October 7, 1993

Docket Nos. 50-498
and 50-499

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

Dear Mr. Cottle:

SUBJECT: ISSUANCE OF AMENDMENT NOS. 55 AND 44 TO FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80 - SOUTH TEXAS PROJECT, UNITS 1 AND 2 (TAC NOS. M77380, M77381, M77455, AND M77456)

The Commission has issued the enclosed Amendment Nos. 55 and 44 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 (TSs) in response to your application dated August 10, 1992 (ST-HL-AE-4176), as supplemented by letter dated September 14, 1993.

The amendments change the Appendix A Technical Specifications by revising Technical Specifications 3/4.4.4 and 3/4.4.9 to incorporate the recommendations provided in Generic Letter 90-06. Additional changes to improve clarity and accuracy are also included.

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

Lawrence E. Kokajko, Senior Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 55 to NPF-76
2. Amendment No. 44 to NPF-80
3. Safety Evaluation

cc w/enclosures:
See next page

DF01

Office	PDIV-2/LA	PDIV-2/PE	PDIV-2/PM	OGC	EMEB	PDIV-2/D	
Name	EPeyton	DSkay;nb	LKokajko		JNorberg	SBlack	
Date	9/20/93	9/20/93	9/20/93	9/30/93	9/21/93	10/7/93	
Copy	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No

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cc w/enclosures:
Mr. J. Tapia
Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 910
Bay City, Texas 77414

Mr. J. C. Lanier/M. B. Lee
City of Austin
Electric Utility Department
721 Barton Springs Road
Austin, Texas 78767

Mr. K. J. Fiedler
Mr. M. T. Hardt
City Public Service Board
P. O. Box 1771
San Antonio, Texas 78296

Mr. D. E. Ward
Mr. T. M. Puckett
Central Power and Light Company
P. O. Box 2121
Corpus Christi, Texas 78403

INPO
Records Center
700 Galleria Parkway
Atlanta, Georgia 30339-3064

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

Mr. Joseph M. Hendrie
50 Bellport Lane
Bellport, New York 11713

Judge, Matagorda County
Matagorda County Courthouse
1700 Seventh Street
Bay City, Texas 77414

Mr. William J. Jump
General Manager, Nuclear Licensing
Houston Lighting and Power Company
P. O. Box 289
Wadsworth, Texas 77483

Jack R. Newman, Esq.
Newman & Holtzinger, P.C.
1615 L Street, N.W.
Washington, D.C. 20036

Licensing Representative
Houston Lighting and Power Company
Suite 610
Three Metro Center
Bethesda, Maryland 20814

Bureau of Radiation Control
State of Texas
1101 West 49th Street
Austin, Texas 78756

Rufus S. Scott
Associate General Counsel
Houston Lighting and Power Company
P. O. Box 61867
Houston, Texas 77208

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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. 55
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 10, 1992, as supplemented by letter dated September 14, 1993, 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-76 is hereby amended to read as follows:

   2. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 55, 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 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: October 7, 1993

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. 44
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 10, 1992, as supplemented by letter dated September 14, 1993, 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. 44, 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 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: October 7, 1993

ATTACHMENT TO LICENSE AMENDMENT NOS. 55 AND 44

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 areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
3/4 4-10	3/4 4-10
3/4 4-11	3/4 4-11
3/4 4-31	3/4 4-31
B 3/4 4-2	B 3/4 4-2
--	B 3/4 4-2a
B 3/4 4-15	B 3/4 4-15
--	B 3/4 4-16

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 1816 cubic feet, and at least two groups of pressurizer heaters supplied by ESF power each having a capacity of at least 175 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With only one group of pressurizer heaters supplied by ESF power OPERABLE, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor Trip System breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters supplied by ESF power shall be verified by energizing the heaters and measuring circuit current at least once per 92 days.

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 Both power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or both PORV(s) inoperable, because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With both PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore at least one of the PORVs to OPERABLE status or close their associated block valves and remove power from the block valves and be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

d. With one block valve inoperable, within 1 hour restore the block valve to operable status or place its associated PORV in closed position; restore the block valve to operable status within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

e. With both block valves inoperable, within 1 hour restore the block valves to operable status or place the associated PORVs in the closed position; restore at least one block valve to OPERABLE status within the next hour; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

f. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Performing a CHANNEL CALIBRATION on the actuation channel, and

b. Operating the valve through one complete cycle of full travel.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION b. or c. in Specification 3.4.4.

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

  1) All nonplugged tubes that previously had detectable wall penetrations (greater than 20%),

  2) Tubes in those areas where experience has indicated potential problems, and

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H. The results of these examinations shall be used to update Figures 3.4-2, 3.4-3 and 3.4-4.

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL - RV INTERMEDIATE SHELL R-1606-3
COPPER CONTENT: CONSERVATIVELY ASSUMED AS 0.10 WT%

RT_{NDT} INITIAL: 10°F
RT_{NDT} AFTER 32 EFPY
1/4, 91°F
3/4T, 64°F

CURVE APPLICABLE FOR HEATUP RATES UP TO 100°/HR FOR THE SERVICE PERIOD UP TO 32 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS

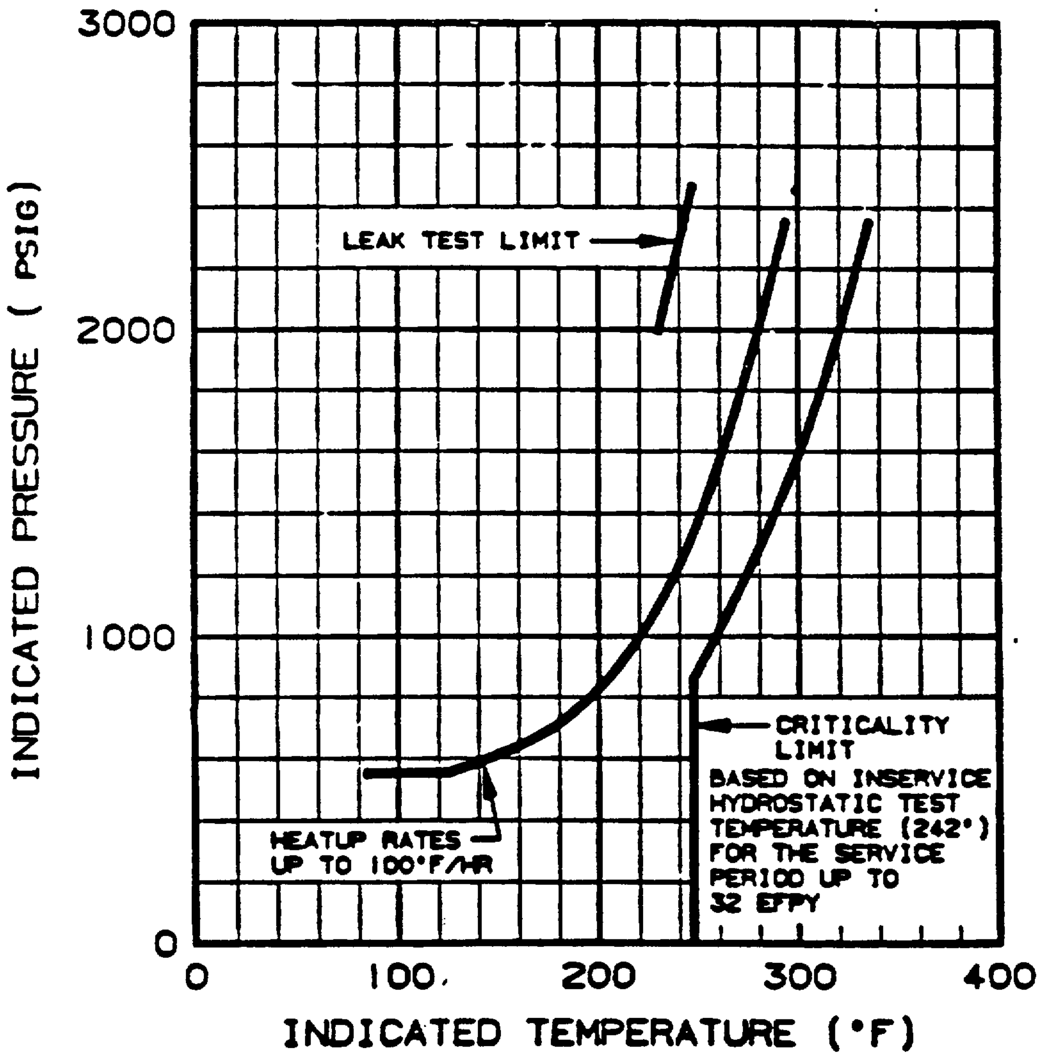

FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 32 EFPY

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 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 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. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the

SAFETY VALVES (Continued)

RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, 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.

RELIEF VALVES (Continued)

C. Manual control of the block valve to: (1) unblock an isolated PORV to allow it to be used for manual control of reactor coolant system pressure (Item A), and (2) isolate the PORV with excessive seat leakage (Item B).

D. Manual control allows a block valve to isolate a stuck-open PORV.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

LOW TEMPERATURE OVERPRESSURE PROTECTION (Continued)

overshoot beyond the PORV Setpoint which can occur as a result of time delays in signal processing and valve opening, instrument uncertainties, and single failure. To ensure that mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require lockout of all high head safety injection pumps while in MODE 5 and MODE 6 with the reactor vessel head on. All but one high head safety injection pump are required to be locked out in MODE 4. Technical Specifications also require lockout of the positive displacement pump and all but one charging pump while in MODES 4, 5, and 6 with the reactor vessel head installed and disallow start of an RCP if secondary temperature is more than 50°F above primary temperature.

Administrative controls and two RHR relief valves will be used to provide cold overpressure protection (COMS) during the ASME stroke testing of two administratively declared inoperable PORVs. During the performance of the PORV function test, two RHR trains will be OPERABLE and in operation with the auto closure interlock bypassed (or deleted) to provide COMS.

With one PORV inoperable, COMS will be provided during the ASME test by the OPERABLE PORV and one RHR relief valve associated with an OPERABLE and operating RHR train which has the auto closure interlock bypassed (or deleted). Each RHR relief valve provides sufficient capacity to relieve the flow resulting from the maximum charging flow with concurrent loss of letdown. The RHR pump design developed head, corresponding to the design flowrate of 3400 gpm, is 205 ft and the actual pump developed pressure is 115 psig. This results in actuation of the RHR relief valves at a RCS pressure of approximately 485 psig (600 psig - 115 psig).

Therefore two OPERABLE and operating RHR trains <u>or</u> one OPERABLE PORV and one OPERABLE and operating RHR train will provide adequate and redundant overpressure protection. Use of the RHR relief valves will maintain the RCS pressure below the low temperature endpoint of the Technical Specification limit curve (550 psig, Ref. Technical Specification Fig. 3.4-2).

With regard to the MODE 6 applicability of this Technical Specification, the statement "with the head on the reactor vessel" means any time the head is installed with or without tensioning the RPV studs.

The Maximum Allowed PORV Setpoint for the COMS will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level

STRUCTURAL INTEGRITY (Continued)

throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Winter 1975.

3/4.4.11 REACTOR VESSEL HEAD VENTS

Reactor vessel head vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of at least two reactor vessel head vent paths ensures that the capability exists to perform this function.

The valve redundancy of the reactor vessel head vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the reactor vessel head vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 55 AND 44 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 10, 1992, as supplemented by letter dated September 14, 1993, 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. The proposed changes would revise Technical Specifications 3/4.4.4 and 3/4.4.9 to incorporate the recommendations provided in Generic Letter 90-06. Additional changes to improve clarity and accuracy are also included.

2.0 BACKGROUND

On June 25, 1990, the staff issued Generic Letter (GL) 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)." The GL represented the technical resolution of the above mentioned generic issues.

Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," involves the evaluation of the reliability of power-operated relief valves (PORVs) and block valves and their safety significance in PWR plants. The GL discussed how PORVs are increasingly being relied on to perform safety-related functions and the corresponding need to improve the reliability of both PORVs and their associated block valves. Proposed staff positions and improvements to the plant's technical specifications were recommended to be implemented at all affected facilities. This issue is applicable to all Westinghouse, Babcock & Wilcox, and Combustion Engineering designed facilities with PORVs. Generic Issue 90, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors," addresses concerns with the implementation of the requirements set forth in the resolution of Unresolved Safety Issue (USI)

A-26, "Reactor Vessel Pressure Transient Protection (Overpressure Protection)." The GL discussed the continuing occurrence of overpressure events and the need to further restrict the allowed outage time for a low-temperature overpressure protection channel in operating modes 4, 5, and 6. This issue is only applicable to Westinghouse and Combustion Engineering facilities.

By letter dated August 10, 1992 (ST-HL-AE-4176), Houston Lighting & Power Company, et. al., (HL&P) requested changes to the South Texas Project, Units 1 and 2 technical specifications in response to GL 90-06.

The licensee had previously responded to GL 90-06 in a letter dated December 21, 1990 (ST-HL-AE-3642). The staff determined that portions of this submittal were acceptable and issued Amendment Nos. 31 and 22 to Facility Operating License Nos. NPF-76 and NPF-80 for the South Texas Project, Units 1 and 2 on November 8, 1991. These amendments consisted of changes to the TS to permit full operability testing of an inoperable PORV following maintenance. These amendments approved only a portion of the requested changes. Additional justification for the unapproved portion of the proposed amendment was requested per teleconference by the staff and was provided in the August 10, 1992, submittal. This safety evaluation addresses the remainder of the changes requested.

3.0 EVALUATION

3.1 Evaluation for Generic Issue 70

The actions proposed by the NRC staff to improve the reliability of PORVs and block valves represent a substantial increase in overall protection of the public health and safety and a determination has been made that the attendant costs are justified in view of this increased protection. The technical findings and the regulatory analysis related to Generic Issue 70 are discussed in NUREG-1316, "Technical Findings and Regulatory Analysis Related to Generic Issue 70 - Evaluation of Power-Operated Relief Valve Reliability in PWR Nuclear Power Plants."

The technical specification (TS) changes in response to Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," consist of the following changes to TS 3/4.4.4, "Relief Valves." An assessment of the proposed TS against the model TS of GL 90-06 for a Westinghouse plant follows.

1. LCO 3.4.4 and Action Statement a. are revised to change all references to PORVs from "all" to "both" since the design of STP includes only two PORVs.

2. Action Statement a. is revised by adding a statement that requires power to be maintained to block valves which have been closed due to excessive PORV leakage. This requirement ensures that the block valves are not rendered inoperable and is consistent with GL guidance.

3. Action Statements a., b., and c. are revised such that they terminate in HOT SHUTDOWN within six hours of the preceding action instead of

terminating in COLD SHUTDOWN within 30 hours of the preceding action. This is necessary since the applicability requirements of the LCO do not extend past the hot standby mode and is consistent with GL 90-06 guidance.

4. Action Statement c. is revised to require that when both PORVs are inoperable, either of the PORVs be restored within an hour rather than both. The GL recommends this action to provide for the removal of power from a closed block valve as additional assurance to preclude any inadvertent block valve opening at a time when the PORV may not be closed due to maintenance to restore it to an operable condition.

5. Action Statement d. is split into two action statements for clarification: one action statement applies when one block valve is inoperable and another action statement applies when both valves are inoperable.

   The change submitted for Action Statement d. deviates slightly from the guidance in the GL in the direction for positioning of the PORV switches in the event of inoperable block valves. The GL guidance is to place the PORV in manual control to preclude its automatic opening and subsequent potential for a stuck-open PORV. The licensee proposes to place the PORV switches in the "close" position in such circumstances. This will likewise preclude automatic PORV opening and the subsequent potential for a stuck-open PORV when the block valve is inoperable and not closed. This is consistent with the intent of the GL and is acceptable.

6. The licensee originally proposed to revise Surveillance Requirement 4.4.4.1.b by adding the statement "during modes 3, 4, or 5" to the requirement to operate the valve through one complete cycle. This deviated from the GL guidance which recommends that Mode 3 or Mode 4 is the preferred test mode. The reasons for Mode 3 or 4 testing are: (1) to verify the capability of the valves to function in an environment more representative of operating conditions, and (2) to perform the test prior to establishing conditions where the PORVs are required for low-temperature overpressure protection. After discussions with the staff, the licensee supplemented its response by letter dated September 14, 1993, to remove the specific mode requirements and return to the original language. In this letter, the licensee committed to administratively control the pressurizer PORV testing and modify procedures to include a specific temperature range to meet the intent of GL 90-06. The licensee will ensure operability of the PORVs by stroke testing the PORVs prior to establishing conditions when the PORVs are used for low temperature over pressure protection. This is consistent with the intent of the GL and is acceptable.

7. Surveillance Requirement 4.4.4.1.a is revised by adding the statement "on the PORV actuation channel" to enhance clarity and provide consistency with Surveillance Requirement 4.4.9.3.1.b. There is no change to the existing requirements.

8. Surveillance Requirement 4.4.4.2 is revised by deleting the statement "with power removed". This statement is not required since it is incorporated into the requirements of ACTION b. and c. of 3.4.4. The revised surveillance requirement is consistent with GL 90-06 guidance.

9. Surveillance Requirements 4.4.4.1.b and 4.4.4.3, as provided in the changes recommended in GL 90-06, are not incorporated in the STP technical specifications. Surveillance Requirement 4.4.4.1.b applies to plants with air-operated PORVs while STP uses solenoid operated PORVs. Surveillance Requirement 4.4.4.3 applies to plants with non-safety grade power while the STP PORVs are powered from class 1E buses.

10. Bases 3/4.4.4 is expanded to identify the major functions for which operability of the PORV and block valves are determined. The proposed Bases deviates from the guidance in GL 90-06 in that automatic control of PORVs is not listed as a function on which operability of the PORV is based. Inoperability of the PORVs automatic function during normal operation does not result in inoperability of the PORV manual operation. This clarification is consistent with the STP design.

The staff has reviewed the licensee's proposed modifications to the STP technical specifications. Since the proposed modifications are consistent with the staff's position previously stated in the GL and found to be justified in the above mentioned regulatory analysis, the staff finds the proposed modifications to be acceptable.

3.2 Evaluation for Generic Issue 94

The actions proposed by the NRC staff to improve the availability of the low-temperature overpressure protection (LTOP) system represents a substantial increase in the overall protection of the public health and safety and a determination has been made that the attendant costs are justified in view of this increased protection. The technical findings and the regulatory analysis related to Generic Issue 94 are discussed in NUREG-1326, "Regulatory Analysis for the Resolution of Generic Issue 94, Additional Low-Temperature Overpressure Protection for Light-Water Reactors."

By letter dated December 21, 1990, the licensee submitted its original response to GL 90-06 and requested changes to the technical specifications. By letter dated November 8, 1991, the NRC issued Amendment Nos. 31 and 22 to the STP license approving some of these requests. These amendments revised Technical Specification 3.4.9.3 to resolve the concerns addressed by Generic Issue 94 and to address a conflict in the TS between TS 3.4.9.3 and TS 4.0.5.

The staff concluded in its safety evaluation that RHR relief valves are an acceptable alternative to the PORVs for LTOP protection for a period not to exceed seven days. In the August 10, 1992, submittal the licensee requested a revision to TS Bases 3/4.4.9 to add details concerning the use of RHR relief valves to provide cold overpressure mitigation system during the stroke testing of inoperable PORVs consistent with TS 3.4.9.3 as approved in Amendment Nos. 31 and 22. This clarification is consistent with operation as described in the Safety Evaluation related to issuance of Amendment Nos. 31 and 22 and is acceptable.

An additional revision to TS 3/4.4.9 was proposed. Technical Specification Figure 3.4-4, "Nominal Maximum Allowable PORV Setpoint for the Cold Overpressure System," is added to the figures listed in Surveillance Requirement 4.4.9.1.2 that are updated based on the results of the reactor pressure vessel (RPV) irradiation surveillance program. The change is made to emphasize the need to update the figure and the allowable PORV setpoint based on results of the RPV irradiation surveillance program. This change is acceptable.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (58 FR 32384). 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.

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

Principal Contributor: Donna Skay, PDIV-2/NRR

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