October 24, 1991

Docket No. 50-445

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Mr. William J. Cahill, Jr.
Executive Vice President
TU Electric
400 North Olive Street, L.B. 81
Dallas, Texas 75201

Dear Mr. Cahill:

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION, UNIT 1 - AMENDMENT NO. 6 TO FACILITY OPERATING LICENSE NO. NPF-87 (TAC NO. M80510)

The Commission has issued the enclosed Amendment No. 6 to Facility Operating License No. NPF-87 for the Comanche Peak Steam Electric Station, Unit 1. The amendment consists of changes to the Technical Specifications in response to your application dated May 24, 1991, as supplemented by letters dated July 30, 1991, September 23, 1991, and October 21, 1991.

The amendment changes the Appendix A Technical Specifications (TS) by replacing the radial peaking factor (Fxy) surveillance with a heat flux hot channel factor (F_Q) surveillance, revising the TS to allow operation with a positive moderator temperature coefficient (PMTC), and implementing Generic Letter 88-16, which allowed the removal of cycle-specific parameter limits from the TS with a reference to a Core Operating Limits Report for the values of those limits.

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,

Original Signed by

Thomas A. Bergman, Acting Project Manager
Project Directorate IV-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 6 to NPF-87
2. Safety Evaluation

cc w/enclosures:
See next page
*For previous concurrences see attached ORC

OFC	:PDIV-2/LA	:PDIV-2/PM*	:OGC*	:PDIV-2/D*	:	:	:
NAME	:EPeyton	:TBergman	:MYoung	:SBlack	:	:	:
DATE	:10/22/91	:10/9/91	:10/09/91	:10/18/91	:	:	:

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

TEXAS UTILITIES ELECTRIC COMPANY, ET AL.*

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT 1

DOCKET NO. 50-445

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 6
License No. NPF-87

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

   A. The application for amendment by Texas Utilities Electric Company (TU Electric) acting for itself and as agent for Texas Municipal Power Agency (licensees) dated May 24, 1991, as supplemented by letters dated July 30, 1991, September 23, 1991, and October 21, 1991, 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.

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

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. <u>Technical Specifications</u>

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 6, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of the date of issuance to be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Suzanne C. Black

Suzanne C. Black, Director
Project Directorate IV-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 24, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 6

FACILITY OPERATING LICENSE NO. NPF-87

DOCKET NO. 50-445

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

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UNIDENTIFIED LEAKAGE

1.37 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

1.38 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee, for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTING

1.39 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

WASTE GAS HOLDUP SYSTEM

1.40 A WASTE GAS HOLDUP SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System offgases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

CORE OPERATING LIMITS REPORT

1.41 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.6. Unit operation within these operating limits is addressed in individual specifications.

TABLE 1.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
SR	At least once per 9 months.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.

SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% Δk/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1.3% Δk/k, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7,000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% Δk/k:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and

b. At least once per 24 hours by consideration of the following factors:

   1) Reactor Coolant System boron concentration,

   2) Control rod position,

   3) Reactor Coolant System average temperature,

   4) Fuel burnup based on gross thermal energy generation,

   5) Xenon concentration, and

   6) Samarium concentration.

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the CORE OPERATING LIMITS REPORT (COLR). The maximum upper limit shall be less than or equal to $+0.5 \times 10^{-4}$ $\Delta k/k/^{\circ}F$ for power levels up to 70% RATED THERMAL POWER with a linear ramp to 0 $\Delta k/k/^{\circ}F$ at 100% RATED THERMAL POWER.

APPLICABILITY: Beginning of Cycle Life (BOL) limit - MODES 1 and 2* only**.
End of Cycle Life (EOL) limit - MODES 1, 2, and 3 only**.

ACTION:

a. With the MTC more positive than the BOL limit specified in the COLR, operation in MODES 1 and 2 may proceed provided:

   1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the BOL limit specified in the COLR within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

   2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

   3. A Special Report is prepared and submitted to the Commission, pursuant to Specification 6.9.2, within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the EOL limit specified in the COLR, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.

**See Special Test Exceptions Specification 3.10.3.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit specified in the COLR prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading; and

b. The MTC shall be measured at any THERMAL POWER and compared to the 300 ppm surveillance limit specified in the COLR (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than the 300 ppm surveillance limit specified in the COLR, the MTC shall be remeasured, and compared to the EOL MTC limit specified in the COLR, at least once per 14 EFPD during the remainder of the fuel cycle.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to 551°F.

APPLICABILITY: MODES 1 and 2* #.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 551°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 551°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 561°F with the T_{avg}-T_{ref} Deviation Alarm not reset.

---

*With K_{eff} greater than or equal to 1.

#See Special Test Exceptions Specification 3.10.3.

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration in the water,

2) Verifying the indicated borated water volume of the water source, and

3) Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is either less than 40°F or greater than 120°F.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All shutdown and control rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With one rod trippable but inoperable due to causes other than addressed by ACTION a., above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within 1 hour:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

   3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions;

      b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours;

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours; and

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

c. With more than one rod trippable but inoperable due to a rod control urgent failure alarm or obvious electrical problem in the rod control system, POWER OPERATION may continue provided that:

1. Within 1 hour, the remainder of the rods in the bank(s) with the inoperable rods are aligned to within ± 12 steps of the inoperable rods while maintaining the rod sequence and insertion limits of Specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, and

2. The inoperable rods are restored to OPERABLE status within 48 hours.

d. With more than one rod misaligned from its group step counter demand height by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the rod position deviation monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Decrease in Reactor Coolant Inventory

- Inadvertent opening of a pressurizer safety or relief valve
- Break in instrument line or other lines from reactor coolant pressure boundary that penetrate containment
- Steam generator tube rupture
- Loss of coolant accidents resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary

Increases in Heat Removal by the Secondary System (steam system piping failure)

Spectrum of Rod Cluster Control Assembly Ejection Accidents

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual (shutdown and control) rod drop time from the physical fully withdrawn position shall be less than or equal to 2.4 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 551°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the drop time of any rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods, and

c. At least once per 18 months.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With a maximum of one shutdown rod inserted beyond the insertion limit specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

a. Restore the rod to within the insertion limit specified in the COLR, or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be within the insertion limit specified in the COLR:

a. Within 15 minutes prior to withdrawal of any rods in Control Bank A, B, C, or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

---

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

#With K_{eff} greater than or equal to 1.

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1* and 2* **.

ACTION:

With the control banks inserted beyond the insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2:

a. Restore the control banks to within the limits within 2 hours, or

b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the insertion limits specified in the COLR, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the rod insertion limit monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

**With K_{eff} greater than or equal to 1.

(DELETED)

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the target band (flux difference units) about the target flux difference. The target band is specified in the CORE OPERATING LIMITS REPORT (COLR).

The indicated AFD may deviate outside the required target band at greater than or equal to 50% but less than 90% of RATED THERMAL POWER provided the indicated AFD is within the Acceptable Operation Limits specified in the COLR and the cumulative penalty deviation time does not exceed 1 hour** during the previous 24 hours.

The indicated AFD may deviate outside the required target band at greater than 15% but less than 50% of RATED THERMAL POWER provided the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

APPLICABILITY: MODE 1, above 15% of RATED THERMAL POWER.*

ACTION:

a. With the indicated AFD outside of the required target band and with THERMAL POWER greater than or equal to 90% of RATED THERMAL POWER, within 15 minutes either:

   1. Restore the indicated AFD to within the target band limits, or

   2. Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

---

* See Special Test Exceptions Specification 3.10.2.

**Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1 provided the indicated AFD is maintained within the Acceptable Operation Limits specified in the COLR. A total of 16 hours operation may be accumulated with the AFD outside of the required target band during testing without penalty deviation.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. With the indicated AFD outside of the required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours or outside the Acceptable Operation Limits specified in the COLR and with THERMAL POWER less than 90% but equal to or greater than 50% of RATED THERMAL POWER:

   1. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes, and

   2. Reduce the Power Range Neutron Flux - High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

c. With the indicated AFD outside of the required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours and with THERMAL POWER less than 50% but greater than 15% of RATED THERMAL POWER, the THERMAL POWER shall not be increased equal to or greater than 50% of RATED THERMAL POWER until the indicated AFD is within the required target band.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

   1) At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

   2) At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

SURVEILLANCE REQUIREMENTS

4.2.1.2 The indicated AFD shall be considered outside of its target band when two or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the above required target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined in conjunction with the measurement of $F_Q^C(Z)$ as defined in Specification 4.2.2.2e. The provisions of Specification 4.0.4 are not applicable.

(DELETED)

POWER DISTRIBUTION LIMITS

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{F_Q^{RTP}}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq \left[\frac{F_Q^{RTP}}{0.5}\right] [K(Z)] \text{ for } P \leq 0.5$$

Where: F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

$K(Z)$ = the normalized $F_Q(Z)$ as a function of core height specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ that exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower N-16 Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ that exceeds the limit; and

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

(DELETED)

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2. $F_Q(Z)$ shall be evaluated to determine if it is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Determining the computed heat flux hot channel factor, $F_Q^C(Z)$ as follows:

Increase the measured $F_Q(Z)$ obtained from the power distribution map by 3% to account for manufacturing tolerances and further increase the value by 5% to account for measurement uncertainties.

c. Verifying that $F_Q^C(Z)$, obtained is Specification 4.2.2.2b. above, satisfies the relationship in Specification 3.2.2.

d. The $F_Q^C(Z)$ obtained in 4.2.2.2b above shall satisfy the following relationship at the time of the taget flux determination:

$$F_Q^C(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{P \times W(Z)} \qquad \text{for } P > 0.5$$

$$F_Q^C(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{0.5 \times W(Z)} \qquad \text{for } P \leq 0.5$$

where $F_Q^C(Z)$ is obtained in Specification 4.2.2.2b. above, F_Q^{RTP} is the F_Q limit, K(Z) is the normalized $F_Q(Z)$ as a function of core height, P is the fraction of RATED THERMAL POWER, and W(Z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP}, K(Z) and W(Z) are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.9.1.6.

e. Measuring $F_Q(Z)$ according to the following schedule:

1. Upon achieving equilibrium condition after exceeding by 20% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(Z)$ was last determined*, or
2. At least once per 31 Effective Full Power Days, whichever occurs first.

*Power level may be increased until the THERMAL POWER for extended operation has been achieved.

SURVEILLANCE REQUIREMENTS (Continued)

f. With measurements indicating

$$\underset{\text{over } Z}{\text{maximum}} \left(\frac{F_Q^C(Z)}{K(Z)} \right)$$

has increased since the previous determination of $F_Q^C(Z)$ either of the following actions shall be taken:

1) Increase $F_Q^C(Z)$ by 2% and verify that this value satisfies the relationship in Specification 4.2.2.2d, or

2) $F_Q^C(Z)$ shall be measured at least once per 7 Effective Full Power Days until two successive maps indicate that

$$\underset{\text{over } Z}{\text{maximum}} \left(\frac{F_Q^C(Z)}{K(Z)} \right) \text{ is not increasing.}$$

g. With the relationships specified in Specification 4.2.2.2d above not being satisfied:

1) Calculate the percent that $F_Q(Z)$ exceeds its limits by the following expression:

$$\left\{ \left(\underset{\text{over } Z}{\text{maximum}} \left[\frac{F_Q^C(Z) \times W(Z)}{\frac{F_Q^{RTP}}{P} \times K(Z)} \right] \right) -1 \right\} \times 100 \text{ for } P > 0.5$$

$$\left\{ \left(\underset{\text{over } Z}{\text{maximum}} \left[\frac{F_Q^C(Z) \times W(Z)}{\frac{F_Q^{RTP}}{0.5} \times K(Z)} \right] \right) -1 \right\} \times 100 \text{ for } P \leq 0.5, \text{ and}$$

2) The following action shall be taken.

Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the AFD limits specified in the CORE OPERATING LIMITS REPORT by 1% AFD for each percent $F_Q(Z)$ exceeds its limits as determined in Specification 4.2.2.2g.1. Within 8 hours, reset the AFD alarm setpoints to these modified limits.

SURVEILLANCE REQUIREMENTS (Continued)

h. The limits specified in Specification 4.2.2.2c are applicable in all core plans regions, i.e. 0 - 100%, inclusive.

i. The limits specified in Specifications 4.2.2.2d, 4.2.2.2f, and 4.2.2.2g, above are not applicable in the following core plane regions:

1. Lower core region from 0 to 15%, inclusive.
2. Upper core region from 85 to 100%, inclusive.

4.2.2.3 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2 an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq F^{RTP}_{\Delta H}[1.0 + PF_{\Delta H} \ (1.0 - P)]$$

Where:

$F^{RTP}_{\Delta H}$ = the $F^N_{\Delta H}$ limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

$PF_{\Delta H}$ = the power factor multiplier for $F^N_{\Delta H}$ specified in the COLR, and

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

APPLICABILITY: MODE 1.

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Within 2 hours either:

1. Restore $F^N_{\Delta H}$ to within the above limit, or
2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. Within 24 hours of initially being outside the above limit, verify through incore flux mapping that $F^N_{\Delta H}$ has been restored to within the above limit, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated, through incore flux mapping, to be within its limit prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,
2. A nominal 75% of RATED THERMAL POWER, and
3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no loading operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.6% $\Delta k/k$ is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, a SHUTDOWN MARGIN of 1.3% $\Delta k/k$ provides adequate protection and is based on the results of the boron dilution accident analysis.

Since the actual overall core reactivity balance comparison required by 4.1.1.1.2 cannot be performed until after criticality is attained, this comparison is not required (and the provisions of Specification 4.0.4 are not applicable) for entry into any Operational Mode within the first 31 EFPD following initial fuel load or refueling.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC . lues at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC) was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections

MODERATOR TEMPERATURE COEFFICIENT (Continued)

involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting End of Cycle Life (EOL) MTC value. The 300 ppm surveillance limit MTC value represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting EOL MTC value.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within it analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.6% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 15,700 gallons of 7000 ppm borated water from the boric acid storage tanks or 70,702 gallons of 2000 ppm borated water from the refueling water storage tank (RWST).

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the minimum DNBR in the core greater than or equal to 1.30 during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods; and

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of the F_Q limit specified in the CORE OPERATING LIMITS REPORT (COLR) times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

AXIAL FLUX DIFFERENCE (Continued)

Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1-hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits specified in the COLR while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density

HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

The heat flux hot channel factor $F_Q(Z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to Constant Axial Offset Control (CAOC) operation, W(Z), to provide assurance that the limit on the heat flux hot channel factor, $F_Q(Z)$, is met. W(Z) accounts for the effects of normal operation transients within the AFD band and was determined from expected power control maneuvers over the range of burnup conditions in the core. The W(Z) function is provided in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.6.

When $F^N_{\Delta H}$ is measured, an adjustment for measurement uncertainty must be included for a full-core flux map taken with the Incore Detector Flux Mapping System.

$F_Q(Z)$ should be measured with the reactor core at, or near, equilibrium conditions. Therefore, the effects of transient maneuvers, such as power increases, should be permitted to decay to the extent possible while assuring that flux maps are taken in accordance with the specified surveillance schedules.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient. The indicated T_{avg} value of 592.7°F (conservatively rounded to 592°F) and the indicated pressurizer pressure value of 2207 psig correspond to analytical limits of 594.7°F and 2193 psig respectively, with allowance for measurement uncertainty. The indicated uncertainties assume that the reading from four channels will be averaged before comparing with the required limit.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation, and to detect any significant flow degradation of the Reactor Coolant System (RCS).

The additional surveillance requirements associated with the RCS total flow rate are sufficient to ensure that the measurement uncertainties are limited to 1.8% as assumed in the Improved Thermal Design Procedure Report for CPSES.

Performance of a precision secondary calorimetric is required to precisely determine the RCS temperature. The transit time flow meter, which uses the N-16 system signals, is then used to accurately measure the RCS flow. Subsequently, the RCS flow detectors (elbow tap differential pressure detectors) are normalized to this flow determination and used throughout the cycle.

ANNUAL REPORTS (Continued)

b. The results of specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.7. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration (μCi/gm) and one other radioidine isotope concentration (μCi/gm) as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.3 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR 50.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**

6.9.1.4 The Semiannual Radioactive Effluent Release Report covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR 50.

MONTHLY OPERATING REPORTS

6.9.1.5 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the Director, Office of Resource

*A single submittal may be made for a multiple unit station.

**A single submittal may be made for a multi-unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

MONTHLY OPERATING REPORTS (Continued)

Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.6 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle for the following:

a. Moderator temperature coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

b. Shutdown Rod Insertion Limit for Specification 3/4.1.3.5,

c. Control Rod Insertion Limits for Specification 3/4.1.3.6,

d. AXIAL FLUX DIFFERENCE Limits and target band for Specification 3/4.2.1.,

e. Heat Flux Hot Channel Factor, K(Z), W(Z), and F_Q^{RTP} for Specification 3/4.2.2,

f. Nuclear Enthalpy Rise Hot Channel Factor Limit and the Power Factor Multiplier for Specification 3/4.2.3.

The analytical methods used to determine the core operating limits shall be those previously approved by the NRC in:

WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary). (Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2. - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT," September 1974 (W Proprietary). (Methodology for Specification 3.2.1. - Axial Flux Difference, [Constant Axial Offset Control].)

T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC January 31, 1980--Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package. (Methodology) for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)

NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981. (Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)

CORE OPERATING LIMITS REPORT (Continued)

WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION," June 1983 (W Proprietary). (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor (W (z) surveillance requirements for F_Q Methodology).)

WCAP-8200, "WFLASH, A FORTRAN-IV COMPUTER PROGRAM FOR SIMULATION OF TRANSIENTS IN A MULTI-LOOP PWR," Revision 2, July 1974 (W Proprietary). (Methodology for Specification 3.2.2. - Heat Flux Hot Channel Factor.)

WCAP-9220-P-A, "Westinghouse ECCS Evaluation Model, February 1978 Version," February 1978 (W Proprietary). (Methodology for Specification 3.2.2. - Heat Flux Hot Channel Factor.)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety;

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 6 TO FACILITY OPERATING LICENSE NO. NPF-87

TEXAS UTILITIES ELECTRIC COMPANY, ET AL.

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT 1

DOCKET NO. 50-445

1.0 INTRODUCTION

By application dated May 24, 1991, as supplemented by letters dated July 30, 1991, September 23, 1991, and October 21, 1991, Texas Utilities Electric Company (the licensee) requested changes to the Technical Specifications (Appendix A to Facility Operating License No. NPF-87) for the Comanche Peak Steam Electric Station, Unit No. 1. The proposed changes would modify the Appendix A Technical Specifications (TS) by replacing the radial peaking factor (Fxy) surveillance with a heat flux hot channel factor (F_Q) surveillance, revising the TS to allow operation with a positive moderator temperature coefficient (PMTC), and implementing Generic Letter 88-16, which allowed the removal of cycle-specific parameter limits from the TS with a reference to a Core Operating Limit Report (COLR) for the values of those limits. The proposed changes also include the definition of the COLR to the Definitions section of the TS, and to the reporting requirements of the Administrative Controls sections of the TS. The July 30, 1991, September 23, 1991, and October 21, 1991, submittals provided additional clarifying information to the original May 24, 1991, application and did not change the initial no significant hazards consideration determination.

Guidance on the proposed changes was developed by the NRC on the basis of the review of a lead-plant proposal submitted on the Oconee plant docket by Duke Power Company. This guidance was provided to all power reactor licensees by Generic Letter 88-16, dated October 4, 1988 (Ref. 4).

2.0 EVALUATION

The licensee's proposed changes to the TS are in accordance with the guidance provided by Generic Letter 88-16 and are addressed below.

a. The Definition section of the TS was modified to include a definition of the COLR that requires cycle/reload-specific parameter limits to be established on a unit specific basis in accordance with NRC approved methodologies that maintain the limits of the safety analysis. The definition notes that plant operation within these limits is addressed by individual specifications.

b. The following specifications were revised to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits.

(1) Specification 3.1.1.3 and Surveillance Requirement 4.1.1.3

The moderator temperature coefficient (MTC) limits for this specification and for this surveillance requirement are specified in the COLR.

The impact of the positive MTC on the LOCA and non-LOCA accident analyses presented in the Comanche Peak Unit 1 Final Safety Analysis Report (FSAR) has been assessed. Those events that were found to be sensitive to a positive MTC were reanalyzed. These transient events included the control rod assembly withdrawal from subcritical, control rod assembly withdrawal at power, loss of reactor coolant flow, locked rotor, turbine trip, control rod ejection, and RCS depressurization events.

Westinghouse has performed these analyses to support the proposed TS change to allow a +5pcm/°F MTC below 70 percent of rated power, ramping down to 0 PCM/°F at 100 percent. The staff has previously approved both Shearon Harris and Callaway PMTC specifications similar to those proposed in this request. We have reviewed and found it acceptable.

(2) Specification 3.1.3.5 and Surveillance Requirement 4.1.3.5

The shutdown rod insertion limit for this specification and for this surveillance requirement is specified in the COLR.

(3) Specification 3.1.3.6

The control rod insertion limits for this specification is specified in the COLR.

(4) Specification 3.2.1

The axial flux difference limits and the target band for this specification are specified in the COLR.

(5) Specification 3.2.2 and Surveillance Requirement 4.2.2

The heat flux hot channel factor F_Q limit at rated thermal power, the normalized F_Q limit as a function of core height K(z), and a cycle dependent function W(z) to account for power distribution transient encountered during normal operation for this specification and for this surveillance are specified in the COLR.

(6) Specification 3.2.3

The nuclear enthalpy rise hot channel factor ($F^N_{\Delta H}$) for this specification is specified in the COLR.

The bases of affected specifications have been modified by the licensee to include appropriate reference to the COLR. Based on our review, we conclude that the changes to these bases are acceptable.

b. Specification 6.9.1.6 is revised to delete a previous reporting requirement on Peaking Factor Limit Report and to add the Core Operating Limits Report to the Administrative Controls section of the TS including currently proposed TS changes and NRC approved methodologies to support the values of cycle-specific parameter limits that are applicable for the current fuel cycle. The approved methodologies are the following:

(1) WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 (W Proprietary).

(Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control]).

(2) WCAP-8385, "Power Distribution Control and Load Following Procedures - Topical Report," September 1974 (W Proprietary).

(Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control]).

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

(Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control]).

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

(Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control]).

(5) WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification," June 1983 (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor [W(z) surveillance requirements for F_Q Methodology]).

(6) WCAP-8200, "WFLASH, A FORTRAN-IV Computer Program for Simulation of Transients in a Multi-Loop PWR," Revision 2, June 1974 (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).

(7) WCAP-9220-P-A, "Westinghouse ECCS Evaluation Model, February 1978 Version," February 1978 (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).

Items (2) WCAP-8385 and (6) WCAP-8200 have been approved by the staff (Refs. 5 and 6).

Finally, the specification requires that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted upon issuance to NRC.

On the basis of the review of the above items, the NRC staff concludes that the licensee provided an acceptable response to those items as addressed in the NRC guidance in Generic Letter 88-16 on modifying cycle-specific parameter limits in TS. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using NRC approved methodologies, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds that the proposed changes are acceptable.

We have reviewed the request by the TU Electric Company to modify the Technical Specifications of the Comanche Peak Steam Electric Station, Unit 1 that would remove the specific values of some cycle-dependent parameters from the specifications and place the values in a Core Operating Limits Report that would be referenced by the specifications. Based on this review, we conclude that these Technical Specification modifications are acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (56 FR 47242). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). This amendment also involves changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, with respect to these items, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

REFERENCES

1. Letter (TXX-91179) from William J. Cahill (TU Electric) to NRC, dated May 24, 1991.

2. Letter (TXX-91287) from William J. Cahill (TU Electric) to NRC, dated July 30, 1991.

3. Letter (TXX-91350) from William J. Cahill (TU Electric) to NRC, dated September 23, 1991.

4. Generic Letter 88-16, "Removal of Cycle-specific Parameter Limits from Technical Specifications," dated October 4, 1988.

5. Letter from John F. Stolz (NRC) to C. Eicheldinger (W), dated January 31, 1978.

6. Letter from D. B. Vassallo (NRC) to C. Eicheldinger (W), dated May 30, 1975.

Principal Contributor: T. Huang, SRXB/DST

Date: October 24, 1991