February 18, 1987

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. 14 TO FACILITY OPERATING LICENSE NO. NPF-38 FOR WATERFORD 3

The Commission has issued the enclosed Amendment No. 14 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 applications transmitted by letters dated June 24, July 15, and August 29, 1986, as supplemented by letters dated October 3, November 3, and November 12, 1986. The November 3 and November 12 letters were merely explanatory and not substantive changes.

The amendment revises the Appendix A Technical Specifications by: bypassing the non-safety related high steam generator level trip below 20% of rated thermal power; adding the Reactor Vessel Level Monitoring System; and changing the location of the seismic monitors inside containment.

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. 14 to NPF-38
2. Safety Evaluation

cc: See next page

8703020219 870218
PDR ADOCK 05000382
P PDR

*See previous concurrence

cc: See next page

PD7	PD7	OGC	DIR:PD7	PEICSB
*JWilson/es	*JLee	*SETurk	*GWKnighton	*CLMiller
2/5 /87	2/5 /87	2/6 /87	2/ /87	2/17/87

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

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Senior Vice President - Nuclear Operations
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The amendment revises the Appendix A Technical Specifications by: bypassing the non-safety related high steam generator level trip below 20% of rated thermal power; adding the Reactor Vessel Level Monitoring System; and changing the location of the seismic monitors inside containment.

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

Sincerely,

James H. Wilson

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

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1. Amendment No. 14 to NPF-38
2. Safety Evaluation

cc: See next page

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Louisiana Power & Light Company

Waterford 3

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

LOUISIANA POWER AND LIGHT COMPANY

DOCKET NO. 50-382

WATERFORD STEAM ELECTRIC STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 14
License No. NPF-38

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

   A. The applications for amendment, dated June 24, July 15, and August 29, 1986, as supplemented by letters dated October 3, November 3, and November 12, 1986 by Louisiana Power and Light Company (licensee), comply 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 applications, 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. 14, 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: February 18, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 14

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 3-3	-
3/4 3-4	-
3/4 3-36	3/4 3-35
3/4 3-37	3/4 3-38
3/4 3-44	3/4 3-43
3/4 3-45	-
3/4 3-45a	-
3/4 3-46	-
B 3/4 3-3	-
B 3/4 3-3a	-

Pages 3/4 2-1, 3/4 2-2 and 3/4 2-2a are reissued for pagination purposes, and Page B 3/4 3-4 is reissued without change.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2 sets of 2 2 sets of 2	1 set of 2 1 set of 2	2 sets of 2 2 sets of 2	1, 2 3*, 4*, 5*	1 8
2. Linear Power Level - High	4	2	3	1, 2	2#, 3#
3. Logarithmic Power Level-High					
a. Startup and Operating	4 4	2(a)(d) 2	3 3	1, 2 3*, 4*, 5*	2#, 3# 8
b. Shutdown	4	0	2	3, 4, 5	4
4. Pressurizer Pressure - High	4	2	3	1, 2	2#, 3#
5. Pressurizer Pressure - Low	4	2(b)	3	1, 2	2#, 3#
6. Containment Pressure - High	4	2	3	1, 2	2#, 3#
7. Steam Generator Pressure - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
8. Steam Generator Level - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
9. Local Power Density - High	4	2(c)(d)	3	1, 2	2#, 3#
10. DNBR - Low	4	2(c)(d)	3	1, 2	2#, 3#
11. Steam Generator Level - High	4/SG	2/SG(g)	3/SG	1, 2	2#, 3#
12. Reactor Protection System Logic	4	2	3	1, 2 3*, 4*, 5*	5 8
13. Reactor Trip Breakers	4	2(f)	4	1, 2 3*, 4*, 5*	5 8
14. Core Protection Calculators	4	2(c)(d)	3	1, 2	2#, 3# and 7
15. CEA Calculators	2	1	2(e)	1, 2	6 and 7
16. Reactor Coolant Flow - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(c) Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 10^{-4}% of RATED THERMAL POWER. During testing pursuant to Special Test Exception 3.10.3, trip may be manually bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 5% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) Each channel shall be comprised of two trip breakers; actual trip logic shall be one-out-of-two taken twice.

(g) High steam generator level trip may be manually bypassed in Modes 1 and 2, at 20% power and below.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6k. The channel shall be returned to OPERABLE status prior to STARTUP following the next COLD SHUTDOWN.

INSTRUMENTATION

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 Each of the above seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the above seismic monitoring instruments which is accessible during power operation and which is actuated during a seismic event (one or more basemat accelerations of 0.05 g or greater) shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 5 days. Data shall be retrieved from the accessible actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety. Each of the above seismic monitoring instruments which is actuated during a seismic event (one or more basemat accelerations of 0.05 g or greater) but is not accessible during power operation shall be restored to OPERABLE status and a CHANNEL CALIBRATION performed the next time the plant enters MODE 3 or below. A supplemental report shall then be prepared and submitted to the Commission within 10 days pursuant to Specification 6.9.2 describing the additional data from these instruments.

TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelerograph System		
a. Accelerometer (YT-SM 6000) Adjacent to RB -35 ft MSL	0.02-1.0 g	1
b. Accelerometer (YT-SM 6001) RB +46 ft MSL	0.02-1.0 g	1
c. Accelerometer (YT-SM 6002) Free Field Yard Area	0.02-1.0 g	1
d. Starter Unit (YS-SM 6000) Adjacent to RB -35 ft MSL	0.01-0.02 g	1
e. Starter Unit (YS-SM 6001) RB +51 ft MSL	0.01-0.02 g	1
f. Recorder (YR-SM 6000) Control Room RAB +46 ft MSL	0.02-1.0 g	1
g. Control Unit (YZ-SM 6000) Control Room RAB +46 ft MSL	0.02-1.0 g	1*
h. Playback Unit (YR-SM 6001) Control Room RAB +46 ft MSL	0.02-1.0 g	1
2. Triaxial Peak Accelerographs		
a. YR-SM 6020 RB +56 ft MSL	0-2 g	1
b. YR-SM 6021 RB +8'2" MSL	0-2 g	1
c. YR-SM 6022 RAB +21 ft MSL	0-2 g	1
3. Triaxial Seismic Switches		
a. Seismic Swtich (YS-SM 6060) RB -35 ft MSL	0.1-0.25 g	1
b. Control Unit (YZ-SM 6060) Control Room RAB +46 ft MSL	0.1-0.25 g	1*
4. Triaxial Response-Spectrum Recorders		
a. YR-SM 6040 RB +10 ft MSL	1-32 Hz, 0-2 g	1
b. YR-SM 6041 RAB -35 ft MSL	1-32 Hz, 0-2 g	1
c. YR-SM 6042 RAB +21 ft MSL	1-32 Hz, 0-2 g	1
d. Peak Shock Annunciator (YR-SM 6045) RB -35 ft MSL	1-32 Hz, 0-2 g	1
e. Peak Shock Annunciator Control Unit (YZ-SM 6045) Control Room RAB +46 ft MSL	1-32 Hz, 0-2 g	1

*With reactor control room annunciation.

TABLE 4.3-4
SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Triaxial Time-History Accelerograph System			
a. Accelerometer (YT-SM 6000) Adjacent to RB -35 ft MSL	N.A.	R	SA
b. Accelerometer (YT-SM 6001) RB +46 ft MSL	N.A.	R	SA
c. Accelerometer (YT-SM 6002) Free Field Yard Area	N.A.	R	SA
d. Starter Unit (YS-SM 6000) Adjacent to RB -35 ft MSL	M	R	SA
e. Starter Unit (YS-SM 6001) RB +51 ft MSL	M	R	SA
f. Recorder (YR-SM 6000) Control Room RAB +46 ft MSL	M	R	SA
g. Control Unit (YZ-SM 6000) Control Room RAB +46 ft MSL	M	R	SA*
h. Playback Unit (YR-SM 6001) Control Room RAB +46 ft MSL	N.A.	R	SA
2. Triaxial Peak Accelerographs			
a. YR-SM 6020 RB +56 ft MSL	N.A.	R	N.A.
b. YR-SM 6021 RB +8'2" MSL	N.A.	R	N.A.
c. YR-SM 6022 RAB +21 ft MSL	N.A.	R	N.A.
3. Triaxial Seismic Switches			
a. Seismic Switch YS-SM 6060 RB -35 ft MSL	M	R	SA
b. Control Unit YZ-SM 6060 Control Room RAB +46 ft MSL	M	R	SA*
4. Triaxial Response-Spectrum Recorders			
a. YR-SM 6040 RB +10 ft MSL	N.A.	R	N.A.
b. YR-SM 6041 RAB -35 ft MSL	N.A.	R	N.A.
c. YR-SM 6042 RAB +21 ft MSL	N.A.	R	N.A.
d. Peak Shock Annunciator YR-SM 6045 RB -35 ft MSL	N.A.	R	N.A.
e. Peak Shock Annunciator Control Unit YZ-SM 6045 Control Room RAB +46 ft MSL	N.A.	R	SA

*With reactor control room annunciation.

METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.4 The meteorological monitoring instrumentation channels shown in Table 3.3-8 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more required meteorological monitoring channels inoperable for more than 7 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.4 Each of the above meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-5.

TABLE 4.3-6

REMOTE SHUTDOWN INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Neutron Flux	M	R*
2. Reactor Trip Breaker Indication	M	N.A.
3. Reactor Coolant Temperature- Cold Leg (T_{Cold})	M	R
4. Reactor Coolant Temperature - Hot Leg (T_{Hot})	M	R
5. Pressurizer Pressure	M	R
6. Pressurizer Level	M	R
7. Steam Generator Level	M	R
8. Steam Generator Pressure	M	R
9. Shutdown Cooling Flow Rate	M	R
10. Emergency Feedwater Flow Rate	M	R
11. Condensate Storage Pool Level	M	R

*Neutron detector may be excluded from CHANNEL CALIBRATION.

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-10, take the action identified in Table 3.3-10.

b. With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10, take the action identified in Table 3.3-10.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
1. Containment Pressure	2	1	29,30
2. Reactor Coolant Outlet Temperature - T_{Hot} (Wide Range)	2	1	29,30
3. Reactor Coolant Inlet Temperature - T_{Cold} (Wide Range)	2	1	29,30
4. Reactor Coolant Pressure - Wide Range	2	1	29,30
5. Pressurizer Water Level	2	1	29,[illegible]
6. Steam Generator Pressure	2/steam generator	1/steam generator	29,30
7. Steam Generator Water Level - Narrow Range	2/steam generator	1/steam generator	29,30
8. Steam Generator Water Level - Wide Range	1/steam generator**	1/steam generator**	29,30
9. Refueling Water Storage Pool Water Level	2	1	29,30
10. Emergency Feedwater Flow Rate	1/steam generator**	1/steam generator**	29,30
11. Reactor Cooling System Saturation Margin Monitor	2	1	29,30
12. Safety Valve Position Indicator	1/valve	1/valve	29,30
13. Containment Water Level (Narrow Range)	1***	1***	29,30
14. Containment Water Level (Wide Range)	2	1	29,30
15. Core Exit Thermocouples	4/core quadrant	2/core quadrant	29,30
16. Containment Isolation Valve Position Indicators*	1/valve#	1/valve#	29,[illegible]
17. Condensate Storage Pool Level	2	1	29,30
18. Reactor Vessel Level Monitoring System	2****	1	31,32

#If the containment isolation valve is declared inoperable and the provisions of Specification 3.6.3 are complied with, position indicators may be inoperable; otherwise, comply with the provisions of Specification 3.3.3.6.

*Containment isolation valves listed in Table 3.6-2 (Category 1).

**These corresponding instruments may be substituted for each other.

***Operation may continue for up to 30 days with less than the Minimum Channels OPERABLE requirement.

****A channel is eight sensors in a probe. A channel is operable if four or more sensors, one or more in the upper three and three or more in the lower five, are operable.

TABLE 3.3-10

ACTION STATEMENTS

ACTION 29 - With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-10, either restore the inoperable channel to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

ACTION 30 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10; either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 31 - With the number of OPERABLE accident monitoring channels, less than the Required Number of Channels, either restore the system to OPERABLE status within 7 days if repairs are feasible without shutting down or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 32 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE in Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1. Initiate an alternate method of monitoring the reactor vessel inventory;

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status; and

3. Restore the system to OPERABLE status at the next scheduled refueling.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Reactor Coolant Outlet Temperature - T_{Hot} (Wide Range)	M	R
3. Reactor Coolant Inlet Temperature -T_{Cold} (Wide Range)	M	R
4. Reactor Coolant Pressure - Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Generator Pressure	M	R
7. Steam Generator Water Level - Narrow Range	M	R
8. Steam Generator Water Level - Wide Range	M	R
9. Refueling Water Storage Pool Water Level	M	R
10. Emergency Feedwater Flow Rate	M	R
11. Reactor Coolant System Saturation Margin Monitor	M	R
12. Safety Valve Position Indicator	M	R
13. Containment Water Level (Narrow Range)	M	R
14. Containment Water Level (Wide Range)	M	R
15. Core Exit Thermocouples	M	R
16. Containment Isolation Valve Position	M	R
17. Condensate Storage Pool Level	M	R
18. Reactor Vessel Level Monitoring System	M	R

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1980 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations." Table 3.3-10 includes Regulatory Guide 1.97 Category I key variables. The remaining Category I variables are included in their respective specifications.

The Subcooled Margin Monitor (SMM), the Heated Junction Thermocouple (HJTC), and the Core Exit Thermocouples (CET) comprise the Inadequate Core Cooling (ICC) instrumentation required by Item II.F.2 NUREG-0737, the Post TMI-2 Action Plan. The function of the ICC instrumentation is to enhance the ability of the plant operator to diagnose the approach to existence of, and recovery from ICC. Additionally, they aid in tracking reactor coolant inventory. These instruments are included in the Technical Specifications at the request of NRC Generic Letter 83-37. These are not required by the accident analysis, nor to bring the plant to Cold Shutdown.

In the event more than four sensors in a Reactor Vessel Level channel are inoperable, repairs may only be possible during the next refueling outage. This is because the sensors are accessible only after the missile shield and reactor vessel head are removed. It is not feasible to repair a channel except during a refueling outage when the missile shield and reactor vessel head are removed to refuel the core. If only one channel is inoperable, it should be restored to OPERABLE status in a refueling outage as soon as reasonably possible. If both channels are inoperable, at least one channel shall be restored to OPERABLE status in the nearest refueling outage.

3/4.3.3.7 CHEMICAL DETECTION SYSTEMS

The OPERABILITY of the chemical detection systems ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chemical release. This capability is required to protect control room personnel and is consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," February 1975 and the recommendations of Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," June 1974.

3/4.3.3.8 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.9 LOOSE-PART DETECTION INSTRUMENTATION

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and Surveillance Requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.3.10 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.11 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the WASTE GAS HOLDUP SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment, or structures.

3/4.2 POWER DISTRIBUTION LIMITS

3/4 2.1 LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate limit (of Figure 3.2-1) shall be maintained by one of the following methods as applicable:

a. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on linear heat rate (when COLSS is in service); or

b. Operating within the region of acceptable operation of Figure 3.2-1a using any operable CPC channel (when COLSS is out of service).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With the linear heat rate limit not being maintained as indicated by:

1. COLSS calculated core power exceeding COLSS calculated core power operating limit based on linear heat rate; or

2. When COLSS is out of service, operation outside the region of acceptable operation in Figure 3.2-1a;

within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:

a. Restore the linear heat rate to within its limits within 1 hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the linear heat rate, as indicated on any OPERABLE Local Power Density channel, is within the limits shown on Figure 3.2-1a.

4.2.1.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on kW/ft.

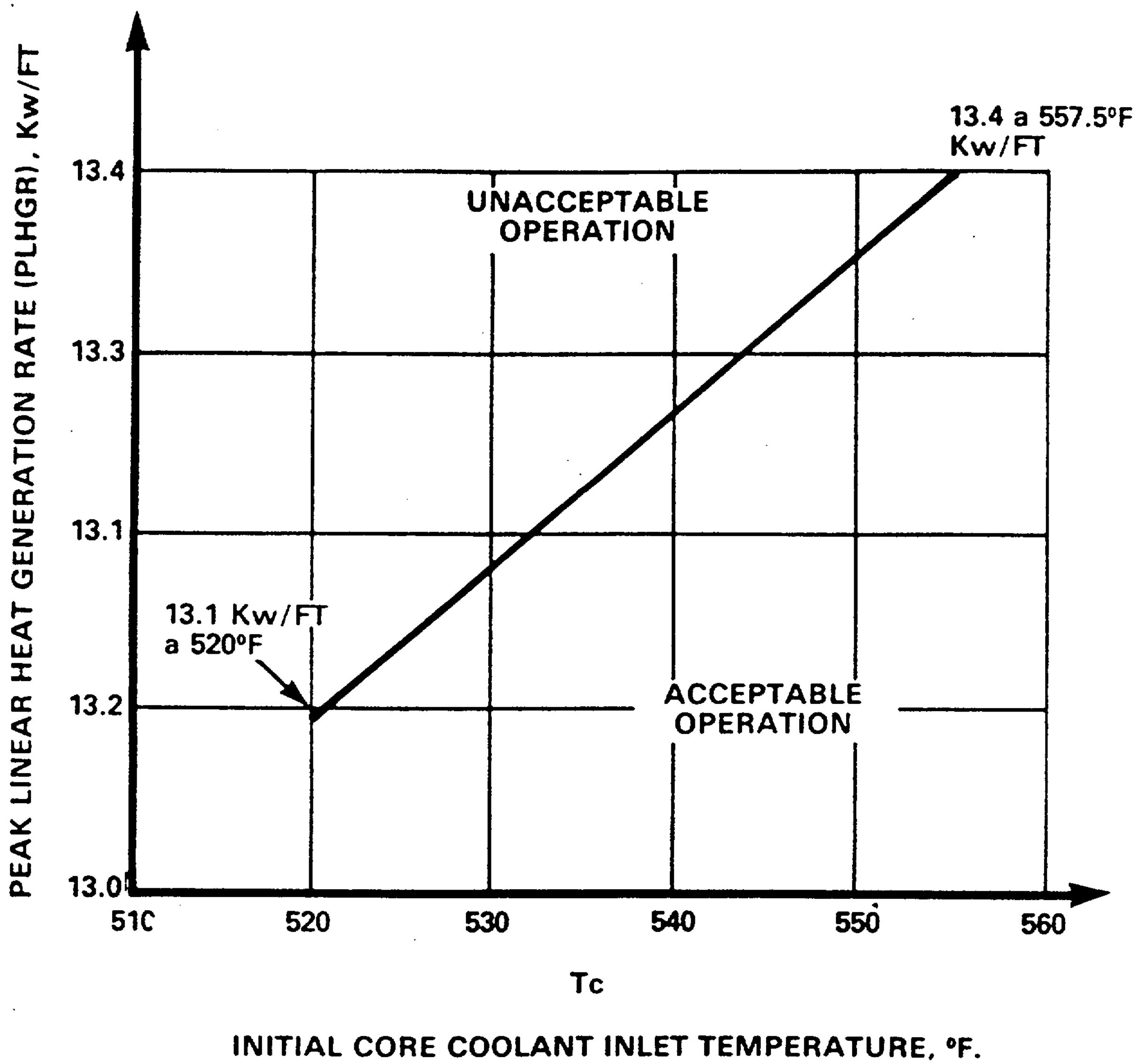

FIGURE 3.2-1

ALLOWABLE PEAK LINEAR HEAT RATE VS Tc

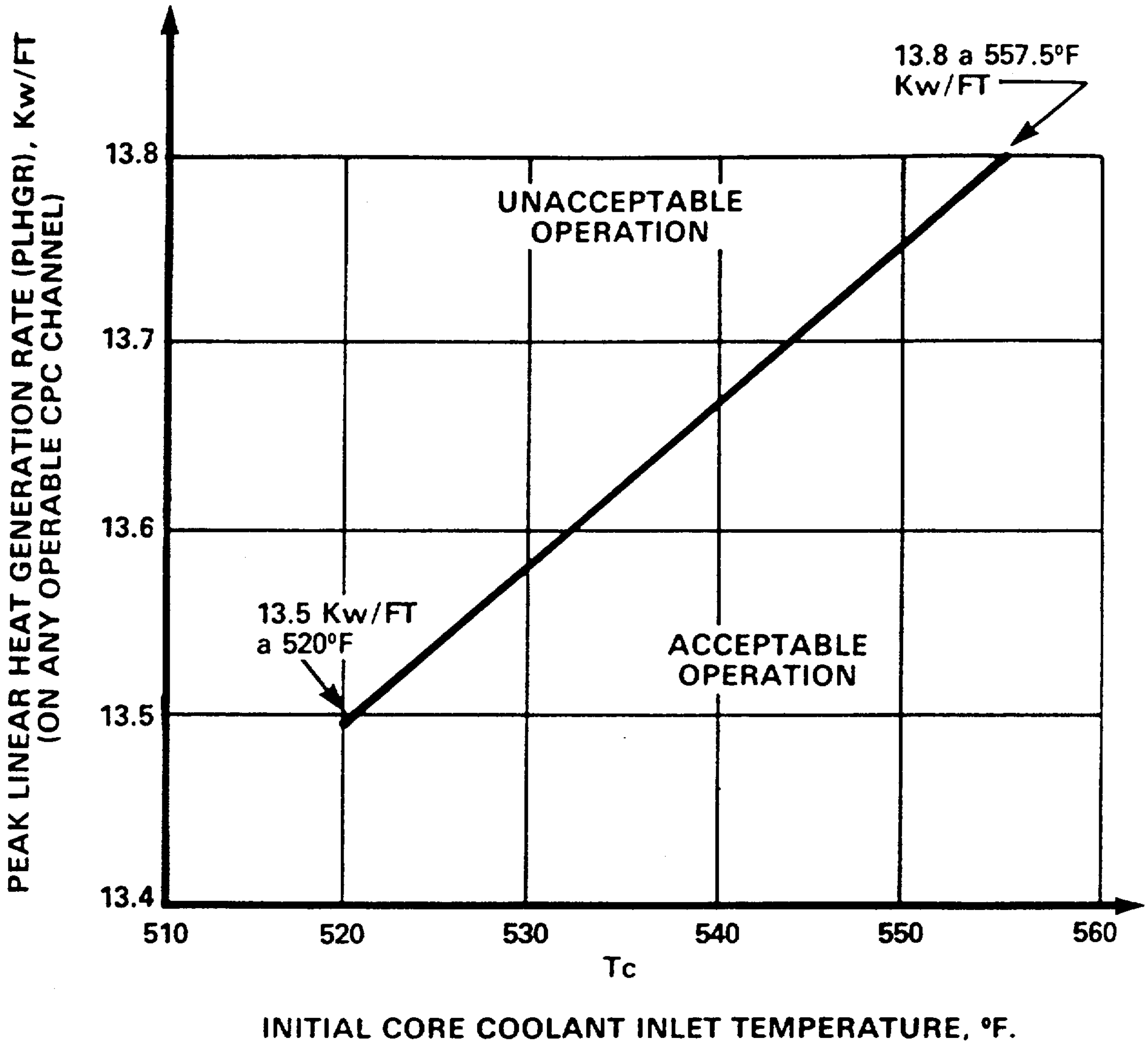

FIGURE 3.2-1a

ALLOWABLE PEAK LINEAR HEAT RATE VS Tc
FOR COLSS OUT OF SERVICE

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 14 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 applications dated June 24, July 15, and August 29, 1986, as supplemented by letters dated October 3, November 3, and November 12, 1986 , Louisiana Power and Light Company (the licensee or LPL) 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) bypass the non-safety related high steam generator level trip; (2) add the Reactor Vessel Level Monitoring System; and (3) change the location of the seismic monitors inside containment.

2.0 DISCUSSION

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

2.1 Steam Generator Level Hi Trip (NPF-38-23)

The proposed change would revise Table 3.3-1 of the Waterford 3 Technical Specifications to include the capability of manually bypassing the high steam generator level trip during Modes 1 and 2 at power levels of 20% full power or less. The revision would be made by adding footnote (g) to the Table on Reactor Protective Instrumentation.

2.2 Reactor Vessel Level Monitoring System (NPF-38-28)

The proposed change would revise Technical Specification 3.3.3.6, "Accident Monitoring Instrumentation", to add the Reactor Vessel Level Monitoring System (RVLMS) to Tables 3.3-10 and 4.3-7, and the associated Bases.

2.3 Seismic Monitors (NPF-38-37)

The proposed change would reflect the relocation of seismic monitor YR-SM 6020 and correct an error in the location of another seismic monitor, YR-SM 6021. YR-SM 6020 and YR-SM 6021 are passive devices that allow for evaluation of reactor coolant system response after a seismic event.

During the first cycle of operation of Waterford 3, YR-SM 6020 suffered heat damage. The licensee proposes to relocate this monitor during the first refueling outage, moving it from the high temperature environment at its present location on the pressurizer to Safety Injection Tank (SIT) 1B. The proposed relocation uses a mount virtually identical to the original mount and which performs the same function.

3.0 EVALUATION

The proposed changes to the Technical Specifications requested by the licensee and described in three areas above, are evaluated below.

3.1 Steam Generator Level Hi Trip (NPF-38-23)

In Waterford 3, a high steam generator water level trip is provided to trip the reactor when measured steam generator water level rises to a high preset value, nominally 87.7 percent of the distance between the lower and upper instrument nozzles. Since the turbine is automatically tripped when the reactor is tripped, the high steam generator water level trip is provided to protect the turbine from excessive moisture carryover. The trip is an equipment protective trip only and, thus, is non-safety related. No credit was taken for operation of this trip in the plant safety analyses nor does the trip setpoint correspond to a Technical Specification safety limit. Likewise, the design and reliability of the reactor protection system is unaffected by the proposed change.

The high steam generator water level trip bypass will be administratively controlled such that the bypass cannot be enabled above 20% power. For power levels at or below 20%, existing automatic and manual controls assure that the main steam piping does not become water filled.

Since the potential does exist for the initial steam generator water level to be as high as 80.5% (narrow range), LPL has analyzed the potential impact of the increased steam generator inventory on the radiological consequences of an inadvertent opening of a steam generator atmospheric dump valve. This event was previously analyzed for Cycle 1 at zero power and with an initial steam generator water level just below the high steam generator water level trip setpoint (87.7%) in order to maximize secondary side water inventory and the radiological consequences. The reanalysis confirms that the total dose remains well within the guidelines of 10 CFR Part 100.

The consequences of a steam line break event with a higher steam generator inventory could be more severe than the consequences with a lower inventory, particularly with regard to peak containment temperature and pressure. LPL analyzed steam line break events for Cycle 2 occurring at full power initial conditions, with a loss of off-site power coincident with the reactor trip and an initial steam generator water level governed by the high water level trip setpoint (87.7%).

These results of the analysis indicate that the highest containment pressure and temperature occur within 60 seconds following a break in the main steam line and blowdown of the affected steam generator ends at 200 seconds. The peak containment pressure and temperature, therefore, occur well before the end of steam generator blowdown. Thus, the additional inventory that may be present in the steam generator at 20% power or less will not change the maximum containment pressure or temperature presented in the FSAR.

Based on the above, the staff finds that the capability of manually bypassing the high steam generator level trip at power levels of 20% or less is not a safety concern and is acceptable.

3.2 Reactor Vessel Level Monitoring System (NPF-38-28)

In response to NRC Generic Letter No. 83-37, "NUREG-0737 Technical Specifications," dated November 1, 1983, the Combustion Engineering (CE) Owners Group proposed a generic Technical Specification for the RVLMS in their letter from R. W. Wells to H. L. Thompson (NRC), dated February 19, 1985 (RWW-85-12). The proposed Technical Specification was reviewed by the NRC staff and it was found to be acceptable for application to CE designed reactors such as Waterford 3 which uses the heated junction thermocouple system (HJTCS). For non-System 80 CE plants, a channel (eight sensors in a probe) is defined to be operable if at least four of its eight sensors are operable (one or more of the upper three [upper head] and three or more of the lower five [plenum region]). The accepted Action Statements permit seven days to repair a failed channel if repairs can be made without shutting down. If repairs are not feasible during operation, a special report must be filed with the NRC within 30 days. If less than one channel is operable, then 48 hours are permitted to restore operability or an alternate method of monitoring reactor vessel inventory must be initiated, a special report filed with the NRC within 30 days, and operability restored at the next refueling shutdown.

The proposed change to the Waterford Technical Specification 3.3.3.6 incorporates the above provisions which have been approved previously by the staff. In addition, since the specific purpose of the proposed change is to enhance accident and transient monitoring capability, the staff finds the proposal acceptable.

3.3 Seismic Monitors (NPF-38-37)

The requirements for seismic monitors in the containment are given in 10 CFR Part 100, Appendix A, Section VI(a)(3): "Required Seismic Instrumentation. Suitable instrumentation shall be provided so that the seismic response of nuclear power plant features important to safety can be determined promptly to permit comparison of such response with that used as the design basis. Such a comparison is needed to decide whether the plant can continue to be operated safely and to permit such timely action as may be appropriate."

The proposed relocation of YR-SM 6020 is due to damage induced by high temperatures experienced by the monitor during the first cycle of operation. On the pressurizer, temperatures exceeded the 555°F rating of the thermal barrier installed with the monitor, resulting in damage to the monitor. For the new monitor location, the SIT maximum design temperature is 200°F - well below the thermal barrier rating. The expected SIT temperature will approximate the containment ambient temperature of 120°F, well below the design rating for the seismic monitor. Other environmental stresses, i.e., acceleration, nuclear irradiation, relative humidity, and chemical exposure, expected at the new location in the containment will be less than the values of these stresses for which the monitor was designed. Thus, the monitor is expected to be available to perform its intended function in the unlikely event of an earthquake.

Because no safety-related equipment is located below the YR-SM 6020 monitor, and there are no electric, hydraulic or pneumatic lines of any kind from YR-SM 6020 to any other system, adverse systems interactions are not a factor in the proposed relocation.

The staff concludes that the proposed relocation of seismic monitor YR-SM 6020 to the lower lifting lug of safety injection tank 1B is acceptable.

The location of seismic monitor YR-SM 6021 is given incorrectly in the Technical Specifications in Tables 3.3-7 and 4.3-4. These tables are revised per this proposed amendment to correctly describe the seismic monitor location.

The staff concludes that Tables 3.3-7 and 4.3-4 of the Technical Specifications for Waterford 3 should be corrected to reflect the actual location of YR-SM 6021.

4.0 CONTACT WITH STATE OFFICIAL

The NRC staff has advised the Administrator, Nuclear Energy Division, Office of Environmental Affairs, 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 occupational 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. 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.

CONCLUSIONS

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 amendment 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.

Principal Contributors: L. Kopp, C. Morris

Dated: February 18, 1987

February 18, 1987

ISSUANCE OF AMENDMENT NO. 14 TO FACILITY OPERATING
LICENSE NP. NPF-38 FOR WATERFORD 3

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