April 2, 1992

Docket Nos. 50-498
and 50-499

Mr. Donald P. Hall
Group Vice-President, Nuclear
Houston Lighting & Power Company
P. O. Box 1700
Houston, Texas 77251

Dear Mr. Hall:

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SUBJECT: ISSUANCE OF AMENDMENT NOS. 35 AND 26 TO FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80 - SOUTH TEXAS PROJECT, UNITS 1 AND 2 (TAC NOS. M81550 AND M81551)

The Commission has issued the enclosed Amendment Nos. 35 and 26 to Facility Operating License Nos. NPF-76 and NPF-80 for the South Texas Project, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated August 30, 1991 (ST-HL-AE-3830) as supplemented by letter dated January 24, 1992 (ST-HL-AE-3958).

The amendments change the Appendix A Technical Specifications by incorporating an additional reference for the methodology used in the Core Operating Limits Report. The referenced methodology was used in the reanalysis of the end of life moderator temperature coefficient (MTC) which resulted in a more negative value for the MTC.

A copy of our Safety Evaluation supporting the amendments is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

The application also included proposed changes to the text of the axial flux difference specification and operability requirements for the incore moveable detection systems. Those changes continue under staff review.

Sincerely,

Original Signed By

George F. Dick, Jr., Senior Project Manager
Project Directorate IV-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 35 to NPF-76
2. Amendment No. 26 to NPF-80
3. Safety Evaluation

cc w/enclosures:
See next page

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OFFICE	PDIV-2/LA	PDIV-2/PM	OGC	PDIV-2/D	
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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. 35
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 August 30, 1991, as supplemented by letter dated January 24, 1992, 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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* 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.

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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. 35, 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 to be implemented within 10 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Suzanne C. Black

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 2, 1992

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

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. 26
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 August 30, 1991, as supplemented by letter dated January 24, 1992, 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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* 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. 26, 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 to be implemented within 10 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Suzanne C. Black

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 2, 1992

ATTACHMENT TO LICENSE AMENDMENT NOS.35 AND 26

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 vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
6-20	6-20
6-20a	6-20a
B 3/4 1-2	B 3/4 1-2

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

in Appendix B shall be supplemented with three additional categories: class of solid wastes (as defined by 10 CFR Part 61), type of container (e.g., LSA, Type A, Type B, Large Quantity) and SOLIDIFICATION agent or absorbent (e.g., cement, urea formaldehyde).

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.* This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figures 5.1-3 and 5.1-4) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time, and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operation." Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM and the ODCM, pursuant to Specifications 6.13 and 6.14, respectively, as well as any major change to Liquid, Gaseous, or Solid Radwaste Treatment Systems pursuant to Specification 6.15. It shall also include a listing of new locations for dose calculations and/or environmental monitoring identified by the Land Use Census pursuant to Specification 3.12.2.

*In lieu of submission with the Semiannual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.3.10 or 3.3.3.11, respectively; and description of the events leading to liquid holdup tanks or gas storage tanks exceeding the limits of Specification 3.11.1.4 or 3.11.2.6, respectively.

MONTHLY OPERATING REPORTS

6.9.1.5 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

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

1. Moderator Temperature Coefficient BOL and EOL limits, and 300 ppm surveillance limit for Specification 3/4.1.1.3,
2. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,
3. Control Bank Insertion Limits for Specification 3/4.1.3.6,
4. Axial Flux Difference limits and target band for Specification 3/4.2.1,
5. Heat Flux Hot Channel Factor, K(Z), Power Factor Multiplier, and (F_{xy}^{RTP}) for Specification 3/4.2.2, and
6. Nuclear Enthalpy Rise Hot Channel Factor, and Power Factor Multiplier for Specification 3/4.2.3.

The CORE OPERATING LIMITS REPORT shall be maintained available in the Control Room.

6.9.1.6.b The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

1. WCAP 9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July, 1985 (W Proprietary).

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

1.A. WCAP 12942-P-A, "SAFETY EVALUATION SUPPORTING A MORE NEGATIVE EOL MODERATOR TEMPERATURE COEFFICIENT TECHNICAL SPECIFICATION FOR THE SOUTH TEXAS PROJECT ELECTRIC GENERATING STATION UNITS 1 AND 2."

CORE OPERATING LIMITS REPORT (Continued)

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient)

2. WCAP 8385, "POWER DISTRIBUTION AND LOAD FOLLOWING PROCEDURES TOPICAL REPORT", September, 1974 (W Proprietary).

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

3. Westinghouse letter NS-TMA-2198, T.M. Anderson (Westinghouse) to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 - Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control). Approved by NRC Supplement No. 4 to NUREG-0422 January, 1981 Docket Nos. 50-369 and 50-370.)

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

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

5. WCAP 9220-P-A, Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL-1981 VERSION", February 1982 (W Proprietary).

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

6. WCAP 9561-P-A, ADD. 3, Rev. 1, "BART A-1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS - SPECIAL REPORT: THIMBLE MODELING W ECCS EVALUATION MODEL", July, 1986 (W Proprietary).

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

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

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

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

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 1.75% Δk/k is required to control the reactivity transient. The 1.75% Δk/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 1.75% Δk/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. 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 SHUTDOWN MARGIN requirements for MODE 5 shall apply.

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.

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 BORATION SYSTEMS

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

With the RCS average temperature above 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.75% Δk/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 27,000 gallons of 7000 ppm borated water from the boric acid storage system or 458,000 gallons of 2500 ppm borated water from the refueling water storage tank (RWST). The RWST volume is an ECCS requirement and is more than adequate for the required boration capability.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 35 AND 26 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 August 30, 1991 (ST-HL-AE-3830), Houston Lighting & Power Company, et.al., (the licensee) requested changes to the Technical Specifications (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 incorporate an additional reference in the Technical Specification (TS) for the methodology used for calculations included in the Core Operating Report. Specifically, the use of the methodology would result in the calculation of a more negative end of life (EOL) moderator temperature coefficient (MTC) and the associated 300 ppm surveillance requirement (SR) limits specified in the Core Operating Limits Report (COLR). The purpose of the 300 ppm SR is to ensure that the most negative MTC at EOL remains within the bounds of the STP safety analysis, in particular for those transients and accidents that assume a constant value for the moderator density coefficient (MDC) of 0.43 Delta k per gm/cc. The methodology is included in Westinghouse Report WCAP-12942, "Safety Evaluation Supporting a More Negative EOL Moderator Temperature Coefficient Technical Specification for the South Texas Project, Units 1 and 2." In its letter of January 24, 1992, the licensee requested a 10-day implementation period following the date of issuance of the license amendment.

2.0 BACKGROUND

The current STP TS 3.1.1.3 states that:

> "The moderator temperature coefficient (MTC) shall be within the beginning of cycle (BOC) and EOC limit specified in the COLR."

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The corresponding action for exceeding this limiting condition for operation (LCO) is to be in hot shutdown within 12 hours. The STP SR involves an MTC measurement at any thermal power within 7 effective full power days (EFPD) after reaching an equilibrium primary coolant boron concentration of 300-ppm.

After appropriate corrections are made, the measured value is compared to the 300-ppm SR limit value specified in the COLR at the all rods out (ARO) rated thermal power (RTP) condition. In the event that the measured MTC is more negative than the 300-ppm SR limit, the MTC must be remeasured and compared with the EOC MTC LCO value at least once per 14 EFPD during the remainder of the operating cycle. The STP Units 1 and 2 300-ppm SR and end-of-cycle (EOC) LCO values for the most negative MTC are conservative (less negative) when compared to the value of the MTC which is used in the safety analyses.

STP proposed to revise the current method for determining the 300-ppm surveillance and the EOC MTC limits specified in the COLR. The revised method for determining the COLR MTC limits will result in the addition of a reference to WCAP-12942 in the TS and in a change to the Technical Specification Bases Section B 3/4.1.1.3. This revised method and the COLR MTC limit changes do not affect the maximum moderator density coefficient (MDC) value of -56 pcm/°F. These changes apply to the current and future reload cycles for STP Units 1 and 2, and are supported by an evaluation provided by Westinghouse methodology (WCAP-12942). The analysis applies only to STP and is similar to that approved for use at other nuclear power plants.

3.0 EVALUATION

3.1 Amendment to End of Cycle Moderator Temperature Coefficient.

The current method used to determine the most negative MTC is described in Bases Section 3/4.1.1.3 of the TS for STP Units 1 and 2. This method is based on incrementally correcting the conservative MDC used in the safety analyses to obtain the most negative MTC value or, equivalently, the most positive MDC at the nominal hot full power (HFP) core conditions. The corrections involve subtracting the incremental change in the MDC, which is associated with the core condition of all control rods inserted (ARI), to an ARO core condition. The MTC is then equal to the product of multiplying the MDC by the rate of change of the moderator density with the temperature at RTP conditions.

The TS Bases provide a method of determining the most negative MTC LCO value which results in an ARO MTC value that is significantly less negative than the MTC used in the safety analysis and which may even be less negative than the best estimate EOC ARO MTC for extended burnup reload cores. This could result in the plant being required by TS 3.1.1.3 to be placed in a hot shutdown condition even though it would retain a substantial margin to the safety analysis MDC. The problem with the current method is caused by adjusting the MDC from an HFP ARI condition to an HFP ARO condition in defining the most negative MTC. The TS on control rod positions does not allow the HFP ARI condition for allowable power operation in which the shutdown banks are completely withdrawn from the core and the control banks must meet the rod insertion limits (RILs).

Westinghouse has provided the most negative feasible (MNF) MTC as an alternative method for adjusting the safety analysis MDC to obtain a most negative MTC. The MTC method seeks to determine the conditions for which a core will exhibit the most negative value that is consistent with operation allowed by the TS. For example, the MNF MTC method would not require the conversion assumption of the ARI HFP condition, but would require the conversion assumption that all control rod banks are inserted to the maximum amount that is permitted by the TS. Westinghouse uses the MNF MTC method to determine EOC MTC sensitivities to those design and operational parameters that directly affect the MTC in such a way that the sensitivity to one parameter depends on the assumed values for the other parameters.

The parameters considered with this MNF MTC method include:

(1) soluble boron concentration in the primary coolant
(2) moderator temperature and pressure,
(3) control rod insertion,
(4) axial power shape, and
(5) transient xenon concentration

The MNF MTC approach uses this sensitivity information to derive an EOC ARO HFP MTC LCO value based on the safety analysis value of the MDC.

Westinghouse stated that this MNF MTC approach has a number of advantages over the previous method for determining the most negative MTC LCO value. The MNF MTC will be sufficiently negative so that repeated MTC measurements from a concentration of 300-ppm of boron in the core to EOC would not be required. The MNF MTC method does not change the moderator feedback assumption or the value of the MDC in the safety analysis. The MNF MTC method is a reasonable basis to assume for an MTC value of a reload core and is consistent with plant operation defined by other TS. Finally, the MNF MTC method retains the SR on MTC at the 300-ppm core condition to verify that the core is operating within the bounds of the safety analysis.

Westinghouse has determined the sensitivity of the above parameters on the EOC MTC based on six reload designs representative of the future STP Units 1 and 2 reloads. These reload designs include fuel designs, discharge burnups, and cycle lengths which are typical of those expected for STP. The concentration of soluble boron was not used in the sensitivity analysis because the TS value for the MTC at the EOC HFP ARO conditions is assumed to be at 0-ppm of boron, the definition of EOC, and because the most negative MTC occurs at 0-ppm of boron in the coolant.

The sensitivity study did not include the radial power distribution which can vary under normal operation and can affect the MTC. The operational activities that affect the radial power distribution do so through the movement of control rods and other activities that affect the xenon concentration. The allowed changes in the radial power distribution are implicitly included in the MTC sensitivity to control rod insertion and xenon concentration.

Westinghouse stated that the SR MTC value would be obtained in the same manner as currently described in the Westinghouse Standard Technical Specification (STS) Bases. The SR MTC value is obtained from the EOC ARO MTC value by making corrections for burnup and boron at a core condition of 300-ppm of boron.

The staff has reviewed the assumptions and basis for the MNF MTC method described and concludes that they are acceptable because they will result in the most negative MTC SR and EOC values that could result from allowed operation of STP Units 1 and 2 from nominal conditions and because the MTC measurement at 300-ppm of boron core condition will ensure, using the SR value of MTC, that the safety analysis MDC will not be exceeded.

3.1 MDC Assumption Used in the South Texas Project Units 1 and 2 Accident Analysis

Westinghouse uses an MDC for performing accident analyses. To perform an analysis for events sensitive to maximum negative moderator feedback, Westinghouse uses a constant value of the MDC of 0.43 delta-K/gm/cc as an assumption throughout the analysis. The average temperature and pressure for HFP and full flow nominal operating conditions are 593.5°F and 2235 psig, respectively. At these conditions, the MTC, equivalent to the MDC of 0.43 delta-K/gm/cc, is -57.6 pcm/°F. The staff reviewed these assumptions and concludes that the evaluation of the MTC from MDC is acceptable because it conforms to the physical relationship of MTC to MDC; that is, the MTC is equal to the MDC times the rate of density with temperature at the nominal pressure and temperature of the coolant at rated thermal power conditions.

3.2 Sensitivity Results

STP Units 1 and 2 TS 3.2.5 provides the LCO of the departure from nucleate boiling (DNB) parameters; reactor coolant systems average temperature (T avg) and pressurizer pressure. The minimum allowable indicated pressurizer pressure is 2201 psig and the maximum allowable (T avg) is 598.0 °F. To account for expected future fuel designs and possible power update conditions, bounding values for RCS pressure of 2201 psig and for RCS temperature of 598.0°F were used for the Westinghouse analyses. The current nominal design (T avg) for STP Units 1 and 2 is 593.0°F so that the safety analysis represents a 5.0°F maximum allowable increase over (T avg) nominal conditions. The current nominal design pressure is 2235 psig, so that the safety analysis represents a 34.0 psi maximum allowable decrease from nominal pressurizer pressure. Based on these maximum allowed system variations, a maximum allowable limit is placed on the moderator density variation. Using the sensitivity of the MTC to temperature and pressure, derived from the analysis of six reload designs, Westinghouse obtained for STP Units 1 and 2 a bounding delta MTC (a proprietary value) associated with these maximum allowable coolant temperature and pressure deviations from nominal conditions.

TS 3.1.3.6 limits control bank insertion by RILs in Modes 1 and 2. All control rods can be inserted at hot zero power (HZP) simultaneously with a reactor trip. In general, control rod insertion results in a more negative

MTC if all other parameters are held constant. However, greater control rod insertion will also reduce the core power and (T avg) which causes the MTC to become more positive. This effect is more pronounced at lower power with the positive change being more important than the negative change in the MTC.

Westinghouse determined that the MTC will be more negative at HFP with control rods inserted to the RIL. Westinghouse analyzed a typical reload core design, using a bounding value of control bank insertion at HFP with no soluble boron in the coolant. This analysis gave a bounding delta MTC associated with the control bank inserted to the RIL for STP Units 1 and 2.

All of the delta MTC values described above are summed to provide a total delta MTC for STP Units 1 and 2 based on the allowed deviations of the various factors from nominal values.

The staff has reviewed the discussion and analysis of the primary factors of the MNF MTC method and concludes that the results obtained are acceptable because approved methods and assumptions were used to generate the results.

3.3 Effect of the MNF MTC on the Safety Analysis

Changes in the parameters discussed previously could take place during a transient to make the MTC more negative than allowed during normal operation. The most adverse conditions seen in the affected transient events will not result in a reactivity insertion that would invalidate the conclusions of the FSAR accident analyses. Thus, the MDC used as a basis for the MNF MTC TS will not change. The reload safety analysis process will include verification that the MDC safety analysis value remains valid. The staff concludes that this verification process for the safety analysis MDC is acceptable.

4.0 SUMMARY

The staff concludes that the proposed change to the method of determining the EOC MTC and 300-ppm SR limit values specified in the COLR is acceptable based on the following considerations:

(1) The most negative feasible MTC method considered the important factors affecting the MTC and the limits on these factors.

(2) Westinghouse used approved methods and computer codes in the analysis.

(3) Measuring the MTC at or near 300-ppm of boron will provide assurance that the MTC at EOC HFP ARO conditions will be less negative than the safety analysis.

(4) The licensee will analyze future reloads for STP Units 1 and 2 to confirm the most negative MTC TS at EOC and SR on MTC at a core condition of 300-ppm of boron.

(5) The licensee will analyze future reloads for STP Units 1 and 2 to confirm that the safety analysis value of the MDC applies.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (56 FR 51926). 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 amendment.

7.0 CONCLUSION

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

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