Mr. William T. Cottle
President and Chief Executive Officer
STP Nuclear Operating Company
South Texas Project Electric
Generating Station
P. O. Box 289
Wadsworth, TX 77483

September 29, 1998

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 - AMENDMENT NOS. 97 AND 84 TO FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80 (TAC NOS. MA0350 AND MA0351)

Dear Mr. Cottle:

The Commission has issued the enclosed Amendment Nos. 97 and 84 to Facility Operating License Nos. NPF-76 and NPF-80 for the South Texas Project, Units 1 and 2 (STP). The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated December 31, 1997, as supplemented by letters dated June 30, August 6, August 18, and August 27, 1998.

The amendments revise TS 2.1 (Safety Limits), 2.2 (Limiting Safety System Settings), and 3/4.2.5 (Departure from Nucleate Boiling Parameters) by including alternate operating criteria to allow continued plant operation with a reduced measured reactor coolant system flow rate, if necessary.

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 W. Alexion, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosures:
1. Amendment No. 97 to NPF-76
2. Amendment No. 84 to NPF-80
3. Safety Evaluation

cc w/encls: See next page

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

September 29, 1998

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cc w/encls: See next page

Mr. William T. Cottle
STP Nuclear Operating Company

South Texas, Units 1 & 2

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

STP NUCLEAR OPERATING COMPANY

DOCKET NO. 50-498

SOUTH TEXAS PROJECT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 97
License No. NPF-76

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

   A. The application for amendment by STP Nuclear Operating Company* acting on behalf of itself and for Houston Lighting & Power Company (HL&P), the City Public Service Board of San Antonio (CPS), Central Power and Light Company (CPL), and City of Austin, Texas (COA) (the licensees), dated December 31, 1997, as supplemented by letters dated June 30, August 6, August 18, and August 27, 1998, 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.

---

*STP Nuclear Operating Company is authorized to act for Houston Lighting & Power Company (HL&P), the City Public Service Board of San Antonio, Central Power and Light Company and City of Austin, Texas and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

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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-76 is hereby amended to read as follows:

   2. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 97 , and the Environmental Protection Plan contained in Appendix B, 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 its date of issuance to be implemented prior to startup from the October 1998 refueling outage for Unit 2.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas W. Alexion

Thomas W. Alexion, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 29, 1998

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

STP NUCLEAR OPERATING COMPANY

DOCKET NO. 50-499

SOUTH TEXAS PROJECT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 84
License No. NPF-80

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

   A. The application for amendment by STP Nuclear Operating Company* acting on behalf of itself and for Houston Lighting & Power Company (HL&P), the City Public Service Board of San Antonio (CPS), Central Power and Light Company (CPL), and City of Austin, Texas (COA) (the licensees), dated December 31, 1997, as supplemented by letters dated June 30, August 6, August 18, and August 27, 1998, 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.

---

*STP Nuclear Operating Company is authorized to act for Houston Lighting & Power Company (HL&P), the City Public Service Board of San Antonio, Central Power and Light Company and City of Austin, Texas and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

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-80 is hereby amended to read as follows:

   2. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 84, and the Environmental Protection Plan contained in Appendix B, 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 its date of issuance to be implemented prior to startup from the October 1998 refueling outage for Unit 2.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas W. Alexion

Thomas W. Alexion, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 29, 1998

ATTACHMENT TO LICENSE AMENDMENT NOS. 97 AND 84

FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80

DOCKET NOS. 50-498 AND 50-499

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
iii	iii
2-1	2-1
---	2-2(A)
2-4	2-4
2-7	2-7
2-8	2-8
2-9	2-9
2-10	2-10
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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1, or in Figure 2.1-2 when operating under alternate operating criteria consistent with reduced Reactor Coolant System flow as addressed in Technical Specification 3.2.5.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, AND 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

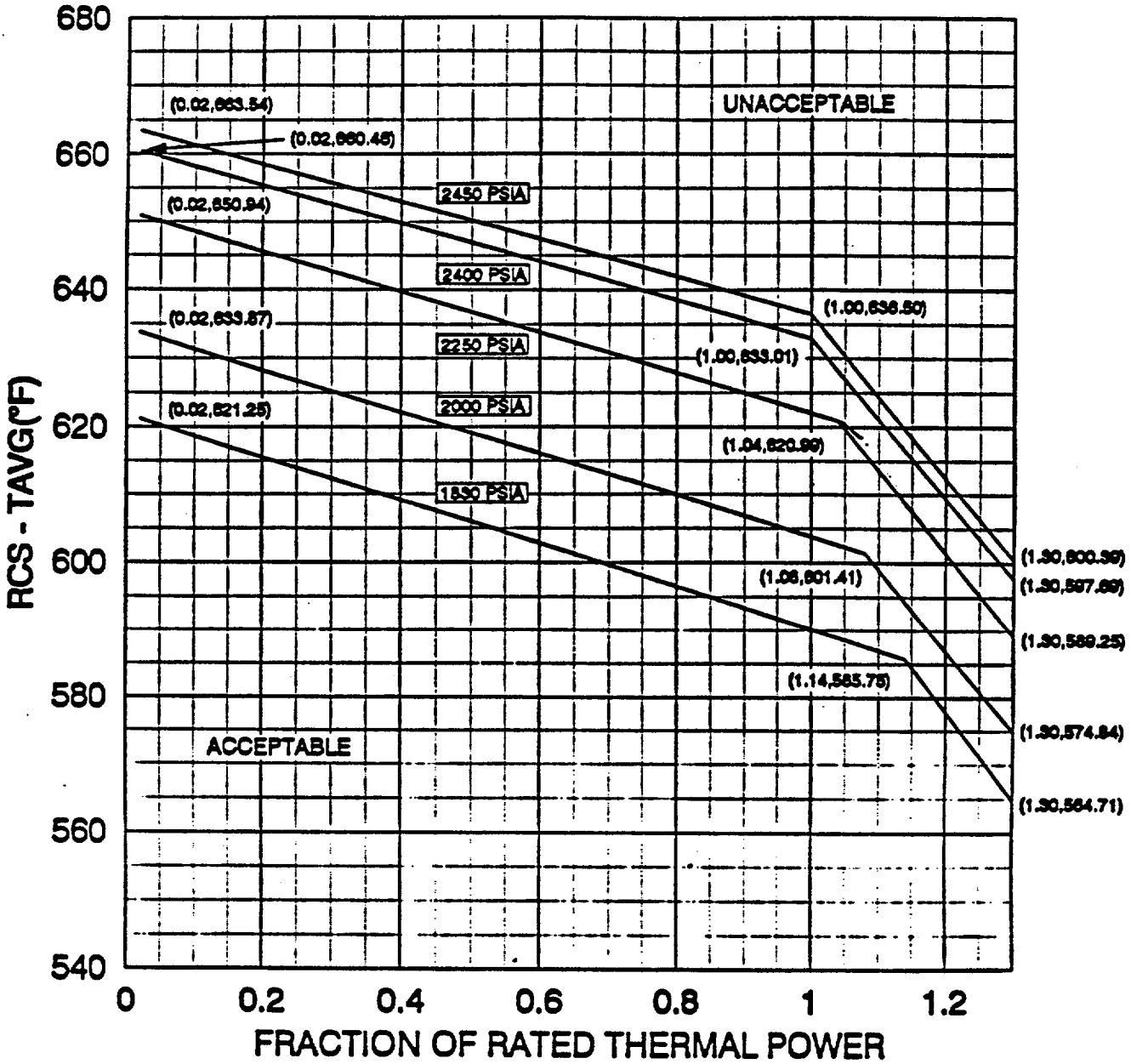

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

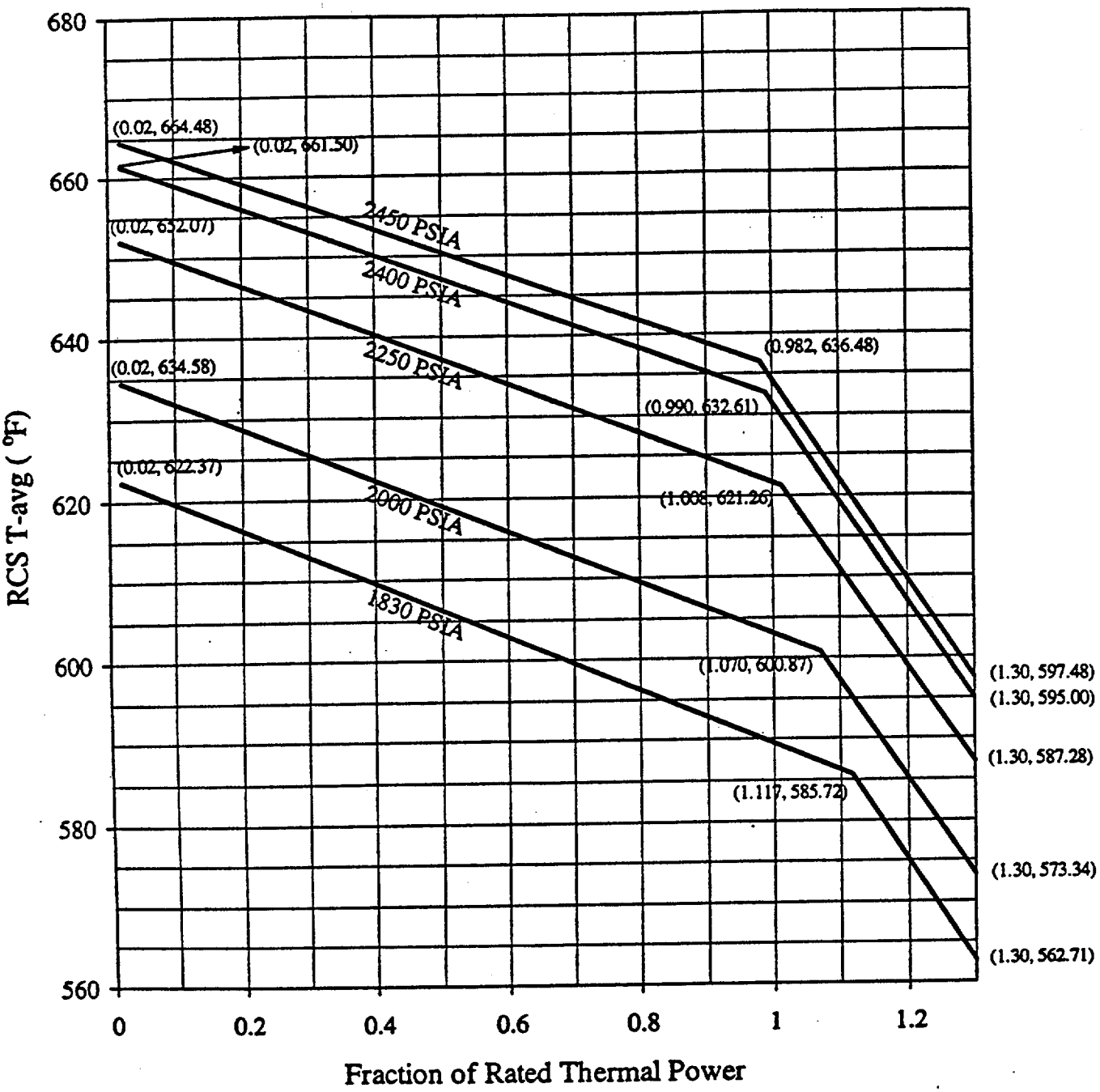

Figure 2.1-2

Reactor Core Safety Limit - Four Loops in Operation (Alternate)

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1, adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Value column of Table 2.2-1, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

Equation 2.2-1 $$Z + R + S \leq TA$$

Where:

Z = The value from Column Z of Table 2.2-1 for the affected channel,

R = The "as-measured" value (in percent span) of rack error for the affected channel,

S = Either the "as-measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 2.2-1 for the affected channel, and

TA = The value from Columh TA (Total Allowance) of Table 2.2-1 for the affected channel.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip N.A.	N.A.	N.A.		N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	6.1	0	≤109% of RTP**	≤110.7% of RTP**
b. Low Setpoint	8.3	6.1	0	≤25% of RTP**	≤27.7% of RTP**
3. Power Range, Neutron Flux, High Positive Rate 2.1	0.5	0		≤5% of RTP** with a time constant ≥2 seconds	≤6.7% of RTP** with a time constant ≥2 seconds
4. Deleted					
5. Intermediate Range, Neutron Flux 16.7	8.4	0		≤25% of RTP**	≤31.1% of RTP**
6. Source Range, Neutron Flux	17.0	10.0	0	≤10^5 CPS	≤1.4 X 10^5 cps
7. Overtemperature ΔT	10.7	8.7	1.5 + 1.5#	See Note 1	See Note 2
8. Overpower ΔT	4.7	2.1	1.5	See Note 3	See Note 4
9. Pressurizer Pressure-Low	5.0	2.3	2.0	≥1870 psig	≥1860 psig
10. Pressurizer Pressure-High	5.0	2.3	2.0	≤2380 psig	≤2390 psig
11. Pressurizer Water Level-High	7.1	4.3	2.0	≤92% of instrument span	≤94.1% of instrument span
12. Reactor Coolant Flow-Low	4.0	2.1	0.6	≥91.8% of loop design flow*	≥90.5% of loop design flow*

* Loop design flow = 95,400 gpm (or 92,500 gpm for alternate operation with reduced RCS flow)
**RTP = RATED THERMAL POWER
1.5% span for ΔT; 1.5% span for Pressurizer Pressure

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \left(\frac{1 + \tau_1 S}{1 + \tau_2 S}\right)\left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_1 - K_2 \left(\frac{1 + \tau_4 S}{1 + \tau_5 S}\right)\left(T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right) + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:

ΔT	= Measured ΔT by RCS Instrumentation;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	= Lead-lag compensator on measured ΔT;
τ_1, τ_2	= Time constant utilized in lead-lag compensator for ΔT, $\tau_1 = 8$ sec, $\tau_2 = 3$ sec;
$\frac{1}{1 + \tau_3 S}$	= Lag compensator on measured ΔT;
τ_3	= Time constant utilized in the lag compensator for ΔT, $\tau_3 = 0$ sec;
ΔT_0	= Indicated ΔT at RATED THERMAL POWER;
K_1	= 1.14, or 1.13 for alternate operation with reduced RCS flow;
K_2	= 0.028/°F;
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	= The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	= Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_4 = 28$ sec, $\tau_5 = 4$ sec;
T	= Average temperature, °F;
$\frac{1}{1 + \tau_6 S}$	= Lag compensator on measured T_{avg};
τ_6	= Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ sec;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T'	≤ 593.0°F (Nominal T_{avg} at RATED THERMAL POWER) OR 590.0°F for alternate operation with reduced RCS flow;
K_3	= 0.00143/psig;
P	= Pressurizer pressure, psig;
P'	= 2235 psig (Nominal RCS operating pressure);
S	= Laplace transform operator, sec^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For $q_t - q_b$ between -70% and + 8%, or +6% for alternate operation with reduced RCS flow, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds -70%, the ΔT Trip Setpoint shall be automatically reduced by 0.0% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds +8%, or +6% for alternate operation with reduced RCS flow, the ΔT Trip Setpoint shall be automatically reduced by 2.65% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.6% ΔT span.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \left(\frac{1 + \tau_1 S}{1 + \tau_2 S}\right)\left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right)\left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left(T \left(\frac{1}{1 + \tau_6 S}\right) - T''\right) - f_2(\Delta I)\right\}$$

Where:

ΔT = As defined in Note 1.

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = As defined in Note 1.

τ_1, τ_2 = As defined in Note 1.

$\frac{1}{1 + \tau_3 S}$ = As defined in Note 1.

τ_3 = As defined in Note 1.

ΔT_0 = As defined in Note 1.

K_4 = 1.08, or 1.07 for alternate operation with reduced RCS flow.

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature.

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation.

τ_7 = Time constant utilized in the rate-lag compensator for T_{avg}, τ_7 = 10 sec.

$\frac{1}{1 + \tau_6 S}$ = As defined in Note 1.

τ_6 = As defined in Note 1.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6 = 0.002/°F for T > T" and K_6 = 0 for T ≤ T",

T = As defined in Note 1,

T" = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤ 593.0°F, or ≤590.0°F for alternate operation with redcued RCS flow),

S = As defined in Note 1, and

$f_2(\Delta I)$ = 0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.9% ΔT span.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the limits following:

a. Reactor Coolant System T_{avg}, ≤ 598°F
(or ≤ 595°F with reduced RCS flow of 3.2.5.c)

b. Pressurizer Pressure, > 2189 psig*

c. Reactor Coolant System Flow, ≥ 392,300 gpm**
(or ≥380,500 gpm** with reduced RCS T_{avg} of 3.2.5.a)

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters shown above shall be verified to be within its limits at least once per 12 hours. The provisions of Specification 4.0.4 are not applicable for verification that RCS flow is within its limit.

4.2.5.2 The RCS flow rate indicators shall be subjected to a channel calibration at least once per 18 months.

4.2.5.3 The RCS total flow rate shall be determined by precision heat balance measurements at least once per 18 months. The provisions of Specification 4.0.4 are not applicable.

---

* Limit not applicable during either a Thermal Power ramp in excess of 5% of RTP per minute or a Thermal Power step in excess of 10% RTP.

**Includes a 2.8% flow measurement uncertainty.

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 Radial Peaking Factor, $F_{xy}(Z)$, is measured periodically to provide assurance that the Hot Channel Factor, $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as provided in the Core Operating Limits Report (COLR) per Specification 6.9.1.6 was determined from expected power control manuevers over the full range of burnup conditions in the core.

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. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

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

3/4.2.5 DNB PARAMETERS (Continued)

initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of greater than or equal to the design limit throughout each analyzed transient. The T_{avg} value of 598°F and the pressurizer pressure value of 2189 psig are analytical values. The readings from four channels will be averaged and then adjusted to account for measurement uncertainties before comparing with the required limit. The flow requirement (392,300 gpm) includes a measurement uncertainty of 2.8%.

Technical Specification 3.2.5 provides for an alternate minimum measured Reactor Coolant System flow limit consistent with plugging up to 10% of steam generator tubes and Departure from Nucleate Boiling requirements. When using the alternate minimum flow limit, the T_{avg} limit is reduced to 595°F for Reactor Coolant System flow no less than 380,500 gpm. Setpoint and constant values for OPΔT and OTΔT are also revised accordingly when this alternate mode of operation is entered.

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 97 AND 84 TO

FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80

STP NUCLEAR OPERATING COMPANY

DOCKET NOS. 50-498 AND 50-499

SOUTH TEXAS PROJECT, UNITS 1 AND 2

1.0 INTRODUCTION

By application dated December 31, 1997, as supplemented by letters dated June 30, August 6, August 18, and August 27, 1998, STP Nuclear Operating Company, et.al., (STPNOC, the licensee) requested changes to the Technical Specifications (TSs) (Appendix A to Facility Operating License Nos. NPF-76 and NPF-80) for the South Texas Project, Units 1 and 2 (STP). The proposed changes would revise TS 2.1 (Safety Limits), 2.2 (Limiting Safety System Settings), and 3/4.2.5 (Departure from Nucleate Boiling Parameters) by including alternate operating criteria to allow continued plant operation with a reduced measured reactor coolant system flow rate, if necessary.

The June 30, August 6, August 18, and August 27, 1998, supplements provided clarifying information and did not change the initial no significant hazards consideration determination.

2.0 DISCUSSION

The following proposed TS changes are related to the reduction in RCS flow rate for alternate operation:

1. Addition of Figure 2.1-2, Reactor Core Safety Limit-Four Loops in Operation (Alternate) to account for reduction in TS RCS flow.

2. Change to Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints as follows because of reduction of TS RCS flow rate:

   a. For Functional Unit 12, Reactor Coolant Flow - Low trip setpoint, by noting that the applicable Reactor Coolant System loop design flow is 92,500 gpm;

   b. For Functional Unit 7, Overtemperature ΔT ($OT\Delta T$), the values in Note 1 are modified for K_1, T', and $f_1(\Delta I)$ offset and gain;

   c. For Functional Unit 8, Overpower ΔT ($OP\Delta T$), the values in Note 3 are modified for K_4 and T";

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3. The Limiting Condition for Operation 3.2.5.(a) is revised by adding 595 °F as an alternate lower Reactor Coolant system T_{avg} limit; and

4. The Limiting Condition for Operation 3.2.5.(c) is revised by adding 380,500 gpm as an alternate lower Reactor Coolant System flow.

TS Bases Section 3/4.2.5 is also to be revised to reflect the alternate departure from nucleate boiling (DNB) criteria.

3.0 BACKGROUND

TS 3.2.5 for STP requires that RCS flow be maintained at or above the minimum measured flow of 392,300 gpm, which is equivalent to the thermal design flow of 381,600 gpm plus 2.8% to account for flow measurement uncertainties. Units 1 and 2 have the potential to fall below this minimum flow limit following a worst case scenario of 10% steam generator tube plugging (current licensing basis limit) combined with the effect of hot leg streaming on the precision calorimetric flow measurement. If this minimum measured flow cannot be met within two hours, the TS requires that the unit be in Mode 2 within the next four hours. In order to minimize the impact on continued plant operation for this scenario, STPNOC proposes to allow a 3% reduction in thermal design flow to 370,000 gpm, or a minimum measured flow of 380,500 gpm (with 2.8% added to account for flow measurement uncertainties), on an as needed basis, while maintaining a full power level of 3817 megawatts-thermal (MWt).

Power capability parameters for a Nuclear Steam Supply System (NSSS) power level of 3817 MWt, based on a 3% reduction in thermal design flow, were evaluated for four cases, using RCS T_{avg} values of 583.2°F (proposed lower bound) and 590 °F (proposed upper bound), and steam generator tube plugging levels of zero and 10%. The increase of 0.9°F in the lower bound from 582.3°F to 583.2°F allows the minimum core inlet design temperature from the T-hot Reduction Program to be maintained at 549.0 °F. These parameters were used to evaluate the operation of Units 1 and 2 at 3817 MWt with the proposed 3% reduction in minimum measured RCS flow. All of the accident analyses previously evaluated in the Updated Final Safety Analysis Report (UFSAR) for STP were reevaluated to determine the effect of this 3% flow reduction.

4.0 REACTOR CORE SAFETY LIMITS, REACTOR TRIP SETPOINTS, AND NON-LOSS OF COOLANT ACCIDENT (LOCA) AND LOCA SAFETY ANALYSIS

4.1 Evaluation

Core safety limits and axial offset limits were recalculated based on alternative operation with a 3% reduction in minimum measured RCS flow. These were incorporated into an alternate TS Figure 2.1-2 for reactor core safety limits when operating under the reduced flow conditions. Available DNB margin and the reduction in the upper end of the nominal T_{avg} range (593 °F to 590 °F) were used in the recalculation of the OTΔT and OPΔT reactor trip setpoints due to the revised safety limits, resulting in a reduction of approximately 1% in the margin to trip. However, the margin between the nominal and maximum values of the reference setpoints (i.e.,

K_1 and K_4 for the OTΔT and the OPΔT setpoints, respectively) were maintained. Therefore, the instrument accuracy uncertainties are unaffected by the change. The revised OTΔT and OPΔT setpoints were calculated using methods reviewed and approved by the NRC as documented in WCAP-8745-P-A.

The f(ΔI) function is used in the OTΔT trip equation and is currently defined for STP such that when the percent power difference between the top and bottom halves of the core is less than +8%, f(ΔI) is zero. If the percent power difference between the top and bottom halves of the core exceeds +8%, the trip setpoint is automatically reduced by 2.65% of its value at rated thermal power. In order to reflect the setpoint changes, the onset of the OTΔT setpoint reduction was lowered to an axial offset of greater than +6% for alternate operation with reduced RCS flow. Since the axial offset is not expected to be as high as +6% during normal full power operation, this reduction should not affect plant operability.

The UFSAR transient and accident analyses were evaluated by the licensee using the 3.0% reduction in flow for the events below.

Event	UFSAR Section
Feedwater Malfunction at Hot Full Power Conditions	15.1.1 - 15.1.2
Excessive Load Increase	15.1.3
Loss of Load/Turbine Trip	15.2.2 - 15.2.5
Reactor Coolant System Depressurization	15.6.1
Rod Withdrawal at Power	15.4.2
Steamline Break	15.1.5
Loss of Flow	15.3.1 - 15.3.2
Locked Rotor	15.3.3 - 15.3.4
Rod Withdrawal from Subcritical	15.4.1
Dropped Rod	15.4.3
Steamline Depressurization	15.1.4
Feedwater Malfunction at Hot Zero Power Conditions	15.1.2
Loss of Normal Feedwater/Station Blackout	15.2.6 - 15.2.7
Feedline Break	15.2.8
Startup of an Inactive Reactor Coolant Loop	15.4.4
Chemical & Volume Control System Malfunction	15.5.2
Boron Dilution	15.4.6
Rod Ejection	15.4.8
Inadvertent Emergency Core Cooling System Actuation	15.5.1
Large Break LOCA	15.6.5
Small Break LOCA	15.6.5
Hot Leg Switchover	6.3.2
Post-LOCA Long-Term Core Cooling	Cycle specific calculation
LOCA Hydraulic Forces	Used as input into analyses contained in Chapter 3
Rod Ejection Mass & Energy Releases	15.4.8
Steam Generator Tube Rupture	15.6.3

The evaluation by the licensee of the events listed above was by recalculation, sensitivity studies or engineering judgement. For those events where the Departure from Nucleate Boiling Ratio (DNBR) or Pressure Limit was the limiting parameter, these requirements were found to be within the acceptable limits.

The limiting DNBR accident is the complete loss of forced reactor coolant flow from simultaneous loss of electrical power to all reactor coolant pumps. The licensee noted that the DNBR is a function of reactor core design, and varies somewhat with the specific parameters of each new core load. The required DNBR value for this application for 3% flow reduction is 1.26 for typical cells and 1.24 for thimble cells. This is based on the Revised Thermal Design Procedure (RTDP) and the WRB-1 DNB correlation; and for both 17x17XL STD and V5H fuel. The safety analysis limit DNBR values of 1.43 for typical cells and 1.38 for thimble cells are used, conservatively increasing the DNBR limit to provide for effects of rod bow and other DNB penalties that may occur. For the limiting accident, assuming four loops in operation, the DNBR is currently 1.514 for the full flow case. With 3% flow reduction, the DNBR is 1.497 and is acceptable. The THINC code is used to calculate DNBR for the transient, based on heat flux from the FACTRAN code and flow from the LOFTRAN code.

The limiting pressure event is the Loss of Load/Turbine Trip (Without Immediate Reactor Trip). In this transient there is a step decrease in turbine load from full power (turbine trip) without immediately initiating a reactor trip and represents the most severe pressure transient on the RCS under upset conditions. The reactor eventually trips as a consequence of a high pressurizer level trip initiated by the Reactor Protection System. The sensitivity study for Loss of Load/Turbine Trip event was performed with the LOFTRAN code which was approved by the NRC in WCAP-79007-P-A. The associated pressure for the limiting high pressure event is 2748 psia which is 110% of the RCS design pressure of 2485 psig. The sensitivity study for the reduced RCS flow calculated a peak pressure of 2747 psia which is 1 psi lower and therefore acceptable.

The current maximum peak clad temperature (PCT) for the large break LOCA is 2132 °F for Unit 1 and 2136 °F for Unit 2. These values are acceptable as they are below the 2,200 °F requirement in 10 CFR 50.46. The licensee stated (letter dated August 27, 1998) that a recalculation of the large break LOCA with updated information, including the effects of a 3% reduction in RCS flow rate indicated a PCT of 2090 °F for Unit 1 and 2. This confirmed that the predicted peak clad temperature remained bounded by the current maximum PCT for the lower loop temperatures and flow rate.

The current PCT temperature for the small break LOCA is 1849 °F for a 1.5 inch break for Unit 1 and Unit 2. This value is acceptable as it is below the acceptance limit of 2,200 °F. The licensee's evaluation confirmed that the predicted peak clad temperature remained bounded by the current PCT for the lower loop temperatures and flow rate.

4.2 Summary

The proposed alternative operating conditions for STP with a 3% RCS flow reduction were evaluated. Based on the staff's evaluation, full power operation with the proposed alternate TS for the reactor core safety limits with a reduction in the RCS T_{avg} range and the OTΔT and

OPΔT reactor trip setpoints remains bounded by the current licensing basis analysis and the proposed changes are acceptable. In addition, non-LOCA and LOCA safety analyses and evaluations were performed to confirm the acceptability of a 3.0% reduction in thermal design flow. The evaluation by the licensee was by recalculation, sensitivity studies or engineering judgement. Approved codes were used in the recalculations. All the applicable acceptance criteria for each event were found to be met. Therefore, the inclusion of the provisional TSs described above to allow full power operation with minimum measured flow and thermal design flow reduced by up to 3% is acceptable.

5.0 STRUCTURAL INTEGRITY OF THE RCS AND RCS COMPONENTS

5.1 Evaluation

As part of the justification to support a 3% reduction in the measured RCS flow, the licensee performed an evaluation of the structural integrity for the RCS and components. The components evaluated include the reactor vessel, reactor internals, control rod drive mechanisms (CRDMs), reactor coolant pumps, pressurizer, steam generators, auxiliary system equipment, and reactor coolant piping, components and their supports.

The licensee indicated that Westinghouse had previously performed bounding analyses for STP over the ranges of operation parameters provided in Table 1 of its June 30, 1998, letter. The bounding analyses consisted of a postulated limiting case associated with a 10°F reduction in vessel hot leg temperature. The postulated T-hot reduction resulted in higher stresses and cumulative fatigue usage factors (CUFs) than the original licensing basis values for certain components. The affected components are the reactor vessel (CRDM housing and rotolock inserts), pressurizer (spray nozzle) and steam generators (auxiliary feedwater nozzle). Other components evaluated were not affected by the hot leg temperature reduction. In its August 18, 1998, letter, the licensee provided a summary table of maximum stresses and CUFs for the limiting RCS components resulting from the current licensing basis bounding analyses. The stress ratios and CUF values provided in the table were found to be within the code allowable limits. The bounding analyses include STP plant operation at an RCS pressure of 2250 psia, a range of steam pressure from 975 psia to 1100 psia, a range of hot leg temperature from 615.6° F to 625.6° F, and a range of RCS average temperature from 549.0° F to 560.4°F, as documented in the STP UFSAR.

In its December 31, 1997, application, the licensee evaluated the effects of a 3% reduction in the RCS flow rate on the structural integrity of the reactor coolant systems and components, by comparing operational conditions used in the existing analyses against the plant new operating conditions at a proposed thermal design flow of 380,500 gpm. The input parameters for the proposed new operating condition are provided in Table 1 of the December 31, 1997, application. The licensee indicated that STP will operate with the proposed 3% RCS flow reduction, at an RCS pressure of 2250 psia, within a range of steam pressure from 979 psia to 1067 psia, a range of hot leg temperature from 617.4°F to 623.8°F, and a range of RCS average temperature from 549.0°F to 556.2°F.

The staff finds that the operational conditions used in the previous evaluation are bounding for the proposed operation under the new conditions with a 3% reduction in RCS flow rates, and