

September 5, 1995

Mr. William T. Cottle
Group Vice-President, Nuclear
Houston Lighting & Power Company
South Texas Project Electric
Generating Station
P. O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 - AMENDMENT NOS. 79
AND 68 TO FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80
(TAC NOS. M92573 AND M92574)

Dear Mr. Cottle:

The Commission has issued the enclosed Amendment Nos. 79 and 68 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 May 31, 1995, as supplemented by letter dated August 2, 1995.

The amendments modify (by relocation to the Technical Requirements Manual) TS 3/4.1.2.1, Boration Systems/Flow Paths - Shutdown, TS 3/4.1.2.2, Boration Systems/Flow Paths - Operating, TS 3/4.1.2.3, Charging Pumps - Shutdown, TS 3/4.1.2.4, Charging Pumps - Operating, TS 3/4.1.2.5, Borated Water Sources - Shutdown, TS 3/4.1.2.6, Borated Water Sources - Operating, TS 3/4.4.2.1, Safety Valves - Shutdown, and the associated Bases.

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. 79 to NPF-76
2. Amendment No. 68 to NPF-80
3. Safety Evaluation

cc w/encls: See next page

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*signed by TWA
on 09/01/95*

Document Name: ST92573.AMD

*See previous concurrence re: P5V6

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

September 5, 1995

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Group Vice-President, Nuclear
Houston Lighting & Power Company
South Texas Project Electric
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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,

A handwritten signature in cursive script that reads "Thomas W. Alexion".

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. 79 to NPF-76
2. Amendment No. 68 to NPF-80
3. Safety Evaluation

cc w/encls: See next page

Mr. William T. Cottle
Houston Lighting & Power Company

South Texas, Units 1 & 2

cc:

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Wadsworth, TX 77483



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

HOUSTON LIGHTING & POWER COMPANY
CITY PUBLIC SERVICE BOARD OF SAN ANTONIO
CENTRAL POWER AND LIGHT COMPANY
CITY OF AUSTIN, TEXAS
DOCKET NO. 50-498
SOUTH TEXAS PROJECT, UNIT 1
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 79
License No. NPF-76

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Houston Lighting & Power Company* (HL&P) acting on behalf of itself and for the City Public Service Board of San Antonio (CPS), Central Power and Light Company (CPL), and City of Austin, Texas (COA) (the licensees), dated May 31, 1995, as supplemented by letter dated August 2, 1995, 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.

*Houston Lighting & Power Company is authorized to act for 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-76 is hereby amended to read as follows:

2. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 79, 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 within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



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 5, 1995



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

HOUSTON LIGHTING & POWER COMPANY
CITY PUBLIC SERVICE BOARD OF SAN ANTONIO
CENTRAL POWER AND LIGHT COMPANY
CITY OF AUSTIN, TEXAS
DOCKET NO. 50-499
SOUTH TEXAS PROJECT, UNIT 2
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 68
License No. NPF-80

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Houston Lighting & Power Company* (HL&P) acting on behalf of itself and for the City Public Service Board of San Antonio (CPS), Central Power and Light Company (CPL), and City of Austin, Texas (COA) (the licensees), dated May 31, 1995, as supplemented by letter dated August 2, 1995, 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.

*Houston Lighting & Power Company is authorized to act for 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. 68, 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 within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



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 5, 1995

ATTACHMENT TO LICENSE AMENDMENT NOS. 79 AND 68

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

3/4 1-9
3/4 1-10
3/4 1-11
3/4 1-12
3/4 1-13
3/4 1-14
3/4 1-15
3/4 4-7
B 3/4 1-2
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B 3/4 4-1
B 3/4 4-2

INSERT

3/4 1-9

3/4 4-7
B 3/4 1-2
B 3/4 1-3
B 3/4 4-1
B 3/4 4-2

Pages 3/4 1-9 through 3/4 1-15 have been deleted.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full-length 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 full-length 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 full-length 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 as specified in the Core Operating Limits Report (COLR). 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.

Page 3/4 4-7 has been deleted.

REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting¹ of 2485 psig \pm 2%.²

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by Specification 4.0.5.

¹The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

²The as left lift setting shall be within \pm 1% following valve testing.

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} . In MODES 1 and 2, the most restrictive condition occurs at EOL, with T_{avg} at no load 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 1.3% $\Delta k/k$ is required to control the reactivity transient. The 1.3% $\Delta k/k$ SHUTDOWN MARGIN is the design basis minimum for the 14-foot fuel using silver-indium-cadmium and/or Hafnium control rods (Ref. FSAR Table 4.3-3). Accordingly, the SHUTDOWN MARGIN requirement for MODES 1 and 2 is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. In MODES 3, 4, and 5, the most restrictive condition occurs at BOL, when the boron concentration is the greatest. In these modes, the required SHUTDOWN MARGIN is composed of a constant requirement and a variable requirement, which is a function of the RCS boron concentration. The constant SHUTDOWN MARGIN requirement of 1.3% $\Delta k/k$ is based on an uncontrolled RCS cooldown from a steamline break accident. The variable SHUTDOWN MARGIN requirement is based on the results of a boron dilution accident analysis, where the SHUTDOWN MARGIN is varied as a function of RCS boron concentration, to guarantee a minimum of 15 minutes for operator action after a boron dilution alarm, prior to a loss of all SHUTDOWN MARGIN.

The boron dilution analysis assumed a common RCS volume, and maximum dilution flow rate for MODES 3 and 4, and a different volume and flow rate for MODE 5. The MODE 5 conditions assumed limited mixing in the RCS and cooling with the RHR system only. The MODE 5 SHUTDOWN MARGIN curve (Figure 3.1-2) can be used to provide the required C_b in the rapid refueling condition (MODE 5 with ARO). The cycle-specific reload safety analysis verifies this curve to be bounding in this condition.

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 values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

REACTIVITY CONTROL SYSTEMS

BASES

MODERATOR TEMPERATURE COEFFICIENT (Continued)

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 analysis to nominal operating conditions. These corrections involved: (1) a conversion of the MDC used in the FSAR analysis to its equivalent MTC, based on the rate of change of moderator density with temperature at RATED THERMAL POWER conditions, and (2) subtracting from this value the largest differences in MTC observed at EOL, all rods withdrawn, RATED THERMAL POWER conditions, and those most adverse conditions of moderator temperature and pressure, rod insertion, axial power skewing, and xenon concentration that can occur in nominal operation and lead to a significantly more negative EOL MTC at RATED THERMAL POWER. These corrections transformed the MDC values used in the FSAR analysis into the limiting EOL MTC value specified in the CORE OPERATING LIMITS REPORT (COLR). The 300 ppm surveillance MTC value specified in the COLR 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 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 561°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its 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 DELETED

REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. Verification that the Digital Rod Position Indicator agrees with the demanded position within ± 12 steps at 24, 48, 120, and 259 steps withdrawn for the Control Banks and 18, 234, and 259 steps withdrawn for the Shutdown Banks provides assurances that the Digital Rod Position Indicator is operating correctly over the full range of indication. Since the Digital Rod Position Indication System does not indicate the actual shutdown rod position between 18 steps and 234 steps, only points in the indicated ranges are picked for verification of agreement with demanded position.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 561°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

Page B 3/4 1-3 has been deleted.

SOUTH TEXAS - UNITS 1 & 2

B 3/4 1-3 Unit 1 - Amendment No. ~~51,54,62~~,79
Unit 2 - Amendment No. ~~40,43,51~~,68

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the design limit during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The boron dilution analysis assumed a common RCS volume, and maximum dilution flow rate for MODES 3 and 4, and a different volume and flow rate for MODE 5. The MODE 5 conditions assumed limiting mixing in the RCS and cooling with the RHR system only. In MODES 3 and 4, it was assumed that at least one reactor coolant pump was operating. If at least one reactor coolant pump is not operating in MODE 3 or 4, then the maximum possible dilution flow rate must be limited to the value assumed for MODE 5.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting an RCP with one or more RCS cold legs less than or equal to 350°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 504,950 lbs per hour of saturated steam at the valve setpoint of 2500 psia.

REACTOR COOLANT SYSTEM

BASES

SAFETY VALVES (Continued)

During Modes 1, 2, and 3, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no Reactor trip until the first Reactor Trip System Trip Setpoint is reached (i.e., no credit is taken for a direct Reactor trip on the turbine trip resulting from loss-of-load) and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4. RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relief RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer Code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

- A. Manual control of PORVs is used to control reactor coolant system pressure. This is a function that is used for the steam generator tube rupture accident and for plant shutdown. Manual control of PORVs is a safety related function.
- B. Maintaining the integrity of the reactor coolant pressure boundary. This is a function that is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NOS. 79 AND 68 TO
FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80
HOUSTON LIGHTING & POWER COMPANY
CITY PUBLIC SERVICE BOARD OF SAN ANTONIO
CENTRAL POWER AND LIGHT COMPANY
CITY OF AUSTIN, TEXAS
DOCKET NOS. 50-498 AND 50-499
SOUTH TEXAS PROJECT, UNITS 1 AND 2

1.0 INTRODUCTION

By application dated May 31, 1995, Houston Lighting & Power Company, et al., (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 modify (by relocation to the Technical Requirements Manual (TRM)) TS 3/4.1.2.1, Boration Systems/Flow Paths - Shutdown, TS 3/4.1.2.2, Boration Systems/Flow Paths - Operating, TS 3/4.1.2.3, Charging Pumps - Shutdown, TS 3/4.1.2.4, Charging Pumps - Operating, TS 3/4.1.2.5, Borated Water Sources - Shutdown, TS 3/4.1.2.6, Borated Water Sources - Operating, TS 3/4.4.2.1, Safety Valves - Shutdown, and the associated Bases.

By letter dated August 2, 1995, the licensee committed to incorporate the TRM, by reference, into the Updated Final Safety Analysis Report (UFSAR). Accordingly, changes to the TRM would be controlled in accordance with the approved station procedures and the requirements of 10 CFR 50.59. The August 2, 1995, supplement is clarifying information that does not change the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

Section 182a of the Atomic Energy Act (the "Act") requires that applicants for nuclear power plant operating licenses state TSs and that these TSs be included as a part of the license. The Commission's regulatory requirements related to the content of TSs are set forth in 10 CFR 50.36. That regulation requires that the TSs include items in five specific categories, including (1) safety limits, limiting safety system settings and limiting control

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settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls and states also that the Commission may include such additional TSs as it finds to be appropriate. However, the regulation does not specify the particular TSs to be included in a plant's license.

The Commission has provided guidance for the contents of TSs in its "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" ("Final Policy Statement"), 58 FR 39132 (July 22, 1993), in which the Commission indicated that compliance with the Final Policy Statement satisfies Section 182a of the Act. In particular, the Commission indicated that certain items could be relocated from the TSs to licensee-controlled documents, consistent with the standard enunciated in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). In that case, the Atomic Safety and Licensing Appeal Board indicated that "technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety."

Consistent with this approach, the Final Policy Statement identified four criteria to be used in determining whether a particular matter is required to be included in the TSs, as follows: (1) installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary; (2) a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (3) a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (4) a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety. As a result, existing limiting condition for operation (LCO) requirements which fall within or satisfy any of the criteria in the Final Policy Statement must be retained in the TSs, while those LCO requirements which do not fall within or satisfy these criteria may be relocated to other, licensee-controlled documents. The Commission recently amended 10 CFR 50.36 to codify and incorporate these four criteria (60 FR 36953). The change to 10 CFR 50.36 is effective as of August 18, 1995.

3.0 EVALUATION

3.1 Chemical Volume and Control System

The boration subsystem of the Chemical Volume and Control System (CVCS) is addressed by the following TSs:

- TS 3/4.1.2.1 Boration Systems/Flow Paths - Shutdown
- TS 3/4.1.2.2 Boration Systems/Flow Paths - Operating
- TS 3/4.1.2.3 Charging Pumps - Shutdown

- TS 3/4.1.2.4 Charging Pumps - Operating
- TS 3/4.1.2.5 Borated Water Sources - Shutdown
- TS 3/4.1.2.6 Borated Water Sources - Operating

The boration subsystem of the CVCS, including the charging pumps, is designed to maintain and control the chemical neutron absorber (boron) concentration in the Reactor Coolant System (RCS) to ensure adequate shutdown margin. The current TSs have requirements for sources of borated water, flow paths to inject this borated water into the RCS, and charging pumps to provide the necessary charging head.

The boration subsystem of the CVCS, including the charging pumps, is not installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary. In addition, it is not a process variable, design feature, or operating restriction that is an initial condition of the type of design basis accident or transient analysis discussed in the background section (above). Shutdown margin is a process variable which is an initial condition of various design basis accidents and transient analyses, however operability of the CVCS is not. Limitations on shutdown margin are established and maintained by other TSs not affected by this proposed change.

The boration subsystem of the CVCS, including the charging pumps, is not a structure, system or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Control and maintenance of boron concentration in the RCS is not part of the primary success path for mitigation of a design basis accident or transient. This function is provided by either the Emergency Core Cooling System (ECCS) or through maintenance of shutdown margin, as established by TSs not affected by the proposed change, which is adequate for the required safety function.

In summary, the requirements specified in these existing TSs do not satisfy the criteria for TSs, and may be relocated to the UFSAR such that future changes to these provisions may be made pursuant to 10 CFR 50.59.

3.2 Pressurizer Code Safety Valves - Shutdown

The pressurizer code safety valves (PSVs) operate to prevent the RCS from being pressurized above its safety limit of 2735 psig. The limiting condition for operation and surveillance requirements of the PSVs is addressed in TS 3/4.4.2.1.

The PSVs are not installed instrumentation that detects significant degradation of the reactor coolant pressure boundary, nor are they a process variable, design feature, or operating restriction that is an initial condition of the type of design basis accident or transient analysis discussed in the background section (above).

In Modes 4, 5, and 6, when the head is on the reactor, the RCS is protected by the overpressure protection systems, which is addressed in TSs not affected by this proposed change. The low temperature overpressure protection requirements ensure that the RCS is protected from pressure transients that could exceed the limits of 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements." Therefore, the PSVs in Modes 4, 5, and 6, when the head is on the reactor, are not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

In summary, the requirements specified in these existing TSs do not satisfy the criteria for TSs, and may be relocated to the UFSAR such that future changes to these provisions may be made pursuant to 10 CFR 50.59.

On this basis, the staff finds that these requirements are not required to be in the TSs under 10 CFR 50.36 or Section 182a of the Atomic Energy Act, and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the health and safety of the public. Further they do not fall within any of the four criteria set forth in the Commission's Final Policy Statement. In addition, the staff finds that this change does not alter the limiting conditions for operation or surveillance requirements because the subject TSs and Bases will be relocated in their entirety to the TRM, and sufficient regulatory controls exist under 10 CFR 50.59 to assure continued protection of the public health and safety and that future changes to these requirements are acceptable. Accordingly, the staff finds that these requirements may be relocated from the TSs to the TRM.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. 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 (60 FR 39441). Accordingly, the amendment meets 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.