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

October 31, 1996

DISTRIBUTION·
PUBLIC
PTressler
WBeckner
EAdensam (EGA1)
CGrimes
LHurley, RIV
JKilcrease, RIV f/r

[redacted] File
PDIV-1 r/f
TAlexion (2)
GHill (4)
OGC
ACRS
JDyer, RIV
TMarsh

TCollins
JCalvo
CMiller
AThadani
JRoe
BSheron
GHolahan
EButcher

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 - AMENDMENT NOS. 85 AND 72 TO FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80 (TAC NOS. M92169 AND M92170)

Dear Mr. Cottle:

The Commission has issued the enclosed Amendment Nos. 85 and 72 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 1, 1995, as supplemented by letters dated June 22, August 28, November 22, and December 19, 1995, and January 4, 8 (two letters), and 23, June 27, July 9, August 8, and September 23, 1996.

The amendments allow extension of the standby diesel generator allowed outage time to 14 days, and extension of the essential cooling water loop and the essential chilled water loop allowed outage times to 7 days. The amendments also add to Administrative Controls a description of the Configuration Risk Management Program (CRMP) used to assess changes in core damage probability resulting from applicable plant configurations.

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

DF01 1/1

Docket Nos. 50-498 and 50-499

Enclosures:
1. Amendment No. 85 to NPF-76
2. Amendment No. 72 to NPF-80
3. Safety Evaluation

050075

cc w/encls: See next page

AT 10/18
(A)DD/NRR
AThadani

#1 10/16

WDB 10/31/96
D/PD4-1
WBeckner

DOCUMENT NAME: STP92169.AMD

OFC	PM/PD4-1	LA/PD4-1	BC/EEL	BC/SPLB	(A)BC/SRXB	BC/SCSB	BC/PERB
NAME	TAlexion	PTressler	JCalvo	TMarsh	TCollins	CBerlinger	CMiller
DATE	9/24/96	9/24/96	9/24/96	9/26/96	9/27/96	9/15/96	9/30/96
COPY	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO

OFC	BC/SPSB	BC/TSB	D/DE	D/DSSA	ADT	OGC
NAME	EButcher	CGrimes	BSheron	GHolahan	AThadani	
DATE	9/24/96	10/2/96	10/8/96	10/10/96	10/16/96	10/17/96
COPY	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO

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

October 31, 1996

Mr. William T. Cottle
Executive Vice-President &
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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. 85 AND 72 TO FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80 (TAC NOS. M92169 AND M92170)

Dear Mr. Cottle:

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Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosures:
1. Amendment No. 85 to NPF-76
2. Amendment No. 72 to NPF-80
3. Safety Evaluation

cc w/encls: See next page

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

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. 85
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 1, 1995, as supplemented by letters dated June 22, August 28, November 22, and December 19, 1995, and January 4, 8 (two letters), and 23, June 27, July 9, August 8, and September 23, 1996, 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. 85, 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

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: **October 31, 1996**

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. 72
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 1, 1995, as supplemented by letters dated June 22, August 28, November 22, and December 19, 1995, and January 4, 8 (two letters), and 23, June 27, July 9, August 8, and September 23, 1996, 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. 72, 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

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: October 31, 1996

ATTACHMENT TO LICENSE AMENDMENT NOS. 85 AND 72

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

DOCKET NOS. 50-498 AND 50-499

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

REMOVE	INSERT
3/4 7-13	3/4 7-13
3/4 7-33	3/4 7-33
3/4 8-1	3/4 8-1
3/4 8-2	3/4 8-2
3/4 8-3	3/4 8-3
3/4 8-4	3/4 8-4
3/4 8-8	3/4 8-8
B 3/4 7-3	B 3/4 7-3
--	B 3/4 7-3a
B 3/4 7-7	B 3/4 7-7
B 3/4 8-2	B 3/4 8-2
B 3/4 8-5	B 3/4 8-5
B 3/4 8-8	B 3/4 8-8
6-18a	6-18a

3/4.7.4 ESSENTIAL COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least three independent essential cooling water loops shall be OPERABLE.

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

ACTION:

With only two essential cooling water loops OPERABLE, restore at least three loops to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 At least three essential cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position;

b. At least once per 18 months during shutdown, by verifying that:

   1) Each automatic valve servicing safety-related equipment actuates to its correct position on a Safety Injection, ECW pump start, screen wash booster pump start and essential chiller start test signals, as applicable,

   2) Each Essential Cooling Water pump starts automatically on a Safety Injection or a Loss of Offsite Power test signal, and

   3) Each screen wash booster pump and the traveling screen start automatically on a Safety Injection test signal.

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink shall be OPERABLE with:

a. A minimum water level at or above elevation 25.5 feet Mean Sea Level, USGS datum, and

b. An Essential Cooling Water intake temperature of less than or equal to 99°F.

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

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION is applicable to both units simultaneously.

SURVEILLANCE REQUIREMENTS

4.7.5 The ultimate heat sink shall be determined OPERABLE at least once per 24 hours by verifying the intake water temperature and water level to be within their limits.

3/4.7.14 ESSENTIAL CHILLED WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.14 At least three independent Essential Chilled Water System loops shall be OPERABLE.

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

ACTION:

With only two Essential Chilled Water System loops OPERABLE, restore three loops to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.14 The Essential Chilled Water System shall be demonstrated OPERABLE by:

a. Performance of surveillances as required by Specification 4.0.5, and

b. At least once per 18 months by demonstrating that the system starts automatically on a Safety Injection test signal.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E Distribution System(1), and

b. Three separate and independent standby diesel generators, each with a separate fuel tank containing a minimum volume of 60,500 gallons of fuel.

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

ACTION:

a. With one offsite circuit of the above-required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. Restore the offsite circuit to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With a standby diesel generator inoperable, demonstrate the OPERABILITY of the above-required A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the standby diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE standby diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.2) for each such standby diesel generator separately within 8 hours, unless it can be demonstrated there is no common mode failure for the remaining diesel generator(s). Restore the inoperable standby diesel generator to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one offsite circuit of the above-required A.C. electrical power sources and one standby diesel generator inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Specification 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; and if the standby diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventive

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE standby diesel generators by performing Surveillance Requirement 4.8.1.1.2a.2) within 8 hours, unless it can be demonstrated there is no common mode failure for the remaining diesel generator(s); restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two offsite circuits to OPERABLE status within 72 hours and three standby diesel generators to OPERABLE status within 14 days from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With one standby diesel generator inoperable in addition to ACTION b. or c. above, verify that:

   1. All required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and

   2. When in MODE 1, 2, or 3, the steam-driven auxiliary feedwater pump is OPERABLE.

   If these conditions are not satisfied within 24 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With two of the above required offsite A.C. circuits inoperable, restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

f. With two or three of the above required standby diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing the requirements of Specification 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; restore at least one standby diesel generator to OPERABLE status within 2 hours and at least two standby diesel generators to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least three standby diesel generators to OPERABLE status within 14 days from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the Onsite Class 1E Distribution System shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring the unit power supply from the normal circuit to each of the alternate circuits.

4.8.1.1.2 Each standby diesel generator shall be demonstrated OPERABLE:(2)(11)

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

   1) Verifying the fuel level in its associated fuel tank,

   2) Verifying the diesel starts from standby condition and accelerates to 600 rpm (nominal) in less than or equal to 10 seconds.(3) The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 10 seconds(3) after the start signal. The diesel generator shall be started for this test by using one of the following signals:

      a) Manual, or

      b) Simulated loss-of-offsite power by itself, or

      c) Simulated loss-of-offsite power in conjunction with a Safety Injection test signal, or

      d) A Safety Injection test signal by itself.

   3) Verifying the generator is synchronized, loaded to 5000 to 5500 kW, and operates with a load of 5000 to 5500 kW for at least 60 minutes,(4)(6) and

   4) Verifying the standby diesel generator is aligned to provide standby power to the associated emergency busses.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from its associated fuel tank;

c. Maintain properties of new and stored fuel oil in accordance with the Fuel Oil Monitoring Program.

SURVEILLANCE REQUIREMENTS (Continued)

d. Deleted.

e. At least once per 18 months, during shutdown, by:

1)[10] Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service;

2) Verifying the generator capability to reject a load of greater than or equal to 785.3 kW while maintaining voltage at 4160 ± 416 volts and frequency at 60 ± 4.5 Hz;[4][5]

3) Verifying the generator capability to reject a load of 5500 kW without tripping. The generator voltage shall not exceed 5262 volts during and following the load rejection;[4][5]

4) Simulating a loss-of-offsite power by itself, and:

a) Verifying deenergization of the ESF busses and load shedding from the ESF busses, and

b) Verifying the diesel starts on the auto-start signal within 10 seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the ESF busses shall be maintained at 4160 ± 416 volts and 60 ± 1.2 Hz during this test.

5) Verifying that on a Safety Injection test signal, without loss-of-offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test;

6) Simulating a loss-of-offsite power in conjunction with a Safety Injection test signal, and:

a) Verifying deenergization of the ESF busses and load shedding from the ESF busses;

b) Verifying the diesel starts on the auto-start signal within 10 seconds, energizes the auto-connected ESF (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator

Table 4.8-1

DIESEL GENERATOR TEST SCHEDULE

NUMBER OF FAILURES IN LAST 20 VALID TESTS[8]	NUMBER OF FAILURES IN LAST 100 VALID TESTS[8]	TEST FREQUENCY
≤ 1	≤ 4	Once per 31 days
≥ 2[9]	≥ 5	Once per 7 days

SPECIFICATION NOTATIONS

(1) Loss of one 13.8 kV Standby Bus to 4.16 kV ESF bus line constitutes loss of one offsite source. Loss of two 13.8 kV Standby busses to 4.16 kV ESF bus lines constitutes loss of two offsite sources.

(2) All diesel generator starts for the purpose of these surveillances may be preceded by a prelube period.

(3) A diesel generator start in less than or equal to 10 seconds (fast start) shall be performed every 184 days. All other diesel generator starts for the purpose of this surveillance may be modified starts involving reduced fuel (load limit) and/or idling and gradual acceleration to synchronous speed.

(4) Generator loading may be accomplished in accordance with vendor recommendations, including a warmup period prior to loading.

(5) The diesel generator start for this surveillance may be a modified start (see SR 4.8.1.1.2a.2)).

(6) Momentary transients outside this load range due to changing conditions on the grid shall not invalidate the test.

(7) If Specification 4.8.1.1.2a.2) is not satisfactorily completed, it is not necessary to repeat the preceding 24-hour test. Instead, the standby diesel generator may be operated at 5000-5500 kW for a minimum of 2 hours or until operating temperature has stabilized.

(8) Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, but determined on a per diesel generator basis.

For the purpose of determining the required test frequency, the previous test failure count may be reduced to zero if a complete diesel overhaul to like-new condition is completed, provided that the overhaul, including appropriate post-maintenance operation and testing, is specifically approved by the manufacturer <u>and</u> if acceptable reliability has been demonstrated. The reliability criterion shall be the successful completion of 14 consecutive tests in a single series. Ten of these tests shall be in accordance with the routine Surveillance Requirements

4.8.1.1.2a.2 and 4.8.1.1.2a.3 and four tests in accordance with the 184-day testing requirement of Surveillance Requirements 4.8.1.1.2a.2 and 4.8.1.1.2a.3. If this criterion is not satisfied during the first series of tests, any alternate criterion to be used to transvalue the failure count to zero requires NRC approval.

(9) The associated test frequency shall be maintained until seven consecutive failure free demands have been performed <u>and</u> the number of failures in the last 20 valid demands has been reduced to one.

(10) This test may be performed during power operation provided that the other two diesel generators are operable.

(11) Credit may be taken for events that satisfy any of these Surveillance Requirements.

3/4.7.1.6 ATMOSPHERIC STEAM RELIEF VALVES

The atmospheric steam relief valves are required for decay heat removal and safe cooldown in accordance with Branch Technical Position RSB 5-1. In the safety analyses, operation of the atmospheric steam relief valves is assumed in accident analyses for mitigation of small break LOCA, feedwater line break, loss of normal feedwater and loss-of-offsite power.

3/4.7.1.7 FEEDWATER ISOLATION VALVES

The OPERABILITY of the feedwater isolation valves ensures that no more than one steam generator will blow down in the event of a steam line or feedwater line rupture. The operability of the Feedwater Isolation valves will minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and limit the pressure rise within containment. The OPERABILITY of the feedwater isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analysis.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 10°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Component Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.4 ESSENTIAL COOLING WATER SYSTEM

The OPERABILITY of the Essential Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

When a risk-important system or component (for example Essential Cooling Water) is taken out of service, it is important to assure that the impact on plant risk of this and other equipment simultaneously taken out of service can be assessed. The Configuration Risk Management Program evaluates the impact

ESSENTIAL COOLING WATER SYSTEM (Continued)

on plant risk of equipment out of service. A brief description of the Configuration Risk Management Program is in Section 6.8.3 (administration section) of the Technical Specification.

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available either: (1) provide normal cooldown of the facility or (2) mitigate the effects of accident conditions within acceptable limits.

The limitations on minimum water level and maximum temperature are based on providing a 30-day cooling water supply to safety-related equipment without exceeding its design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974.

3/4.7.6 (Not used)

3/4.7.7 CONTROL ROOM MAKEUP AND CLEANUP FILTRATION SYSTEM

The OPERABILITY of the Control Room Makeup and Cleanup Filtration System ensures that: (1) the ambient air temperature does not exceed the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system, and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rems or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

3/4.7.8 FUEL HANDLING BUILDING EXHAUST AIR SYSTEM

The OPERABILITY of the Fuel Handling Building Exhaust Air System ensures that radioactive materials leaking from the ECCS equipment within the FHB following a LOCA are filtered prior to reaching the environment. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

3/4.7.9 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance

BASES

3/4.7.13 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The temperature limits include an allowance for instrument error of ± 3°F maximum.

3/4.7.14 ESSENTIAL CHILLED WATER SYSTEM

The OPERABILITY of the Essential Chilled Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

When a risk-important system or component (for example Essential Chilled Water) is taken out of service, it is important to assure that the impact on plant risk of this and other equipment simultaneously taken out of service can be assessed. The Configuration Risk Management Program evaluates the impact on plant risk of equipment out of service. A brief description of the Configuration Risk Management Program is in Section 6.8.3 (administration section) of the Technical Specification.

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, D.C. SOURCES, AND ONSITE POWER DISTRIBUTION

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for: (1) the safe shutdown of the facility, and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix A to 10 CFR Part 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least two redundant sets of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of the other onsite A.C. source. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. The term, verify, as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the Surveillance Requirements needed to demonstrate the OPERABILITY of the component.

BACKGROUND

The unit Class 1E AC Electrical Power Distribution System AC sources consist of the offsite power sources [preferred power sources, normal and alternate(s)], and the onsite standby power sources [Train A, Train B and Train C diesel generators (DGs)]. As required by 10 CFR 50, Appendix A, GDC 17, the design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) systems.

The onsite Class 1E AC Distribution System is divided into redundant load groups (trains) so that the loss of any one group does not prevent the minimum safety functions from being performed. Each train has connections to two preferred offsite power sources and a single DG.

Offsite power is transmitted to the plant switchyard at 345 kV by multiple circuits on four separate rights-of-way. The two unit standby transformers are energized from separate busses in the switchyard via independent feeders. Each standby transformer has the capacity to supply the Class 1E loads of both units. In normal operation, the Class 1E loads of each unit can be supplied by the standby transformers and/or its auxiliary unit transformer. In the event of a loss of power from its normal source that unit's Class 1E loads are

A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION (Continued)

manually transferred to the unit's auxiliary transformer or to the standby transformers.

In the event of a loss of preferred power, the ESF electrical loads are automatically connected to the DGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident (DBA) such as a loss of coolant accident (LOCA).

Ratings for Train A, Train B and Train C DGs satisfy the requirements of Regulatory Guide 1.108. The continuous service rating of each DG is 5500 kW with 10% overload permissible for up to 2 hours in any 24 hour period.

Refer to UFSAR Chapter 8 for a more complete description.

APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in the FSAR, Chapter 6 and Chapter 15, assume ESF systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded.

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the Accident analyses and is based upon meeting the design basis of the unit. This results in maintaining at least two trains of the onsite or one train of the offsite AC sources OPERABLE during Accident conditions in the event of:

a. An assumed loss of all offsite power or all onsite AC power; and

b. A worst case single failure.

The AC sources satisfy Criterion 3 of NRC Policy Statement.

A single train onsite AC source can effectively mitigate all but the most severe events with operator action in some cases. The events that cannot be mitigated by a single train onsite AC source are highly unlikely. When a risk-important system or component (for example a Standby Diesel Generator) is taken out of service, it is important to assure that the impact on plant risk of this and other equipment simultaneously taken out of service can be assessed. The Configuration Risk Management Program evaluates the impact on plant risk of equipment out of service. A brief description of the Configuration Risk Management Program is in Section 6.8.3 (administration section) of the Technical Specification.

LCO

Two qualified circuits between the offsite transmission network and the onsite Class 1E Electrical Power System and separate and independent DGs for each train ensure availability of the required power to shut down the reactor and maintain in a safe shutdown condition after an anticipated operational occurrence (AOO) or a postulated DBA.

Qualified offsite circuits are those that are described in the FSAR and are part of the licensing basis for the unit.

A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION (Continued)

TS 3.8.1.1 Action e.

Operation may continue for a period that should not exceed 24 hours. This level of degradation means that the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite AC sources have not been degraded. This level of degradation generally corresponds to a total loss of the immediately accessible offsite power sources. With both of the required offsite circuits inoperable, sufficient onsite AC sources are available to maintain the unit in a safe shutdown condition in the event of a DBA or transient.

TS 3.8.1.1 Action f.

With two or three of the standby diesel generators inoperable, there is insufficient or no remaining standby AC sources. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources are available to power the minimum required ESF functions. A single train onsite AC source can effectively mitigate all but the most severe events with operator action in some cases. The events that cannot be mitigated by a single train onsite AC source are highly unlikely. Since the offsite electrical power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

Surveillance Requirements

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with 10 CFR 50, Appendix A, GDC 18. Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The Technical Specification Surveillance Requirements (SRs) for demonstrating the OPERABILITY of the standby diesel generators are in accordance with the recommendations of Regulatory Guide 1.108, Regulatory Guide 1.137, as addressed in the FSAR and NUREG-1431.

Where the SRs discussed herein specify voltage and frequency tolerances, the following is applicable. The minimum steady state output voltage of 3744 is 90% of the nominal 4160 V output voltage. This value, which is specified in ANSI C84.1, allows for voltage drop to the terminals of 4000 V motors with minimum operating voltage specified as 90% or 3600 V. It also allows for voltage drops to motors and other equipment down through the 120 V level where minimum operating voltage is also usually specified as 90% of name plate rating. The specified maximum steady state output voltage of 4576 V is less

A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION (Continued)

than the maximum operating voltage of 4756 specified for 4000 V motors. It ensures that for a lightly loaded distribution system, the voltage at the terminals of 4000 V motors is less than the maximum rated operating voltages. The specified minimum and maximum frequencies of the standby diesel generators are 58.8 Hz and 61.2 Hz, respectively. These values are equal to plus or minus 2% of the 60 Hz nominal frequency and are derived from the recommendations given in Regulatory Guide 1.108 and NUREG-1431.

SR 4.8.1.1.1.a

This SR ensures proper circuit continuity for the offsite AC electrical power supply to the onsite distribution network and availability of offsite AC electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure that distribution busses and loads are connected to their preferred power source, and that appropriate independence of offsite circuits is maintained. The 7 day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and because its status is displayed in the control room.

SR 4.8.1.1.1.b

Transfer of each 4.16 kV ESF bus power supply from the normal offsite circuit to the alternate offsite circuit demonstrates the OPERABILITY of the alternate circuit distribution network to power the shutdown loads. The 18 month Frequency of the Surveillance is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that the components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 4.8.1.1.2.a.1

This SR provides verification that the level of fuel oil in the fuel tank is at or above the required level.

SR 4.8.1.1.2.a.2

This SR helps to ensure the availability of the standby electrical power supply to mitigate DBAs and transients and to maintain the unit in a safe shutdown condition.

To minimize the wear on moving parts that do not get lubricated when the engine is not running, these SRs are modified by a Note (Note 2) to indicate that all DG starts for these Surveillances may be preceded by an engine prelube period and followed by a warmup period prior to loading.

A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION (Continued)

For purposes of this testing, the DGs are started from standby conditions. Standby condition for a DG mean that the diesel engine coolant and oil are being continuously circulated and temperature is being maintained consistent with manufacturer recommendations.

In order to reduce stress and wear on diesel engines, some manufacturers recommend a modified start in which the starting speed of DGs is limited, warmup is limited to this lower speed, and the DGs are gradually accelerated to synchronous speed prior to loading. In addition, the modified start may involve reduced fuel (load limit). These start procedures are the intent of Note 3, which is only applicable when such modified start procedures are recommended by the manufacturer.

Once per 184 days the DG starts from standby conditions and achieves required voltage and frequency within 10 seconds. The 10 second start requirement supports the assumptions of the design basis LOCA analysis in the FSAR.

The 10 second start requirement is not applicable (see Note 3) when a modified start procedure as described above is used.

The normal 31 day Frequency for SR 3.8.1.2 (see Table 4.8-1, "Diesel Generator Test Schedule," in the accompanying LCO) is consistent with Regulatory Guide 1.108. The 184 day Frequency in Note 3 is a reduction in cold testing consistent with Generic Letter 84-15. These Frequencies provide adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

SR 4.8.1.1.2.a.3

This Surveillance verifies that the DGs are capable of synchronizing with the offsite electrical system and accepting loads greater than or equal to the equivalent of the maximum expected accident loads. A minimum run time of 60 minutes is required to stabilize engine temperature, while minimizing the time that the DG is connected to the offsite source.

The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

This SR is modified by two Notes. Note 4 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 6 states that momentary transients, because of changing bus loads, do not invalidate this test.

A successful DG start under SR 4.8.1.1.2.a.2 must precede this test to credit satisfactory performance.

A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION (Continued)

SR 4.8.1.1.2.b

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel oil tanks once every 31 days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137. This SR is for preventative maintenance. The presence of water does not necessarily represent failure of the SR, provided the accumulated water is removed during the performance of this Surveillance.

SR 4.8.1.1.2.c

The requirements will be controlled and administered by the Diesel Fuel Oil Testing Program located in section 6.8.3 of Administrative Controls.

SR 4.8.1.1.2.e.1

This inspection is conducted once per cycle to ensure unexpected degradation is discovered.

SR 4.8.1.1.2.e.2

Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG load response characteristics and capability to reject the largest single load (785.3 kW) without exceeding predetermined voltage and frequency. The 18 month Frequency is consistent with the recommendation of Regulatory Guide 1.108.

This SR is modified by two Notes. Note 4 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 5 allows the diesel start for this surveillance to be a modified start as stated in SR 4.8.1.1.2.a.2.

PROCEDURES AND PROGRAMS (Continued)

j) Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program", dated September 1995.

Peak calculated primary containment internal pressure for the design basis loss of coolant accident (LOCA), P_a is 41.2 psig.

The maximum allowable primary containment leakage rate, L_a, is 0.3% of primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Primary containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit start-up following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ as-left and $\leq 1.0\ L_a$ as-found for Type A tests.

b. Air lock testing acceptance criteria for the overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

The provisions of Surveillance Requirement 4.0.2 do not apply to the test intervals specified in the Containment Leakage Rate Testing Program.

The provisions of Surveillance Requirement 4.0.3 apply to the Containment Leakage Rate Testing Program.

k) Configuration Risk Management Program (CRMP)

A program to assess changes in core damage frequency and cumulative core damage probability resulting from applicable plant configurations. The program should include the following:

1) training of personnel,

2) procedures for identifying plant configurations, the generation of risk profiles and the evaluation of risk against established thresholds; and

3) provisions for evaluating changes in risk resulting from unplanned maintenance activities.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 85 AND 72 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 1, 1995, as supplemented by letters dated June 22, August 28, November 22, and December 19, 1995, and January 4, 8 (two letters), and 23, June 27, July 9, August 8, and September 23, 1996, 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 allow extension of the standby diesel generator (SDG) allowed outage time (AOT) to 14 days, and extension of the essential cooling water (ECW) loop and the essential chilled water (ECHW) loop allowed outage times to 7 days. The amendments also add to Administrative Controls a description of the Configuration Risk Management Program (CRMP) used to assess changes in core damage probability resulting from applicable plant configurations. The purpose of these proposed changes is to obtain greater flexibility in the scheduling of preplanned preventive maintenance of the SDGs and the ECW and ECHW systems.

The August 8 and September 23, 1996, supplements provided clarifying information and did not change the initial no significant hazards consideration determination.

2.0 BACKGROUND

The licensee's May 1, 1995, application up to and including the January 23, supplement, proposed allowing the extension of an SDG AOT for a cumulative of 21 days, once per train per cycle. In addition, it proposed extending the AOT on each ECW loop for a cumulative of 7 days, once per train per cycle.

Based on feedback from NRC, the licensee revised their application by letters dated June 27, July 9, August 8, and September 23, 1996. The proposed TS changes would now allow extension of the SDG allowed outage time from 72 hours to 14 days, and extension of the ECW loop and the ECHW loop allowed outage times from 72 hours to 7 days. In the unlikely event that a second SDG should become inoperable during the 14-day AOT, there is also a proposed change to extend the allowed outage time for 2 inoperable SDGs from 2 to 24 hours. If all 3 SDGs should become inoperable, the AOT would be the same as in the current TSs (2 hours).

The new amendments would also add to TS Administrative Controls a description of the CRMP used to assess changes in core damage probability resulting from applicable plant configurations. Revised Bases were also proposed consistent with the proposed TS changes.

3.0 PROPOSED CHANGES

TS and Bases 3/4.7.4 - ECW System

The licensee proposes extending the TS AOT for one inoperable ECW loop from 72 hours to 7 days. The proposed Bases indicates that when an ECW loop is taken out of service, the impact on plant risk can be assessed by the licensee's CRMP.

TS and Bases 3/4.7.14 - ECHW System

The licensee proposes extending the AOT for one inoperable ECHW loop from 72 hours to 7 days. The proposed Bases indicates that when an ECHW loop is taken out of service, the impact on plant risk can be assessed by the licensee's CRMP.

TS and Bases 3/4.8.1 - AC Sources

The licensee proposes extending the AOT for one inoperable SDG from 72 hours to 14 days, and extending the AOT for 2 inoperable SDGs from 2 to 24 hours. The licensee proposes that currently existing footnote 10 (which allows testing to be performed during power operation provided the other 2 SDGs are operable) would also apply to the 18-month SDG inspection, and proposes that a new footnote 11 (which allows that credit may be taken for events that satisfy any of these surveillance requirements) would apply to the demonstration of SDG operability.

The proposed Bases indicates that when a SDG is taken out of service, the impact on plant risk can be assessed by the licensee's CRMP. The proposed Bases also indicates that a single train onsite AC source can effectively mitigate all but the most severe events with operator action in some cases, that the events that cannot be mitigated by a single train onsite AC source are highly unlikely, and that the 18-month SDG inspection be conducted once per cycle.

Administrative Controls TS 6.8.3.k - CRMP

The licensee proposes that the CRMP would assess changes in core damage frequency and cumulative core damage probability resulting from applicable plant configurations. The CRMP would include training of personnel, procedures for identifying plant configurations, the generation of risk profiles and the evaluation of risk against established thresholds, and provisions for evaluating changes in risk resulting from unplanned maintenance activities.

4.0 STAFF EVALUATION

The staff evaluated the licensee's proposed amendment to the TSs using both deterministic analysis and probabilistic risk analysis (PRA) methods. The staff's deterministic analysis evaluated the capabilities of a single train of engineered safety features (ESF) equipment at STP to mitigate all design basis events. The results of this deterministic evaluation were then used by the staff to determine the safety impact of extending the AOTs for one SDG, for one train of ECW, and for two trains of ESF equipment. The results of this deterministic evaluation showed that with only one train of ESF equipment available and allowing for some operator actions, the licensee would be able to mitigate all design basis events except for one particular large break loss-of-coolant accident (LOCA) scenario.

4.1.a Evaluation of the SDG AOT Extension

STP is a two unit site originally built with three separate and redundant safety-related electrical power trains per unit. Each of these three safety-related power trains is backed up with its own onsite SDG, and any one of the three SDGs can provide sufficient power to safely shutdown its associated reactor and remove the reactor's decay heat for all risk significant core damage frequency (CDF) sequences identified in the STP plant-specific probability safety assessment (PSA). In addition to the six Category I SDGs, the licensee also has available onsite other diesel generators which can be used to supply emergency power to the Technical Support Centers and balance of plant equipment. In addition to the normal 345 kV sources of offsite power, the STP electrical design includes a 138 kV source of power from a radial line out of Central Power and Light Company's Blessing Substation which can supply emergency power to Units 1 and 2 through a separate emergency transformer. This emergency transformer is physically separated from both the Unit 1 and Unit 2 standby transformers by a minimum of 800 feet.

The licensee has stated that the SDGs will have performance goals set in accordance with 10 CFR 50.65, the Maintenance Rule, and that both diesel generator reliability and availability will be monitored and controlled in accordance with its maintenance rule performance goals. The licensee's monitoring program will be used then to periodically evaluate the overall reliability and availability of the SDGs and to ensure that neither measure of performance significantly decreases before remedial actions are taken. The licensee's station blackout (SBO) reliability goal for SDGs is 0.975.

Each STP unit has three trains of ESF each backed up by its own SDG. Any one of the three Class 1E SDGs per unit can be designated as an alternate AC power source (NRC SBO Safety Evaluation for STP dated July 24, 1995). The licensee states that the circuit between the 138 kV offsite power source, via the emergency transformer, and the onsite Class 1E distribution system, and the technical support center diesel generator, will be functional and available prior to removing an SDG from service. The 138 kV offsite power source, via the emergency transformer, has a capacity greater than any SDG. The licensee states that it will verify at least once per shift that the emergency power transformer breaker alignment is correct and that power is available from the transformer. The above will be satisfied no matter which SDG the licensee removes from service.

The licensee stated that the maintenance activities in the switchyard which could directly cause a loss of offsite power event will be prohibited unless required to ensure the continued reliability and availability of the offsite power sources. Transmission and Distribution personnel will be involved in this planning process to ensure all work to be performed is preplanned and no risk significant work is scheduled in the switchyard during the AOT. The licensee also stated that "current plant procedures will prevent voluntary entry into this LCO [limiting condition for operation] during expected adverse weather conditions." The weather conditions included are hurricane, tornado, and flood watches and warnings.

In view of the capability of the design to mitigate all design basis events with two SDGs and the compensatory measures taken, during the AOT extension, as discussed above, the staff considers acceptable on a deterministic basis performing maintenance on the onsite emergency power sources during power operation.

4.1.b Evaluation of the AOT Extension for Two Inoperable SDGs

In addition to requesting an AOT extension for one inoperable SDG, the licensee also has requested that the AOT for two inoperable SDGs be extended from 2 hours to 24 hours. The staff finds the proposed change to be acceptable for the following reason. In the case of a more typical two train plant design if all onsite AC power is unavailable, then a 2 hour AOT is allowed as long as both offsite power sources continue to be available. At STP, with two of the three SDGs inoperable (for up to 24 hours), an almost fully capable train of ESF equipment will be available to mitigate the consequences of postulated events. The staff evaluated the potential effects of having only one SDG available in Sections 4.3 through 4.6.

4.1.c Conclusions Regarding the SDG AOT Extension

On the basis of its review, the staff finds that the licensee's request for an SDG AOT extension of 14 days for each inoperable SDG is deterministically acceptable. The results of the staff's evaluation in Sections 4.3 through 4.6 show that in almost all cases, the safety systems powered from only one train of onsite AC power are capable of mitigating the consequences of design basis events. For a very few cases, proper and timely operator actions would be required to assure that selected safety systems performed their function. Based on these findings, the licensee's request to extend the AOT for two inoperable SDGs from 2 hours to 24 hours is also deterministically acceptable.

4.2 Evaluation of the ECW and ECHW System AOT Extensions

The ECW system consists of three independent trains, any one of which can successfully mitigate all design basis accidents except for certain LOCA break sizes and locations as defined in Section 4.3.a. Although the ECW system does not directly support high head safety injection (HHSI) pump operation, the ECW system does provide cooling water to the ECHW system which provides cooling water to the HHSI pump room coolers. Thus, the unavailability of an ECW train affects the unavailability of an ECHW train and a HHSI pump.

The ECW and ECHW systems can meet almost all their safety design bases with only one operable train, including all of the more probable accidents and anticipated operational occurrences as defined in Appendix A to 10 CFR Part 50. Therefore, the proposed AOT of 7 days is conservative and an appropriate value for the capability of the three train ECW and ECHW systems.

Based on its evaluation as described above and in Sections 4.3 through 4.6, the staff finds that the proposed change to increase the ECW and ECHW system AOTs is consistent with previously approved AOTs for cooling systems of similar capability (extra redundancy), provides added operational flexibility without compromising plant safety, is adequately supported by the licensee's risk assessment associated with the proposed change, and could potentially result in an overall decrease in plant shutdown risk due to increased availability of the ECW and ECHW systems during shutdown. These findings are also supported by the staff's review and evaluation of the licensee's PSA findings in Section 4.6 of this evaluation. The staff, therefore, concludes that the proposed changes to extend the ECW and ECHW system AOTs are acceptable.

4.3 Evaluation of Safety Related Functions Affected by the SDG and ECW AOT Extensions

The staff has reviewed the licensee's submittal for compliance with the requirements contained in 10 CFR 50.46, noting that a single failure of the emergency core cooling system (ECCS) does not have to be assumed to occur while one ECCS train is out-of-service under the AOT allowed by the TS action statement. However, since the licensee's proposed changes involved extending some AOTs significantly beyond that contained in standard TSs, the staff has evaluated the capability of a single train of ESF equipment at STP to mitigate the consequences of a design basis event to determine the safety impact of the proposed amendment. The events causing the greatest concern during the proposed extended AOTs are the LOCA and the main steamline break (MSLB) accident. A deterministic assessment of these events assumes that there is a loss of offsite power coincident with the LOCA and that the breaks occur in the piping location that results in the greatest loss of inventory. The staff's deterministic evaluation of the proposed amendment included a review of the licensee's LOCA and MSLB accident analyses, and a review of the capability of the residual heat removal (RHR) system to provide long term cooling. The staff also evaluated how the auxiliary feedwater system, the component cooling water system, the fuel handling building filtration system, and equipment qualification would be affected by the licensee's proposed amendment request.

4.3.a Loss-of-Coolant Accident (LOCA)

As per the January 4, 1996, letter from the licensee, the design basis for STP is three ESF trains with consideration of a single failure. The LOCA analysis of record assumes the flow from one train of safety injection (SI) fails (single failure), flow from one train of SI goes out the break, and flow from the third train of SI injects into the reactor coolant system (RCS).

The proposed AOT extensions basically use the same action statement wording as in the current TSs, with the exception that the time to restore the inoperable component to operable status is increased. For deterministic assessment a single failure is not considered when the plant is operating in an action statement. However, the impact of excluding the single failure while in the action statement needs to be reviewed to ensure that there is no significant increase in the risk estimate for the plant.

The large break and small break LOCA analysis of record assumes a loss of offsite power, the break occurs in the cold leg of the RCS, one safety train fails to start, one train of safety injection flow goes out the break into containment and one train provides the required ECCS flow. For the 14-day AOT, if a single failure is not assumed to occur when one ECCS train is inoperable, then the second ECCS train feeds the break, and the third ECCS train is available to inject sufficient ECCS flow into the RCS, consistent with STP's LOCA analysis. Therefore, the 14-day AOT being proposed will not impact the ability of the SI system to provide adequate cooling when no single failure is assumed and the SI system will continue to satisfy the ECCS cooling requirements of 10 CFR 50.46 for all RCS break sizes. The staff agrees with the licensee's assessment because with no additional single failure being assumed for the ECCS, the ECCS cooling capability is consistent with STP's LOCA analyses.

For the 24-hour AOT period, or the 14-day AOT period assuming a single failure of one diesel, circumstances exist wherein the available ECCS pumps may not be able to maintain core cooling. Assuming a loss of offsite power, as required by 10 CFR 50.46, only one train of ECCS would remain available for cooling during the 24-hour AOT period (or the 14-day period assuming the single failure of the diesel). If the break is assumed to occur in the cold leg that the ECCS train is injecting into, all of the safety injection would go out the break into the containment and no core cooling would occur. Thus, sufficient ECCS cooling to meet the requirements of 10 CFR 50.46 cannot be assumed.

Therefore, using deterministic approach and postulating a LOCA, an extended AOT may not be supported.

Realistically, if a small break LOCA occurs while only one ECCS train is in operation (the 24-hour AOT or the 14-day AOT with a single failure), the licensee can depressurize and cooldown the plant using non-safety grade equipment by employing emergency operating procedures for post-accident depressurization and cooldown. Based on the core exit thermocouple temperatures, 0POP05-EO-FRC1, "Response to Inadequate Core Cooling" and 0POP05-EO-FRC2, "Response to Degraded Core Cooling", the operator is directed to depressurize the RCS so that the accumulator and low head safety injection system can provide ECCS flow to supply core cooling. These actions can be taken from the control room in a timely manner. The specific break sizes for which the procedures would be acceptable have not been specified by the licensee.

4.3.b Main Steamline Break (MSLB)

The licensee's current Updated Final Safety Analysis Report (UFSAR) indicates that two trains of safety injection are required to mitigate the return to power and to prevent the fuel from experiencing departure from nucleate boiling (DNB) following a MSLB. As per the licensee, with only one train available the possibility of return to power is increased slightly above the analyzed value. Assuming two safety injection trains are operable, the current calculated departure from nucleate boiling ratio (DNBR) for Unit 1, Cycle 5 and Unit 2, Cycle 5 is 2.61 and 2.04, respectively. The acceptance limit is 1.495. There is significant amount of DNB margin to offset the DNB penalty associated with the potential increase in the reactor power due to only one operable safety injection train following a MSLB.

In a letter dated January 23, 1996, the licensee indicated that the results of its reevaluation confirms that DNB is not expected to occur following a MSLB with only one safety injection train operable. Therefore, the acceptance criteria for a MSLB can be satisfied by only one safety injection train operable.

4.3.c Loss of Charging Pumps

There are only two trains of charging pumps at STP, powered from safety trains A and C. If safety train B is the only safety train available, then the charging pumps would be unavailable. A loss of charging pumps would lead to the loss of seal injection flow to the reactor coolant pump (RCP) seals. Based on the licensee's submittal, if there is a loss of RCP seal injection during a loss of offsite power, the seal cooling can be accomplished by component cooling water (CCW) through the RCP thermal barrier and the CCW associated with safety train B will provide adequate CCW flow to the RCP thermal barrier to prevent seal damage. Based on the above, we find that the RCP seal integrity could be maintained by only one operable diesel generator train.

4.3.d RHR Heat Exchanger Cooling via CCW and RHR System Long-Term Cooling

The licensee indicated that failure to isolate a non-essential header (see Section 4.3.f) would lead to reduced RHR heat exchanger cooling via CCW. The flow rate to the RHR heat exchangers is projected to be 85 percent to 90 percent of design flow and the peak CCW supply temperature to the ECCS components would be approximately 130°F. The licensee concluded that one train of CCW is sufficient to achieve safe shutdown of the plant because the STP units are designed so that hot standby is safe shutdown. This conclusion is consistent with the licensee's assessment documented in the STP docket. Since the recovery of off-site power to the ESF bus is expected within 8 hours per the licensee's SBO analysis, more than one train of CCW will be available to reduce plant conditions to cold shutdown.

STP has the ability to provide long term cooling via the RHR system, which is not a part of the ECCS, or via the steam generator using auxiliary feedwater and atmospheric dump valves (ADVs). During power operation, the RHR system is isolated from the RCS by two motor operated valves. These valves prevent overpressurization of the RHR system, by the RCS, during power operation. If there is a loss of AC power during power operation, the valves will remain closed. RHR function is only required during low pressure operation, at which time, the valves are required to be opened and re-closure of these valves is not a required safety function.

If a LOCA occurs with loss of offsite power during the AOT, long term cooling is accomplished through the low-head safety injection (LHSI) system with suction from the containment sump, for a large break LOCA, or via the auxiliary feedwater (AFW) and ADVs with heat removal through the steam generator, for a small break LOCA prior to RHR initiation. The above long term cooling function could be achieved with only one operable SDG. The AFW is supplied from a safety grade AFW storage tank with a minimum of 495,000 gallons of water. This water storage capacity is sufficient to support 12 hours of AFW operation during which the plant can remain in a hot standby condition for 4 hours and cooldown the RCS to 350°F, at the rate of 25°F/hour. After 12 hours, a minimum of two SDGs must be made operable or offsite must be restored in order to initiate the RHR system for plant cold shutdown. Since off-site power (and therefore RHR) is expected to be recovered within 8 hours, the capabilities of AFW to perform heat removal is sufficient.

4.3.e Auxiliary Feedwater System (AFW)

One of the worst-case design basis feedwater system pipe break (with loss of offsite power (LOOP)) scenarios results in only the Train B AFW pump automatically feeding one steam generator (B). The AFW system has four AFW pumps (A, B, C and D) with the D pump being turbine driven while the other three are motor driven from their respective safety (SDG) bus (A, B and C). The automatic actuation circuits for both A and D pumps are powered from the A electrical train. The actuation circuits for the B and C pumps are powered from the B and C electrical trains, respectively. Therefore, under the analyzed scenario, SDG A is assumed to fail resulting in only AFW Trains B and C automatically starting and feeding Steam Generators B and C, respectively (note that the turbine-driven pump [D] could still be manually initiated). The feedwater system pipe break was assumed to occur at the C steam generator resulting in all the AFW flow from Pump C being fed to the faulty steam generator (exiting through the break). The acceptance limit for this scenario is that the pressurizer does not go solid within 30 minutes. If SDG C is the only diesel being assumed available (under the 24-hour action statement of the proposed AOT period or with a single failure during the 14-day action statement) no AFW flow would be automatically delivered to an intact steam generator. However, if this situation were to occur, ample time exists to realign the C pump to feed one of the intact steam generators to ensure no increase in consequences from this event. In its July 9, 1996, submittal, the licensee indicated it would take 5 minutes for an operator to be dispatched to the isolation valve cubicle and another 10 minutes to perform the necessary valve manipulations to feed an intact steam generator. The staff concludes that these operator actions are acceptable to prevent a water solid pressurizer.

4.3.f <u>Component Cooling Water System (CCW)</u>

The CCW system is a three train system. However, situations could occur with only one train available where the remaining train may not be able to supply design basis flow to all components without operator action. In addition to the accident heat loads, there are two non-essential headers fed by all three trains of the CCW system. One is referred to as the common header and the other, the nonsafety header. The common header provides cooling to the spent fuel pool cooling (SFPC) system while the nonsafety header provides cooling water to the boron recycle system (BRS), the letdown heat exchanger and other small nonessential loads, including the charging pumps. None of the loads supplied by either of these non-essential headers are required for safe shutdown or to mitigate the consequences of any design basis accidents. There are two series motor operated isolation valves (MOVs) for each of these headers. One MOV on each line is powered by Train C, while the other is powered by Train A on the nonsafety header and Train B on the common header. These valves are normally open and automatically close on an ESF signal to assure adequate flow to required accident loads (RHR pumps and heat exchangers, and reactor containment fan coolers) in the event of accident conditions. Unless the one remaining operable electrical train is Train C, one of the headers would not isolate in the event of an ESF signal (assuming only one SDG is available). With only one CCW pump available (which occurs during the 24-hour SDG AOT, or if a single active failure is assumed during the 14-day SDG AOT) and one of these headers unisolated, enough flow is diverted so that the accident loads may not receive the design CCW flow. However, operator actions can be taken to manually close the affected MOV. Since the operators will be aware of being in the LCO, they would also be aware of, and have procedures for necessary operator actions in the event an accident or transient occurs. This should increase the potential to restore design CCW flow to the required components. Even if operator actions are not taken to restore design basis CCW flow to the accident loads, the results are acceptable as described in Sections 4.3.c, 4.3.d and 4.4.a of this evaluation.

Therefore, the staff concludes that the CCW system is capable of handling all postulated events with one CCW train if credit is given for successful operator actions, i.e., manual valve operation. Note that operator actions are not required if the operable SDG is Train C since both nonessential headers would still automatically isolate on an ESF signal.

4.3.g Fuel Handling Building (FHB) Filtration System

The licensee has identified that only two trains of FHB filtration are diesel generator backed (Trains A and B). Thus, if SDG C is the only available diesel generator, the capability for FHB filtration is assumed lost. During power operation, the primary safety function of the FHB filtration system is to mitigate the consequences of a design basis LOCA by ensuring that radioactive materials leaking from the ECCS equipment within the FHB following a LOCA are filtered prior to reaching the environment. Assuming that the entire FHB filtration system is made up of only two trains is a conservative assumption. The actual design has only two filter trains but uses three 50 percent trains of exhaust fans to provide the required air flow through the filters. Each train of exhaust fans is powered from a separate diesel generator backed bus (Trains A, B and C). Therefore, if SDG C was the only diesel generator available, one train of filtration is still available with the exception that there would be no power to the either of the heaters (Train A and B) in the flow path to the filter units. With no heaters available, the efficiency of the filtration units could be reduced if the moisture in the air stream reached a 70-percent-relative-humidity level or higher. Thus, calculated offsite dose limits could potentially be exceeded if the worst case LOCA were assumed. However, procedures are available to energize the Train B heaters from the C SDG. The current TSs provide an AOT of 7 days if one of the filter trains is inoperable for any reason, e.g., loss of one heater or loss of any one fan. This basically means that, for 7 days, the system does not meet the single failure criterion even without a LOOP. The 7 days is based on a pure two train system (100 percent each) plus the fact that FHB filtration is not an ECCS function and, therefore, even a complete loss of function does not affect the CDF or the amount of core damage that can occur. Based on the above, the staff concludes that AOTs extensions for the SDGs are acceptable from the standpoint of potential FHB filtration system failure effects.

4.3.h Equipment Qualification (EQ)

For a main steam line break inside containment or a large break LOCA, the existing EQ licensing basis assumes two trains of containment spray and two trains of reactor containment fan coolers (RCFCs). During the extended AOTs for the SDGs (one operable for analysis purposes) this could be reduced to one train. The licensee reviewed the relevant analyses and concluded that adequate EQ margin exists to accommodate the resulting increase in temperature and pressure. Based on the licensee's conclusions, the required equipment is qualified for the resulting temperature and pressure profiles. Therefore, with only one SDG operating, a main steam line break inside containment or a large break LOCA does not result in exceeding the pressure and temperature EQ limits of the necessary equipment. However, in its August 28, 1995, submittal, the licensee indicated that for the large break LOCA, the EQ radiation dose limits inside containment may be exceeded if only one train of electrical equipment is available. Given the staff's experience that the radiation dose for which equipment is usually qualified is much higher than actual doses that are calculated to occur following a LOCA, the staff requested further clarification of the licensee's statement. The licensee clarified that the calculated post LOCA doses would increase (however, dose limits would not be exceeded) if only one train of containment spray and one train of RCFCs were assumed to operate. By letter dated November 22, 1995, the licensee verified that the resulting expectant dose would still be below the EQ limits, and this is consistent with the staff's experience in this area. Therefore, the staff concludes that with only one diesel generator available, the conditions inside containment following a steam line break or a LOCA will not result in exceeding the EQ limits of equipment necessary to mitigate either of the assumed accidents.

4.3.i Anticipated Transients Without Scram (ATWS)

The licensee indicated that two trains of AFW are included in the analysis of secondary heat removal following an ATWS. Three of the trains, A, B, C, are motor operated pump trains and one is a turbine driven pump train D. The three motor driven pump trains receive an ESF signal to start from their respective safety trains, while AFW train D receives a start signal from safety train A.

Since the start signals from the safety trains are backed up by plant batteries, an inoperable SDG would not render inoperable its associated safety train for generating a start signal to the AFW pump. Therefore, under the conditions that there is only one operable SDG, a motor driven AFW pump and the turbine driven AFW pump will be operable to satisfy secondary heat removal requirements following an ATWS.

4.3.j Conclusions Regarding Safety Related Functions Affected by the SDG and ECW AOT Extensions

The licensee has proposed to allow continued operation of STP for a period of up to 14 days with only two SDGs operable, for a period of 7 days with only two trains of ECW and ECHW operable, and for a period of up to 24 hours with only one train of ESF equipment available. With only one train of ESF equipment out-of-service at STP, and provided that the two trains which remained available could mitigate all design basis events assuming a single failure, then an essentially unlimited AOT could be allowed for one train of ESF equipment based on the general design criteria for light water reactors contained in 10 CFR Part 50, Appendix A. The design of STP, however, is such that if a particular small break LOCA in the cold leg should occur while only one train of ESF equipment is in operation, then the licensee would have to rely on operator actions to depressurize and cooldown the RCS, and if a particular large break LOCA in the cold leg were to occur, then the licensee would not be able to mitigate the postulated accident.

In past cases involving individual plant designs, where the designs have included more redundancy than the required two train minimum (e.g., three pumps, where each pump is fully capable and redundant to the other two pumps), a 30-day AOT has been allowed. In the case of a more typical two train plant design, an AOT of 72 hours is allowed when redundant ESF equipment becomes inoperable. If both trains of ESF equipment for a two train plant design become inoperable, then an immediate shutdown is required. For any plant design if all onsite AC power is unavailable, then a 2-hour AOT is allowed as long as offsite power continues to remain available. Therefore, with two operable trains of ESF equipment available, AOTs from between 72 hours and 30 days can be justified for the STP design. This is based on STP's similarity to previously accepted designs and the AOTs allowed for similar designs and the more typical two-train plant designs. The licensee's proposed AOT extensions are within the time limits previously allowed. Likewise, since an almost fully capable train of ESF equipment will be available to mitigate the consequences of any postulated event, the licensee's request to extend the AOT for operating with only one train of ESF equipment (one SDG) can also be justified based on STP's design.

The staff has performed a deterministic evaluation of the licensee's proposed amendment, using engineering judgement to evaluate the risk associated with single train operation of STP, and determined that the proposed amendment is acceptable. Based on its review, the staff has concluded that the STP design has sufficient redundancy to allow the proposed AOT extensions and that the STP design will continue to meet the requirements of 10 CFR 50.46. The results of the staff's probabilistic risk analysis evaluation (Section 4.6) supports this deterministic evaluation and also finds that the proposed AOT extensions are acceptable from an overall risk perspective.

4.4 Impact on Containment Design Basis and Safety Functions

4.4.a Design Basis Accidents

The design pressure and temperature for the containment structure is established based on consideration of a spectrum of pipe break sizes, break locations, initial reactor power levels, and possible single failures. For STP, loss of coolant accidents are limiting for peak containment pressure, and main steam line break accidents are limiting for maximum containment temperature. In support of the requested TS amendment, the licensee considered the impact on containment response for the spectrum of accidents if the design basis pipe break were to occur at STP while one of the SDGs for that unit were in an extended outage, and showed that the design basis accident remains limiting.

The licensee confirmed that the containment spray trains share only the Refueling Water Storage Tank, the spray ring headers, and some piping to the spray ring headers, and that there are no dependencies in electrical power, instrumentation, or support systems across trains for the containment spray system. Thus, for any combination of two diesel generators failed or out of service, one train of containment spray will remain available. Two RCFCs would be available given any one train of electrical power is in service. However, TS 3.6.2.3 allows one RCFC to be inoperable. In assessing the potential impact of the requested AOTs on the design basis, the licensee conservatively assumed one of the two RCFCs to be inoperable.

In addition to reducing the number of available containment spray trains and RCFCs from that assumed in the design basis accident (DBA), unavailability of a second diesel will result in a temporary degradation in the performance of the operable spray train and RCFC. Specifically, the spray initiation time would increase by about 50 seconds because the same volume of spray pipe and spray ring headers would be filled by one pump rather than two. Also, if the diesel generator failures include the C train one of two CCW headers will not automatically isolate, resulting in a 10 percent decrease in flowrate and a 15°F increase in supply temperature to the operable RCFC (from design values) until the header is manually isolated. The licensee accounted for this degradation in safety system performance in their assessment of containment pressure and temperature response described below.

The DBA for peak containment pressure is a double-ended pump suction guillotine break with maximum safety injection and minimum containment heat removal (two of three trains of containment spray and three of six RCFCs operate). The peak calculated containment pressure for the DBA is 37.5 psig, and the containment design pressure is 56.5 psig. With only one train of containment spray and one RCFC in operation (and degraded as described above), the licensee estimates the peak containment pressure for the design basis break to be 46 psig. The margin between peak calculated and design pressure is sufficient that peak pressure will remain below design for the entire spectrum of pipe breaks considered in the UFSAR if only one safety train is available. The staff concludes that the containment design pressure remains valid and would not be exceeded if the design basis pipe break were to occur at STP while one of the SDGs for that unit was in an extended outage, and a second SDG was unavailable.

The DBA for maximum containment temperature is a double-ended rupture of the main steam line with main steam isolation valve (MSIV) failure and maximum containment heat removal (three trains of containment spray and five of six RCFCs operate). Peak calculated containment vapor temperature for the DBA is 323°F. The design temperature for the containment structure is 286°F, based on an analysis that assumes the peak vapor temperature is maintained for approximately 500 seconds. With only one train of containment spray and one RCFC in operation (and degraded as described above), the licensee estimates the maximum vapor temperature for the design basis break to be 329°F. The licensee also estimates that the calculated vapor temperatures exceed the design temperature for the containment structure (286°F) for approximately 300 seconds. Although the maximum vapor temperature with one train of sprays operating is higher than assumed in the design basis assessment, the period of time that the vapor temperature exceeds the structure design temperature is much less than in the design basis assessment (300 versus 500 seconds), and offsets the slightly higher vapor temperature. Because the temperature profile with one train operating is less severe than the temperature profile on which the structure design temperature is based, the staff concludes that the structure design temperature of 286°F remains bounding and would not be exceeded if the design basis pipe break were to occur at STP while one of the SDGs for that unit was in an extended outage, and a second SDG was unavailable.

4.4.b Containment Isolation

Containment isolation of various systems is accomplished in STP by two of the three safety trains. By letters dated August 28, and November 22, 1995, the licensee provided information regarding the ability to isolate containment given the loss of any two diesel generators. The licensee's assessment was based on a screening analysis of all containment penetrations that are required to isolate in the event of an accident, and a quantitative assessment of the containment isolation failure frequency with and without the requested AOTs.

Through the screening analysis, the licensee determined that most of the penetrations have an air operated valve which will be closed by ESF actuation or fail closed on a loss of instrument air. The only required containment isolation lines with MOVs both inboard and outboard are the containment radiation monitoring line (supplied by safety Trains A and B) and RCP seal return lines (supplied by safety Trains B and C). In both cases, these lines are small and emergency operating procedures contain instructions to manually isolate the lines using local, manually operated valves in the event of a loss of all AC power.

The licensee provided a comparison of the containment isolation failure frequency with and without the TS changes requested in their May 1, 1995, application (i.e., a cumulative SDG AOT of 21 days once per train per cycle, and a cumulative ECW AOT of 7 days once per train per cycle). Containment isolation failure is defined in the PSA as a failure to close at least one valve in each containment penetration. Failure to isolate small diameter and large diameter penetrations is modelled by separate top events in the PSA. Risk from containment isolation failures is dominated by failure to isolate large diameter lines. The frequency of core damage with concurrent failure to isolate large diameter lines is 1.3 E-7 per reactor-year in the base case PSA (without the requested AOTs). With the originally-requested AOTs, the failure frequency would increase to 1.9 E-7 per reactor-year. Although this represents a 50 percent increase in isolation failure frequency, the increase is small in absolute terms. The licensee subsequently revised their application to limit the SDG AOT to 14 days rather than 21 days, as described in Section 3.0. The impact of the revised TS changes on containment isolation failure frequency were not requantified but these changes would tend to reduce the frequency of containment failure below 1.9E-7 with the proposed AOTs. The staff concludes that the increase in isolation failure frequency would remain small and is therefore acceptable.

In order to provide heightened awareness among the operating staff during the requested AOTs and to prevent entry into the AOTs while in an action statement associated with containment integrity or containment purge valves, the licensee committed to a number of compensatory measures related to plant operations prior to and during the requested AOTs (Attachment 4 of the May 1, 1995, submittal). This includes a commitment that prior to commencement of maintenance under the proposed AOTs, containment integrity will be verified to ensure containment isolation penetrations are in their proper alignments and the reactor containment building supplemental purge valves will be verified to be operable and in their proper alignment. Additionally, containment purges that may be required during the AOTs will be strictly controlled. In a letter dated January 4, 1996, the licensee indicated that the requirement to perform this compensatory action is included in the administrative procedure that will be performed prior to each entry into the AOT action statements.

The staff concludes that the containment isolation function and the design criteria of General Design Criteria (GDC) 56, Appendix A to 10 CFR Part 50 will continue to be met if an accident occurs at STP while one SDG is in an extended AOT. If an accident were to occur while two SDGs were inoperable, then automatic isolation of the containment radiation monitoring line (supplied by safety Trains A and B) or the RCP seal return lines (supplied by safety Trains B and C) is not guaranteed. In both these cases, however, the lines are small and emergency operating procedures contain instructions to manually isolate the lines using local, manually operated valves in the event of a loss of all AC power. Based on this fact, the staff concludes that the estimated increase in containment isolation failure frequency associated with the requested AOTs would not represent a significant increase in the total risk for STP.

4.4.c Hydrogen Control

STP has two trains of hydrogen recombiners. Backup power to the recombiners is supplied by safety Trains B and C. By letter dated November 22, 1995, the licensee provided an assessment of the ability to power the hydrogen recombiners from alternate power sources, given the loss of preferred power sources. Entry into the proposed AOT action statements requires the Emergency Transformer and associated 138 kV transmission line to be available. In the event of a loss of the preferred offsite power sources, the Emergency Transformer secondary may be aligned to any one of the three 4.16 kV ESF busses, including either bus powering a hydrogen recombiner. In the event of an extended loss of all offsite power sources, including the Emergency Transformer, two ESF busses may be powered by a single standby diesel generator. Plant Procedure 0POP04-AE-0001 defines the procedural steps for cross-connecting ESF Train B to either ESF Train A or C, and similar procedural steps would apply to cross-connection of any one ESF bus to either remaining ESF bus.

The hydrogen recombiners would not be needed for at least 11 days following a postulated DBA. This would allow the licensee considerable time to either restore offsite power or to complete the necessary procedural steps needed to realign the hydrogen recombiners to an operable SDG before hydrogen recombiner operation was required. Based on the ability to power the hydrogen recombiners from alternate power sources, and the considerable time available to the licensee to realign the hydrogen recombiners to an operable power source before operation is required, the staff concludes that the hydrogen recombiner system would be available to perform its function if an accident were to occur at STP even if two SDGs for one unit were unavailable.

4.5.a Potential Radiological Consequences of the Proposed Amendment

Extending the allowed outage times for the SDGs increases the chance that only one train of containment spray will be available during a large break LOCA. The current DBA analysis assumes that a single failure will result in no less than two trains of spray remaining operable to remove radioactive iodine from the source term in the post-LOCA containment atmosphere (i.e., reduce the amount of radioactive iodine available for leakage from containment). With one of the three trains of containment spray out of service during an extended allowed outage, a single failure disabling a second train leaves only one train operable. The resulting reduced flow would negatively impact the ability of the Containment Spray System to mitigate the thyroid dose of individuals at the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ) as well as thyroid doses to operators within the control room. Since containment spray is not credited for removing radioactive noble gas releases during a LOCA, the calculated whole body doses to these individuals are unchanged.

4.5.b Offsite Radiological Consequences

Operation of the Containment Spray System with a single train (e.g., single pump) will reduce the system pressure and flow. The effectiveness of the containment spray at removing iodine from the containment atmosphere is inversely proportional to the mean diameter of the liquid droplets in the spray. The current analysis is based on the droplet size distribution measured during spray nozzle testing at system design pressure and flow. In lieu of providing a revised spray droplet size, the licensee's evaluation of the possible radiological impact of the extended SDG AOTs takes no credit for the removal of elemental iodine by the containment spray. As discussed in a letter dated January 8, 1996, only iodine removal by wall deposition was included in the licensee's evaluation.

An elemental iodine deposition coefficient of 4.5 per hour was calculated using the methods and assumptions on page 6.5.2-10 of the NUREG-0800 Standard Review Plan, Rev. 2 (SRP). A conservatively bounded surface area of 92,900 m^2, and a conservatively bounded net free volume of 100,808 m^3 were used as the inputs to the elemental iodine removal coefficient calculation. This initial deposition removal rate is assumed to continue until a decontamination factor (DF) of 100 is reached (i.e., the airborne concentration is one percent of its initial value). The removal rate was then assumed to continue at a rate that is five percent of the initial removal rate until a decontamination factor of 200 is reached, at approximately 4.1 hours after initiation of the DBA. No further elemental iodine removal is assumed. Section 6.5.2 of the SRP limits the DF for elemental iodine to a maximum value of 200. This model for elemental iodine removal by surface wall deposition is consistent with the model for elemental iodine removal by wall deposition currently in the STP UFSAR and is conservative compared with the staff's guideline as specified in SRP 6.5.2, Revision 2. Therefore, it is acceptable to the staff.

The staff performed an independent analysis of the thyroid doses resulting from a postulated LOCA during an extended diesel outage using the licensee's iodine removal model and the methods and assumptions in the SRP. The thyroid dose to individuals at the EAB and LPZ listed in the table below were calculated with the HABIT computer code. Input parameters were taken from Table 15.5 on the "Safety Evaluation Related to the Operation of South Texas Project, Units 1 and 2" (NUREG-0781).

4.5.c Control Room Operator Doses

By letter dated September 26, 1991, the NRC issued Amendment Nos. 28 and 19 to the STP operating licenses for Units 1 and 2, respectively. These amendments addressed an identified single failure of a heater in the control room ventilation system that resulted in control room operator DBA doses in excess of those previously analyzed. As described in the accompanying staff's Safety Evaluation (SE), the postulated heater failure reduces the iodine removal efficiency of the ventilation system charcoal filtration units.

The staff independently evaluated the radiological impact of the licensee's request to extend the SDG AOTs on the habitability of the control room during a postulated DBA. Using input parameters taken from Table 6.1 of NUREG-0781, as modified by the September 26, 1991, SE, the thyroid doses to control room operators during the course of a postulated DBA were calculated with the HABIT computer code. The results are listed in the table below with the current DBA results and the associated acceptance criteria for comparison.

RADIOLOGICAL CONSEQUENCES OF A LOCA WITH ONE SDG INOPERABLE			
	CURRENT BASIS WHOLE BODY/ THYROID DOSE (REM)	PROPOSED WHOLE BODY/ THYROID DOSE (REM)	ACCEPTANCE CRITERIA
EAB (0-2 hr)	3.6 / 165	3.6 / 199	25 / 300
LPZ (0-30 day)	1.3 / 74	1.3 / 88	25 / 300
CR (0-30 day)	3.1 / 17	3.1 / 30	5 / 30

By letter dated January 4, 1996, the licensee identified that if a LOCA during the proposed extended AOTs resulted in a single operable train of control room ventilation, the current design could not maintain the required 1/8 inch (water gauge) of positive pressure in the control room envelope with respect to adjacent areas. The staff questioned a statement in the January 4, 1996, letter regarding the ability of the ventilation system to maintain a positive pressure in the control room. The licensee conceded that the entire control room ventilation envelope may not be maintained at a positive pressure during single train operation. Testing of the control room ventilation system with only one train running in October 1994, resulted in a negative relative pressure (0.04 inch) in one equipment room within the control room envelope.

The requirement that the control room envelope be maintained at a positive pressure during an accident insures that any leakage will be clean air out of the control room. Without the assurance that the control room will remain at a relative positive pressure, the 10 cubic feet per minute (CFM) of unfiltered in-leakage (a standard assumption to account for opening and shutting doors) used in the staff's evaluation of this request (and the DBA in NUREG-0781) is invalid. The possibility exists therefore that operator thyroid doses could exceed the acceptance criteria of GDC 19 in Appendix A to 10 CFR Part 50 if a LOCA/LOOP occurred and only a single train of control room ventilation was available. However, the current TSs provide an AOT of 72 hours if two trains of control room ventilation are inoperable. Therefore, operation for up to 24 hours with only a single onsite power source, and its effect on control room ventilation, is bounded by current TSs.

4.5.d <u>Conclusions Regarding Radiological Consequences</u>

The results of the staff's evaluation indicate that if a LOCA were to occur at one of the STP units while one of the SDGs for that unit was inoperable the siting criteria for radiation doses at the Exclusion Area Boundary and the Low Population Zone in 10 CFR Part 100 would still be met. If a LOCA were to occur when only a single train of control room ventilation was operable, the possibility exists that operator thyroid doses would exceed the acceptance criteria of GDC 19 in Appendix A to 10 CFR Part 50. The staff has reviewed this possibility and determined it to be acceptable.

4.6 <u>Evaluation of the Probabilistic Safety Analysis (PSA) Used to Support the Proposed Amendment</u>

The staff used a three-element approach in its evaluation of the risk associated with the proposed TS changes.

- The first element was an evaluation of the impact on plant risk as expressed by the change in core damage frequency (delta CDF), the incremental core damage probability (delta CDP) and the impact on the large early release frequency (LERF) resulting from the increased AOTs.

- The second element was an evaluation of the licensee's process used to address potentially high risk configurations that could exist if equipment in addition to that associated with the changed AOTs were to be taken out of service simultaneously, or other risk significant operational factors such as concurrent system or equipment testing were also involved. The objective of this part of the staff's review was to assure that appropriate restrictions on dominant risk-significant configurations associated with the changed AOT were considered in appropriate procedures.

- The third element was an evaluation of the licensee's overall configuration risk management system to assure that adequate programs and procedures would be in place to identify and compensate for other potentially lower probability, but none the less risk significant, configurations resulting from maintenance and other operational activities.

Each of these three elements of the staff's evaluation is discussed separately below.

Element 1 - PSA Evaluation of AOT Extension

The licensee stated in their submittals that the incremental risk associated with the proposed AOT extensions has been determined to be small. In addition, the licensee stated that a risk assessment would be performed in accordance with the STP Configuration Risk Management Program, to determine if further restrictions are warranted while not meeting the LCO. The STP PSA, therefore, plays an important role in understanding and implementing the proposed AOT extensions at STP.

(a) Evaluation of PSA Model, Data and Assumptions

The STP PSA includes a Level 1 and Level 2 analysis, with external events. The Level 1 analysis used the large event tree/small fault tree methodology, which explicitly addressed system dependencies in the event trees. Small fault trees were used to quantify the likelihood of system failure, which then provided input to the event tree nodes. RISKMAN software was used to quantify the CDF.

The current PSA used fault trees for all system logic modelling, which is an improvement compared to the use of reliability block diagrams with some fault trees used in earlier versions of the PSA. Additionally, in the current PSA, common cause failures due to system dependencies were incorporated directly in the event tree logic using the Multiple Greek Letter (MGL) method.

Data collection was performed through examination of generic and plant-specific sources. The latest set of data includes plant-specific experience related to plant trips at both units. These data were incorporated into the PSA through a Bayesian updating process that utilized the PLG-0500 generic database. Additionally, where there was scarce initiating event frequency data, models and expert judgement were used to supplement available information. Finally, the PSA credited the emergency transformer which had not been included in the previous PSA.

STP personnel participated in evolution of the PSA, which included model development, data collection, and requantification of models with updated plant-specific data. In addition to reviews performed by inhouse PSA engineers, reviews were also performed by an independent internal team consisting of personnel from the operations, training, and engineering backgrounds.

In 1991, the NRC staff completed an in-depth review of the STP PSA (see NUREG/CR-5606), and found the level of detail of the models "quite high and consistent with current start-of-the-art." A subsequent update of the PSA included a variety of CDF estimates for various assumptions regarding the rolling maintenance schedule and combinations of modified TS AOTs and Surveillance Test Intervals (STIs). The PSA was again updated in March of 1995 to include the NRC staff-approved risk-based TS AOTs and STIs, plant-specific equipment failure rate data, and incorporation of the emergency transformer into the model.

The staff concludes that the licensee's PSA can appropriately evaluate the impact of the proposed TS change on CDF and containment performance.

(b) PSA Insights and Findings

For each LCO the licensee evaluated AOT-induced changes in the plant CDF, which also allowed for the determination of the corresponding incremental conditional core damage probability (ICCDP) during the AOT period. The Large Early Release Probability (LERP), defined in the licensee's submittal as a large (>3" diameter hole) and early containment failure or bypass that possesses a significant potential for short term health impact, was calculated for the modified AOT conditions of each LCO.

The licensee's PSA estimated the total STP-CDF to be approximately 2E-5/yr based on 3 day AOTs for the SDGs. To this total, station blackout (a subset of loss of offsite power events) contributed approximately 18%, or 4E-6/yr. Sensitivity studies indicate that extending and completely utilizing SDG AOTs of 14 days, and 21 days (a conservative assumption) would only increase the SBO-CDF contribution by approximately 7E-7/yr and 2E-6/yr respectively. The resultant change in total CDF of less than 10% would continue to keep STP at the low end of the PWR spectrum, for both SBO induced CDF and total CDF. Likewise the relatively small LERF of approximately 5E-7/yr and small ICCDP of 7E-7/yr when utilizing the expected average time associated with the AOT extension, indicate that change would have minimal risk significance. The lack of sensitivity of SBO risk to SDG-AOT extensions is primarily due to the redundancy in the STP emergency AC power design. Other "qualitative" factors which further reduce the SBO contribution to plant risk are discussed below.

Element 2 - Evaluation of LCO Configuration Restrictions

The licensee has produced approximately 300 pre-calculated configurational conditional CDF estimates used to estimate a large number of configuration risks. From this process, certain plant configurations were identified as being potentially risk-significant if entered during the AOT. These configurations dictated which equipment must be available while in the AOT. This process is the second element of the three-element process, and specifies additional procedures that apply during the AOTs.

Licensee procedures state that, for entry into the proposed LCOs with the proposed AOT extensions, certain actions need to be taken, or certain maintenance activities precluded. These activities, or conditions, are stipulated in the licensee's procedures for the AOT and listed below.

(a) The requirements for two (2) of the onsite power sources specified in Specification 3.8.1.1.b and the two (2) supporting ECW loops specified in Specification 3.7.4 are operable;

(b) The circuits required by Specification 3.8.1.1.a are operable;

(c) The equipment specified by Action 3.8.1.1.d is operable;

(d) The circuit between the 138 kV offsite transmission network, via the emergency transformer, and the onsite Class 1E Distribution System shall be functional and available;

(e) The technical support center diesel generator and the positive displacement pump are functional and available;

(f) Planned maintenance on the equipment specified in Action 3.8.1.1.d is suspended;

(g) Maintenance activities in the switchyard which could directly cause a loss of offsite power event will be prohibited unless required to ensure the continued reliability and availability of the offsite power sources.

In reflecting the additional element 2 constraints into the PSA, the top event for the positive displacement charging pump was modified by adding a new set of split fractions that apply only when the AOT is in effect. These split fractions reflect the element 2 restrictions that during the AOT, scheduled maintenance would not be performed on the positive displacement charging pump and the technical support center diesel generator, i.e., programmatic requirements will be in effect as a prerequisite to the AOT to prevent deliberate unavailability of this equipment during this period. (Note: there still remains a failure rate for failure to start on demand to include any standby failures that may occur during this period.)

<u>Element 3 - Other Configuration Management Provisions</u>

As required by the Maintenance Rule, the licensee will assess the overall impact on safety functions of performing maintenance activities, including the removal of any equipment from service. That is, prior to entry into the AOT, a PSA analysis of the "planned work configuration" will be performed, taking into account the actual configuration of associated systems and trains. Furthermore, the licensee (under element 2) has generated approximately 300 pre-calculated, configuration-specific conditional CDF (CCDF) estimates, which can be used to estimate a large number of configuration risks. These pre-calculated estimates will be augmented with additional configuration-specific CCDF estimates on an as-needed basis.

The licensee's proposed Bases for the proposed AOT relaxations states that their Configuration Risk Management Program (CRMP) evaluates the impact on plant risk of equipment out of service. The licensee's CRMP will specify the process for assessing and monitoring changes in the core damage probability, or large early release probability, while in certain planned and unplanned maintenance configurations. Procedures are in place to ensure that, immediately before and during entry into the subject Actions, the status of all associated systems and trains are reviewed for their impact on safety, taking into consideration the conditions expected as a result of modifying the AOTs.

As part of the three-element approach, the licensee will perform risk profile analyses in conjunction with its CRMP, to ensure adequacy of safety functions before performing maintenance activities including removal of any equipment from service. This is addressed in the proposed Administrative Controls TS 6.8.3.k. Administrative procedures require maintenance planners and schedular reviewers to meet at the beginning of each schedular week, to provide preliminary and adjusted interactive schedule inputs for risk profile generation prior to the initiation of planned maintenance activities. The procedure is to ensure minimal temporary CDF impact due to schedular planning.

The staff concludes that the actions taken by the licensee are appropriate for addressing the concerns that simultaneous equipment outage and other operational considerations during the AOT could potentially result in risk-significant configurations.

<u>Conclusions Regarding the Licensee's Probabilistic Safety Analysis Used To Support the Proposed Amendment</u>

The staff finds that the licensee has:

1. demonstrated that the calculated ICCDP and LERF are both relatively small, primarily because of substantial redundancy in system design, and robustness in containment severe accident mitigation capability,

2. implemented procedural restrictions that preclude entry into dominant risk-significant configurations during the extended AOT, and

3. provided the necessary assurances that appropriate assessments of the overall impacts on safety functions will be performed prior to any maintenance or other operational activities, including removal of equipment from service and documented these assurances in the TSs and Bases.

In addition to the above restrictions, the licensee will remain aware of any potential severe weather conditions which could result in an extended loss of offsite power. Because of the STP site exposure to severe weather (ESW5 category under SBO Rule categorization), the licensee will utilize plant-specific pre-hurricane shutdown requirements and procedures which meet the guidelines of Section 4.2.3 of NUMARC 87-00. These guidelines have been implemented according to the STP's Severe Weather Guidelines, 0POP04-ZO-0002, Revision 9.

The staff therefore, finds that the AOT of TS 3/4.8.1 (A.C. Sources) may be extended to 14 days subject to procedural requirements without a significant increase in risk. The staff also finds that the AOTs of TS 3/4.7.4 (ECW System) and of 3/4.7.14 (ECHW System) may be extended to 7 days without a significant increase in risk.

The staff's findings have taken into consideration the licensee's commitment to the above discussed compensatory measures, including maintenance of an up-to-date PSA model, and adequacy of relevant portions of the licensee's program to meet the requirements of the Maintenance Rule.

5.0 SUMMARY

The staff has evaluated the licensee's proposed changes for compliance with regulatory requirements as documented in this evaluation, and determined that they are acceptable. This determination had been based on:

(1) the need to maintain reliable safety systems;

(2) consideration of the number of redundant trains of onsite emergency AC power available at STP (i.e., STP's three train design where in almost all cases the safety systems powered from only one train of onsite emergency AC power are capable of mitigating the consequences of design-basis events, as compared to a typical two train design system);

(3) implementation of compensatory measures to offset any reduction in defense-in-depth.

In addition, PRA insights indicate that the risk associated with extending the SDG, ECW, and ECHW system allowed outage times is small, and offset by the licensee's CRMP. The CRMP evaluates the impact on plant risk of equipment out of service, and ensures the availability of safety functions before performing maintenance activities including removal of any equipment from service.

6.0 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.

7.0 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 (61 FR 40019). 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.

8.0 CONCLUSION

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

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