October 16, 1986

Docket No.: 50-382

Mr. J. G. Dewease
Senior Vice President - Nuclear Operations
Louisiana Power and Light Company
317 Baronne Street, Mail Unit 17
New Orleans, Louisiana 70160

Dear Mr. Dewease:

Subject: Issuance of Amendment No. 6 to Facility Operating License No. NPF-38 for Waterford 3

The Commission has issued the enclosed Amendment No. 6 to Facility Operating License No. NPF-38 for the Waterford Steam Electric Station, Unit 3. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated August 1, 1985, and supplemented by letter dated October 8, 1986.

The amendment revises the Appendix A Technical Specifications by correcting three typographical errors in Table 3.8-1, changing Technical Specification 3/4.9.7 so that use of the spent fuel handling machine is not required for movement of new fuel outside the spent fuel pool, and revising Technical Specification 3/4.7.2, "Steam Generator Pressure/Temperature Limits".

A copy of the Safety Evaluation supporting the amendment is also enclosed.

Sincerely,

/S/

James H. Wilson, Project Manager
PWR Project Directorate No. 7
Division of PWR Licensing-B

Enclosures:
1. Amendment No. 6 to NPF-38
2. Safety Evaluation

cc: See next page

PBD7	PBD7	AD:PWR-B	OGC	DIR:PBD7
JWilson/yt	JLee	DCrutchfield		GWKnighton
10/8/86	10/8/86	10/8/86	10/9/86	10/16/86

Mr. Jerrold G. Dewease
Louisiana Power & Light Company

Waterford 3

cc:
W. Malcolm Stevenson, Esq.
Monroe & Leman
1432 Whitney Building
New Orleans, Louisiana 70103

Mr. E. Blake
Shaw, Pittman, Potts and Trowbridge
2300 N Street, NW
Washington, D.C. 20037

Mr. Gary L. Groesch
P. O. Box 791169
New Orleans, Louisiana 70179-1169

Mr. F. J. Drummond
Project Manager - Nuclear
Louisiana Power and Light Company
317 Baronne Street
New Orleans, Louisiana 70160

Mr. K. W. Cook
Nuclear Support and Licensing Manager
Louisiana Power and Light Company
317 Baronne Street
New Orleans, Louisiana 70160

Resident Inspector/Waterford NPS
P. O. Box 822
Killona, Louisiana 70066

Mr. Ralph T. Lally
Manager of Quality Assurance
Middle South Services, Inc.
P. O. Box 61000
New Orleans, Louisiana 70161

Chairman
Louisiana Public Service Commission
One American Place, Suite 1630
Baton Rouge, Louisiana 70804

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
Office of Executive Director
for Operations
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

Carole H. Burstein, Esq.
445 Walnut Street
New Orleans, Louisiana 70118

Mr. Charles B. Brinkman, Manager
Washington Nuclear Operations
Combustion Engineering, Inc.
7910 Woodmont Avenue, Suite 1310
Bethesda, Maryland 20814

Mr. William H. Spell, Administrator
Nuclear Energy Division
Office of Environmental Affairs
P. O. Box 14690
Baton Rouge, Louisiana 70898

President, Police Jury
St. Charles Parrish
Hahnville, Louisiana 70057

OCT 1 6 1986

ISSUANCE OF AMENDMENT NO. 6 TO FACILITY OPERATING
LICENSE NP. NPF-38 FOR WATERFORD 3

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JLee (5)
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LOUISIANA POWER AND LIGHT COMPANY

DOCKET NO. 50-382

WATERFORD STEAM ELECTRIC STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 6
License No. NPF-38

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment, dated August 1, 1985 by Louisiana Power and Light Company (licensee), complies with standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the 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 amendment will not be inimical to the common defense and security or to the health and safety of the public;

   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-38 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 6, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in this license. LP&L shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James H. Wilson

James H. Wilson, Project Manager
PWR Project Directorate No. 7
Division of PWR Licensing-B

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 16, 1986

October 16, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 6

TO FACILITY OPERATING LICENSE NO. NPF-38

DOCKET NO. 50-382

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Also to be replaced are the following overleaf pages to the amended pages.

Amendment Pages	Overleaf Pages
3/4 7-10	3/4 7-9
3/4 8-24	3/4 8-23
3/4 8-39	3/4 8-40
3/4 8-41	3/4 8-42
3/4 9-7	3/4 9-8
B 3/4 7-3	B 3/4 7-4

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

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

ACTION:

MODE 1

With one main steam line isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, be in at least HOT STANDBY within the next 6 hours.

MODES 2, 3, and 4

With one main steam line isolation valve inoperable, subsequent operation in MODE 2, 3, or 4 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 3.0 seconds when tested pursuant to Specification 4.0.5.

PLANT SYSTEMS

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperature of the secondary coolant in the steam generators shall be greater than 115°F when the pressure of the secondary coolant is greater than 210 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure to less than or equal to 210 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure of the steam generators shall be determined to be less than 210 psig at least once per hour when the temperature of the secondary coolant is less than 115°F.

TABLE 3.8-1 (Continued)

480 VOLTS POWER FROM MCCs

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES				REMARKS
			DEVICE	TYPE	TRIP SETPOINT (NOTE 1)	TIME-CURRENT CHARACTERISTIC	
1	Safety Inj. Tank 1A Iso. Val. 1SI-V1505 Tk 1A (SI-331A)	Primary	Breaker	EF	61	Notes 2, 3	
		Backup	Fuse	TRS	61	Note 4	
2	Safety Inj. Tank 2A Iso. Val. 1SI-V1507 Tk 2A (SI-332A)	Primary	Breaker	EF	61	Notes 2, 3	
		Backup	Fuse	TRS	61	Note 4	
3	LP-311	Primary	Breaker	EF	62	Notes 2, 3	
		Backup	Fuse	TRS	62	Note 4	
4	RCS Loop 2 SDC Iso. Val. 1SI-V1504A (SI-401A)	Primary	Breaker	EF	63	Notes 2, 3	
		Backup	Fuse	TRS	63	Note 4	
5	CARS Suction Val. 2HV-F253A (CARS-201A)	Primary	Breaker	EF	64	Notes 2, 3	
		Backup	Fuse	TRS	64	Note 4	
6	Hydraulic Pump For Val. 1SI-V1503A (SI-405A)	Primary	Breaker	EF	64	Notes 2, 3	
		Backup	Fuse	TRS	64	Note 4	

TABLE 3.8-1 (Continued)

480 VOLTS POWER FROM MCCs (Continued)

			OVERCURRENT PROTECTIVE DEVICES				
ITEM NO.	SYSTEM POWERED	PROTECTION	DEVICE	TYPE	TRIP SETPOINT (NOTE 1)	TIME-CURRENT CHARACTERISTIC	REMARKS
7	Safety Inj. Tank 1B Iso. Val. 1SI-V1506 Tk 1B (SI-331B)	Primary	Breaker	EF	65	Notes 2, 3	
		Backup	Fuse	TRS	65	Note 4	
8	Safety Inj. Tank 2B Iso. Val. 1SI-V1508 Tk 2B (SI-332B)	Primary	Breaker	EF	65	Notes 2, 3	
		Backup	Fuse	TRS	65	Note 4	
9	LP-310	Primary	Breaker	EF	66	Notes 2, 3	
		Backup	Fuse	TRS	66	Note 4	
10	RCS Loop 1 SDC Iso. Val. 1SI-V1502B (SI-401B)	Primary	Breaker	EF	67	Notes 2, 3	
		Backup	Fuse	TRS	67	Note 4	
11	CARS Suction Val. 2HV-F254B (CAR-201B)	Primary	Breaker	EF	68	Notes 2, 3	
		Backup	Fuse	TRS	68	Note 4	
12	Hydraulic Pump For Val. 1SI-V1501B (SI-405B)	Primary	Breaker	EF	68	Notes 2, 3	
		Backup	Fuse	TRS	68	Note 4	

TABLE 3.8-1 (Continued)

120 VOLTS CONTROL POWER FROM PDPs OR MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES TRIP SETPOINT (NOTE 1) SHEET NO.	CIRCUIT NO.	DEVICE	TYPE/TIME-CURRENT CHARACTERISTIC (NOTE 2)	REMARKS
52	Motor Htr. Leads	Primary	121	Ckt. 15	Breaker	EE	
	AH-1 (3D-SB)	Backup	121A	F4	Fuse	TRS	
53	Motor Htr. Leads	Primary	CWD 1141		Breaker	TED	120/208V SWGR heater bus, double breaker protection.
	E-16 (3B)	Backup	CWD 1141		Breaker	TED	
54	Motor Htr. Leads	Primary	CWD 1142		Breaker	TED	120/208V SWGR heater bus, double breaker protection.
	E-16 (3D)	Backup	CWD 1142		Breaker	TED	
55	Cont. Fan Coolers	Primary	121	Ckt. 17	Breaker	EE	
	Dampers	Backup	121A	F5	Fuse	TRS	
56	Samp. Sys. Sol. Valve	Primary	148A	Ckt. 49	Breaker	CD	
	2SL-F601 (PSL-404A)	Backup	148A	Ckt. 49	Fuse	FRN	
57	Samp. Sys. Sol. Valve	Primary	148A	Ckt. 45	Breaker	CD	
	2SL-F603 (PSL-404B)	Backup	148A	Ckt. 45	Fuse	FRN	
58	Samp. Sys. Recorder	Primary	133	Ckt. 35	Breaker	EE	
	Panel	Backup	133A	F12	Fuse	TRS	

TABLE 3.8-1 (Continued)

120 VOLTS CONTROL POWER FROM PDPs OR MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES TRIP SETPOINT (NOTE 1) SHEET NO.	CIRCUIT NO.	DEVICE	TYPE/TIME-CURRENT CHARACTERISTIC (NOTE 2)	REMARKS
59	Cont. Purge Exh. Damper SV-D22 (CAP-202) and SV-D23 (CAP-201)	Primary	133	Ckt. 1	Breaker	EE	
		Backup	133A	F5	Fuse	TRS	
		Primary	134	Ckt. 1	Breaker	EE	
		Backup	134A	F2	Fuse	ATM	
60	Sol. Valve 7RC-F604 (RC-323)	Primary	133	Ckt. 8	Breaker	EE	
		Backup	133A	F3	Fuse	TRS	
61	Sol. Valve 7RC-F605 (RC-325)	Primary	133	Ckt. 10	Breaker	EE	
		Backup	133A	F4	Fuse	TRS	
62	Sol. Valve 1CH-E2504B (CVC-218B)	Primary	148	Ckt. 29	Breaker	CD	
		Backup	148A	Ckt. 29	Fuse	FRN	
63	Sol. Valve 1CH-E2503A (CVC-218A)	Primary	147	Ckt. 27	Breaker	CD	
		Backup	147A	Ckt. 27	Fuse	FRN	
64	Sol. Valves 3CC-P1501A1 (CC-665A) & 3CC P1505A1 (CC-679A)	Primary	150	Ckt. 25	Breaker	TEB	
		Backup	CWD 280	F1	Fuse	ATM	

TABLE 3.8-1 (Continued)

120 VOLTS CONTROL POWER FROM PDPs OR MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES TRIP SETPOINT (NOTE 1) SHEET NO.	CIRCUIT NO.	DEVICE	TYPE/TIME-CURRENT CHARACTERISTIC (NOTE 2)	REMARKS
65	Sol. Valves 3CC-P1503A2 (CC-666A) & 3CC-P1507A2 (CC-680A)	Primary	150	Ckt. 27	Breaker	TEB	
		Backup	CWD 282	F1	Fuse	ATM	
66	RCP1A Instrumentation and Accessories	Primary	CWD 220		Fuse	OTS	Two fuses in series, one each, + and - poles.
		Backup	CWD 220		Fuse	OTS	
67	RCP2A Instrumentation and Accessories	Primary	CWD 240		Fuse	OTS	Two fuses in series, one each, + and - poles.
		Backup	CWD 240		Fuse	OTS	
68	CEDM Cool. Valves & Dampers	Primary	149	Ckt. 14	Breaker	TEB	
		Backup	CWD 1145	F2	Fuse	ATM	
69	CEDM Cool. Units Inlet Damper	Primary	150	Ckt. 20	Breaker	TEB	
		Backup	CWD 1145	F1	Fuse	ATM	
70	Sol. Valve 2CH-F1514AB (RC-602)	Primary	150	Ckt. 5	Breaker	TEB	
		Backup	CWD 326	F2	Fuse	ATM	
71	Sol. Valve 7BM-P237 (GWM-101)	Primary	135	Ckt. 11	Breaker	EE	
		Backup	CWD 401	F1	Fuse	ATM	

TABLE 3.8-1 (Continued)

120 VOLTS CONTROL POWER FROM PDPs OR MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES TRIP SETPOINT (NOTE 1) SHEET NO.	CIRCUIT NO.	DEVICE	TYPE/TIME-CURRENT CHARACTERISTIC (NOTE 2)	REMARKS
72	Sol. Valve 5SI-F1563 (SI-342)	Primary	150	Ckt. 1	Breaker	TEB	
		Backup	CWD 499	F3	Fuse	ATM	
73	Sol. Valves 3CC-P1502B1 (CC-665B) & 3CC-P1506B1 (CC-679B)	Primary	150	Ckt. 26	Breaker	TEB	
		Backup	CWD 281	F2	Fuse	ATM	
74	Sol. Valves 3CC-P1504B2 (CC-666B) & 3CC-P1508B2 (CC-680B)	Primary	150	Ckt. 28	Breaker	TEB	
		Backup	CWD 283	F2	Fuse	ATM	
75	RCP1B Instrumentation and Accessories	Primary	CWD 230		Fuse	OTS	Two fuses in series, one each, + and - poles.
		Backup	CWD 230		Fuse	OTS	
76	RCP2B Instrumentation and Accessories	Primary	CWD 250		Fuse	OTS	Two fuses in series, one each, + and - poles.
		Backup	CWD 250		Fuse	OTS	
77	Sol. Valve 2CA-E604B (ARM-109)	Primary	148	Ckt. 26	Breaker	CD	
		Backup	148A	Ckt. 26	Fuse	FRN	

3/4.9.7 CRANE TRAVEL - FUEL HANDLING BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Cranes in the fuel handling building shall be restricted as follows:

a. The spent fuel handling machine shall be used* for the movement of fuel assemblies (with or without CEAs) and shall be OPERABLE with:

1. A minimum hoist capacity of 1800 pounds, and

2. An overload cutoff limit of less than or equal to 1900 pounds, and,

b. Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the spent fuel pool.

APPLICABILITY: During movement of fuel assemblies in the fuel handling building, or with fuel assemblies in the spent fuel pool.

ACTION:

a. With the spent fuel handling machine inoperable, suspend the use of the spent fuel handling machine for movement of fuel assemblies and place the crane load in a safe position.

b. With loads in excess of 2000 pounds over fuel assemblies in the spent fuel pool, place the crane load in a safe position.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7.1 The spent fuel handling machine shall be demonstrated OPERABLE within 72 hours prior to the start of fuel assembly movement and at least once per 7 days thereafter by performing a load test of at least 1800 pounds and demonstrating the automatic load cutoff when the hoist load exceeds 1900 pounds.

4.9.7.2 The electrical interlock system which prevents crane main hook travel over fuel assemblies in the spent fuel pool shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

4.9.7.3 Administrative controls which prevent crane auxiliary hook travel with loads in excess of 2000 pounds over the fuel assemblies in the spent fuel pool shall be enforced during crane operations.

*Not required for movement of new fuel assemblies outside the spent fuel pool.

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one shutdown cooling train shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is greater than or equal to 23 feet.

ACTION:

With no shutdown cooling train OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling train to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one shutdown cooling train shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm** at least once per 12 hours.

*The shutdown cooling loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

**The minimum flow may be reduced to 3000 gpm after the reactor has been shut down for greater than or equal to 175 hours or by verifying at least once per hour that the RCS temperature is less then 135°F.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss-of-offsite electrical power. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVE

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analyses.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator secondary pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitation to 115°F and 210 psig is based on a steam generator RT_{NDT} of 40°F and is sufficient to prevent brittle fracture. Below this temperature of 115°F the system pressure must be limited to a maximum of 20% of the secondary hydrostatic test pressure of 1375 psia (corrected for instrument error). Should steam generator temperature drop below 115°F an engineering evaluation of the effects of the overpressurization is required. However, to reduce the potential for brittle failure the steam generator temperature may be increased to a limit of 200°F while performing the evaluation. The limitations on the primary side of the steam generator are bounded by the restrictions on the reactor coolant system in Specification 3.4.8.1.

3/4.7.3 COMPONENT COOLING WATER AND AUXILIARY COMPONENT COOLING WATER SYSTEMS

The OPERABILITY of the component cooling water system and its corresponding auxiliary 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 these systems, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.4 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level, temperature, and number of fans ensure that sufficient cooling capacity is available to either (1) provide normal cooldown of the facility, or (2) to 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 their design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974.

3/4.7.5 FLOOD PROTECTION

The limitation on flood protection ensures that facility protective actions will be taken in the event of flood conditions. The limit of elevation 27.0 ft Mean Sea Level is based on the maximum elevation at which the levee provides protection, the nuclear plant island structure provides protection to safety-related equipment up to elevation +30 ft Mean Sea Level.

3/4.7.6 CONTROL ROOM AIR CONDITIONING SYSTEM

The OPERABILITY of the control room air conditioning 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. 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 rem 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.

Operation of the system with the heaters on for at least 10 hours continuous over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. Obtaining and analyzing charcoal samples after 720 hours of adsorber operation (since the last sample and analysis) ensures that the adsorber maintains the efficiency assumed in the safety analysis and is consistent with Regulatory Guide 1.52.

3/4.7.7 CONTROLLED VENTILATION AREA SYSTEM

The OPERABILITY of the controlled ventilation area system ensures that radioactive materials leaking from the penetration area or the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 6 TO FACILITY OPERATING LICENSE NO. NPF-38

LOUISIANA POWER AND LIGHT COMPANY

WATERFORD STEAM ELECTRIC STATION, UNIT 3

DOCKET NO. 50-382

1.0 INTRODUCTION

By letter dated August 1, 1985, as supplemented by letter dated October 8, 1986, Louisiana Power and Light Company (the licensee), requested changes to the Technical Specifications (Appendix A to Facility Operating License No. NPF-38) for the Waterford Steam Electric Station, Unit 3. The proposed changes would: (1) correct three typographical errors in Table 3.8-1, "Containment Penetration Conductor Overcurrent Protective Devices"; (2) change Technical Specification 3/4.9.7, "Crane Travel-Fuel Handling Building" so that use of the spent fuel handling machine is not required for movement of new fuel outside the spent fuel pool; and (3) revise Technical Specification 3/4.7.2, "Steam Generator Pressure/Temperature Limits".

2.0 DISCUSSION

The changes to the technical specifications requested by the licensee are in three areas as described below.

Containment Penetration Conductor Overcurrent Protection Devices

The proposed change revises the Appendix A Technical Specifications by correcting three typographical errors in Table 3.8-1, "Containment Penetration Conductor Overcurrent Protective Devices" of Technical Specification 3/4.8.4, "Electrical Equipment Protective Devices."

Technical Specification 3/4.8.4 delineates the operability and surveillance requirements for the containment penetration conductor overcurrent protective devices listed in Table 3.8-1. The requirements of this Technical Specification ensure these devices will not prevent safety-related valves from performing their function. The proposed change to Table 3.8-1 consists of the following three parts:

a. Item 8, 480 Volts Power from MCCs, of Table 3.8-1 (page 3/4 8-24) currently lists the valve number as 1SI-V1508 TK 1B. The proposed change will correct the typographical error in the tank designation suffix to accurately list the valve number as 1SI-V1508 TK 2B.

b. Item 57, 120 Volts Control Power from PDPs or MCCs, of Table 3.8-1 (page 3/4 8-39) currently lists the Power Distribution and Motor Data (PDMD) sheet number for primary protection as 148. The proposed change will correct the typographical error to accurately list the PDMD sheet number as 148A.

c. Item 71, 120 Volts Control Power from PDPs or MCCs, of Table 3.8-1 (page 3/4 8-41) currently lists the valve number as 2BM-P237. The proposed change will correct the typographical error in the code class prefix to accurately list the valve number as 7BM-P237.

Crane Travel - Fuel Handling Building

The proposed change revises the Appendix A Technical Specifications by changing Technical Specification 3/4.9.7, "Crane Travel-Fuel Handling Building," so that use of the spent fuel handling machine is not required for movement of new fuel outside the spent fuel pool.

The purpose of Technical Specification 3/4.9.7 is to restrict movement of loads in excess of the nominal weight of a fuel assembly, control element assembly (CEA), and associated handling tool over other fuel assemblies in the spent fuel pool to ensure that in the event this load is dropped, (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. The original intent of the Specification, as it relates to new fuel, was to require new fuel within the spent fuel pool be handled by the spent fuel handling machine to protect against damage to irradiated fuel.

The proposed change to Technical Specification 3.9.7 will clarify that the use of the spent fuel handling machine is not required for movement of new fuel assemblies outside the spent fuel pool and will also allow for movement of new fuel assemblies in areas other than the spent fuel pool if the spent fuel handling machine is inoperable.

Along this line, the proposed change will bring Technical Specification 3/4.9.7 into conformance with FSAR Section 9.1.4 which specifies the use of other fuel handling equipment (cask crane, new fuel elevator, etc.) for the movement of new fuel outside the spent fuel pool.

The proposed change consists of the following two parts:

a. Technical Specification 3.9.7 currently states in part:

> "Cranes in the fuel handling building shall be restricted as follows: a. The spent fuel handling machine shall be used for the movement of fuel assemblies (with or without CEAs) and shall be OPERABLE with...."

The proposed change will add the following note of clarification:

> "Not required for movement of new fuel assemblies outside the spent fuel pool."

b. The proposed change will add the following Action Statement to Technical Specification 3.9.7:

> "The provisions of Specification 3.0.4 are not applicable."

Specification 3.0.4 normally prevents entry into the applicable mode or condition (movement of fuel assemblies in this case) unless the conditions of the Limiting Condition for Operation are met. This added ACTION Statement will allow for start of new fuel movement in areas other than the spent fuel pool while ACTION Statement a. is in effect.

Steam Generator Pressure/Temperature Limits

The proposed change revises the Appendix A Technical Specifications by changing Technical Specification 3/4.7.2 "Steam Generator Pressure/Temperature Limits" and reflecting this change in the related section of the Bases.

The purpose of Technical Specification 3/4.7.2 is to ensure that steam generator secondary pressure and temperature is limited so that pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The purpose of Specification 3.7.2(b) is to ensure, in the event of a low temperature overpressurization of the steam generator secondary, that an engineering evaluation is completed and it is determined that the steam generator remains acceptable for continued operation.

The proposed change will allow for steam generator temperatures up to 200°F prior to completion of the engineering evaluation, consistent with the Revision 3 of the CE Standard Technical Specifications. The present temperature value of 115°F, with respect to performing an engineering evaluation, is incorrect. The LIMITING CONDITION FOR OPERATION (LCO) 3.7.2 properly requires that secondary side steam generator temperature be greater than 115°F when secondary side pressure is above 210 psig. The limitation to 115°F and 210 psig is based on a steam generator RT_{NDT} of 40°F, which is sufficient to prevent brittle fracture. However, when the Waterford 3 Technical Specifications were issued, the LCO temperature of 115°F was inadvertently substituted into ACTION statement 3.7.2.b. As noted above, the CE Standard Technical Specification temperature limitation of 200°F prior to completion of the engineering evaluation (the ACTION statement temperature) should not have been stated as 115°F. Raising the ACTION statement temperature limitation to 200°F corrects this error and is more conservative in the event of an overpressure condition.

3.0 EVALUATION

Containment Penetration Conductor Overcurrent Protective Devices

The proposed changes to Table 3.8-1, as described above, are purely administrative and will only correct typographical errors to bring the Technical Specification into conformance with other plant documents. Therefore, the proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated or create the possibility of a new or different kind of accident from any accident previously evaluated or involve a significant reduction in the margin of safety.

The staff finds these proposed changes to be acceptable.

Crane Travel - Fuel Handling Building

The proposed change to Technical Specification 3/4.9.7, as described above, will allow for the use of fuel handling equipment designed and intended for the movement of new fuel outside the spent fuel pool and bring the Technical Specification into conformance with the FSAR.

The Fuel Handling Accident Analysis in FSAR Chapter 15 is based on the Fuel Handling System described in FSAR Subsection 9.1.4. The proposed change only allows for the use of fuel handling equipment as described by FSAR Subsection 9.1.4 and continues to restrict the movement of heavy loads over fuel assemblies in the fuel spent pool. Therefore, the proposed change will not involve any increase in the probability or consequences of any accident previously evaluated.

Operation of the facility will be in accordance with the assumptions made in the FSAR and the Technical Specification that fuel will be handled in accordance with the designed fuel handling system and movement of heavy loads in the spent fuel pool will be restricted. Therefore, the proposed change will not involve any reduction in margin of safety.

Operation of the facility will be in accordance with the assumptions made in the FSAR and the Technical Specification that fuel will be handled in accordance with the designed Fuel Handling System and movement of heavy loads in the spent fuel pool will be restricted. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The staff concludes that the proposed change is acceptable.

Steam Generator Pressure/Temperature Limits

The proposed change will change the temperature value of 115°F by revising Technical Specification 3.7.2(b) to reflect the 200°F temperature value shown in the CE Standard Technical Specifications, which is the temperature value originally intended for this ACTION. Thus, the proposed change will correct an error that occurred during development of the Technical Specifications.

The lowest service temperature for the secondary side of the steam generators is 115°F when pressure is 210 psig or greater. Assuming steam generator temperature drops below 115°F, the Technical Specification as currently written limits temperature to 115°F or below while an engineering evaluation is performed. In so doing, the Technical Specification unnecessarily exposes the steam generators to the potential for brittle fracture in the event of an overpressure condition. The proposed change would allow an increase in steam generator temperature up to 200°F while performing the engineering evaluation, this providing a more conservative condition with respect to brittle fracture should an overpressure condition occur. Therefore, the proposed change will not involve any increase in the probability or consequences of any accident previously evaluated. In fact, the probability of brittle fracture will decrease.

Temperatures less than 200°F do not impact Loss of Coolant Accident or Main Steam Line Break considerations. The proposed change requires temperatures be maintained to 200°F or less until it is determined that the steam generator remains acceptable for continued operation. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The lowest service temperature for the secondary side of the steam generators is 115°F. The Technical Specification, as currently written, limits the temperature to 115°F or below and is nonconservative because it unnecessarily exposes the steam generator to brittle fracture in the event of an overpressure condition. The proposed change allows for temperatures up to 200°F, providing for a more conservative condition by allowing temperatures that will place the steam generator material in the ductile range and making them less susceptible to brittle fracture. Therefore, the proposed change increases rather than decreases the margin of safety.

The brittle fracture resistance of the Waterford 3 steam generator materials increases with increased temperature. Since the increase in temperature (from 115°F to 200°F) would result in an increase in brittle fracture resistance of the steam generator materials, the staff concludes that the proposed change is acceptable.

In order to be consistent with the change made in TS 3/4.7.2, the related section of the Bases is being changed to reference the 200°F temperature limitation while the engineering evaluation is being performed.

4.0 CONTACT WITH STATE OFFICIAL

The NRC staff has advised the Administrator, Nuclear Energy Division, Department of Environmental Quality, State of Louisiana of the proposed determination of no significant hazards consideration. No comments were received.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in the installation or use of facility components located within the restricted area. The staff has determined that the amendment involves no significant increase in the amounts of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupation radiation exposure. The Commission has previously issued proposed findings that the amendment involves no significant hazards consideration, and there has been no public comment on such findings. (The change in the Bases for TS 3/4.7.2 required no prior notice of consideration of issuance of amenemnt.) 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 this amendment.

6.0 CONCLUSION

Based upon our evaluation of the proposed changes to the Waterford 3 Technical Specifications, we have concluded that: there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public. We, therefore, conclude that the proposed changes are acceptable, and are hereby incorporated into the Waterford 3 Technical Specifications.

Dated: October 16, 1986