May 24, 1988

Docket No. 50-498

Mr. J. H. Goldberg
Group Vice-President, Nuclear
Houston Lighting & Power Company
P. O. Box 1700
Houston, Texas 77001

Dear Mr. Goldberg:

SUBJECT: ISSUANCE OF AMENDMENT NO. 1 TO FACILITY OPERATING LICENSE NPF-76 - SOUTH TEXAS PROJECT, UNIT 1 (TAC NO. 67930)

The Commission has issued the enclosed Amendment No. 1 to Facility Operating License No. NPF-76 for the South Texas Project, Unit 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated May 23, 1988 as supplemented May 23, 1988.

The amendment changes the Technical Specifications to delete all references to the excessive cooldown protection and associated items.

A copy of the Safety Evaluation supporting the amendment is also enclosed. Notice of Issuance will be included in the Commission's next Bi-weekly Federal Register notice.

Sincerely,
/s/
George F. Dick, Jr., Project Manager
Project Directorate - IV
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 1 to NPF-76
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION:

Docket File	JPartlow	NRC PDR	TBarnhart (4)
Local PDR	Wanda Jones	PD4 Reading	EButcher
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JCalvo	ARM/LFMB	OGC-Rockville	DHagan
M. Hodges	S. Newberry	EJordan	Plant File

DOCUMENT NAME: STP TAC NO. 67930

PD4/LA	PD4/PM	SRXB	SICB	OGC-Rockville	PD4/D
PNoonan	GDick:sr	MHodges	SNewberry		JCalvo
05/23/88	05/23/88	05/23/88	05/23/88	05/24/88	05/24/88

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P PDR

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DR4A
LRubenstein
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
May 24, 1988

Docket No. 50-498

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Houston Lighting & Power Company
P. O. Box 1700
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Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

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See next page

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

HOUSTON LIGHTING & POWER COMPANY

DOCKET NO. 50-498

SOUTH TEXAS PROJECT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 1
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) dated May 23, 1988 as supplemented May 23, 1988, 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.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. NPF-76 is hereby amended to read as follows:

   2. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 1 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

L S Rubenstein

Lester S. Rubenstein, Assistant Director
for Region IV and Special Projects
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 24, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 1

FACILITY OPERATING LICENSE NO. NPF-76

DOCKET NO. 50-498

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

Remove	Insert
3/4 3-18	3/4 3-18
3/4 3-21	3/4 3-21
3/4 3-22	3/4 3-22
3/4 3-24	3/4 3-24
3/4 3-26	3/4 3-26
3/4 3-29	3/4 3-29
3/4 3-31	3/4 3-21
3/4 3-32	3/4 3-32
3/4 3-34	3/4 3-34
3/4 3-36	3/4 3-36
3/4 3-38	3/4 3-38
3/4 3-40	3/4 3-40
3/4 3-42	3/4 3-42
3/4 3-45	3/4 3-45
3/4 3-47	3/4 4-47
3/4 3-49	3/4 3-49
B 3/4 3-3	B 3/4 3-3

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one train so that:

a. Each logic train is tested at least once per 36 months,

b. Each actuation train is tested at least once per 54 months*, and

c. One channel per function so that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" column of Table 3.3-3.

*If an ESFAS instrumentation channel is inoperable due to response times exceeding the limits of Table 3.3-5, perform an engineering evaluation to determine if the test failure is a result of degradation of the actuation relays. If degradation of the actuation relays is determined to be the cause, increase the ENGINEERED SAFETY FEATURES RESPONSE TIME surveillance frequency such that all trains are tested at least once per 36 months.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Emergency Ventilation, Start Standby Diesel Generators, Reactor Containment Fan Coolers, and Essential Cooling Water).					
a. Manual Initiation	2	1	2	1, 2, 3, 4	19
b. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	14
c. Actuation Relays	3	2	3	1, 2, 3, 4	14
d. Containment Pressure--High-1	3	2	2	1, 2, 3, 4	15
e. Pressurizer Pressure--Low	4	2	3	1, 2, 3#	20
f. Compensated Steam Line Pressure-Low	3/steam line	2/steam line any steam line	2/steam line in each steam line	1, 2, 3#	15

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation					
a. Manual Initiation					
1) Individual	2/steam line	1/steam line	2/operating steam line	1, 2, 3	24
2) System	2	1	2	1, 2, 3	23
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	22
c. Steam Line Pressure - Negative Rate--High	3/steam line	2/steam line any steam line	2/steam line in each steam line	3###	15
d. Containment Pressure - High-2	3	2	2	1, 2, 3	15
e. Compensated Steam Line Pressure - Low	3/steam line	2/steam line any steam line	2/steam line in each steam line	1, 2, 3#	15

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	25
b. Steam Generator Water Level--High-High (P-14)	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2, 3	20
c. Deleted					
d. Deleted					
e. Safety Injection	See Item 1. for all Safety Injection initiating functions and requirements.				
f. T_{avg}-Low coincident with Reactor Trip (P-4)**	4 (1/loop)	2	3	1, 2, 3	20

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater					
a. Manual Initiation	1/pump	1/pump	1/pump	1, 2, 3	26
b. Automatic Actuation Logic	2	1	2	1, 2, 3	22
c. Actuation Relays	3	2	3	1, 2, 3	22
d. Stm. Gen. Water Level--Low-Low Start Motor-Driven Pumps and Turbine-Driven Pump	4/stm. gen.	2/stm. gen. in any stm. gen.	3/stm. gen. in each stm. gen.	1, 2, 3	20
e. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
f. Loss of Power (Motor Driven Pumps Only)	See Item 8. below for all Loss of Power initiating functions and requirements.				
7. Automatic Switchover to Containment Sump****					
a. Automatic Actuation Logic and Actuation Relays	3-1/train	1/train	1/train	1, 2, 3, 4	19
b. RWST Level--Low-Low	3-1/train	1/train	1/train	1, 2, 3, 4	19
Coincident With: Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8. Loss of Power					
a. 4.16 kV ESF Bus Undervoltage-Loss of Voltage	4/bus	2/bus	3/bus	1, 2, 3, 4	20
b. 4.16 kV ESF Bus Undervoltage-Tolerable Degraded Voltage Coincident with SI	4/bus	2/bus	3/bus	1, 2, 3, 4	20
c. 4.16 kV ESF Bus Undervoltage - Sustained Degraded Voltage	4/bus	2/bus	3/bus	1, 2, 3, 4	20
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	21
b. Low-Low T_{avg}, P-12	4	2	3	1, 2, 3	21
c. Reactor Trip, P-4	2	1	2	1, 2, 3	23

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
10. Control Room Ventilation					
a. Manual Initiation	3(1/train)	2(1/train)	3(1/train)	All	27
b. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
c. Automatic Actuation Logic and Actuation Relays	3	2	3	All	27
d. Control Room Intake Air Radioactivity - High	2	1	2	All	28
e. Loss of Power	See Item 8. above for all Loss of Power initiating functions and requirements.				
11. FHB HVAC					
a. Manual Initiation	3(1/train)	2(1/train)	3(1/train)	1, 2, 3, 4 or with irradiated fuel in spent fuel pool	29, 30
b. Automatic Actuation Logic and Actuation Relays	3	2	3	1, 2, 3, 4 or with irradiated fuel in spent fuel pool	29, 30
c. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
d. Spent Fuel Pool Exhaust Radioactivity - High	2	1	2	With irradiated fuel in spent fuel pool	30

TABLE 3.3-3 (Continued)

TABLE NOTATIONS

**Feedwater Isolation only.

***Function is actuated by either actuation train A or actuation train B. Actuation train C is not used for this function.

****Automatic switchover to containment sump is accomplished for each train using the corresponding RWST level transmitter.

#Trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

##During CORE ALTERATIONS or movement of irradiated fuel within containment.

###Trip function automatically blocked above P-11 and may be blocked below P-11 when Low Compensated Steamline Pressure Protection is not blocked.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 16 - (Not Used)

ACTION 17 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 18 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge supply and exhaust valves are maintained closed.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1.	Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Emergency Ventilation, Start Standby Diesel Generators, Reactor Containment Fan Coolers, and Essential Cooling Water)					
a.	Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b.	Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
c.	Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
d.	Containment Pressure--High 1	3.6	0.71	2.0	≤ 3.0 psig	≤ 4.0 psig
e.	Pressurizer Pressure--Low	13.1	10.71	2.0	≥ 1850 psig##	≥ 1842 psig##
f.	Compensated Steam Line Pressure-Low	13.6	10.71	2.0	≥ 735 psig	≥ 714.7 psig*
2.	Containment Spray					
a.	Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b.	Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
c.	Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
d.	Containment Pressure--High-3	3.6	0.71	2.0	≤ 9.5 psig	≤ 10.5 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
3) Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
4) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
b. Containment Ventilation Isolation					
1) Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
2) Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
4) RCB Purge Radioactivity-High	3.1×10^{-4} µCi/cc	1.8×10^{-4} µCi/cc	1.3×10^{-4} µCi/cc	$\leq 5 \times 10^{-4}$ µCi/cc ###	$\leq 6.4 \times 10^{-4}$ µCi/cc
5) Containment Spray - Manual Initiation	See Item 2. above for Containment Spray manual initiation Trip Setpoints and Allowable Values.				
6) Phase "A" Isolation - Manual Initiation	See Item 3.a. above for Phase "A" Isolation manual initiation Trip Setpoints and Allowable Values.				
c. Phase "B" Isolation					
1) Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
2) Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Containment Pressure--High-3	3.6	0.71	2.0	$\leq$ 9.5 psig	$\leq$ 10.5 psig
4) Containment Spray-Manual Initiation	See Item 2. above for Containment Spray manual initiation Trip Setpoints and Allowable Values.				

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
4. Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Line Pressure - Negative Rate--High	2.6	0.5	0	≤ 100 psi	≤ 126.3 psi**
d. Containment Pressure - High-2	3.6	0.71	2.0	≤ 3.0 psig	≤ 4.0 psig
e. Compensated Steam Line Pressure - Low	13.6	10.71	2.0	≥ 735 psig	≥ 714.7 psig*
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Steam Generator Water Level--High-High (P-14)	4.5	2.35	2.0+0.2#	≤ 87.5% of narrow range instrument span.	≤ 88.9% of narrow range instrument span.
c. Deleted					

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
5. Turbine Trip and Feedwater Isolation (Continued)					
d. Deleted					
e. Safety Injection	See Item 1 above for all Safety Injection Trip Setpoints and Allowable Values.				
f. T_{avg}-Low Coincident with Reactor Trip (P-4) (Feedwater Isolation Only)	4.5	1.36	0.8	$\geq$ 574°F	$\geq$ 571.1°F
6. Auxiliary Feedwater					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
c. Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
d. Steam Generator Water Level--Low-Low	15.0	12.75	2.0+0.2#	$\geq$ 33.0% of narrow range instrument span.	$\geq$ 31.5% of narrow range instrument span.
e. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
6. Auxiliary Feedwater (Continued)					
f. Loss of Power (Motor Driven Pumps Only)	See Item 8. below for all Loss of Power Trip Setpoints and Allowable Values.				
7. Automatic Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. RWST Level--Low-Low	5.0	1.21	2.0	≥ 11%	≥ 9.1%
Coincident With: Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
8. Loss of Power					
a. 4.16 kV ESF Bus Undervoltage (Loss of Voltage)	N.A.	N.A.	N.A.	≥ 3107 volts with a ≤ 1.75 second time delay.	≥ 2979 volts with a ≤ 1.93 second time delay.
b. 4.16 kV ESF Bus Undervoltage (Tolerable Degraded Voltage Coincident with SI)	N.A.	N.A.	N.A.	≥ 3835 volts with a ≤ 35 second time delay.	≥ 3786 volts with a ≤ 39 second time delay.
c. 4.16 kV ESF Bus Undervoltage (Sustained Degraded Voltage)	N.A.	N.A.	N.A.	≥ 3835 volts with a ≤ 50 second time delay.	≥ 3786 volts with a ≤ 55 second time delay.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	$\leq$ 1985 psig	$\leq$ 1993 psig
b. Low-Low T_{avg}, P-12	N.A.	N.A.	N.A.	$\geq$ 563°F	$\geq$ 560.1°F
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.
10. Control Room Ventilation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
d. Control Room Intake Air Radioactivity - High	3.7×10^{-5} µCi/cc	2.2×10^{-5} µCi/cc	1.6×10^{-5} µCi/cc	$\leq 6.1\times10^{-5}$ µCi/cc	$\leq 7.8\times10^{-5}$ µCi/cc
e. Loss of Power	See Item 8. above for all Loss of Power Trip Setpoints and Allowable Values.				
11. FHB HVAC					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
11. FHB HVAC (Continued)					
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
d. Spent Fuel Pool Exhuast Radioactivity - High	3.1×10^{-4} µCi/cc	1.8×10^{-4} µCi/cc	1.3×10^{-4} µCi/cc	$\leq 5.0 \times 10^{-4}$ µCi/cc	$\leq 6.4 \times 10^{-4}$ µCi/cc

TABLE 3.3-4 (Continued)

TABLE NOTATIONS

*Time constants utilized in the lead-lag controller for Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

**The time constant utilized in the rate-lag controller for Steam Line Pressure-Negative Rate-High is greater than or equal to 50 seconds. CHANNEL CALIBRATION shall ensure that this time constant is adjusted to this value.

****Loop design flow = 95,400 gpm

#2.0% span for Steam Generator Level; 0.2% span for Reference Leg RTDs

##Until resolution of the Veritrak transmitter uncertainty issue, the trip setpoint will be set at ≥ 1869 psig, with the allowable value at ≥ 1861 psig.

###This setpoint value may be increased up to the equivalent limits of Specification 3.11.2.1 in accordance with the methodology and parameters of the ODCM during containment purge or vent for pressure control, ALARA and respirable air quality considerations for personnel entry.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATION SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Phase "A" Isolation	N.A.
d. Phase "B" Isolation	N.A.
e. Containment Ventilation Isolation	N.A.
f. Steam Line Isolation	N.A.
g. Feedwater Isolation	N.A.
h. Auxiliary Feedwater	N.A.
i. Essential Cooling Water	N.A.
j. Reactor Containment Fan Coolers	N.A.
k. Control Room Ventilation	N.A.
l. Reactor Trip	N.A.
m. Start Diesel Generator	N.A.
2. Containment Pressure--High-1	
a. Safety Injection (ECCS)	$\leq 27^{(1)}/12^{(5)}$
1) Reactor Trip	$\leq 2^{(3)}$
2) Feedwater Isolation	$\leq 12^{(3)}$
3) Phase "A" Isolation	$\leq 33^{(1)}/23^{(2)}$
4) Containment Ventilation Isolation (18-inch lines)	$\leq 23^{(1)}/13^{(2)}$
5) Auxiliary Feedwater	≤ 60
6) Essential Cooling Water	$\leq 62^{(1)}/52^{(2)}$
7) Reactor Containment Fan Coolers	$\leq 38^{(1)}/28^{(2)}$
8) Control Room Ventilation	$\leq 72^{(1)}/62^{(2)}$
9) Start Standby Diesel Generators	≤ 12

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure--Low	
a. Safety Injection (ECCS)	$\leq 27^{(1)}/12^{(5)}$
1) Reactor Trip	$\leq 2^{(3)}$
2) Feedwater Isolation	$\leq 12^{(3)}$
3) Phase "A" Isolation	$\leq 33^{(1)}/23^{(2)}$
4) Containment Ventilation Isolation	N.A.
5) Auxiliary Feedwater	≤ 60
6) Essential Cooling Water	$\leq 62^{(1)}/52^{(2)}$
7) Reactor Containment Fan Coolers	$\leq 38^{(1)}/28^{(2)}$
8) Control Room Ventilation	$\leq 72^{(1)}/62^{(2)}$
9) Start Standby Diesel Generators	≤ 12
4. Deleted	
5. Compensated Steam Line Pressure--Low	
a. Safety Injection (ECCS)	$\leq 22^{(4)}/12^{(5)}$
1) Reactor Trip	$\leq 2^{(3)}$
2) Feedwater Isolation	$\leq 12^{(3)}$
3) Phase "A" Isolation	$\leq 33^{(1)}/23^{(2)}$
4) Containment Ventilation Isolation	N.A.
5) Auxiliary Feedwater	≤ 60
6) Essential Cooling Water	$\leq 62^{(1)}/52^{(2)}$
7) Reactor Containment Fan Coolers	$\leq 38^{(1)}/28^{(2)}$

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
5. Compensated Steam Line Pressure--Low (Continued)	
8) Control Room Ventilation	$\leq 72^{(1)}/62^{(2)}$
9) Start Diesel Generators	$\leq$ 12
b. Steam Line Isolation	$\leq 8^{(3)}$
6. Containment Pressure--High-3	
a. Containment Spray	$\leq 30^{(1)}/20^{(2)}$
b. Phase "B" Isolation	$\leq 28^{(1)}/18^{(2)}$
7. Containment Pressure--High-2	
Steam Line Isolation	$\leq 7^{(3)}$
8. Steam Line Pressure - Negative Rate--High	
Steam Line Isolation	N.A.
9. Steam Generator Water Level--High-High	
a. Turbine Trip	$\leq 3^{(3)}$
b. Feedwater Isolation	$\leq 12^{(3)}$
10. Steam Generator Water Level--Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps	$\leq$ 60
b. Turbine-Driven Auxiliary Feedwater Pump	$\leq$ 60
11. RWST Level--Low-Low Coincident with Safety Injection	
Automatic Switchover to Containment Sump	$\leq 32^{(2)}$
12. Loss of Power	
a. 4.16 kV ESF Bus Undervoltage (Loss of Voltage)	$\leq$ 12
b. 4.16 kV ESF Bus Undervoltage (Tolerable Degraded Voltage Coincident with Safety Injection)	$\leq$ 49

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
12. Loss of Power (Continued)	
c. 4.16 kV ESF Bus Undervoltage (Sustained Degraded Voltage)	$\leq$ 65
13. RCB Purge Radioactivity-High	
a. Containment Ventilation Isolation (48-inch lines)	$\leq 73^{(2)}$
b. Containment Ventilation Isolation (18-inch lines)	$\leq 23^{(2)}$
14. Deleted	
15. Deleted	
16. T_{avg} - Low Coincident with Reactor Trip Feedwater Isolation	N.A.
17. Control Room Intake Air Radioactivity - High Control Room Ventilation	$\leq 78^{(2)}$
18. Spent Fuel Pool Exhaust Radioactivity - High FHB HVAC Emergency Startup	$\leq 42^{(2)}$

TABLE 3.3-5 (Continued)

TABLE NOTATIONS

(1) Diesel generator starting and sequence loading delays included.

(2) Diesel generator starting delay not included, sequence loading delay is included. Offsite power available.

(3) Not dependent upon diesel generator starting or sequence loading delays.

(4) Diesel generator starting and sequence loading delay included. Low Head Safety Injection pumps not included.

(5) Diesel generator starting delays not included, sequence loading delay is included. Low Head Safety Injection pumps not included.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Emergency Ventilation, Start Standby Diesel Generators, Reactor Containment Fan Coolers, and Essential Cooling Water)								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	M(1)	N.A.	N.A.	1, 2, 3, 4
c. Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	M(6)	Q(4,5)	1, 2, 3, 4
d. Containment Pressure-High-1	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
e. Pressurizer Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
f. Compensated Steam Line Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steam Line Isolation								
e. Compensated Steam Line Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(6)	Q(4)	1, 2, 3
b. Steam Generator Water Level-High-High (P-14)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Deleted								
d. Deleted								
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
f. T_{avg}-Low Coincident with Reactor Trip (P-4) (Feedwater Isolation Only)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
6. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	M(1)	N.A.	N.A.	1, 2, 3
c. Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	M(6)	Q	1, 2, 3
d. Steam Generator Water Level--Low-Low	S	R	M	N.A.	N.A.	N.A	N.A	1, 2, 3
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
f. Loss of Power	See Item 8. below for all Loss of Power Surveillance Requirements.							
7. Automatic Switchover to Containment Sump								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(6)	M(6)	Q	1, 2, 3, 4
b. RWST Level--Low-Low	S	R	M	N.A.	N.A.	N.A.	N.A	1, 2, 3, 4
Coincident With: Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
8. Loss of Power								
a. 4.16 kV ESF Bus Undervoltage (Loss of Voltage)	N.A.	R	N.A	M	N.A.	N.A.	N.A.	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
8. Loss of Power (Continued)								
b. 4.16 kV ESF Bus Undervoltage (Tolerable Degraded Voltage Coincident with SI)	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4
c. 4.16 kV ESF Bus Undervoltage (Sustained Degraded Voltage)	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4
9. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Low-Low T_{avg}, P-12	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip, P-4	N.A.	N.A	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
10. Control Room Ventilation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	All

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
10. Control Room Ventilation (Continued)								
b. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(6)	N.A.	N.A.	All
d. Control Room Intake Air Radioactivity-High	S	R	M	N.A.	N.A.	N.A.	N.A.	All
e. Loss of Power	See Items 8. above for all Loss of Power Surveillance Requirements.							
11. FHB HVAC								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4, or with irradiated fuel in the spent fuel pool
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(6)	N.A.	N.A.	1, 2, 3, 4, or with irradiated fuel in the spent fuel pool

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
11. FHB HVAC (Continued)								
c. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
d. Spent Fuel Pool Exhaust Radio-activity-High	S	R	M	N.A.	N.A.	N.A.	N.A.	With irradiated fuel in spent fuel pool.

TABLE NOTATION

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(2) Deleted

(3) Deleted

(4) Except relays K807, K814, K829 (Train B only), K831, K845, K852 and K854 (Trains B and C only) which shall be tested at least once per 18 months during refueling and during each COLD SHUTDOWN exceeding 24 hours unless they have been tested within the previous 92 days.

(5) Except relay K815 which shall be tested at indicated interval only when reactor coolant pressure is above 700 psig.

(6) Each actuation train shall be tested at least every 92 days on a STAGGERED TEST BASIS. Testing of each actuation train shall include master relay testing of both logic trains. If an ESFAS instrumentation channel is inoperable due to failure of the Actuation Logic Test and/or Master Relay Test, increase the surveillance frequency such that each train is tested at least every 62 days on a STAGGERED TEST BASIS unless the failure can be determined by performance of an engineering evaluation to be a single random failure.

*During CORE ALTERATIONS or movement of irradiated fuel within containment.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING FOR PLANT OPERATIONS

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels for plant operations shown in Table 3.3-6 shall be OPERABLE with their Alarm/Trip Setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel Alarm/Trip Setpoint for plant operations exceeding the value shown in Table 3.3-6, adjust the Setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels for plant operations inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel for plant operations shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and DIGITAL CHANNEL OPERATIONAL TEST for the MODES and at the frequencies shown in Table 4.3-3.

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip via P-16, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level, allows Safety Injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 On increasing pressurizer pressure, P-11 automatically reinstates Safety Injection actuation on low pressurizer pressure or Low Compensated Steamline Pressure signals, reinstates steam line isolation on Low Compensated Steamline Pressure signals, and opens the accumulator discharge isolation valves. On decreasing pressure, P-11 allows the manual block of Safety Injection actuation on low pressurizer pressure or Low Compensated Steamline Pressure signals, allows the manual block of steamline isolation on Low Compensated Steamline Pressure signals, and enables steam line isolation on high negative steam line pressure rate.

P-12 On increasing reactor coolant loop temperature, P-12 automatically provides an arming signal to the Steam Dump System. On decreasing reactor coolant loop temperature, P-12 automatically removes the arming signal from the Steam Dump System.

P-14 On increasing steam generator water level, P-14 automatically trips the turbine and the main feedwater pumps, and closes all feedwater isolation valves and feedwater control valves.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance. The radiation monitors for plant operations sense radiation levels in selected plant systems and locations and determine whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Ventilation Systems.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the Excore Neutron Flux Detection System, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range channel is inoperable.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM

The OPERABILITY of the Remote Shutdown System ensures that sufficient capability is available to permit safe shutdown of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 1 TO

FACILITY OPERATING LICENSE NO. NPF-76

HOUSTON LIGHTING & POWER COMPANY

SOUTH TEXAS PROJECT, UNIT 1

DOCKET NO. 50-498

1.0 INTRODUCTION

By letters dated May 23, 1988, Houston Lighting & Power Company, (HL&P, the licensee) requested an amendment to the Technical Specifications (TSs) appended to Facility Operating License No. NPF-76 for South Texas Project, Unit 1 (STP-1). The proposed amendment would delete all references to Excessive Cooldown Protection and associated items. Background information was contained in HL&P letter dated April 18, 1988. Additional information was provided in HL&P letter of May 18, 1988.

2.0 DISCUSSION

Excessive cooldown protection, as presently installed on South Texas Project, consists of Safety Injection actuation and steamline isolation from two out of three low-low compensated T-cold signals from any loop with the reactor tripped or below 10% power, feedwater isolation and turbine trip from two out of three low compensated T-cold signals in any loop with reactor tripped or below 10% power or from two out of three high feedwater flow signals in any loop with the reactor tripped or below 10% power, interlocked with two out of four RCS low flow signals or two out of four low T-avg signals.

Excessive cooldown protection was in the original design of South Texas Project to prevent the Reactor from returning critical subsequent to a steam system piping failure or inadvertent opening of steam generator relief or safety valve, or excessive main feedwater addition. South Texas Project has subsequently adopted NRC approved licensing criterion which permits return to criticality following the above mentioned events. The analyses for these events as described in Chapter 15 of the FSAR shows the possibility of return to criticality following these events. Two portions of the original excessive cooldown protection, emergency boration system and main steam isolation on any safety injection, were deleted prior to issuance of the operating license for South Texas Project, Unit 1.

On March 30, 1988, STP-1 experienced a loss of offsite power, a reactor trip, and safety injection event. In reviewing the event, the licensee determined that the Low-Low Compensated T-Cold Excessive Cooldown Protection circuitry will initiate a safety injection actuation if charging flow is maintained after the Reactor Coolant Pumps stop or trip. This condition is unique to the STP design as a result of the inclusion of excessive cooldown protection circuitry. This condition is considered to be undesirable since it results in unwarranted cycling of safeguards equipment and complicates the response to less significant events. The licensee concludes that anytime the Reactor Coolant Pumps are stopped while charging flow is maintained, a safety injection actuation will occur due to excessive cooldown protection.

Not only is this an undesirable situation during normal operation but, the condition creates a special problem for conducting two required tests; the shutdown from outside the Control Room test, and the loss of offsite power (LOOP) test. During both of these tests, the conditions will be present in which the excessive cooldown protection can be expected to cause a safety injection (SI) actuation. Conducting the tests with the excessive cooldown protection in place will cause the operators to mitigate a safety injection as part of the tests. This is beyond the scope of the tests and significantly complicates plant response.

3.0 EVALUATION

The staff has reviewed the licensee's evaluation of removal of the excessive cooldown protection on the appropriate accident analyses.

3.1 Inadvertent Opening of a Steam Generator Relief or Safety Valve causing a Depressurization of the Main Steam System (FSAR Chapter 15.1.4)

Although safety injection will no longer actuate from two out of three low-low compensated T-cold in any loop, it will actuate from two out of three low compensated steamline pressure signals from any loop or from two out of four low pressurizer pressure signals. In addition, redundant action will close the main feedwater valves following a reactor trip and a Safety Injection signal will rapidly close all feedwater control valves and feedwater isolation valves and trip the main feedwater pumps. Closure of the fast-acting main steam isolation valves (MSIVs) will be accomplished from either low compensated steamline pressure above the P-11 setpoint, or from high negative steamline pressure rate signal below the P-11 setpoint. The original analyses for these events show that safety injection is initiated by low pressurizer pressure. No credit is taken in the original analysis for mitigation from the excessive cooldown protection.

3.2 Steam System Piping Failures Inside Or Outside Containment (FSAR Chapter 15.1.5)

Although Safety Injection will no longer actuate from two out of three low-low compensated T-cold in any loop, it will actuate from 2 out of 3 low compensated steamline pressure signals from any loop, from two out of four low pressurizer pressure signals, or from two out of three high containment pressure signals. In addition, redundant isolation of the main feedwater flow is provided, in that normal control action will close the main feedwater valves following a reactor trip and a Safety Injection signal will rapidly close all feedwater control valves and feedwater isolation valves and trip the main feedwater pumps. Closure of the fast-acting main steam isolation valves (MSIVs) will be accomplished from either low compensated steamline pressure above the P-11 setpoint, from high negative steamline pressure rate signal below the P-11 setpoint, or from two out of three High-2 containment pressure signals. The original analyses for these events show that safety injection is initiated by low steam line pressure. No credit is taken in the original analysis for mitigation from the excessive cooldown protection.

3.3 Mass and Energy Release for Postulated Secondary System Pipe Ruptures Inside the Containment

No credit was taken in the original analysis for mitigation of the consequences from actuation of excessive cooldown protection.

The deletion of excessive cooldown protection (which results in a protection system functionally equivalent to RESAR 3S Protection Systems) does not have any effect upon the probability of occurrence of a malfunction of equipment important to safety in that the only physical changes on equipment important to safety is the deletion of the actuation signals from the protection system. The reduction in unnecessary cycling of Engineered Safeguards Equipment will have a positive effect upon reducing the potential of malfunction of equipment important to safety.

3.4 Implementation of Circuitry Changes

During a meeting on May 6, 1988, the licensee proposed that the simplest method to delete the Excessive Cooldown Protection is by cutting the signal wires from the Process Instrument Cabinet to the ESFAS Cabinet. All of the logic circuit boards within the ESFAS cabinet will not be replaced until the first refueling. All the surveillance test provision will not be changed except the monthly analog Channel functional test procedure will be modified to indicate the disconnection between the process instrument cabinet and the ESFAS Cabinet. The T-cold analog signal which provides monitoring function will be maintained. The intertie between the process instrument cabinet and the ESFAS Cabinet is the relay to contact connection. Cutting signal wires will not affect the logic circuit operation inside the ESFAS Cabinet. Any malfunction within the ESFAS Cabinet still can be detected by the surveillance test provision. No jumpers or lifting leads are required to accomplish this modification. Therefore, the staff finds that the proposed circuitry changes are acceptable.

4.0 EMERGENCY CIRCUMSTANCES

After the March 30, 1988 event, an analysis determined the root cause. The licensee then directed the vendor, Westinghouse, to consider the options and propose a solution. This required a review of the original design basis for the excessive cooldown actuation circuitry and the impact of its removal on the FSAR analyses. Westinghouse completed its review and made a recommendation on May 14, 1988. The licensee expedited the TS change request review through both the Plant Operations Review Committee and the Nuclear Safety Review Board. Approval of the TS change is needed in order to avoid a delay in the plant testing and startup. The affected power ascension tests, LOOP and shutdown outside the control room are scheduled to begin by midnight, May 24, 1988 with the reactor at 30% power. Attempting to conduct the tests prior to the removal of the excessive cooldown protection is expected to result in SI actuation which will complicate the conductance of the tests, may obscure some of the results. The SI actuation will cause an additional challenge to the system and an additional transient on the plant. Using the normal procedures for processing the TS change will result in a delay in the startup schedule.

5.0 NO SIGNIFICANT HAZARDS CONSIDERATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if the operation of the facility in accordance with the amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or
(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or
(3) Involve a significant reduction in a margin of safety.

The evaluation in Section 3.0 shows that deletion of the excessive cooldown protection will have no effect on the probability and no significant effect on the consequences of any of the accidents previously evaluated. The proposed change does not create a possibility of a new or different accident, and does not affect any margins of safety.

Based on the above evaluation, the staff concludes that operation of the facility in the proposed manner would not involve a significant increase in the probability or consequences of an accident previously evaluated, would not create the possibility of a new or different kind of accident from any accident previously evaluated, and would not involve a significant reduction in a margin of safety.

Accordingly, we conclude the amendment involves no significant hazards consideration.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, consultation was held with the State of Texas by telephone. The State expressed no concern from both the standpoint of safety and the standpoint of the no significant hazards consideration determination.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment involves a deletion of the excessive cooldown protection circuitry. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposures. The Commission has made a final no significant hazards consideration finding with respect to this amendment. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

8.0 CONCLUSION

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: May 24, 1988

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