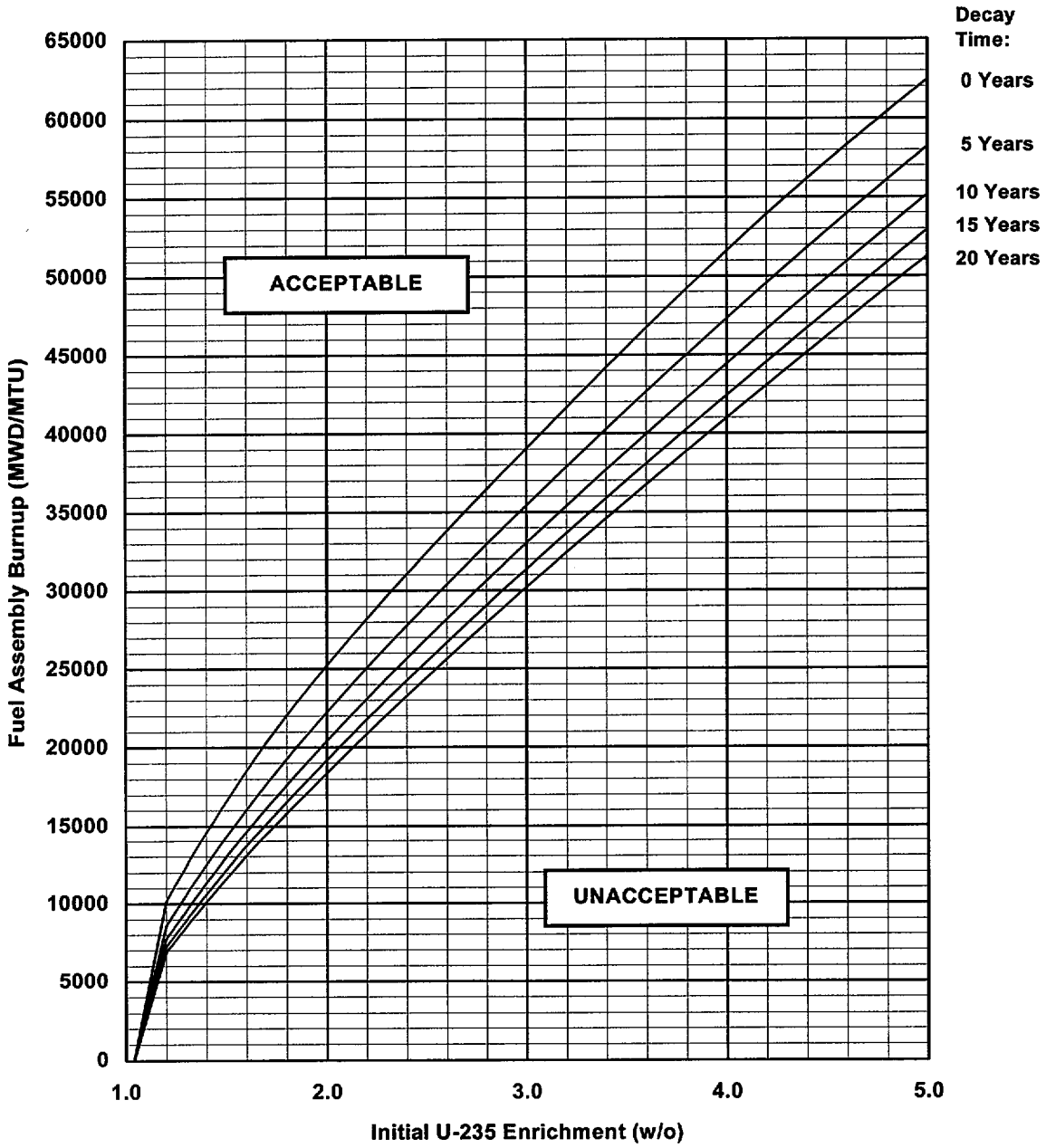

FIGURE 3.7.17-1
Fuel Assembly Burnup vs. U-235 Enrichments vs. Decay Time Limits
For a 4 out of 4 Storage Configuration in Region II Racks

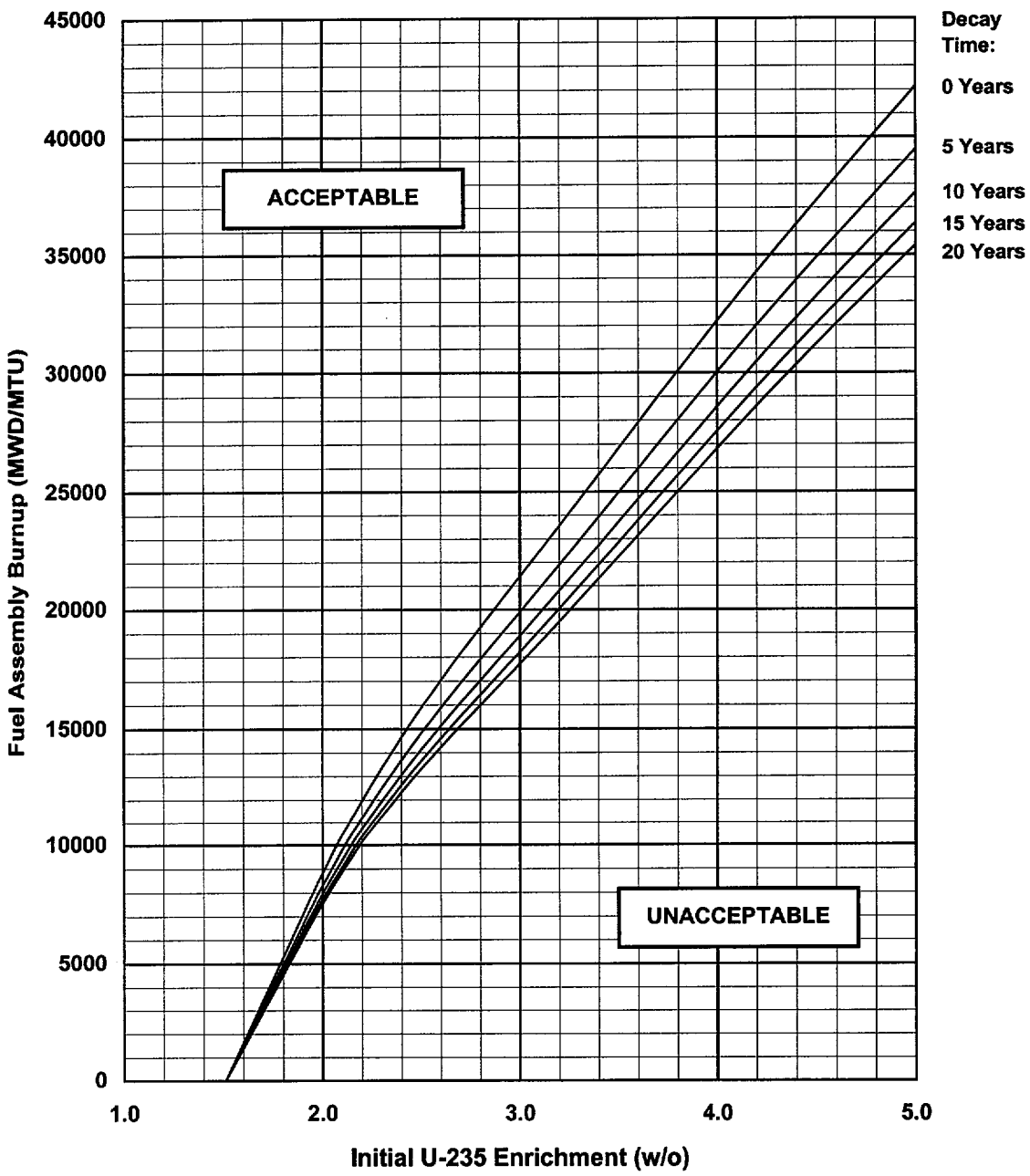

Figure 3.7.17-2
Minimum Burnup vs. Initial U-235 Enrichment vs. Decay Time
For a 3 out of 4 Storage Configuration in Region II Racks

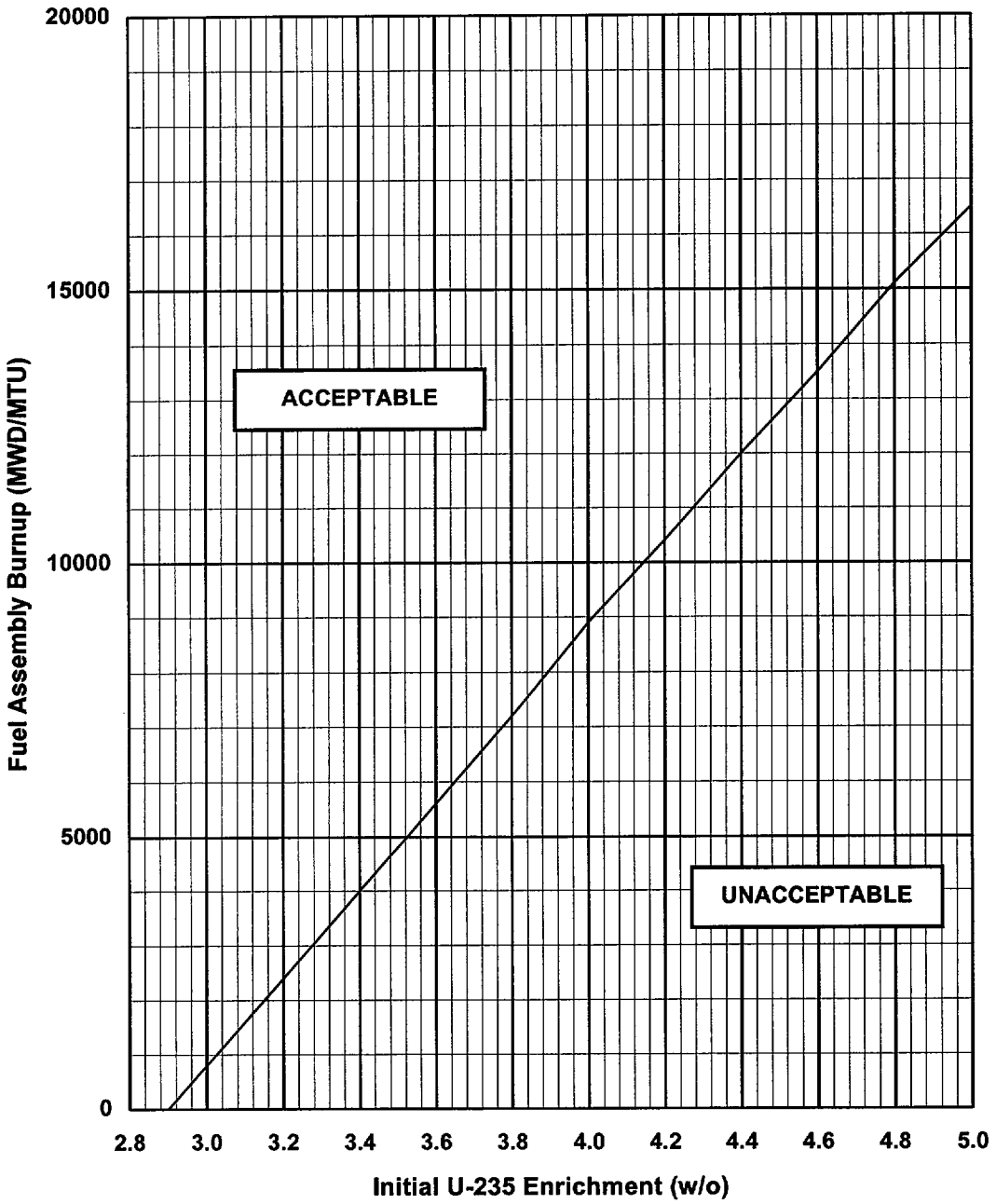

Figure 3.7.17-3
Minimum Burnup vs. Initial U-235 Enrichment
For a 2 out of 4 Storage Configuration in Region II Racks

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 Region II (4/4), new or partially spent fuel assemblies in the "acceptable" domain of Figure 3.7.17-1.

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

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

D Region II (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 Region II 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.

Region I and Region II interface restrictions: The Region II 1 out of 4 configuration shall be oriented such that the single fuel assembly resides in the internal row with the empty cells facing Region I. There are no interface restrictions between the Region II (2/4, 3/4, 4/4) and Region I configurations.

Figure 3.7.17-4
Storage Configurations (4/4, 3/4, 2/4, 1/4) in Region II Racks

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 800 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 Region II fuel storage racks;

e. A nominal 10.6 inch by nominal 11 inch center to center distance between fuel assemblies placed in Region I fuel storage racks;

f. New or partially spent fuel assemblies may be allowed restricted storage in a 1 out of 4 configuration in Region II fuel storage racks (as shown in Figure 3.7.17-4) or unrestricted storage in Region I 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 storage in a 4 out of 4 configuration in Region II 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 Region II fuel storage racks as shown in Figure 3.7.17-4; and

(continued)

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 Region II 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 3373 fuel assemblies.

ATTACHMENT 6 to TXX-00144

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B 3.7 PLANT SYSTEMS

B 3.7.16 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND

A common Fuel Building houses facilities for storage and transfer of new and spent fuel. Two pools are provided for CPSES spent fuel storage. Each pool may be used to store fuel from either or both of the CPSES units.

In the Region II rack (References 1 and 2) design, the spent fuel storage pool numbers 1 and 2 (SFP1 and SFP2) permit four different configurations (as shown in Figure 3.7.17-4) which, for the purpose of criticality considerations, are considered as separate pools. Region II racks, with 1462 and 1470 storage positions in SFP1 and SFP2 respectively (2932 total), are designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups and decay times 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) a 1 out of 4 configuration as shown in Figure 3.7.17-4.

Region I racks, with 222 and 219 storage positions located in SFP1 and SFP2 respectively (441 total), constitute a fifth configuration within the pools. These Region I racks are designed to accommodate new fuel with a maximum enrichment of 5.0 w/t % U-235 or spent fuel regardless of the discharge fuel burnup or decay time. Soluble boron is not credited for the storage of spent fuel assemblies within the Region I racks, and there are no storage pattern restrictions associated with the Region I racks. The neutron absorber material Boral is credited for the storage of spent fuel assemblies within the Region I racks to maintain k_{eff} less than or equal to 0.95 (Reference 2).

Criticality analyses have been performed (References 2 and 3) which demonstrate that the multiplication factor, k_{eff}, of the fuel and spent fuel storage racks is less than or equal to 0.95. In order to maintain k_{eff} less than or equal to 0.95, the presence of fuel pool soluble boron is credited.

A description of how credit for fuel storage pool soluble boron is used under normal storage configuration conditions is found in Reference 4. The storage configuration is defined using calculations to ensure that k_{eff} will be less than 1.0 with no soluble boron under normal storage conditions including tolerances and uncertainties. Soluble boron credit is then used to maintain k_{eff} less than or equal to 0.95. The pools require 800 ppm of soluble boron to maintain k_{eff} less than or equal to 0.95 for all allowed combinations of storage configurations, enrichments, burnups, and decay

(continued)

BACKGROUND (continued)

time limits. The effect of B-10 depletion on the boron concentration for maintaining k_{eff} less than or equal to 0.95 is negligible.

Criticality analyses considering accident conditions have also been performed (References 2 and 3). These analyses establish the amount of soluble boron necessary to ensure that k_{eff} will be maintained less than or equal to 0.95 should pool temperatures fall outside the assumed range or a fuel assembly misload occur. The amount of soluble boron required to mitigate these events is 1900 ppm.

For an occurrence of the above postulated accident condition, the double contingency principle of ANSI/ANS 8.1-1983 (Reference 6) can be applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for these postulated accident conditions, the presence of additional soluble boron in the storage pool water (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.

A boron concentration equal to or greater than 2000 ppm assures that a dilution event which will result in a k_{eff} greater than 0.95 is not credible. This is demonstrated by a boron dilution analysis performed for the CPSES Spent Fuel pools. This conclusion is based on the following: (1) a substantial amount of water is needed in order to dilute the SFP to the design k_{eff} of 0.95, (2) since such a large water volume turnover is required, a SFP dilution event would be readily detected by plant personnel via alarms, flooding in the fuel and auxiliary buildings or by normal operator rounds through the SFP area, and (3) evaluations indicate that, based on the flow rates of non-borated water normally available to the SFP, taken in conjunction with significant operator errors, and equipment failures, sufficient time is available to detect and respond to a dilution event. In addition, there is significant conservatism built into this evaluation; for example, the cooling of the spent fuel pools can be performed by one train supplying common water to both pools. This cooling configuration would allow credit of the volume of both pools and substantially increase the dilution time estimates presented. However, because the flexibility exists for the cooling system to be totally dedicated to one pool, only one pool volume is considered in this evaluation. It should be noted that this boron dilution evaluation considered the boron dilution volumes required to dilute the SFP from 1900 ppm to 800 ppm. The 800 ppm end point was utilized to

(continued)

BACKGROUND (continued)

ensure that k_{eff} for the spent fuel racks would remain less than or equal to 0.95. However, as discussed above, calculations for Region II have been performed on a 95/95 basis to show that the spent fuel rack k_{eff} remains less than 1.0 with non-borated water in the pool. Thus, even if the SFP were diluted to concentrations approaching zero ppm, the fuel in the Region II racks would remain subcritical and the health and safety of the public would be protected.

The storage of fuel with initial enrichments up to and including 5.0 weight percent U-235 in the Comanche Peak fuel storage pools has been evaluated. The resulting enrichment, burnup, and decay time limits for the pool are shown in Figures 3.7.17-1 through 3.7.17-4.

APPLICABLE SAFETY ANALYSES

Most fuel storage pool accident conditions will not result in a significant increase in k_{eff}. Examples of such accidents are the drop of a fuel assembly on top of a rack, and the drop of a fuel assembly outside but adjacent to the rack modules.

A dropped assembly accident occurs when a fuel assembly is dropped onto the storage racks. The rack structure is not excessively deformed. An assembly, in its most reactive condition, is considered in the criticality evaluation. Accident analyses have been performed which demonstrate that the dropped assembly which comes to rest horizontally on top of the rack has sufficient water separating it from the active fuel height of stored assemblies to preclude neutronic interaction. This is true even with unborated water. For the borated water condition, the potential for interaction is even less since the water contains boron which is an additional thermal neutron absorber.

However, three accidents can be postulated for each storage configuration that could increase reactivity beyond the analyzed condition. The first postulated accident would be a change in pool temperature to outside the range of normal operating temperatures assumed in the criticality analyses (50°F to 150°F). The second accident would be dropping a fuel assembly into an already loaded cell. The third would be the misloading of a fuel assembly within the racks into a cell for which the restrictions on location, enrichment, burnup, or decay time are not satisfied or adjacent to but outside the racks.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Variations in the temperature of the water passing through the stored fuel assemblies outside the normal operating range were considered in the criticality analysis. The reactivity effects of a temperature range from 32°F to 212°F were evaluated. The increase in reactivity due to the change in temperature is bounded by the misloading accident.

For the accident of dropping a fuel assembly into an already loaded cell, the upward axial leakage of that cell will be reduced; however, the overall effect on the rack reactivity will be insignificant. This is because minimizing the upward-only leakage of just a single cell will not cause any significant increase in reactivity. Furthermore, the neutronic coupling between the dropped assembly and the already loaded assembly will be low due to several inches of assembly nozzle structure which would separate the active fuel regions. Therefore, this accident would clearly be bounded by the misloading accident.

The fuel assembly misloading accident involves placement of a fuel assembly in a location for which it does not meet the requirements for enrichment, burnup, or decay time including the placement of an assembly in a location that is required to be left empty. The result of the misloading is to add positive reactivity, increasing k_{eff} toward 0.95. The maximum required boron to compensate for this event is 1900 ppm, which is below the LCO limit of 2000 ppm.

The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of the 10CFR50.36(c)(2)(ii) .

LCO

The fuel storage pool boron concentration is required to be $\geq$ 2000 ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential criticality accident scenarios as described in Reference 5. The amount of soluble boron required to offset each of the above postulated accidents was evaluated for all of the proposed storage configurations. The specified minimum boron concentration of 2000 ppm assures that the concentration will remain above these values.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel storage pool

(continued)

BASES (continued)

ACTIONS

A.1, A.2

When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This action is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. Prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This requirement does not preclude movement of a fuel assembly to a safe position.

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply. If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 9.1.

2. License Amendment Requests 94-22, 98-08, and 00-05, Spent Fuel Storage Capacity Increase, Docket NOS 50-445 and 50-446, CPSES.

3. Comanche Peak High Density Spent Fuel Rack Criticality Analysis using Soluble Boron Credit and No Outer Wrapper Plate, dated April, 2000 (Enclosure 2 to TXX-00144).

4. WCAP-14416 NP-A, Rev. 1, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," November 1996.

5. FSAR, Section 15.7.4.

6. American Nuclear Society, "American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," ANSI/ANS-8.1-1983, October 7, 1983.

B 3.7 PLANT SYSTEMS

B 3.7.17 Spent Fuel Assembly Storage

BASES

BACKGROUND

A common Fuel Building houses facilities for storage and transfer of new and spent fuel. Two pools are provided for CPSES spent fuel storage. Each pool may be used to store fuel from either or both of the CPSES units.

In the Region II rack (References 1 and 2) design, the spent fuel storage pool numbers 1 and 2 (SFP1 and SFP2) permit four different configurations (as shown in Figure 3.7.17-4) which, for the purpose of criticality considerations, are considered as separate pools. Region II racks, with 1462 and 1470 storage positions in SFP1 and SFP2 respectively (2932 total), are designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups and decay times 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) a 1 out of 4 configuration as shown in Figure 3.7.17-4.

Region I racks, with 222 and 219 storage positions located in SFP1 and SFP2 respectively (441 total), constitute a fifth configuration within the pools. These Region I racks are designed to accommodate new fuel with a maximum enrichment of 5.0 w/t % U-235 or spent fuel regardless of the discharge fuel burnup. Soluble boron is not credited for the storage of spent fuel assemblies within the Region I racks, and there are no storage pattern restrictions associated with the Region I racks. The neutron absorber material Boral is credited for the storage of spent fuel assemblies within the Region I racks to maintain k_{eff} less than or equal to 0.95 (Reference 2).

A discussion of how soluble boron is credited for the storage of spent fuel assemblies is contained in the BACKGROUND for B 3.7.16.

APPLICABLE SAFETY ANALYSES

Most fuel storage pool accident conditions will not result in a significant increase in k_{eff}. Examples of such accidents are the drop of a fuel assembly on top of a rack, and the drop of a fuel assembly outside but adjacent to the rack modules. However, accidents can be postulated for each rack storage configuration which could increase reactivity beyond the analyzed condition. A discussion of these accidents is contained in B 3.7.16.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents may be limited to a small fraction of the total operating time.

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, in accordance with Figures 3.7.17-1 through 3.7.17-4, in the accompanying LCO, ensures the k_{eff} of the spent fuel storage pool will always remain $\leq$ 0.95, assuming the pool to be flooded with borated water.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in Region II racks of the fuel storage pool.

ACTIONS

A.1

When the configuration of fuel assemblies stored in Region II racks of the spent fuel storage pool is not in accordance with Figures 3.7.17-1 through 3.7.17-4, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figures 3.7.17-1 through 3.7.17-4 .

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR verifies, by administrative means, that the initial enrichment, burnup and decay time of the fuel assembly is in accordance with Figures 3.7.17-1 through 3.7.17-4 in the accompanying LCO.

REFERENCES

1. FSAR Section 9.1.

2. License Amendment Requests 94-22, 98-08, and 00-05, Spent Fuel Storage Capacity Increase, Docket NOS 50-445 and 50-446, CPSES.

ATTACHMENT 7 to TXX-00144

Commitments

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

Commitment Number	Commitment
27209	The Westinghouse criticality analysis methodology used for the Region II (3/4 and 4/4) storage configurations include a generically determined reactivity bias to account for axial, or three-dimensional, burnup effects. In March, 2000, Duke Power notified the NRC of a potential non-conservatism in the Westinghouse generic axial burnup reactivity bias. A description of the NRC approved generic axial burnup reactivity bias is described in WCAP 14416. This generic issue is currently under evaluation by Westinghouse, and should it be determined that there is an impact on the analyses presented in this submittal, TXU Electric will submit revised information as appropriate. (Page 7-8 of Attachment 2)
27210	Comanche Peak's corrective action program has identified that the impact of spent fuel pool heat loads and heat loads of previous modifications were not properly maintained current in the CCW thermal loads analysis. The analysis will be updated prior to implementation of this License Amendment Request (LAR). The CCW System will continue to perform its intended safety functions. This LAR will be supplemented should these conclusions change. (Page 15 of Attachment 2)
27211	Comanche Peak's corrective action program has identified that the various plant heat loads were not properly considered (collectively) in the ultimate heat sink analysis. This issue does not involve the Spent Fuel Pool Cooling System's ability to reject heat to the SSI. The analysis will be updated prior to implementation of License Amendment 74 (Credit for Boron). The ultimate heat sink will continue to perform its intended safety functions. This LAR will be supplemented should these conclusions change. (Page 16 of Attachment 2)

27212	Comanche Peak's corrective action program has identified that the effect of the revised building responses on the balance of Fuel Building structures, systems, and components was not properly considered. The analyses will be updated prior to implementation of this modification. It is anticipated that the design of the Fuel Building structures, systems, and components will remain adequate and that any necessary modifications to the balance of the Fuel Building (none are expected) would be completed prior to implementation of the proposed reracking. This LAR will be supplemented should these conclusions change. (Page 20 of Attachment 2)

The commitment number is used by TXU Electric for the internal tracking of CPSES commitments.

ENCLOSURE 4 to TXX-00144

Affidavit (request to withhold proprietary information in Enclosure 1)

AFFIDAVIT PURSUANT TO 10CFR2.790

I, Charles W. Bullard II, being duly sworn, depose and state as follows:

(1) I am an employee of Holtec International and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in Holtec report HI-2002402, "Licensing Report for Spent Fuel Rack Installation at Comanche Peak Steam Electric Station." The proprietary material in this document is contained within square brackets (i.e., [ ]) or delineated by proprietary designation on specific pages.

(3) In making this application for withholding of proprietary information of which it is the owner, Holtec International relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4) and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10CFR Part 9.17(a)(4), 2.790(a)(4), and 2.790(b)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by Holtec's competitors without license from Holtec International constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

c. Information which reveals cost or price information, production, capacities, budget levels, or commercial strategies of Holtec International, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future Holtec International customer-funded development plans and programs of potential commercial value to Holtec International;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs 4.a and 4.b above.

(5) The information sought to be withheld is being submitted to the NRC in confidence. The information (including that compiled from many sources) is of a sort customarily held in confidence by Holtec International, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by Holtec International. No public disclosure has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within Holtec International is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his designee), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside Holtec International are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information classified as proprietary was developed and compiled by Holtec International at a significant cost to Holtec International. This information is classified as proprietary because it contains detailed historical data and analytical results not available elsewhere. This information would provide other parties, including competitors, with information from Holtec International's technical database and the results of evaluations performed using codes developed by Holtec International. Release of this information would improve a competitor's position without the competitor having to expend similar resources for the development of the database. A substantial effort has been expended by Holtec International to develop this information.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to Holtec International's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of Holtec International's comprehensive spent fuel storage technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology, and includes development of the expertise to determine and apply the appropriate evaluation process.

The research, development, engineering, and analytical costs comprise a substantial investment of time and money by Holtec International.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

AFFIDAVIT PURSUANT TO 10CFR2.790

Holtec International's competitive advantage will be lost if its competitors are able to use the results of the Holtec International experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to Holtec International would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive Holtec International of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

STATE OF NEW JERSEY)
) ss:
COUNTY OF BURLINGTON)

Charles W. Bullard II, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at Marlton, New Jersey, this 15th day of September 2000.

Charles W. Bullard II
Holtec International

Subscribed and sworn before me this 15 day of September, 2000.

Maria C. Pepe

MARIA C. PEPE
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires April 25, 2005

ENCLOSURE 2 to TXX-00144

Criticality Analysis