October 2, 2001

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES), UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: INCREASE IN SPENT FUEL STORAGE CAPACITY TO 3,373 FUEL ASSEMBLIES (TAC NOS. MB0207 AND MB0208)

Dear Mr. Terry:

The Commission has issued the enclosed Amendment No. 87 to Facility Operating License No. NPF-87 and Amendment No. 87 to Facility Operating License No. NPF-89 for CPSES, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated October 4, 2000, as supplemented by letters dated April 30, June 18, and July 18, 2001.

The amendments revise the TSs to increase the spent fuel storage capacity from 2,026 to 3,373 fuel assemblies in the spent fuel pool.

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

Sincerely,

/RA/

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

Docket Nos. 50-445 and 50-446

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

cc w/encls: See next page

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TXU ELECTRIC

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT NO. 1

DOCKET NO. 50-445

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 87
License No. NPF-87

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

   A. The application for amendment by TXU Electric dated October 4, 2000, as supplemented by letters dated April 30, June 18, and July 18, 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

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

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

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 87, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. TXU Electric shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented no later than January 31, 2002.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachment: Changes to the Technical Specifications

Date of Issuance: October 2, 2001

TXU ELECTRIC

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT NO. 2

DOCKET NO. 50-446

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 87
License No. NPF-89

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

   A. The application for amendment by TXU Electric dated October 4, 2000, as supplemented by letters dated April 30, June 18, and July 18, 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

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

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

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 87, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. TXU Electric shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented no later than January 31, 2002.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachment: Changes to the Technical Specifications

Date of Issuance: October 2, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 87

TO FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 87

FACILITY OPERATING LICENSE NO. NPF-89

DOCKET NOS. 50-445 AND 50-446

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

Remove	Insert
3.7-36	3.7-36
3.7-37	3.7-37
3.7-38	3.7-38
3.7-39	3.7-39
3.7-40	3.7-40
3.7-41	3.7-41
4.0-2	4.0-2
4.0-3	4.0-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 87 TO

FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 87 TO

FACILITY OPERATING LICENSE NO. NPF-89

TXU ELECTRIC

COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-445 AND 50-446

1.0 INTRODUCTION

By application dated October 4, 2000, as supplemented by letters dated April 30, June 18, and July 18, 2001 (References 1, 2, 3, and 4, respectively), TXU Electric (the licensee) requested changes to the Technical Specifications (TSs) for the Comanche Peak Steam Electric Station (CPSES), Units 1 and 2. The proposed changes would revise the TSs to increase the spent fuel storage capacity from 2,026 to 3,373 fuel assemblies in the spent fuel pool (SFP). The letters dated April 30, June 18, and July 18, 2001, provided clarifying or additional information that did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's proposed no significant hazards consideration determination.

2.0 BACKGROUND

The spent fuel storage facility at CPSES consists of two SFPs designated as SFP1 and SFP2. The requested increase in fuel storage capacity would be achieved by replacing the existing twenty low-density spent fuel storage racks (racks) in SFP1 with three high-density racks manufactured by Holtec International (Holtec) and nine high-density racks manufactured by Westinghouse Electric Company LLC (Westinghouse), and adding three Holtec high-density racks to the existing nine high-density racks in SFP2. This modification will increase total storage capacity of the plant from 2,026 to 3,373 fuel assemblies. The Holtec racks, designated as Region I racks, can accommodate 222 fuel assemblies in SFP1 and 219 in SFP2. The Westinghouse racks, designated as Region II racks, can accommodate 1,470 fuel assemblies in SFP1 and 1,462 fuel assemblies in SFP2. The Region I racks contain Boral neutron absorbing material and can store fuel with no restrictions on its burnup or decay times. The Region II racks do not have any neutron absorbing material and have some restrictions on the fuel assemblies stored in them. Both Holtec and Westinghouse racks are constructed from stainless steel.

3.0 EVALUATION

3.1 Evaluation of Spent Fuel Rack Materials

The Region I racks consist of freestanding modules containing prismatic storage cells interconnected through longitudinal welds. Panels of Boral provide neutron attenuation between adjacent storage cells. The following structural materials are used in the racks:

- All sheet metal stock, baseplate, internally threaded support legs, and bearing pads are made from American Society of Mechanical Engineers (ASME) SA240-304 stainless steel

- The externally threaded support spindles are made from ASME SA564-630 precipitation hardened stainless steel (heat treated to 1,100 °F)

- Weld material is in accordance with ASME specification: SFA [spent fuel assembly] 5.9 ER308L

All these materials have been previously used in many other applications. In the past, they were exposed to an environment similar or more severe than those in the SFPs of CPSES without experiencing any observable corrosion damage. In this regard, the TS for CPSES requires that boron concentration in the SFPs be equal to or greater than 2,000 parts per million (ppm). The environment to which the racks are exposed is, therefore, acidic with a pH of approximately 4.7.

Boral is used as a poison material in the Region I racks for absorbing neutrons generated by the stored fuel. Boral is a cermet composite material made of Type 1100 aluminum and boron carbide. The composite panel consists of boron carbide particles imbedded in Type 1100 aluminum matrix clad in Type 1100 aluminum sheets. The Type 1100 aluminum materials impart sufficient pitting and general corrosion resistance by forming an aluminum oxide layer on its surface when exposed to an oxidizing environment. The oxide is stable in environments with a pH of 4.5 to 8.5. The boron carbide particles in Boral panels have shown good structural stability with the Type 1100 aluminum matrix material. Despite these corrosion resistant properties of Boral, some small degree of corrosion is expected. Although this will not result in a significant depletion of boron and resulting degradation of its neutron absorbing properties, some generation of hydrogen from corrosion of aluminum can occur when Boral is exposed to SFP water. This effect is more pronounced in new panels which do not yet have a well formed protective oxide film. This hydrogen, when not vented, could produce pressure, causing swelling of the metal sheath holding the Boral panels and resultant deformation of the storage cells. In order to prevent swelling from occurring, the racks manufactured by Holtec will have vented Boral metal sheaths allowing the generated hydrogen to escape. Production of this hydrogen will significantly decrease as aluminum surfaces develop protective oxide film.

The Region II racks in SFP1 are new racks and the racks in SFP2 are the existing racks. Neither of these racks contain neutron absorbing material. They are similar in design. However, the racks in Region II of SFP1 originally contained Boraflex neutron absorber. This material was removed and, in order to satisfy structural requirements, the wrappers which served to attach Boraflex panels to the walls of the fuel cells were replaced with spacer plates. The racks consist of freestanding modules containing prismatic cells, which are welded to one

another and to the base plate to produce a matrix structure. The rack assembly consists of two major sections which are the cell assembly and the base support assembly. Both the existing and the new Region II racks are fabricated from the same materials. The following structural materials are used in the racks:

- ASME SA-479 or SA-240 type 304 stainless steel complying with NF-2000, Class 3, Subsection III, ASME Boiler and Pressure Vessel Code

- Type 17-4 pH stainless steel used for leveling screws

These materials are resistant to the SFP environment, as was demonstrated by the existing racks, which were exposed to this environment for a number of years.

Based on its evaluation, the NRC staff concludes that the materials in the Region I and Region II racks are compatible with the environment in the SFPs. These racks will not undergo material degradation that could affect their ability to safely store spent and new fuel. A vented design of the metal sheaths holding Boral panels in the Holtec racks prevents the corrosion-generated hydrogen from building up pressures, that could cause distortion of the fuel assembly cells. The NRC staff concludes, therefore, that the racks are constructed of materials that are acceptable for use in the SFP environment.

3.2 Structural Evaluation

The SFP racks are seismic Category I equipment and are required (in order to satisfy General Design Criterion (GDC) 62 of Appendix A to 10 CFR Part 50) to remain functional during and after a safe-shutdown earthquake (SSE) under all applicable loading conditions. The licensee's contractor, Holtec, performed the design, fabrication, and safety analysis of the new high-density SFP racks. The principal construction materials for the Region I and Region II racks are SA240-Type 304 stainless steel and plate stock, and SA564-630 (precipitation-hardened stainless steel) for the adjustable support spindles.

The overall design of the new racks at CPSES is similar to Holtec racks that the NRC staff has approved for service at many other nuclear power plants. The key design criteria are based on NRC letter dated April 14, 1978 (Reference 5), to all power reactor licensees as modified by NRC letter dated January 18, 1979 (Reference 6). The key design criteria of the CPSES SFP racks are described in Section 2.2 of Enclosure 1 of Reference 1. Briefly, the following criteria are applicable from the structural safety point of view: (1) all new rack modules are required to be free-standing; (2) all free-standing modules are required to be kinematically stable (against tipping or overturning) when a seismic event that is 150% of the postulated operating-basis earthquake (OBE) or 110% of the postulated SSE is imposed on any module; (3) all primary stresses in the modules satisfy the limits postulated in Section III, Subsection NF of the 1980 ASME Boiler and Pressure Vessel Code with addenda through 1982; (4) the spatial average bulk pool temperature is required to remain under 150 °F during and following a normal refueling and, for the abnormal design conditions, no bulk pool boiling occurs; and (5) the ability of the reinforced concrete structure of the SFP to withstand the effects of the load combinations set forth in the CPSES Updated Final Safety Analysis Report (UFSAR) Section 3.8.4 is demonstrated.

3.2.1 Structural / Seismic Analysis

The licensee presented the analysis and results of a dynamic evaluation of SFP1's rack array with a conservatively loaded capacity of 1698 fuel assemblies in Section 6 of Enclosure 1 to Reference 1. A survey of both SFPs revealed that the dimensions of SFP1 are slightly more restrictive (i.e., smaller) than those of SFP2. Accordingly, the analysis performed for SFP1 is deemed to apply to SFP2 as well. As discussed in detail in Reference 1, Holtec, the licensee's consultant, modeled the entire assemblage of rack modules in one comprehensive simulation known as the Whole-Pool-Multi-Rack (WPMR) analysis. In order to maintain continuity with previous analytical methods, both single rack (SR) and WPMR analyses were performed to assure conformance with established acceptance criteria (e.g., stress and displacement). Holtec used an NRC-accepted computer program, DYNARACK, for the dynamic analysis to demonstrate the structural adequacy of the SFP rack design under the earthquake loading conditions. The DYNARACK program (which can perform simultaneous simulation of all racks in the SFP for the WPMR analysis) has been used for other rack analyses for several nuclear power plants. The DYNARACK program utilizes a nonlinear analytical model consisting of inertial mass elements, spring elements, gap elements, and friction elements to simulate the 3-dimensional (3-D) dynamic behavior of the rack and the SFAs, including the frictional and hydrodynamic effects. The DYNARACK computer code accurately simulates the friction, impact, and other nonlinear dynamic events. The code models the beam characteristics of the rack including shear, flexibility, and torsion effects appropriately, by modeling each rack as a 3-D structure having the support pedestals and the SFAs in proper locations. The potential interaction between the SFA and storage cell walls is simulated by permitting the impact at any of the four facing walls followed by rebound and impact at the opposite wall. Further, the rack pedestals can lift off or slide to satisfy the instantaneous dynamic equilibrium of the system throughout the seismic event. The rack structure can undergo overturning, bending, twisting, and other dynamic motion modes as dictated by the interaction between the seismic inertia, impact, friction, and fluid coupling forces. The DYNARACK code calculates the nodal forces and displacements at the nodes, and then obtains the detailed stress field in the rack elements from the calculated nodal forces.

The lateral motion of the rack due to earthquake ground motion is resisted by the pedestal-to-pool slab interface friction, and is amplified or retarded by the fluid coupling forces produced by the close position of the rack to other structures. The seismic analyses of the racks were performed utilizing the direct integration time-history method. One set of three artificial time-histories (two horizontal and one vertical acceleration time-histories) was generated in accordance with the provisions of Standard Review Plan (SRP) 3.7.1, "Seismic Design Parameters." A preferred criterion for the time-history generation given in SRP 3.7.1 calls for both the response spectrum and the power spectral density corresponding to the generated acceleration time-history to envelope their target (design basis) counterparts with only finite enveloping infractions.

The licensee provided the following information, in Attachment 3 to Reference 3, regarding the validation of DYNARACK program:

> Data supplied by Scavuzzo to simulate the experiment using the pre-processor CHANBP6 and the solver DYNARACK was used. The results of the comparisons have been incorporated into the Holtec validation manual for DYNARACK (HI-91700) as an additional confirmation of the fluid coupling methodology. This validation manual, along

with additional supporting documentation and discussions, have been previously submitted to the NRC in April, 1992, under dockets 50-315 and 50-316 for D.C. Cook station and for Waterford 3. The submittal for Waterford contained the evaluation of the Scavuzzo theory and experiment and demonstrated that the WPMR general formulation was in agreement with the experimental results.

In view of the above information provided by the licensee, the NRC staff accepts the use of the DYNARACK computer program for the structural analysis of the racks at CPSES.

The licensee considered the applicable loads and their combinations in the seismic analysis of the racks, and performed parametric simulations for both the SR and WPMR analyses. The parameters, which were varied in the different computer runs, consisted of the rack/SFP interface coefficient of friction, the extent of storage locations occupied by spent fuel (ranging from nearly empty to full) and the type of seismic input (SSE or OBE). The results of these analyses show the maximum rack displacement to be 0.324 inches (for the SSE condition). For this case, a rack overturning evaluation indicated the factor of safety against overturning to be 111 (Reference 1, Section 6.0 in Enclosure 1), which is much higher than the acceptance criteria of 1.1 for the SSE condition. These results show that there are large safety margins against overturning of the racks, and that the structural integrity and stability of the racks and SFAs will be maintained.

Using the results of parametric evaluations, the licensee computed the maximum values of pedestal vertical forces, rack impact loads (between a single pedestal and bearing pad) and rack displacements. Using these data, the licensee performed the rack impact evaluation, as well as the stress limit evaluation of the rack structure satisfying the ASME Code, Section III, Subsection NF, for normal and upset conditions (Level A or Level B), and Section F-1334 (ASME Section III, Appendix F) for the Level D condition. The calculated results show that there are no rack-to-wall impacts, and no rack-to-rack impacts at the top of the rack during a postulated design basis seismic event (OBE or SSE). However, there is some minor impact between adjacent racks at the baseplate level, as baseplate gaps are initially set to zero. The licensee evaluated the stresses imposed by the instantaneous impacts on the steel baseplate, and indicated that all stresses were well below the corresponding ASME Code Section III, Subsection NF limits (Reference 1, Section 6.13 in Enclosure 1).

In summary, the licensee's parametric study (e.g., varying coefficients of friction, different geometries and fuel loading conditions of the rack) involving both SR and WPMR analyses showed that: (1) there are large factors of safety for the induced stresses of the rack when compared to the corresponding allowable values provided in the ASME Code, Section III; and (2) there are no rack-to-wall and rack-to-rack impacts (except some insignificant impact at the rack baseplate level as described above). Therefore, the NRC staff concludes that the racks will perform their safety function and maintain their structural integrity under postulated loading conditions and are, therefore, acceptable.

3.2.2 Spent Fuel Pool (SFP) Structural Integrity

The SFPs consist of reinforced concrete slab and walls that are lined with a stainless steel liner. The nominal inner dimensions of the pools are: 40 feet, 3 inches in east-west (E-W) direction, 30 feet, 0 inches in the north-south (N-S) direction, and 40 feet, 5-1/4 inches deep from the operating deck to the top of the steel liner at elevation 819 feet, 6-3/4 inches. The SFP slabs,

as an integral part of the Fuel Building (FB) foundation, are directly supported by the underlying soil for the most part. The southern end of SFP2 has a 12 feet, 0 inches wide tunnel below the reinforced concrete slab. Reinforced concrete slabs located north and south of the SFPs provide horizontal support to the pool walls. The building exterior wall that bounds the transfer canal from west and the walls located east of the pool are extended upwards to the building roof, providing horizontal bracing to the pool structure in the E-W direction. The 2 inch wide seismic gap between the reactor containment shell and the FB walls provides for both free seismic movements and thermal expansion of the SFP structure towards the north and west.

The structural integrity of the SFPs was previously evaluated and approved for installation of high-density racks in support of License Amendment Request (LAR) 94-022 (Reference 7). The analysis methodology and criteria used in the seismic analysis for the present LAR are the same as those used for Reference 7.

The licensee analyzed pool regions using the NRC-accepted industry codes and standards. The results for individual load components were combined using factored load combinations. In addition to the dead and live loads, the analysis considered the seismic, thermal, and hydrodynamic loadings. Tables 8.2.4 and 8.2.5 in Enclosure 2 of Reference 3 show the analytical results from potentially limiting load combinations for the bending moments and axial forces of the slab and walls, respectively. Using these results, the licensee determined the safety margins and found them to range from 1.11 to 1.35 for the slab, and 1.12 to 3.28 for the walls. In determining these safety margins, the licensee has considered a reduction in concrete strength due to high temperature and thermal cycling (Reference 3, which supplemented the results given in Reference 2).

The licensee has determined that the maximum design basis bulk pool temperature will be less than 150 °F for normal operation in accordance with American Concrete Institute Code 349. For abnormal conditions, the maximum bulk pool temperature will be less than 212 °F (per UFSAR Table 9.1-1) which is acceptable for a short duration.

The NRC staff has reviewed the licensee's analytical approach for evaluating the structural integrity of the SFP and the summary of the results, and has concluded that the structural integrity of the SFP structure under full fuel loading and SSE loading conditions is acceptable.

3.3 Auxiliary Systems

The SFP cooling and cleanup system (SFPCCS) which provides cooling to both SFPs consists of two cooling trains each primarily equipped with one pump and one heat exchanger. Heat is removed from the SFPCCS heat exchangers by the component cooling water system.

In the safety evaluation (SE) prepared by the NRC staff in support of Amendment No. 46 to License No. NPF-87 and Amendment No. 32 to License No. NPF-89 (Reference 8), the NRC staff stated that the licensee's thermal-hydraulic analysis was based on a planned full core off-load of 193 SFAs, 3,386 SFAs stored in the SFP, and a single failure of one cooling train. Under these conditions, the SFP bulk water was calculated to be 191 °F compared to CPSES UFSAR design value of 200 °F. With both cooling trains available, the licensee calculated the maximum bulk SFP temperature to be 139 °F. The NRC staff concluded that the design of the SFPCCS met the intent of the guidance described in SRP 9.1.3, "Spent Fuel Pool Cooling and Cleanup System," with regard to providing adequate cooling for the postulated spent fuel

inventory under planned and unplanned full core offload conditions and, therefore, was acceptable.

Per Amendment No. 72 (Reference 9), CPSES, Unit 2, was approved to operate at an increase of 1% in reactor power level, 3445 Megawatts Thermal (MWt) compared to the previous licensed power level of 3411 MWt. In the SE, to support its approval of Amendment No. 72, the NRC staff concluded that plant operations at the 1% increase in power would have an insignificant or no impact on SFPCCS.

In the submittal for this amendment request, the licensee stated that the decay heat for the bounding SFP thermal-hydraulic analysis had been updated to consider the effect of increasing the core thermal power to 3565 MWt for each unit, based on an assumed total SFP storage capacity of 3,386 SFAs. The increase in the reactor thermal power of both units will result in a slight increase of decay heat load in the SFP for any specific fuel discharge scenario. In Reference 3, the licensee stated that the calculated bulk pool temperature for maximum design heat load (planned full core off-load), with a single failure of one cooling train, increases from 191 °F to 195 °F; therefore, the conclusions of the previous bounding thermal analyses remain acceptable.

Based on it's review, the NRC staff concludes that the proposed increase of SFA storage capacity in the SFPs at CPSES does not change the design and the operational aspects of SFPCCS, and the calculated peak SFP water temperature of 195 °F remains below the design value of 200 °F.

With regard to air-handling systems, the SFP ventilation system is a nonsafety-related system designed to maintain a slight negative pressure during normal operations and during a fuel handling accident (FHA) to prevent the outflow of unfiltered, contaminated air to the environment. Operating the primary plant ventilation exhaust filter trains minimizes the release of radioactive particulate and radioiodine to the environment. As stated in Section 9.4.2.1 of the CPSES UFSAR, during emergency conditions (loss-of-coolant accident (LOCA) with a loss-of-offsite power) the ambient temperature in the SFPs heat exchanger and pump rooms shall be maintained below 122 °F, though the temperature may rise to 129 °F for a short duration. Sufficient air changes are provided to maintain the concentration of airborne radioisotopes throughout the area during fuel handling operations below the concentration levels specified in Appendix B to 10 CFR Part 20.

The licensee stated that the pump room coolers are safety-related and designed to maintain air temperatures in the pump rooms below 122 °F during normal and accident conditions. An evaluation performed by the licensee indicates that the FB heating, ventilation and air conditioning (HVAC) system is capable of maintaining the air temperature within specified limits, and that the increase in the SFPs heat loads will not affect the FB HVAC system during normal operation.

In the event of a loss of non-safety ventilation, the temperature and humidity in the FB will increase due to heat generated by storage of SFAs. The NRC staff concludes that these increases in temperature and humidity can be accommodated by the existing Class 1E equipment in accordance with the CPSES UFSAR.

Based on the NRC staff's review, it concludes that the FB HVAC system is adequate to accommodate the additional heat loads resulting from the proposed increase in spent fuel storage in the SFP.

3.4 Load Handling

The NRC staff's evaluation of the licensee's proposed increase in spent fuel storage capacity is focused on the movement of heavy loads associated with removal and installation of racks in SFPs 1 and 2, fuel movement within the SFPs, and movement of the gates that isolate the SFPs from the transfer canal. Considerations are given to the design and operation of the hoisting systems, safe load paths, and the potential impact and consequences of a load drop. The movement of a dry, spent fuel storage cask is not pertinent to the licensee's activities under consideration; therefore, it is not addressed in this SE.

3.4.1 Hoisting Systems

A heavy load, including the SFA and its handling tool, is defined as 2,150 lbs. The CPSES Technical Requirements Manual (TRM) states that loads in excess of 2,150 lbs. are prohibited from travel over fuel assemblies in the storage pools. The maximum lifted weight during spent fuel rack installation is 18,820 lbs., including the rack, the special lifting device, and rigging.

Both the FB overhead crane (130 tons) and the rack handling crane (a.k.a. the Wonder hoist), coupled with special lifting devices, will be used to handle the racks and the gates that isolate the pools from the transfer canal. The FB overhead crane will be used to lift and move racks received from the railway bay to the operating deck. The 20-ton Wonder hoist will be suspended from the bridge of the FB overhead crane and used in conjunction with the rack lifting rig (special lifting device) to lift and move the racks into and out of the SFPs. The use of the 20-ton Wonder hoist will allow the licensee to avoid contamination of the main hook of the FB overhead crane during rack movement in the pools.

The racks will be moved to/from the railway bay of the FB at Elevation (EL) 810 feet to/from the SFP operating deck at EL 860 feet. Appendix U of Reference 10 states that the FB overhead crane and the rack handling crane are designed, fabricated, installed, inspected, tested, and operated in accordance with requirements of the Crane Manufacturers Association of America Specification No. 70 (Reference 11). Both cranes are designed to American National Standards Institute (ANSI) B30.2-1967 (Reference 12) (also know as United States of America Standard (USAS)) instead of ANSI B30.2-1976 (same title, issued in 1976) as recommended in NUREG-0612 (Reference 13). Both cranes are designed to single-failure- proof criteria in accordance with Reference 13 and NUREG-0554 (Reference 14) via Regulatory Guide 1.104 (Reference 15). Reference 15 preceded Reference 14 and was subsequently incorporated into Reference 14.

The FB overhead crane is designed to seismic category I requirements; therefore, it is designed to maintain its structural integrity and the load under the dynamic loading conditions of a SSE. The single-failure-proof design feature enables the licensee to hold and retain the maximum design load in a stable and immobile safe position during an SSE under all postulated seismic loadings. The crane is equipped for manual lowering of the load during emergencies. Also, all the components in the load path of the crane, such as the hook, hoist rope, reeving, and braking mechanisms, either are redundant or have a large factor of safety.

The NRC staff, in Reference 10, approved the load handling systems, including the FB overhead crane and the rack handling crane for providing adequate protection against heavy load drops in accordance with Reference 13. The licensee states that the Region I rack modules will be lifted using a remotely engaged rack lifting rig that is specifically designed to lift the rack modules. The lifting rig is designed, tested to 300% of the maximum weight to be lifted, and inspected in accordance with the guidelines in Sections 5.1.6(1) and 5.1.6(3a) of Reference 13, and the requirements in ANSI N14.6 (1978) (Reference 16).

The lifting device complies with the provisions of Reference 16. It is redundantly designed with four independently loaded lift rods and configured such that failure of a single rod will not result in uncontrolled lowering of the rack. Also, the lifting rig has twice the design safety factor with respect to ultimate strength that is ten times the combined concurrent dynamic and static loads with a single lift point. To satisfy both the stress design and the load testing of the lifting rig, the lift rods are designed as follows: (1) with a stress design factor (as specified in Section 3.2.1 of Reference 16) of three times the combined weight to be lifted, plus the weight of the lifting device, without exceeding the minimum yield strength of the material; (2) five times the lifted weight without exceeding the ultimate strength of the material; and (3) load tested to 300% of the maximum weight to be lifted, hoisted, and suspended for a minimum of ten minutes. After load testing, the licensee will perform an examination of the critical weld joints using a liquid penetrant as is recommended in Section 6.3 of Reference 16.

The NRC staff concludes that the single-failure-proof design of the FB overhead crane and the rack handling crane, coupled with the rack lifting rig (special lifting device), facilitates safe movement of the racks. In addition, the design, inspection, and testing of the cranes and the special lifting device will help to assure safe handling of the racks with little to no risk of an accidental rack drop during rack installation.

3.4.2 Safe Load Paths

Spent fuel will be removed from SFP1 to SFP2 prior to removing the low-density racks from SFP1. Then, the existing low-density Region I racks in SFP1 will be removed from the pool via the rack handling crane and transported to a temporary platform that is located midway between SFP1 and SFP2 over the wet cask transfer area. Transfer of the racks to the FB overhead crane will occur at the temporary platform. The Region I racks are then loaded into special containers that are lowered from the operating level through the equipment hatch, to the railway bay where they are transferred to the processing facility. Conversely, the new high-density racks received in the railway bay are upended, using the FB overhead crane and the special lifting device, and hoisted to the operating level through the equipment hatch. The racks are then moved to the temporary platform over the wet cask transfer area where the load is transferred to the rack handling crane, and subsequently moved and placed into the SFP.

Fuel movement will be completed for each pool prior to the rack change-out. SFP2 fuel is to be moved to the east end of SFP2, and all the SFP1 fuel is to be moved into the east end of SFP2. Rack installation and fuel assembly storage will start and end in SFP2 before proceeding to SFP1 to avoid any potential to traverse a heavy load over fuel in the pool. As stated in the CPSES UFSAR, physical limitations (i.e., interlocks) on the FB overhead crane restricts the crane from passing over the SFP in any mode of operation. Administrative controls and travel limits on the rack handling crane (i.e., rail stops) enable the licensee to stay within the safe load paths, avoiding travel over fuel stored in the racks or over the pool, thus reducing the potential

for load drops on fuel in the pool. In its response to the NRC request for additional information (Reference 3), CPSES stated that the previously established safe load path for the FB overhead crane, as documented in a letter dated June 8, 1983, now includes the new fuel vault. This change in the safe load path has been identified and entered into the CPSES Corrective Action Program. CPSES committed to perform a risk assessment in accordance with the maintenance rule prior to making any heavy load lifts through the new fuel vault area.

Furthermore, in Reference 3, the licensee states that they will adhere to requirements in the TRM that restricts travel of heavy loads over fuel assemblies in the SFP. In addition, the crane and fuel bridge operators and the rack installation crews are to be trained in accordance with plant specific training programs and ANSI/ASME B30.2-1996 (Reference 17), as recommended in Reference 13.

Based on the above discussion, we find that the safe load paths for movement of the racks will be adequately controlled so as to avoid any movement of the racks over spent fuel in the SFP. The physical limitations of the FB overhead crane and the administrative and physical controls on the rack handling crane will enable the licensee to avoid travel over fuel in the racks or over the SFPs.

3.4.3 Analysis of Heavy Load Drop Accidents

The licensee analyzed postulated load drop accidents involving a SFA. The licensee did not analyze postulated load drop accidents involving the use of the FB overhead crane to move a rack. This is due to the single-failure-proof crane design capabilities to retain and hold the load. In addition, as stated in Reference 3, the Wonder-hoist is the only crane used to move SFP1 racks and is capable of travel over either SFP. Accordingly, the licensee states that the Wonder hoist will be administratively and physically restricted (through the use of rail stops) from moving racks over SFP2 during the removal of the existing low-density racks from SFP1. This will reduce the potential for a drop of an SFP1 rack over fuel stored in SFP2. Therefore, the potential for the drop of a rack on fuel during this re-rack evolution is considered an unlikely event.

The licensee also stated that the swing gates to SFP1 and SFP2 will be closed during installation and removal of the Wonder hoist system; therefore, a postulated drop of the crane system into the cask pit or the transfer canal that results in a tear in the SFP liner will not result in a loss of the SFP inventory or the SFP cooling function. The licensee also committed to adhere to CPSES's TRM which restricts heavy loads from travel over fuel assemblies in the storage pools.

Three accident scenarios involving the drop of a fuel assembly were considered: (1) straight vertical drops on top of the racks, (2) inclined drops on top of the racks (bounded by scenario 1), and (3) straight vertical drops to the bottom of the racks. The fuel assembly drops on top of the racks (scenario 1) resulted in localized deformation of the racks to a depth of 17 inches below the top of the rack, with no damage to the fuel. The inclined drop of a fuel assembly (scenario 2) resulted in less damage than in scenario 1. The fuel assembly drops through the cell onto the bottom of the racks (scenario 3) resulted in damage to the baseplate where the baseplate was lowered 2.14 inches. Although the SFP liner will experience some stresses due to the impact of the dropped fuel assembly on the baseplate, the SFP liner is not expected to fail. The underlying concrete slab of the SFPs is also expected to be subjected to

high strain rates during a drop under scenario 3. However, the concrete will not fail and there will be no abrupt loss of water from the pool.

Reference 13 recommends that licensees provide an adequate defense-in-depth approach to maintaining safety during the handling of heavy loads near spent fuel, and cited four major causes of accidents: operator errors, rigging failures, lack of adequate inspection, and inadequate procedures. The licensee stated that they will implement measures using administrative controls and procedures to preclude load drop accidents in these four areas. Accordingly, the licensee plans to: (1) provide comprehensive training to the re-rack installation crew in accordance with Reference 12; (2) use redundantly designed and adequately tested lifting rigs in accordance with Reference 16; (3) perform inspection and maintenance checks on the cranes, lifting devices, and racks themselves prior to and during the re-rack operation; and (4) use specific procedures that cover the entire re-rack effort, including the identification of required equipment, inspection, acceptance criteria prior to load movement, defining safe load paths, and steps and precautions for proper load handling and movement. In addition, the licensee discussed how the pattern of installing the racks will enable the racks to be lifted and inserted without travel over spent fuel in the SFPs during the rack installation operation.

The use of the single-failure-proof crane and the rack handling crane in conjunction with administrative procedures and controls, as discussed above, will enable the licensee to maintain safety during the movement of heavy loads.

The NRC staff has evaluated the drop of a heavy load from a structural standpoint. For scenario 1, the dynamic analysis shows that the top of the impacted rack undergoes localized plastic deformation. The maximum depth of plastic deformation is limited to 17 inches, which is below the design limit of 18.25 inches. The deep drop through an exterior cell does produce some deformation of the baseplate and localized severing of the baseplate/cell wall welds. The fuel assembly support surface is lowered by a maximum of 2.14 inches, which is much less than the distance of 7.5 inches from the baseplate to the liner. Therefore, the pool liner will not be contacted by the deformed baseplate. In response to a staff question about the allowable deformation of the baseplate, the licensee stated that, for scenario 3, the allowable deformation of the baseplate is not determined by its material strength characteristics; it is determined rather by the continued ability of the rack to store fuel in a subcritical configuration. Thus, the baseplate shall not be damaged or deformed to the extent that the criticality acceptance criteria are violated. To that end, the baseplate deformation must not: (a) lead to a gross structural collapse of the rack; or (b) lower the enriched zone of the fuel assembly far enough below the rack's "poisoned" region to violate subcriticality. The licensee has performed detailed analyses (Reference 3) to confirm that neither situation occurs as a result of the postulated deep drop event. The licensee further states that the damage to the rack is of a local nature, and the maximum fuel assembly lowering of 2.14 inches causes no significant changes to the reactivity.

The licensee states that, from a radiological standpoint, the increase in the spent fuel storage capacity does not change the physical consequences of a dropped SFA. For a postulated FHA occurring after the proposed spent fuel storage expansion, only the dropped assembly is damaged because the proposed Region I / Region II racks can withstand the drop of a SFA without any significant deformation that would affect other assemblies stored in the racks. The approved current FHA radiological analysis, as documented in Section 15.7.4 of the CPSES UFSAR, assumes all the rods in the dropped fuel assembly are ruptured; therefore, the assumed amount of damaged fuel remains the same as previously evaluated by the licensee.

In addition, the proposed re-racking and TS changes do not affect the other assumptions for the calculation of the radiological consequences of the postulated FHA outside containment. The licensee did not revise the current UFSAR FHA radiological analysis, which shows that the radiological consequences of postulated FHAs are well within the dose limits given in 10 CFR Part 100. The staff finds that, since the licensee's proposed changes do not affect the assumptions used in calculating the radiological consequences of design basis FHAs, the licensee's current UFSAR Section 15.7.4 FHA analysis remains bounding.

Based on the preceding discussions, the NRC staff concludes that the aforementioned considerations for the movement of heavy loads to increase in the SFP storage capacity are acceptable. The licensee's use of the 130-ton single-failure-proof FB overhead crane, the rack handling crane, the spent fuel rack lifting device, and administrative controls and procedures that are in accordance with References 13 and 16, will help to maintain safety during the installation of the new racks and, therefore, satisfy Appendix A to 10 CFR Part 50, GDC 61. The reliability of the crane, coupled with the design, testing and inspection of the crane, lifting rig, and other lifting devices will enable the licensee to handle safely the racks and other heavy loads during the rack installation process. The postulated accident analyses involving a dropped SFA indicated that the SFP liner would not be breached and the integrity of the SFP slab would be maintained. Therefore, a loss of SFP inventory is unlikely. Accordingly, the NRC staff concludes that the licensee's provisions for control of movement of heavy loads are acceptable.

3.5 Occupational Exposure

The NRC staff has reviewed the licensee's plan for the installation of the replacement rack modules in SFP1 and SFP2 with respect to occupational radiation exposure. A number of facilities have performed similar operations in the past. On the basis of lessons learned from these operations, the licensee estimates that the proposed rack installation can be performed for approximately 2.3 person-rem.

As indicated in Reference 3, all of the operations involved in the rack installation will utilize detailed procedures prepared with full consideration of as low as is reasonably achievable (ALARA) principles. Workers performing the SFP operations will be given pre-job briefings to ensure that they are aware of their job responsibilities and precautions associated with the job. The licensee will monitor and control work, personnel traffic, and equipment movement in the SFP area to minimize contamination and to assure that exposures are maintained ALARA. Personnel will wear protective clothing and respiratory protective equipment, if necessary. The licensee will issue thermoluminescent dosimeters (TLDs) or alarming electronic dosimeters to all personnel. Additional personnel monitoring equipment (such as multi-badging, and alarming and integrating devices) will be issued as required.

The licensee plans on using divers to assist in the removal and installation of the rack modules in the SFPs. Prior to any diving operations, the licensee will conduct radiation surveys of the diving area using two independent survey instruments. As part of these surveys, the licensee will verify the location of all SFAs and irradiated objects located in the SFPs. SFAs and irradiated objects may be shuffled prior to divers entering the SFPs in order to ensure that doses to divers are maintained ALARA during the diving operations; however, the licensee will not change the locations of any SFAs or irradiated objects in the SFPs while the divers are in the water. When practical, the licensee will use physical barriers to prevent divers from

accessing spent fuel elements or other high radiation items or areas. Divers will be equipped with a calibrated electronic alarming dosimeter that will alarm underwater to warn the diver of high dose rates or when they exceed a predetermined dose limit. The licensee may also equip the diver with an underwater detector with a readout on the surface so that the licensee can monitor underwater dose rates in the diver work area and notify the diver of any significant dose rate changes. Continuous radiation protection coverage will be provided at all times while a dive is in progress and divers will have continuous voice communication with surface personnel providing dive support. If needed, the licensee will provide additional overhead and underwater lighting to support re-racking operations and provide a safe working environment for the divers. The licensee will use underwater cameras to monitor the movements of the divers.

The existing SFP filtration system will be used to maintain pool clarity. An installed skimmer will be used to recirculate and clean the SFP surface water. The licensee will use an underwater vacuum system to supplement the SFP purification system. The removal of the low-density fuel racks from SFP1 could result in the generation of small amounts of crud and debris. The vacuum system may be employed to remove extraneous debris, reduce general contamination levels prior to diving operations, and to assist in the restoration of pool clarity following any pressure washing operations.

The dose rate at the surface of the SFP is due to the activity of the SFP water. Since the SFP water activity will not change significantly due to increased spent fuel storage, any changes in dose rates at the SFP surface will be negligible. The licensee has estimated that the anticipated dose rates at the edge of SFP1, at the SFP bridge, and in the general area will be approximately 1 mR/hr with the proposed increased spent fuel storage capacity. This is below the design limit for these areas of 2.5 mR/hr. In Reference 3, the licensee evaluated whether the increased storage capacity of the SFPs would necessitate radiation zoning changes to any of the accessible areas adjacent to the sides or bottoms of the SFPs. The licensee stated that there will be no need to change any existing radiation zones.

There is only one pipe chase room below SFP2 where the dose rates in the overhead area (at a height of 18 feet above the floor) might exceed 2.5 mR/hr for a short time if fresh SFAs were to be placed in the SFP adjacent to this room; however, this portion of the room ceiling area is not generally accessible without radiation protection involvement.

In order to control the spread of contamination, the licensee will spray the low-density racks with water upon removal from the SFP. After the low-density racks have been allowed to drip dry, they will be sealed, loaded into special containers meeting all appropriate Department of Transportation shipping regulations, and transferred to the processor’s facility for volume reduction and disposal. The licensee does not expect the concentrations of airborne radioactivity in the vicinity of the SFP to increase as a result of the expanded spent fuel storage capacity; however, the licensee will conduct airborne radioactivity surveys and will utilize continuous air monitors in all normally occupied plant areas where the likelihood for airborne radioactivity exists during the removal and installation of the rack modules in the SFPs.

The NRC staff concludes that the proposed increase in spent fuel storage capacity at CPSES, Units 1 and 2, can be performed in a manner that will ensure that doses to the workers will be maintained ALARA. The NRC staff concludes that the projected dose for the project of less than 3 person-rem is in the range of doses for similar modifications at other plants and is therefore acceptable.

3.6 Criticality Evaluation

The current CPSES TSs allow the use of 20 low-density racks in SFP1 with a nominal 16 inch center-to-center spacing between the fuel assemblies. In addition, the TSs also allow fuel assemblies to be stored in nine SFP2 Region II racks in either a 1/4 (a square array of four storage locations, one of which contains a fuel assembly) or a 2/4 storage configuration. CPSES has received NRC approval (Reference 18) to credit soluble boron to allow the storage of additional assemblies in SFP2 Region II in either the 3/4 or the 4/4 storage configuration. The nine new SFP 1 racks in Region II that will be installed by these amendments are very similar to the existing nine SFP2 racks in Region II.

3.6.1 Region I and Region II Criticality Analysis

The analyses of the reactivity effects of fuel storage in SFP1 and SFP2 were performed with the 3-D 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 library. Since the KENO-Va code package does not have burnup capability, depletion analyses and the determination of small reactivity increments due to manufacturing tolerances were made with the 2-D transport theory code, PHOENIX-P, which uses a 42 energy group nuclear data library. These codes are widely used for the analysis of rack reactivity and have been benchmarked against results from numerous critical experiments. These experiments simulate the CPSES racks as realistically as possible with respect to parameters important to reactivity such as enrichment and assembly spacing. These two independent methods of analysis (KENO-Va and PHOENIX-P) showed good agreement both with experiment and with each other. The intercomparison between different analytical methods is an acceptable technique for validating calculational methods for nuclear criticality safety. To minimize the statistical uncertainty of the KENO-Va calculations, a large number of neutron histories were accumulated in each calculation. Experience has shown that the use of a large number of histories enhances the convergence process of Monte Carlo calculations. The NRC staff concludes that the analytical methods used are acceptable and capable of predicting the reactivity of the CPSES racks with a high degree of confidence.

The Region I racks utilize boral as a neutron absorber for supplementary reactivity control and are designed to accommodate fuel with a maximum of 5 weight percent (w/o) U-235 enrichment. 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 high-density CPSES racks have been qualified for storage of various Westinghouse 17x17 fuel assembly types with maximum enrichment up to 5 w/o U-235. Other fuel assembly designs were evaluated to assure that fuel of the highest reactivity (bounding case) was used in the analyses.

The criticality analyses conducted by CPSES included postulated abnormal and accident conditions to assure that the CPSES racks will be safe under all credible conditions. Although there is boron in the SFP water, the criticality calculation was performed assuming an accident in which all the boron in the SFP water has been lost. The double contingency principle of ANSI N-16.1-1975 (Reference 19) and in Reference 5 allows credit for soluble boron under

other abnormal or accident conditions since only a single accident need be considered at one time. Increased water temperature and fuel misplacement (misloaded bundle) accidents were also considered.

The analysis for the Region I racks shows that the k_{eff} has a value of 0.9134. The additional biases for temperature and methodology and statistical uncertainties led to a 95/95 k_{eff} of 0.9258. Since this value is less than 1 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.

The Region II racks were analyzed with the intention of taking credit for soluble boron in the criticality calculation. The NRC acceptance criterion for criticality is that the effective neutron multiplication factor (k_{eff}) in the racks when fully flooded with unborated water shall be no greater than 0.95, including uncertainties at a 95/95 probability/confidence level, under all conditions. The criticality analyses were performed with several assumptions which tend to maximize the rack reactivity. These include:

(1) k_{eff} < 1 when flooded with unborated pool water at a 95/95 probability/confidence level as described in WCAP-14416-NP-A, Revision 1 (Reference 20)

(2) k_{eff} $\leq$ or equal to 0.95 when flooded with borated water, including all uncertainties at 95/95 level as described in Reference 20.

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 Region II racks have a nominal 9 inch center-to-center cell spacing. The high-density CPSES racks in Region II have been qualified for storage of various Westinghouse and Siemens Power Corporation 17x17 fuel assembly types with maximum enrichment up to 5 w/o U-235.

The licensee used soluble boron in the SFP criticality calculation to provide a safety limit margin by maintaining $k_{eff} \leq$ 0.95 including uncertainties. The soluble boron credit calculations assumed the SFP1 Region II (4/4) storage configuration to be moderated by water borated to 200 ppm. The resulting 95/95 k_{eff} with individual tolerances, uncertainties, temperature, and methodology biases included, was calculated to be 0.93531 for fuel enriched to 1.04 w/o U-235 in the SFP1 Region II (4/4) storage configuration. This value for k_{eff} meets the acceptance criteria for precluding criticality with credit for soluble boron. The concentration of soluble boron required to maintain k_{eff} is well below the minimum SFP boron concentration value of 2,000 ppm required by CPSES TS 3.7.16.

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

3.6.2 Reactivity Equivalence Analysis

The concept of reactivity equivalencing due to fuel burnup was used to define the conditions under which fresh and irradiated fuel assemblies are interchangeable on an overall reactivity basis. The NRC has previously accepted the use of reactivity equivalencing to equate an array of fresh fuel assemblies and their enrichments, which have been shown to be acceptable for storage, to an array of irradiated assemblies with different initial enrichments and burnable absorber concentrations. To determine the amount of soluble boron required to maintain $k_{eff} \leq 0.95$ for storage of fuel assemblies with enrichments higher than those acceptable for storage of fresh fuel assemblies, 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 racks. These calculations are shown in References 1 and 4, as part of TS 3.7.17-1 and 3.7.17-2 for the SFP1 Region II (4/4) and (3/4) storage configurations, respectively.

The amount of additional soluble boron (above the 200 ppm value required above) that is needed to account for these reactivity equivalencing 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. Adding this amount to the soluble boron credit of 200 ppm required for k_{eff} to be less than or equal to 0.95 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 values of 2,000 ppm required by TS 3.7.16.

Uncertainties associated with burnup credit include a reactivity uncertainty of 0.01 Δk at 30,000 mega-watt days per metric ton of uranium (MWD/MTU) applied linearly to the burnup credit requirement to account for calculational and depletion uncertainties. An uncertainty of 5 percent was also applied to the calculated burnup to account for burnup measurement uncertainty.

The NRC staff concludes that these uncertainties conservatively reflect the uncertainties associated with burnup calculations and are acceptable.

An evaluation of various fuel misloading accidents indicated that the misplacement of a fresh fuel assembly enriched to 5 w/o U-235 results in the highest reactivity increase. However, the minimum SFP boron concentration value of 2,000 ppm required by TS 3.7.16 is more than sufficient to maintain k_{eff} less than or equal to 0.95 for this reactivity increase. In fact, an additional 200 ppm of soluble boron is sufficient to maintain $k_{eff} \leq 0.95$ for this reactivity increase. By virtue of the double contingency principle, which has been endorsed by the NRC staff, two unlikely independent and concurrent events are beyond the scope of the required analysis; therefore, credit for the presence of the entire 2,000 ppm of soluble boron may be assumed in evaluating other accident conditions such as a fuel misloading.

3.6.3 Axial Burnup Credit Bias

The calculations for the burnup credit reactivity equivalencing are done on a radial, 2-D basis with the PHOENIX-P code. Inherent in a 2-D treatment for this calculation is a uniform axial burnup distribution. To account for the varying burnup and reactivity axially along

the assembly, that is, the 3-D burnup effect, a bias term had been defined in Reference 20, using the PHOENIX-P code and the Advanced Nodal Code (ANC).

A recent investigation (Reference 21) concluded that the methodology for calculating axial bias as described in Reference 4 could be nonconservative. Consequently, Westinghouse reevaluated the 2-D bias for the CPSES Region II (3/4 and 4/4) storage configurations. Westinghouse used axial burnup profiles from 3-D ANC reactor core models for recent operating cycles to determine the axial burnup effects. The 3-D depletion calculations were performed with control bank D inserted at 200 steps withdrawn and the burnup shapes were determined for various assembly average burnups. Westinghouse calculated the axial burnup bias using an infinite array (x-y direction) of racks and the PHOENIX-P/ANC codes. These calculations took into account top and bottom axial reflectors with 6 inches of pure water and reflective boundary conditions which essentially eliminated neutron leakage. The axial biases were determined by comparing the reactivity of the nonuniform axial burnup cases to those of the uniform axial burnup cases with the same models. The PHOENIX-P/ANC results were benchmarked against calculations using the Monte Carlo code KENO-Va to confirm the axial burnup bias results. The depletion calculations were performed for the highest enriched fuel (5.0 w/o) in each storage configuration. The use of the 5.0 w/o fuel assemblies results in the highest required burnup where the axial bias is expected to be most limiting. The NRC staff concludes that these calculations are acceptable; however, the NRC staff has notified Westinghouse in a letter dated July 27, 2001 (Reference 22), that, since the axial bias methodology as it is currently described in Reference 20, is known to be non-conservative, this section of Reference 20 is no longer valid. In addition, the staff informed Westinghouse that a revised version of Reference 20, should be submitted to resolve the above stated nonconservatisms.

Based on the review described above, the NRC staff concludes that the criticality aspects of the proposed increase in the storage capacity of the CPSES racks are acceptable and meet the requirements of GDC 62 for the prevention of criticality in fuel storage and handling. The analysis assumed credit for soluble boron, as allowed in Reference 20, but no credit for the Boral neutron absorber panels. The required amount of soluble boron for each analyzed storage configuration is shown in Table 8 of Reference 4. The criticality analysis conforms to the NRC guidance on the regulatory requirements for criticality analysis of fuel storage at light-water reactor power plants (Reference 23).

3.7 Changes to the Technical Specifications

The licensee has proposed changes to TS 3.7.17, “Spent Fuel Assembly Storage,” and TS 4.3, “Fuel Storage,” as follows:

TS 3.7.17, Spent Fuel Assembly Storage

Specification 3.7.17: The licensee has proposed revising TS 3.7.17 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 “high density” to “Region II” racks, and from “all cell” to “4/4.” 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 interface 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 ≤ 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 2,026 to 3,373 fuel assemblies.

The TS changes proposed as a result of the revised criticality analysis are consistent with the NRC-approved methodology given in Reference 20, with the exception of the axial bias treatment. The issues associated with the axial bias section of Reference 20 have been resolved on a plant specific bases. Based on the agreement with the approved portions of the methodology and additional supporting plant specific analysis, the NRC staff finds these TS changes acceptable. The proposed changes to the TS related to the increase in SFP storage capacity are acceptable in that the licensee has demonstrated the structural capability to support the additional loads. In addition, the required auxiliary systems are capable of limiting the temperature of the SFPs to acceptable levels. The remaining proposed changes to the TS are either nomenclature (e.g. changing “high density” to “Region II”) or grammatical and these changes are acceptable in that they do not change any regulatory requirements.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (66 FR 75737, published December 4, 2000). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

7.0 REFERENCES

1. Letter from C. L. Terry, TXU Electric, to U.S. NRC, 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,” October 4, 2000.

2. Letter from C. L. Terry, TXU Electric, to U.S. NRC, Subject: "Comanche Peak Steam Electric Station (CPSES) - Docket Nos. 50-445 and 50-446 - Supplement One to License Amendment Request (LAR) 00-05: Revision to Technical Specification Spent

Fuel Assembly Storage Racks and Fuel Storage Capacity (TAC Nos. MB0207 and MB0208)," April 30, 2001.

3. Letter from C. L. Terry, TXU Electric, to U.S. NRC, Subject: "Comanche Peak Steam Electric Station (CPSES) - Docket Nos. 50-445 and 50-446 - Response to Request for Additional Information and Supplement Two to License Amendment Request (LAR) 00-05: Revision to Technical Specification Spent Fuel Assembly Storage Racks and Fuel Storage Capacity (TAC Nos. MB0207 and MB0208)," June 18, 2001.

4. Letter from C. L. Terry, TXU Electric, to U.S. NRC, Subject: "Comanche Peak Steam Electric Station (CPSES) - Docket Nos. 50-445 and 50-446 - Supplement Three to License Amendment Request (LAR) 00-05: Revision to Technical Specification Spent Fuel Assembly Storage Racks and Fuel Storage Capacity (TAC Nos. MB0207 and MB0208)," July 18, 2001.

5. Nuclear Regulatory Commission, Letter to All Power Reactor Licensees, from B. K. Grimes, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978.

6. Nuclear Regulatory Commission, Letter to All Power Reactor Licensees, from B. K. Grimes, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," January 18, 1979.

7. Letter from C. L. Terry, TU Electric, to U.S. NRC, Subject: "Comanche Peak Steam Electric Station (CPSES) Docket Nos. 50-445 and 50-446 Submittal of License Amendment Request (LAR) 94-022 - Spent Fuel Capacity Increase," December 30, 1994.

8. Letter from T. Polich, U.S. NRC, to C. L. Terry, TU Electric, "Comanche Peak Steam Electric Station, Units 1 and 2 - Amendment Nos. 46 and 32 to Facility License Nos. NPF-87 and NPF-89 (TAC Nos. M91244 and M91245)," February 9, 1996.

9. Letter from D. Jaffe, U.S. NRC, to C. L. Terry, TXU Electric, "Comanche Peak Steam Electric Station (CPSES), Units 1 and 2 - Issuance of Amendments Re: Increase in CPSES, Unit 2 Thermal Power to 3445 Megawatts Thermal (TAC Nos. MA4436 and MA4437)," September 30, 1999.

10. NUREG-0797, Supplement No. 12, "Safety Evaluation Report Related to the Operation of Comanche Peak Steam Electric Station Units 1 and 2," October 1985.

11. Crane Manufacturers Association of America (CMAA) Specification No. 70, "Specifications for Electric Overhead Traveling Cranes," 1971.

12. USAS B30.2-1967, "Overhead and Gantry Cranes," 1967.

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

14. NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," May 1979.

15. Regulatory Guide 1.104, “Single-Failure-Proof Overhead Crane Handling Systems for Nuclear Power Plants,” Draft 3, Revision 1, October 1978 (Withdrawn, see NUREG-0554, see 44 FR 49321, August 22, 1979).

16. ANSI N14.6-1978, “Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials,” 1978.

17. ANSI/ASME B30.2-1996, “Overhead and Gantry Cranes (Top Running Bridge and Multiple Girder),” August 30, 1996.

18. Letter from D. Jaffe, U.S. NRC, to C. L. Terry, TXU Electric, “Comanche Peak Steam Electric Station (CPSES), Units 1 and 2 - Issuance of Amendments Re: Increase in Allowable Spent Fuel Storage Capacity and Credit for Soluble Boron in the Spent Fuel Pool (TAC Nos. MA4841 and MA4842),” February 24, 2000.

19. ANSI N-16.1-1975, “Nuclear Criticality Safety in Operation with Fissionable Materials Outside Reactors,” April 14, 1975.

20. Newmyer, W. D., “Westinghouse Spent Fuel Rack Criticality Analysis Methodology,” WCAP-14416-NP-A, Revision 1, November 1996.

21. Westinghouse letter WPT-16160, Revision 1, from J. S. Wyble to C. L. Terry “NSAL-00-015, Axial Burnup Shape Reactivity Bias,” dated December 6, 2000.

22. Letter from S. Dembek, U.S. NRC, to H. Sepp, Westinghouse Electric Company, “Non-conservativisms in Axial Burnup Biases for Spent Fuel Rack Criticality Analysis Methodology,” July 27, 2001.

23. NRC memorandum to T. Collins from L. Kopp, "Guidance on Regulatory Requirements for Criticality Analysis of Fuel Storage at Light -Water Reactor Plants," August 19, 1998.

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