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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 46 AND 32 TO

FACILITY OPERATING LICENSE NOS. NPF-87 AND NPF-89

TEXAS UTILITIES ELECTRIC COMPANY

COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-445 AND 50-446

1.0 INTRODUCTION

By application dated December 30, 1994, (TXX-94325) (Reference 1), Texas Utilities Electric Company (TU Electric/the licensee) requested changes to the Technical Specifications (Appendix A to Facility Operating License Nos. NPF-87 and NPF-89) for the Comanche Peak Steam Electric Station (CPSES), Units 1 and 2. A request for additional information (RAI) was issued by the NRC staff on May 23, 1995, and during subsequent conversations in October, November, and December 1995 the NRC staff requested additional information. As a result of the staff's request for additional information the licensee supplemented their initial request by letters dated July 28, (TXX-95187); September 14, (TXX-95235); and November 29, 1995, (TXX-95299); and January 2, 1996, (TXX-96-003) (References 2, 3, 4, and 5) respectively. These supplemental letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The proposed changes would revise Technical Specification (TS) 5.6, "Fuel Storage" to reflect installation of high density spent fuel pool storage racks in Spent Fuel Pool No. 2 (SFP2) and adopt the wording, content, and format of the Improved Standard Technical Specifications. The new racks would accommodate an increase in spent fuel assemblies beyond the storage capacity authorized for SFP2.

The current CPSES spent fuel storage configuration has 20 low density racks installed in Spent Fuel Pool No. 1 (SFP1) with a total storage capacity of 556 fuel assemblies. These racks provided adequate capacity for storage of spent fuel through the end of the fourth refueling outage for Unit 1, completed in the spring of 1995. To increase spent fuel storage capacity at CPSES, TU Electric will install nine free standing, high density spent fuel storage racks in SFP2. Although these high density racks originally included Boraflex neutron absorbing material, TU Electric elected to remove the Boraflex because of recent indications of Boraflex degradation at other storage facilities. The reracking will provide an ultimate storage capacity of 1291 assemblies (556 low density fuel assemblies in SFP1 and 735 high density fuel assemblies in SFP2).

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This evaluation addresses the adequacy of the criticality, control of heavy loads, thermal-hydraulics, and structural aspects of the TU Electric license amendment submittal to increase the spent fuel storage capacity at CPSES. The licensee provided additional clarification to the thermal-hydraulic portion of this amendment during a site visit the week of April 16, 1995, and during a conference call on October 6, 1995.

2.0 EVALUATION

2.1 Criticality

The low density racks will remain in SFP1 and provide a total storage capacity of 556 assemblies with a nominal 16-inch center-to-center spacing between assemblies. These racks have been previously found acceptable for unrestricted storage of Westinghouse 17x17 fuel assemblies enriched to a maximum 5.0 weight percent (w/o) U-235.

The high density racks to be installed in SFP2 contain 1470 total storage cell locations with a nominal 9.0-inch center-to-center spacing. These racks originally contained Boraflex as a neutron absorber. However, because of the reported Boraflex deterioration problems observed at other storage facilities, TU Electric has elected to remove the Boraflex and replace it with a spacer plate.

The analysis of the reactivity effects of fuel storage in SFP2 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 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 two-dimensional transport theory code, PHOENIX-P, which uses a 42 energy group nuclear data library. These codes are widely used for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. These experiments simulate the CPSES spent fuel 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 minimum of 60,000 neutron histories were accumulated in each calculation. Experience has shown that this number of histories is sufficient to assure convergence of KENO-Va reactivity calculations. The staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the CPSES storage racks with a high degree of confidence.

The NRC acceptance criterion for criticality is that the effective neutron multiplication factor (k_{eff}) in the spent fuel pool storage racks when fully flooded by 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) Unborated pool water moderation at a density of 1.0 g/cc.

(2) Assumption of infinite array of storage cells in all directions.

(3) Neutron absorption effect of structural material is neglected.

The design basis fuel assembly was a Westinghouse 17x17 Optimized Fuel Assembly (OFA). Calculations have shown that this is the most reactive fuel assembly design at CPSES for the maximum enrichment considered.

Based on the above, the staff concludes that appropriately conservative assumptions were made.

For the nominal storage cell design, uncertainties due to water temperature range, tolerances in cell lattice spacing, cell inner diameter, stainless steel thickness, and fuel enrichment and density were accounted for. These uncertainties were appropriately determined at least at the 95/95 probability/confidence level. In addition, a calculational bias and uncertainty were determined from benchmark calculations as well as an allowance for uncertainty in depletion calculations for those cases where burnup credit is used. These biases and uncertainties meet the previously stated NRC requirements and are, therefore, acceptable.

The licensee's analysis using the acceptable methods discussed above has shown that fresh fuel of 5.0 w/o U-235 nominal enrichment stored in a one-out-of-four (1/4) checkerboard configuration results in a maximum k_{eff} of 0.9379, including calculational and manufacturing uncertainties (95 percent/ 95 percent). This meets the staff's criterion of k_{eff} no greater than 0.95 including all uncertainties at the 95/95 probability/confidence level and is, therefore, acceptable. A 1/4 checkerboard arrangement with empty cells means that no two fuel assemblies may be stored face adjacent or corner adjacent.

Similar calculations have shown that fresh fuel assemblies with a nominal enrichment of 2.9 w/o U-235 stored in a two-out-of-four (2/4) checkerboard arrangement result in a maximum (95 percent/95 percent) k_{eff} of 0.9451. A 2/4 checkerboard arrangement with empty cells means that no two fuel assemblies may be stored face adjacent. They may, however, be stored corner adjacent.

In order to store fuel with nominal enrichment greater than 2.9 w/o U-235, but no greater than 5.0 w/o U-235, in a 2/4 checkerboard pattern, the concept of burnup reactivity equivalencing was used. This concept is based on the reactivity decrease associated with fuel depletion and has been previously found acceptable by the NRC for use in pressurized water reactor (PWR) fuel storage analysis. A series of reactivity calculations is performed to generate a set of enrichment versus burnup ordered pairs which yield an equivalent k_{eff} for fuel stored in the CPSES high density SFP2 racks. The results of these calculations indicate that a fresh 2.9 w/o fuel assembly yields the same rack reactivity as a nominally enriched 5.0 w/o assembly depleted to 16,500 MWD/MTU. In addition to the calculational and manufacturing biases and uncertainties previously described, an uncertainty associated with the burnup dependent reactivities computed with PHOENIX-P was accounted for in the reactivity equivalencing calculations. Based on the good agreement between PHOENIX-P predictions and measurements, the staff concludes that this uncertainty, which increases linearly from zero at zero burnup to 0.01 Δk at 30,000 MWD/MTU, is acceptable. This reactivity equivalencing method is the standard one used for storage rack reactivity evaluations and is acceptable.

Although not included in the burnup dependent criticality analyses, subsequent decay of Pu-241 with long-term storage results in a significant decrease in reactivity. This will provide an increasing subcriticality margin and further compensate for any uncertainty in the depletion calculations.

Most abnormal storage conditions will not result in an increase in the k_{eff} of the racks. However, it is possible to postulate events, such as the inadvertent misloading of an assembly with a burnup and enrichment combination outside of the acceptable areas in TS Figure 5.6-1, which could lead to an increase in reactivity. However, for such events credit may be taken for the presence of soluble boron in the pool water which is assured by administrative procedures during fuel handling operations since the staff does not require the assumption of two unlikely, independent, concurrent events to ensure protection against a criticality accident (Double Contingency Principle). The plant procedures require that the boron concentration in the pool be maintained between 2300 and 2500 ppm during operating modes, which is confirmed by weekly surveillance measurements. The reduction in k_{eff} caused by the boron more than offsets the reactivity addition caused by credible accidents. In fact, the licensee has confirmed that a minimum boron concentration of only 600 ppm boron would be adequate to assure that the limiting k_{eff} of 0.95 is not exceeded.

The following TS changes have been proposed as a result of the requested spent fuel pool reracking:

(1) TS 5.6.1 has been separated into two specifications. New TS 5.6.1.1 reflects the new requirements for fuel storage in Region 1 and Region 2 of the spent fuel pool. New TS 6.5.1.2 reflects the storage requirements for fresh fuel storage in the new (fresh) fuel storage racks.

(2) TS 5.6.3 has been modified to reflect the increased fuel pool storage capacity to 1291 fuel assemblies.

Based on the above evaluation in Section 2.1, the staff finds these changes as well as the associated Bases changes acceptable.

2.2 Control of Heavy Loads

SFP1 currently contains 20 low density racks with a total of 556 storage cells, 389 of which were occupied following the Unit 1 refueling outage in the spring of 1995. SFP2 currently contains no racks and is dry. The licensee will install nine high density, free standing, non-poisoned storage racks in

SFP2. The configuration and type of racks in SFP1 will not be affected. During the reracking, SFP2 will not contain spent fuel or other irradiated materials.

To accomplish the rack installations, the new racks will be lifted from the Fuel Building loading bay and placed on a temporary platform between the two spent fuel pools using the Fuel Building Overhead Crane. The Fuel Building Overhead Crane cannot travel over SFP1 or SFP2. In their licensing report, the licensee states that the Fuel Building Overhead Crane complies with the criteria for single-failure-proof cranes in accordance with the criteria presented in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," July 1980.

The licensee also commits to using a single-failure-proof handling system designed to meet the criteria of Section 5.1.6 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," July 1980, to transfer the racks from the temporary platform to SFP2. In their licensing report, the licensee maintains that the Rack Handling Crane (RHC) conforms to ANSI B30.2 and has a capacity of 30,000 lbs, which exceeds the heaviest rack weight of 20,600 lbs. The RHC will be installed onto the trolley rails currently used by the fuel handling bridge crane. Mechanical stops will be installed on the trolley rails to isolate the refueling bridge and SFP1 from the reracking process. The crane manufacturer will perform a rated load test and full performance test prior to shipment of the RHC and operational tests, which will include a lift test using the heaviest rack, will be performed by the licensee after installation and prior to its use.

Single-failure-proof cranes and associated lifting devices which conform to the criteria of NUREG-0554 and NUREG-0612 satisfy the guidance of Regulatory Guide 1.13 and Section 9.1.5 of the Standard Review Plan (SRP), NUREG-0800, and the requirements of the General Design Criteria 4 and 61 of Appendix A to 10 CFR Part 50 with regard to the design of heavy load handling systems. The staff finds the licensee has committed to employ an acceptable heavy loads handling system in the reracking process.

The licensee commits to employ operator training programs, crane inspection plans, mechanical stops, safe load paths, and use of specific procedures, which comply with the criteria in Section 5.1.1 of NUREG-0612. These plans and commitments are consistent with the approach of NUREG-0612 and the guidance of Section 9.1.5 of the SRP and are, therefore, acceptable.

Based on the above evaluation in Section 2.2, the staff finds the proposed changes as well as the associated Bases changes acceptable.

2.3 Thermal-Hydraulics

2.3.1 Spent Fuel Pool Cooling

The Spent Fuel Pool Cooling and Cleanup System (SFPCCS) is designed to remove the decay heat from the spent fuel that has been discharged from the station's two nuclear reactors to either of the site's two spent fuel pools. The SFPCCS is designed as a common system supporting both spent fuel pools. The system consists of two cooling loops, two purification loops, and one surface skimmer loop. Each cooling loop consists of a 3600 gpm pump, 13.6 Mbtu/hr heat exchanger, and associated piping and valves. A purification loop containing a demineralizer is provided for each cooling loop. Normally, one cooling loop is aligned to a pool to provide cooling to the spent fuel. The licensee is currently making modifications to the SFPCCS and associated support systems to allow the cooling loops to be cross-connected.

In order to evaluate the total decay heat load, an inventory of 2820 fuel assemblies accumulated through scheduled discharges was assumed to be present in the pools. In addition to heat load from this inventory, additional heat loads from the following three scenarios were evaluated: (1) Maximum Design Condition, (2) Maximum Summer Design Condition, and (3) Abnormal Maximum Design Condition. The Maximum Design Condition corresponds to a normal refueling in one unit with the other unit operating. This planned event typically occurs during the fall or the spring. The Maximum Summer Design assumes both units are operating during the hotter months of the year. The Abnormal Maximum Design Condition corresponds to an emergency core offload while the other unit is on line, and can happen at any time of the year. The heat load from the inventory of 2820 fuel assemblies was assumed to be constant for all calculations and a period of 4.5 years of full power operation was assumed for all stored fuel. Discharges were conservatively assumed to start 100 hours after plant shutdown at a rate of 3 assemblies per hour, completing the full core offload 168 hours after shutdown. Convective heat transfer and evaporative cooling were not credited in the licensee's analysis.

The licensee performed transient calculations to evaluate bulk pool temperatures under the previously stated assumptions. The most limiting design basis scenario with regard to bulk pool temperature was found to be a normal full core offload (Maximum Design Condition) of 193 fuel assemblies coincident with a single failure of one cooling train. The pools were assumed to contain the maximum inventory of spent fuel (2820 fuel assemblies) plus 94 assemblies recently discharge from the other unit for a total of 3107 assemblies. For this analysis, the licensee assumed 193 spaces remain available for a full core offload of the other unit, and 86 spare locations for a total inventory of 3386 (TS capacity is 1291). Under these conditions, the licensee calculated the bulk spent fuel pool temperature to be 191°F. The final safety analysis report (FSAR) specifies a design temperature of 200°F for SFP support system components, including the SFP purification system, and the demineralizer resin is rated at 140°F. Even though the bulk pool temperature will exceed the demineralizer resin's rated temperature, the licensee's calculations indicate that the inlet temperature to the SFPCCS purification loop will not exceed the demineralizer resin's rated temperature even under the Maximum Design Condition heat load assuming a single failure. With both cooling trains available under Maximum Design Conditions, the license calculated the maximum temperature to be 139°F.

Although long-term exposure of concrete structures to temperatures in excess of 150°F may result in damage to these structures, the staff does not consider this to be a concern under the Maximum Design Conditions. Based on the transient nature of the SFP temperature, the continuously decreasing decay heat load of the SFP inventory, the conservative approach of the calculation, and the heat transfer that exists through the concrete and through evaporation, the staff concluded that the temperature will not exceed 150°F for a period sufficient to cause structural damage.

Under the Abnormal Maximum Design Conditions, which assumes a heat load from a full core discharged to the spent fuel pools after a 150 hour decay period, 94 fuel assemblies with a 36 day decay period, 94 assemblies with a 66 day decay period, and the remainder of the pool filled with spent fuel from previous discharges, the bulk pool temperature was calculated to be 176°F. A single failure was not assumed coincident with this scenario.

Since the postulated maximum normal SFP bulk temperature has not been found to result in damage to structures or systems, the staff finds that the design of the SFPCCS complies with the guidance in Section 9.1.3 of the SRP with regard to providing adequate cooling for the postulated spent fuel inventory under normal full core offload operations. Likewise, the maximum SFP bulk temperature for the abnormal full core offload condition, assuming both trains of SFP cooling are in operation, was calculated to be below the temperature associated with the onset of bulk boiling and, therefore, meets the guidance of Section 9.1.3 of the SRP for adequate SFP cooling under abnormal conditions.

2.3.2 Decay Heat Calculation

The licensee stated that the previous analysis used heat loads calculated in accordance with NRC Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling," Rev. 2, July 1981, and included other conservative assumptions. The staff performed confirmatory decay heat load calculations to verify the conclusion made by the licensee. Results of our confirmatory calculations indicate that the licensee's decay heat calculations are conservative.

Based on our review and confirmatory calculations, we find the proposed changes that were based on the licensee's maximum decay heat calculations to be acceptable.

2.3.3 Effects of Boiling

The licensing report also evaluated the transient response of the SFP following a complete loss of all forced cooling resulting in the heat-up and eventual boiling of the SFP water. The calculated minimum time from the loss of pool cooling until the pool boils is in excess of 3 hours for the most severe scenario, with a boil off rate of 106 GPM. However, makeup water to the pool can be provided from the demineralized water supply system in excess of the maximum boil-off rate. In addition, the seismic Category I, Safety Class #3 Reactor Water Makeup System (RWMS) is available to provide makeup water to the SFP from either units' RWMS at 210 GPM, and the local fire protection stations can be aligned to the SFPs to provide an additional 126 GPM, if necessary. Therefore, the staff finds that the guidance of Section 9.1.3 of the SRP is met with regard to provision of makeup water.

2.3.4 Fuel Cladding Integrity

In order to verify cladding integrity is not threatened, the licensee developed a model to calculate the maximum local cladding temperature. The licensee's model assumed fuel assembly loading pattern that maximized the need for cooling by natural circulation. Any flow cell blockage under these conditions would have maximized the fuel assembly temperature due to the relationship between coolant velocity and the heat transfer coefficient. The results of the licensee's evaluation showed no boiling inside the cells with an assumed 80 percent flow blockage. The licensee considers complete blockage of the cell unlikely due to the storage rack's configuration of large or multiple flow openings.

The licensee also evaluated the effects of a complete loss of cooling where the pool was allowed to boil and makeup water was available to replace the pool inventory. The results of the licensee's evaluation indicated that, due to the effects of natural circulation, the fuel cladding temperature would remain sufficiently low to preclude structural failure.

Based on the above evaluation in Section 2.3, the staff finds the proposed changes as well as the associated Bases changes acceptable.

2.4 Structural Integrity

2.4.1 High Density Racks

The high density spent fuel storage racks are seismic Category I equipment, and are required to remain functional during and after a safe shutdown earthquake (SSE). TU Electric used a computer program, WECAN, for dynamic analysis to demonstrate the structural adequacy of the CPSES spent fuel rack design under earthquake loading conditions. The proposed spent fuel storage racks are free-standing and self-supporting equipment, and are not attached to the floor of the storage pool. A nonlinear dynamic model consisting of inertial mass elements, beam elements, stiffness elements, gap elements and friction elements, as defined in the program, were used to simulate three dimensional dynamic behavior of the rack and the stored fuel assemblies including frictional and hydrodynamic effects. The program calculated forces and displacements at the nodes, and then obtained the detailed stress field in the rack elements from the calculated nodal forces.

Two model analyses were performed: the 3-D single rack model analysis and the 3-D whole pool multi-rack (WPMR) analysis. For the 3-D single rack model analysis, a rack with the dimensions of 11 ft (width) x 14 ft (length) x 14 ft (height) was considered for the calculation of stresses and displacements. The rack was analyzed with two (fully and partially) loaded conditions and two different coefficients of friction (μ=0.2 and 0.8) between the rack and the pool floor to identify the worst case response for rack movement and for rack member stresses. In the WPMR model analysis, all nine (9) racks were considered to investigate the fluid-structure interaction effects between racks and pool walls as well as those among the racks.

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 time histories) were generated from the design response spectra defined in the FSAR. TU Electric demonstrated the adequacy of the single artificial time history set used for the seismic analyses by satisfying requirements of both enveloping design response spectra as well as matching a target power spectral density (PSD) function compatible with the design response spectra as discussed in SRP Section 3.7.1.

A total of eight (8) 3-D single rack model analyses were performed. The results of the analyses show that the maximum displacement of the rack is at the top corner and is about 0.28 inch assuring that there are no rack-to-wall or rack-to-rack impacts under the load combinations (Level A, B and D service limits). The analyses results indicate that there are large safety margins in the rack design against overturning as evidenced by the small rack movements and, thereby, the structural integrity and stability of the racks and fuel assemblies are maintained. In addition, 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 Boiler and Pressure Vessel Code (1980 edition), Section III, Subsection NF. The results show that all induced stresses under the load combinations are smaller than the corresponding allowable stresses specified in the ASME Code indicating that the rack design is adequate.

In the 3-D WPMR analyses, all nine racks were considered and were subjected to the load combinations. The results of the multi-rack analysis indicate that the calculated stresses on a rack are smaller than the corresponding allowable stresses in the ASME Code as shown in Table 11.1 (Reference 3). In addition, the results show that there are no rack-to-wall or rack-to-rack impacts as the result of a SSE; assuring that the structural integrity and stability of the racks are maintained.

TU Electric also calculated the weld stresses of the rack under the dynamic loading conditions. Table 4.1 of Reference 1 shows the ratio of the calculated weld stress with respect to the allowable stresses specified in the ASME Code. The calculated factors of safety are in the range of 1.13 to 1.95 indicating that the weld connection design of the rack is adequate.

Based on: (1) the TU Electric's comprehensive parametric study, (2) large factor of safety of the induced stresses and strains of the rack when they are compared to the corresponding allowables provided in the ASME Boiler and Pressure Vessel Code, and (3) TU Electric's overall structural integrity and stability conclusions supported by both single rack and multi-rack analyses, the staff concludes that the rack modules will perform their safety function and maintain their structural integrity under postulated loading conditions and, therefore, are acceptable.

However, it is quite likely that the racks will move during or after seismic events. Therefore, TU Electric is required to institute a surveillance program that inspects and maintains the originally installed rack gaps after the occurrence of an earthquake equivalent to or larger than an operating basis earthquake (OBE), if any occurs. In addition, if TU Electric finds any discrepancy after rack installation indicating that the as-built clearances between the storage racks and the spent fuel pool walls are less than those assumed in the analysis of Reference 1, TU Electric is required to perform a subsequent appropriate analysis and submit its analysis results for further NRC review.

2.4.2 Spent Fuel Storage Pool

The spent fuel pool structure is a reinforced concrete structure and is designed as a seismic Category I structure. The dimensions of the CPSES pool structure are approximately 30 feet wide, 40 feet long and 41.5 feet high with 6 feet thick reinforced concrete. The internal surface of the pool structure is lined with stainless steel to ensure water tight integrity.

The pool structure was analyzed by using the finite element computer program, ANSYS, to demonstrate the adequacy of the pool structure with fully loaded high density racks. The pool structure with the racks was subjected to the load combinations specified in the CPSES FSAR including thermal loadings.

The table (page 4-37 of Reference 1) shows the predicted factors of safety varying from 1.23 to 3.44 for the concrete walls and slab. In view of the calculated factors of safety, the staff concludes that the TU Electric pool structural analysis demonstrates the adequacy and integrity of the pool structure under full fuel loading, thermal loading and SSE loading conditions. Thus, the storage fuel pool design is acceptable.

2.4.3 Fuel Handling Accident

The following three refueling accident cases were evaluated by TU Electric: (1) drop of a fuel assembly through a 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) same drop as Case (2) except the impacting mass is at 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 staff reviewed the TU Electric's analyses results submitted (Reference 1), and concurs with its findings. They are acceptable based on the TU Electric's structural integrity conclusions supported by the parametric studies.

Although TU Electric demonstrated the structural integrity of the rack modules due to a drop of fuel assembly and control rod assembly onto the top and bottom of the rack structure, it is, however, quite likely that a liner would be damaged if a fuel assembly is dropped directly on the liner and cause leakage of water through the structurally failed liner. Therefore, TU Electric is required to establish a safe load path that will prevent or will not increase the probability of an accidental dropping of a fuel assembly onto the liner of the spent fuel pool structure.

Based on the above evaluation in Section 2.4, the staff finds the proposed changes as well as the associated Bases changes acceptable.

3.0 CONCLUSION

3.1 Criticality

Based on the review described above in Section 2.1, the staff finds the criticality aspects of the proposed increase in the storage capacity of the CPSES spent fuel pool storage racks are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling. Therefore, the proposed change to the CPSES TS 5.6.3 is acceptable with regard to criticality.

3.2 Control of Heavy Loads and Thermal-Hydraulics

The staff determined that the licensee's commitment to comply with the criteria of NUREG-0612 with regard to the control of heavy loads during the reracking is acceptable. The licensee's analysis demonstrated the adequacy of SFP cooling and makeup water systems in supporting the increased decay heat load permitted by the reracking process. The staff found the analysis acceptable in addressing the potential SFP thermal-hydraulic concerns. The licensee's evaluation of local cladding temperature provides additional assurance that SFP cooling is adequate to protect cladding integrity following the proposed reracking.

Based on the review described above in Sections 2.2 and 2.3, the staff finds the control of heavy loads and thermal-hydraulic aspects of the proposed SFP increase acceptable. The staff also found the proposed change to the CPSES TS 5.6.3 to be acceptable with regard to the total capacity of the spent fuel pools.

However, an issue associated with spent fuel pool cooling adequacy was identified in NRC Information Notice 93-83, "Potential Loss Spent Fuel Pool Cooling Following a Loss of Coolant Accident (LOCA)," October 7, 1993, and in a 10 CFR Part 21 notification, dated November 27, 1992. The staff is evaluating this issue, as well as broader issues associated with spent fuel storage safety, as part of the NRC generic issue evaluation process. If the generic review concludes that additional requirements in the area of spent fuel pool safety are warranted, the staff will address those requirements to the license under separate cover.

3.3 Structural Integrity

Based on the review and evaluation described above in Section 2.4 of the TU Electric's submittal (Reference 1), and additional information and analysis provided by TU Electric (References 2, 3, 4, and 5), the staff concludes that TU Electric's structural analysis and design of the spent fuel rack modules and the SFP structure are adequate to withstand the effects of the required loads. The analysis and design are in compliance with the current licensing basis set forth in the FSAR and applicable provisions of the SRP and therefore, are acceptable provided that TU Electric commits (1) to implement a surveillance program that inspects and maintains the originally installed rack gaps after the occurrence of an earthquake equivalent to or larger than an OBE, (2) to submit analysis results for NRC review if any discrepancy is found after rack installation that the as-built clearances between the storage racks and the spent fuel pool walls are less than those assumed in the analysis of Reference 1, and (3) to establish a safe load path that will prevent or will not increase the probability of an accidental dropping of a fuel assembly onto the liner of the spent fuel pool structure.

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

Pursuant to 10 CFR 51.21. 51.32, and 51.35, an environmental assessment and finding of no significant impact was published in the Federal Register on February 9, 1996 (61 FR 5042). Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

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.

Principal Contributors: Laurence Kopp
Christopher Gratton
Yong Kim

Date: February 9, 1996