December 29, 1999

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

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: REPLACEMENT STEAM GENERATOR WATER LEVEL TRIP SETPOINT DIFFERENCES (TAC NOS. MA2500 AND MA2501)

Dear Mr. Cottle:

The Commission has issued the enclosed Amendment No. 120 to Facility Operating License No. NPF-76 and Amendment No. 108 to Facility Operating License No. NPF-80 for the South Texas Project, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated July 22, 1998, as supplemented by letters dated June 16, October 21 and 27, November 17, and December 9, 1999.

The amendments revise the TSs to reflect the steam generator water level low-low trip setpoint differences between the existing Model E and the replacement Model Delta-94 steam generators for the reactor trip system and the engineered safety features actuation system instrumentation.

Until Amendments 120 and 108 are fully implemented, please submit two sets of TS pages for any proposed pages affected in future amendments by the issuance of these amendments. The TS pages should reflect the conditions before and after full implementation of these amendments so that the correct TS pages can be issued in any future amendments. We also request that you submit a letter informing the staff when Amendments 120 and 108 are fully implemented.

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,

/s/

Thomas W. Alexion, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosures:
1. Amendment No. 120 to NPF-76
2. Amendment No. 108 to NPF-80
3. Safety Evaluation

cc w/encls: See next page

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.

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STP NUCLEAR OPERATING COMPANY

DOCKET NO. 50-498

SOUTH TEXAS PROJECT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 120
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 the City of Austin, Texas (COA) (the licensees), dated July 22, 1998, as supplemented by letters dated June 16, October 21 and 27, November 17, and December 9, 1999, 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.

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*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 the 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-76 is hereby amended to read as follows:

   2. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 120, 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 and shall be implemented following replacement of Unit 1 Model E steam generators with Model Delta-94 steam generators and prior to entry into Operational Mode 3.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

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

Attachment: Changes to the Technical Specifications

Date of Issuance: December 29, 1999

STP NUCLEAR OPERATING COMPANY

DOCKET NO. 50-499

SOUTH TEXAS PROJECT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 108
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 the City of Austin, Texas (COA) (the licensees), dated July 22, 1998, as supplemented by letters dated June 16, October 21 and 27, November 17, and December 9, 1999, 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.

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*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 the 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. 108, 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 and shall be implemented following replacement of Unit 1 Model E steam generators with Model Delta-94 steam generators and prior to entry into Operational Mode 3.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

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

Attachment: Changes to the Technical Specifications

Date of Issuance: December 29, 1999

ATTACHMENT TO LICENSE AMENDMENT NOS. 120 AND 108

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 revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
2-5	2-5
3/4 3-32	3/4 3-32

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 120 AND 108 TO

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

STP NUCLEAR OPERATING COMPANY, ET AL.

SOUTH TEXAS PROJECT, UNITS 1 AND 2

DOCKET NOS. 50-498 AND 50-499

1.0 INTRODUCTION

By application dated July 22, 1998, as supplemented by letters dated June 16, October 21 and 27, November 17, and December 9, 1999, STP Nuclear Operating Company (the licensee) requested changes to the South Texas Project (STP), Units 1 and 2, Technical Specifications (TSs). The proposed changes would revise the TSs to reflect the steam generator (SG) water level low-low trip setpoint differences between the existing Model E and the replacement Model Delta-94 SGs for the reactor trip system and the engineered safety features actuation system instrumentation.

The June 16, October 21 and 27, November 17, and December 9, 1999, supplements provided additional clarifying information that was within the scope of the original application and *Federal Register* notice and did not change the staff's initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

By application dated July 22, 1998, the licensee submitted a request to change the SG water level low-low trip setpoint specified in TS Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints," and Table 3.3-4, "Engineered Safety Features Actuation System Instrumentation Trip Setpoints." The proposed TSs would add new values for the trip setpoint and allowable value ($\geq$ 20.0% and $\geq$ 18.0% of narrow range instrument span, respectively) for the new Westinghouse Model Delta-94 SGs. The existing TS values for the trip setpoint and allowable value ($\geq$ 33.0% and $\geq$ 30.7% of narrow range instrument span, respectively) for the current Westinghouse Model E SGs would remain in the TSs.

The licensee plans to replace the existing Unit 1 Model E SGs with Model Delta-94 SGs at the end of Cycle 9. The Model Delta-94 SG design characteristics incorporate increased SG heat transfer area, increased secondary mass inventory at hot full power, and increased reactor coolant system (RCS) flow and volume. The TS changes are intended to increase the operating margin to the SG water level low-low reactor trip setpoint and to minimize the potential for unnecessary auxiliary feedwater actuation and excessive RCS cooldown following a reactor trip.

In support of the proposed TS, the licensee provided analyses in the July 22, 1998, application and responses to the staff's questions by letters dated June 16, October 27 and November 17, 1999. The October 21, 1999, supplement requested a 30-day implementation period following NRC approval. However, the December 9, 1999, supplement requested that the implementation period be revised to the period following replacement of Unit 1 Model E steam generators with Model Delta-94 steam generators and prior to entry into Operational Mode 3

3.0 EVALUATION

The staff review is to confirm that the licensee performed safety analyses with acceptable methods, to verify that the analytical results meet the required acceptance criteria and to ensure that the proposed TS appropriately reflects the results of the acceptable safety analyses. The following evaluation includes the staff review of the analytical methods, transient analyses, and setpoint methodology.

3.1 Analytical Methods

The licensee used the RETRAN code to perform transient reanalyses. The RETRAN code is a general purpose, thermal-hydraulic code that models the reactor coolant as a single phase or as two equilibrium phases with the exception that a non-equilibrium pressurizer component can be included. It provides a simulation of the system response and calculates system parameters such as core power, RCS flow, and primary and secondary side temperatures and pressure during a transient. The staff finds that the RETRAN code was previously reviewed and approved by the NRC for use in the design-basis transient analysis for existing Westinghouse-designed pressurized water reactors for licensing applications (letter from F. Akstulewicz (NRC) to H. Sepp (Westinghouse Electric Corporation), "Acceptance for Referencing of Licensing Topical Report WCAP-14882, 'RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analysis," dated February 11, 1999, as revised by NRC letter dated March 2, 1999). In response to the staff's request, the licensee evaluated its compliance with conditions specified in the safety evaluation (SE) for the RETRAN code and showed that the SE conditions for RETRAN had been met (in the October 27, 1999, supplement). Therefore, the staff concludes that the licensee adequately addressed the staff's concerns relating to conformance to the SE conditions.

3.2 Transient Analysis

The licensee evaluated the effect of the TS changes on the results of safety analysis presented in the Updated Final Safety Analysis Report (UFSAR). As a result, the licensee identified that three events would be affected by the TS changes because these events require a reactor trip on SG water level low-low trip signal for event mitigation. These events are (1) loss of non-emergency AC power to the plant auxiliaries (LOAC), (2) loss of normal feedwater flow (LONF), and (3) feedwater system pipe break (FLB). The licensee reanalyzed these three events. At the staff's request, the licensee provided the results of the reanalyses (in Attachment 1 of the June 16, 1999, supplement). In the analyses, the licensee made the following assumptions:

- Automatic operation of the pressurizer power-operated relief valves (PORVs) was assumed in the reanalyses of the LOAC, LONF, and FLB events. This assumption is consistent with the UFSAR assumption. However, the current TS 3.4.4.a permits the block valves upstream of their associated PORVs to be closed for unlimited time for PORVs with excessive leakage and thus, the automatic function of the pressurizer PORVs becomes unavailable. In response to the staff's question regarding the adequacy of assuming automatic opening of the pressurizer PORVs in the reanalyses, by letter dated October 27, 1999, the licensee indicated that the LOAC, LONF and FLB events were previously identified as the limiting events with respect to the pressurizer overfill criterion, and that the automatic opening of the PORVs increased the water level swell in the pressurizer and reduced the margin to the pressurizer overfill. The staff has reviewed the licensee's response and the results of the reanalyses and concluded that the assumption related to automatic opening of pressurizer PORVs is acceptable because (1) it is consistent with the UFSAR assumption, and (2) it increases the severity of the transients with respect to pressurizer overfill and is a conservative assumption.

- The minimum measured RCS flow was assumed to be 403,000 gpm [gallons per minute]. The value is consistent with the minimum measured RCS flow for the plant with the Model Delta-94 SGs. A lower RCS flow reduces the heat removal capability from the primary to the secondary systems and results in greater water swell level in the pressurizer for the LOAC, LONF, and FLB events. The assumption is conservative and is acceptable.

- The full power RCS average temperature was assumed to be 593 ^{0}F and the full power main feedwater temperature was assumed to be 440 ^{0}F. The assumed values are consistent with the values used in the current analyses in the UFSAR, and therefore, are acceptable.

- The SG tube plugging level was assumed at 10 percent of the total SG tubes. The assumption is consistent with the current operating limit and is therefore acceptable.

- Credit was taken for automatic operation of the safety-grade SG PORVs for accident mitigation of the LOAC and LONF events. In a separate application, the licensee proposed to revise TS 3/4.7.1.6, "Atmospheric Steam Relief Valves," and proposed to add a new TS for atmospheric steam relief valve instrumentation, to ensure that the automatic feature of the SG PORVs remains operable during Mode 1 and 2. The NRC had previously reviewed and approved these proposed revisions to the TSs (Amendments 114/102 dated August 19, 1999). Because the NRC-approved TSs require the SG PORVs to function automatically, the assumption of the use of the these valves for event mitigation is acceptable.

- The SG water level low-low trip setpoint was assumed to be 20 percent of the narrow range span. The auxiliary feedwater system was assumed to actuate on the SG water level low-low trip signal. The assumptions are consistent with the proposed TS changes and are acceptable.

- The initial pressurizer water level was assumed at no greater than 57 percent span. This assumption is inconsistent with the current TS 3/4.4.3, which requires, in part, that the pressurizer water volume be no greater than 1816 cubic feet (about 92 percent span). The current TS 3/4.4.3 is consistent with Westinghouse Standard TS 3.4.9.a, which requires that the pressurizer water level be less than or equal to 92 percent span. The lower initial pressurizer water volume in the analysis provides a greater initial margin to pressurizer overfill. However, the current TSs do not prohibit the pressurizer water level from exceeding the assumed value. Therefore, this assumption is nonconservative from the point of pressurizer overfill. On July 22, 1999, the NRC met with the licensee to resolve the inconsistency between the TS and the initial pressurizer water volume used in the analysis. The NRC had reviewed the licensee's views expressed in the meeting and concluded, in an August 26, 1999, letter to the licensee, that the use of 57 percent initial pressurizer water volume in the analysis was acceptable and the NRC would not require that the licensee propose revised TSs to support the revised analysis assumption regarding the pressurizer water level with the proposed TS change. The conclusion was based on (1) the relatively low probability of plant operation with pressurizer water levels exceeding safe limits per the analysis assumption (because the operator, by procedure, closely monitors the water level to keep it within the value assumed in the analysis), and (2) the issue of the need for revised TSs to support the analysis assumption regarding the pressurizer water levels is being dealt with adequately on a generic basis.

The licensee presented the analytical results in its June 16, 1999, response to the staff request for additional information. In the June 16, 1999, response, Figures 1 through 5 presented the LOAC transient results while Table 1 included the time sequence of the event. For the LONF event, Figures 6 through 10 illustrated the results and Table 2 presented the sequence of the event.

The licensee reanalyzed a total of four different cases for the FLB event. The four cases were (1) an FLB with maximum reactivity feedback and with offsite power available, (2) an FLB with minimum reactivity feedback and with offsite power available, (3) an FLB with maximum reactivity feedback and without offsite power available, and (4) an FLB with minimum reactivity feedback and without offsite power available. The calculated plant parameters following FLB events were shown in Figures 11 through 26 of the June 16, 1999, response. The calculated sequence of events and results for all FLB cases were given in Tables 4 through 8 of the June 16, 1999, response. During the final stage of the review, the licensee submitted the results of the FLB reanalysis by letter dated November 17, 1999. The reanalysis reflected a correction in the input error and replaced the results of the FLB analyses shown in Figures 11 through 26 and Tables 4 through 8 of the June 16, 1999, supplement.

The results showed that for all cases reanalyzed, no water relief occurred from the pressurizer during the period before operator actions. The pressurizer PORVs would reclose during the transient since the PORVs were designed to open and close in a steam condition. Therefore, the staff concludes that the licensee's reanalyses provide reasonable assurance that the likelihood of a small-break loss-of-coolant accident resulting from stuck-open PORVs is not affected by the proposed TSs and thus, the licensee has addressed the concerns of TMI Action Plan Item II.K.3.10, "Proposed Anticipatory Trip Modification," satisfactorily.

The results of the reanalyses also showed that the calculated peak pressurizer and SG pressure were less than 110 percent of the design pressure, thereby meeting the acceptance criteria of Standard Review Plan (SRP) Sections 15.2.6, 15.2.7, and 15.2.8 for the LOAC, LONF, and FLB events, respectively.

Since the reanalyses used approved methods, made adequate assumptions to maximize the calculated peak pressurizer water level for the limiting cases with respect to the pressurizer overfill, and showed compliance with the requirements of TMI Action Plan Item II.K.3.10 and the applicable SRP acceptance criteria, the staff concludes that the reanalyses are acceptable.

3.3 Setpoint Methodology

SG water level instrumentation trip setpoints have been affected by operational considerations that have been incorporated into analytical assumptions. As a result, the SG water level low-low trip setpoints for the reactor trip system and the engineered safety features actuation system will change with the new SGs.

The proposed trip setpoint TS changes are as follows:

- Table 2.2-1, “Reactor Trip System Instrumentation Trip Setpoints,“ Functional Unit 13, “Steam Generator Water Level Low-Low” - Values of ≥ 20.0% and ≥ 18.0% of narrow range instrument span are added for the trip setpoint and the allowable value, respectively, for the Model Delta-94 SGs.

- Table 3.3-4, “Engineered Safety Features Actuation System Instrumentation Trip Setpoints,” Functional Unit 6d, Auxiliary Feedwater, Steam Generator Water Level Low-Low” - Values of ≥ 20.0% and ≥ 18.0% of narrow range instrument span are added for the trip setpoint and the allowable value, respectively, for the Model Delta-94 SGs.

The licensee indicated that the allowable values were calculated using the methodology in WCAP-11273, Revision 2, “Westinghouse Setpoint Methodology for Protection Systems - South Texas Project, Units 1 and 2,” February 1993, which the staff found acceptable in its safety evaluation issued with Amendments 61/50 dated May 27, 1994. The staff finds that the use of a previously accepted setpoint methodology for the proposed TS setpoint changes is acceptable.

3.4 Evaluation Summary

Based on its review, the staff finds that the licensee’s reanalyses are acceptable and the input parameters assumed in the acceptable reanalyses adequately reflect the proposed TSs. Therefore, the staff concludes that the proposed TS Tables 2.2-1, “Reactor Trip System Instrumentation Trip Setpoints,” and 3.3-4, “Engineered Safety Features Actuation System Instrumentation Trip Setpoints,” with respect to the SG water level low-low trip setpoints are acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (63 FR 48268). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

Principal Contributors: S. Sun
F. Gee

Date: December 29, 1999