January 16, 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. 12 TO FACILITY OPERATING LICENSE NO. NPF-38 FOR WATERFORD 3

The Commission has issued the enclosed Amendment No. 12 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 letter dated October 1, 1986, as supplemented by letters dated October 29 and November 19, 1986.

The amendment revises the Appendix A Technical Specifications by: (1) revising the core protection calculator DNBR setpoint; (2) revising the core operating limit supervisory system out-of-service DNBR limits; (3) revising the peak linear heat rate; (4) revising the reactor protection instrumentation response times; and (5) revising the control element assembly insertion limits.

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

Sincerely,

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

Enclosures:
1. Amendment No. 12 to NPF-38
2. Safety Evaluation

cc: See next page

PD7	PD7	OGC	DIR:PD7
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1/7/87	1/ /87	1/14/87	1/16/87

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J/ 1 6 1987

ISSUANCE OF AMENDMENT NO. 12 TO FACILITY OPERATING
LICENSE NP. NPF-38 FOR 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. 12
License No. NPF-38

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

   A. The applications for amendment, dated October 1, 1986, as supplemented by letters dated October 29 and November 19, 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. 12, 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 January 17, 1987.

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: January 16, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 12

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
2-1	2-2
2-3	-
2-4	-
B 2-1	-
B 2-2	-
B 2-5	-
B 2-6	-
3/4 1-27	3/4 1-28
3/4 2-1	-
3/4 2-2	-
3/4 2-2a	-
3/4 2-6	3/4 2-5
3/4 2-7	-
3/4 2-8	-
3/4 2-9	3/4 2-10
3/4 3-8	3/4 3-7
3/4 3-9	3/4 3-10
B 3/4 1-5	-
B 3/4 2-1	-
B 3/4 2-1a	-
B 3/4 2-3	-
B 3/4 2-4	-

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE

DNBR

2.1.1.1 The DNBR of the reactor core shall be maintained greater than or equal to 1.26.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the DNBR of the reactor has decreased to less than 1.26, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

PEAK LINEAR HEAT RATE

2.1.1.2 The peak linear heat rate (adjusted for fuel rod dynamics) of the fuel shall be maintained less than or equal to 21.0 kW/ft.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the peak linear heat rate (adjusted for fuel rod dynamics) of the fuel has exceeded 21.0 kW/ft, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4, and 5

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High		
Four Reactor Coolant Pumps Operating	≤ 110.1% of RATED THERMAL POWER	≤ 110.7% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	≤ 0.257% of RATED THERMAL POWER	≤ 0.275% of RATED THERMAL POWER
4. Pressurizer Pressure - High	≤ 2365 psia	≤ 2372 psia
5. Pressurizer Pressure - Low	≥ 1684 psia (2)	≥ 1644 psia (2)
6. Containment Pressure - High	≤ 17.1 psia	≤ 17.3 psia
7. Steam Generator Pressure - Low	≥ 764 psia (3)	≥ 748 psia (3)
8. Steam Generator Level - Low	≥ 27.4% (4)	≥ 26.7% (4)
9. Local Power Density - High	≤ 21.0 kW/ft (5)	≤ 21.0 kW/ft (5)
10. DNBR - Low	≥ 1.26 (5)	≥ 1.26 (5)
11. Steam Generator Level - High	≤ 87.7% (4)	≤ 88.4% (4)
12. Reactor Protection System Logic	Not Applicable	Not Applicable
13. Reactor Trip Breakers	Not Applicable	Not Applicable
14. Core Protection Calculators	Not Applicable	Not Applicable
15. CEA Calculators	Not Applicable	Not Applicable
16. Reactor Coolant Flow - Low	≥ 23.8 psid (7)	≥ 23.6 psid (7)

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATIONS

(1) 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.

(2) Value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) Value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and low level instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. 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.

(6) Note 6 has been deleted.

(7) The setpoint may be altered to disable trip function during testing pursuant to Specification 3.10.3.

2.1.1 REACTOR CORE

The restrictions of these safety limits prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by (1) restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature, and (2) maintaining the dynamically adjusted peak linear heat rate of the fuel at or less than 21.0 kW/ft which will not cause fuel centerline melting in any fuel rod.

First, by operating within the nucleate boiling regime of heat transfer, the heat transfer coefficient is large enough so that the maximum clad surface temperature is only slightly greater than the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in higher cladding temperatures and the possibility of cladding failure.

Correlations predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the predicted DNB heat flux at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB. The minimum value of DNBR during normal operational occurrences is limited to 1.26 for the CE-1 correlation and is established as a Safety Limit. This value is based on a statistical combination of uncertainties. It includes uncertainties in the CHF correlation, allowances for rod bow and hot channel factors (related to fuel manufacturing variations) and allowances for other hot channel calculative uncertainties.

Second, operation with a peak linear heat rate below that which would cause fuel centerline melting maintains fuel rod and cladding integrity. Above this peak linear heat rate level (i.e., with some melting in the center), fuel rod integrity would be maintained only if the design and operating conditions are appropriate throughout the life of the fuel rods. Volume changes which accompany the solid to liquid phase change are significant and require accommodation. Another consideration involves the redistribution of the fuel which depends on the extent of the melting and the physical state of the fuel rod at the time of melting. Because of the above factors, the steady state value of the peak linear heat rate which would not cause fuel centerline melting is established as a Safety Limit. To account for fuel rod dynamics (lags), the directly indicated linear heat rate is dynamically adjusted.

Limiting safety system settings for the Low DNBR, High Local Power Density, High Logarithmic Power Level, Low Pressurizer Pressure and High Linear Power Level trips, and limiting conditions for operation on DNBR and kW/ft margin are specified such that there is a high degree of confidence that the specified acceptable fuel design limits are not exceeded during normal operation and design basis anticipated operational occurrences.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The Reactor Coolant System components are designed to Section III, 1974 Edition, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The DNBR - Low and Local Power Density - High are digitally generated trip setpoints based on Limiting Safety System Settings of 1.26 and 21.0 kW/ft, respectively. Since these trips are digitally generated by the Core Protection Calculators, the trip values are not subject to drifts common to trips generated by analog type equipment. The Allowable Values for these trips are therefore the same as the Trip Setpoints.

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density - High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in the latest applicable revision of CEN-305-P, "Functional Design Requirements for a Core Protection Calculator" and CEN-304-P, "Functional Design Requirements for a Control Element Assembly Calculator."

Local Power Density - High (Continued)

The local power density (LPD), the trip variable, calculated by the CPC incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the peak LPD Safety Limit. CPC uncertainties related to peak LPD are the same types used for DNBR calculation. Dynamic compensation for peak LPD is provided for the effects of core fuel centerline temperature delays (relative to changes in power density), sensor time delays, and protection system equipment time delays.

DNBR - Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor of 1860 psia. At this pressure a DNBR - Low trip will automatically occur. This low pressure trip also provides protection against steam generator tube rupture events. The DNBR is calculated in the CPC utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

b. Reactor Coolant System pressure from pressurizer pressure measurement;

c. Differential temperature (Delta T) power from reactor coolant temperature and coolant flow measurements;

d. Radial peaking factors from the position measurement for the CEAs;

e. Reactor coolant mass flow rate from reactor coolant pump speed;

f. Core inlet temperature from reactor coolant cold leg temperature measurements.

The DNBR, the trip variable, calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core DNBR is sufficiently greater than the fuel design limit such that the decrease

DNBR - Low (Continued)

in actual core DNBR after the trip will not result in a violation of the DNBR Safety Limit of 1.26. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

a.	RCS Cold Leg Temperature-Low	$\geq$ 495°F
b.	RCS Cold Leg Temperature-High	$\leq$ 580°F
c.	Axial Shape Index-Positive	Not more positive than +0.5
d.	Axial Shape Index-Negative	Not more negative than -0.5
e.	Pressurizer Pressure-Low	$\geq$ 1860 psia
f.	Pressurizer Pressure-High	$\leq$ 2375 psia
g.	Integrated Radial Peaking Factor-Low	$\geq$ 1.28
h.	Integrated Radial Peaking Factor-High	$\leq$ 4.28
i.	Quality Margin-Low	$\geq$ 0

Steam Generator Level - High

The Steam Generator Level - High trip is provided to protect the turbine from excessive moisture carry over. Since the turbine is automatically tripped when the reactor is tripped, this trip provides a reliable means for providing protection to the turbine from excessive moisture carry over. This trip's setpoint does not correspond to a Safety Limit and no credit was taken in the safety analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

Reactor Coolant Flow - Low

The Reactor Coolant Flow - Low trip provides protection against a reactor coolant pump sheared shaft event and a steam line break event with a loss-of-offsite power. A trip is initiated when the pressure differential across the primary side of either steam generator decreases below a nominal setpoint of 23.8 psid. The specified setpoint ensures that a reactor trip occurs to prevent violation of local power density or DNBR safety limits under the stated conditions.

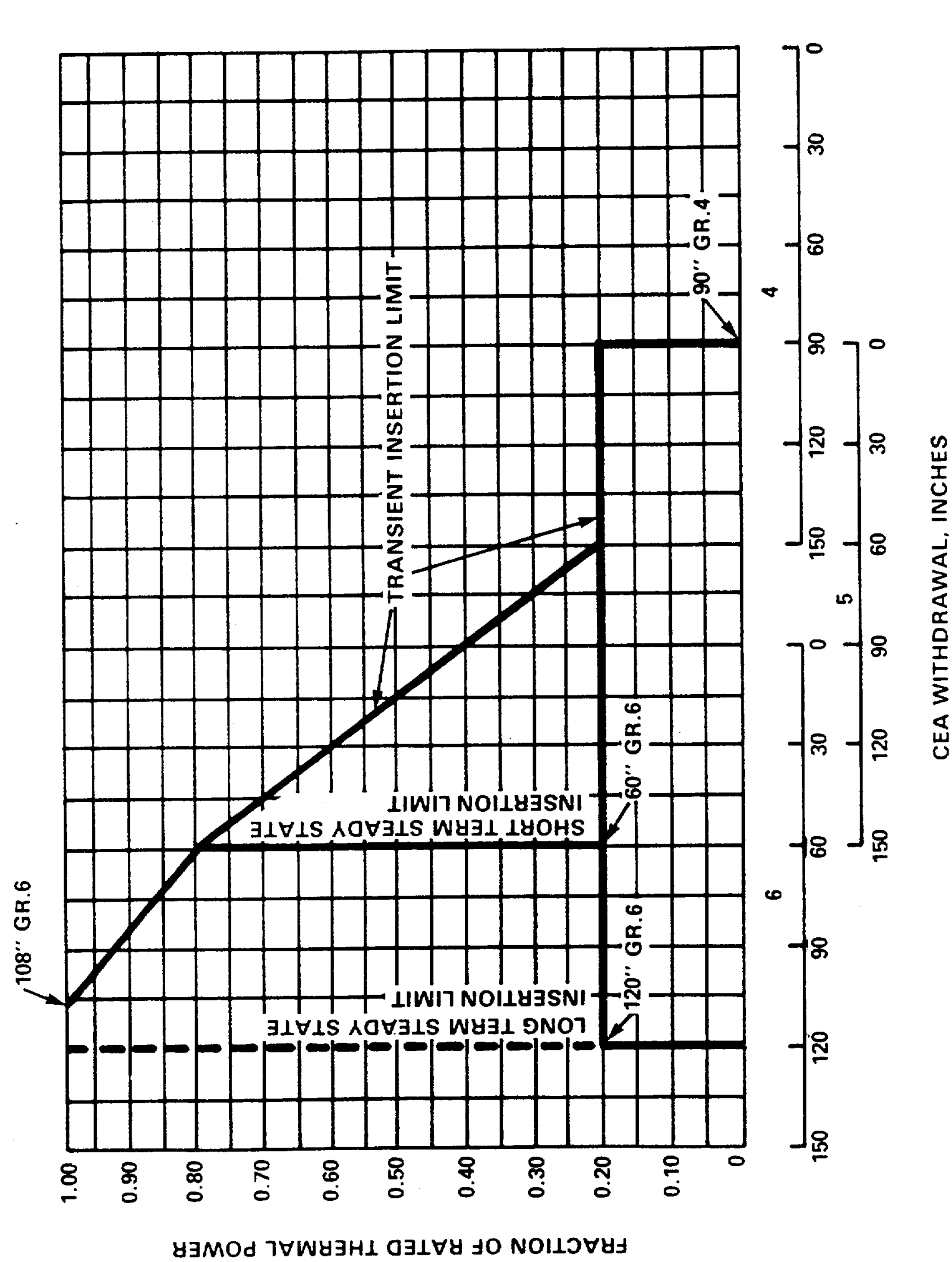

FIGURE 3.1-2
CEA INSERTION LIMITS VS THERMAL POWER

PART-LENGTH CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.7 Part-length CEA groups positioned between 0" - 17" withdrawn shall be restricted to prevent the neutron absorber section of the part-length CEA group from covering the same axial segment of the fuel assemblies for a period in excess of 7 EFPD out of any 30 EFPD period.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the neutron absorber section of the part-length CEA group covering any same axial segment of the fuel assemblies for a period exceeding 7 EFPD out of any 30 EFPD period, either:

a. Reposition the part-length CEA group to ensure no neutron absorber section of the part-length CEA group is covering the same axial segment of the fuel assemblies within 2 hours, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.7 The position of the part-length CEA group shall be determined at least once per 12 hours.

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.

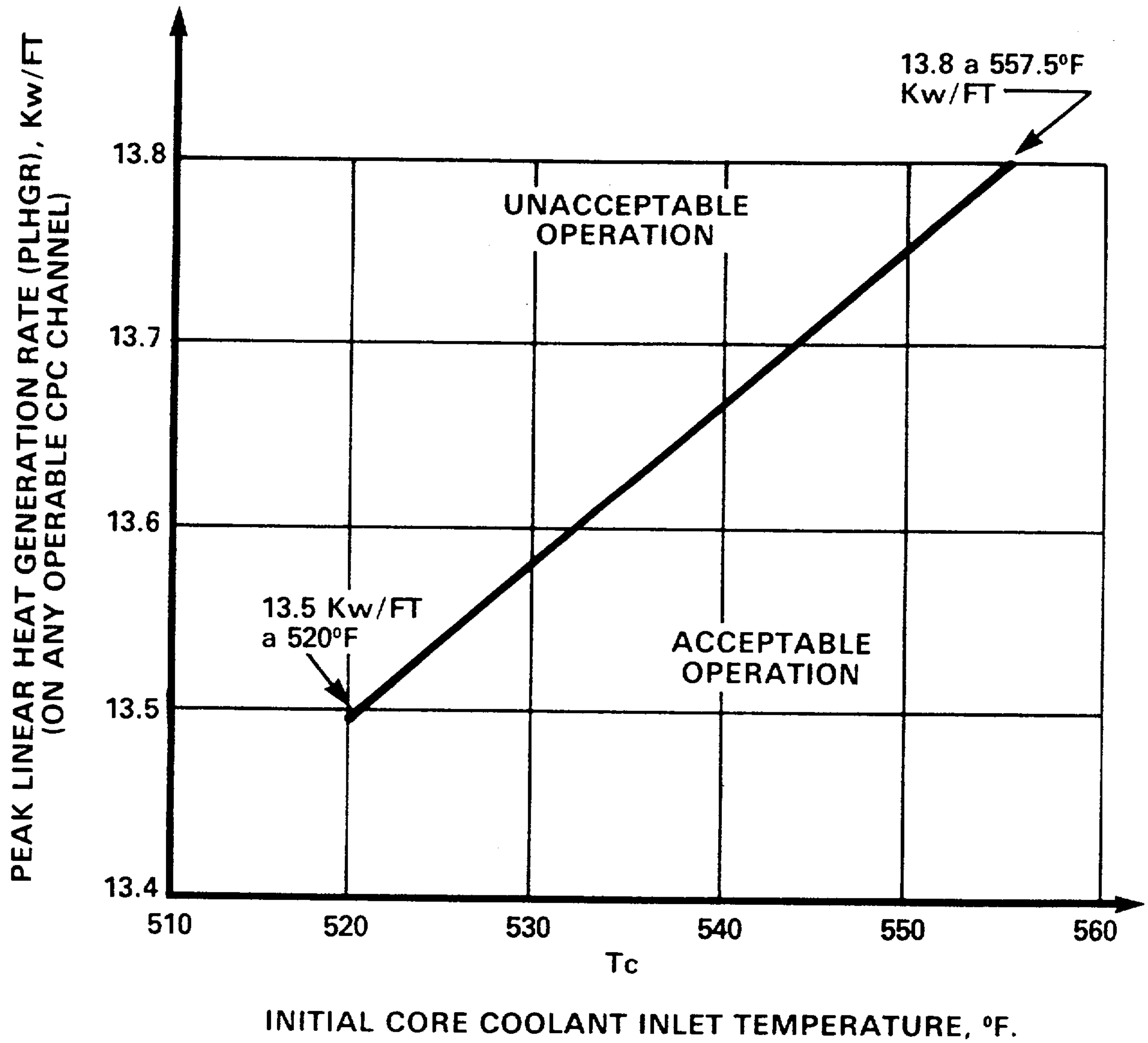

FIGURE 3.2-1a

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

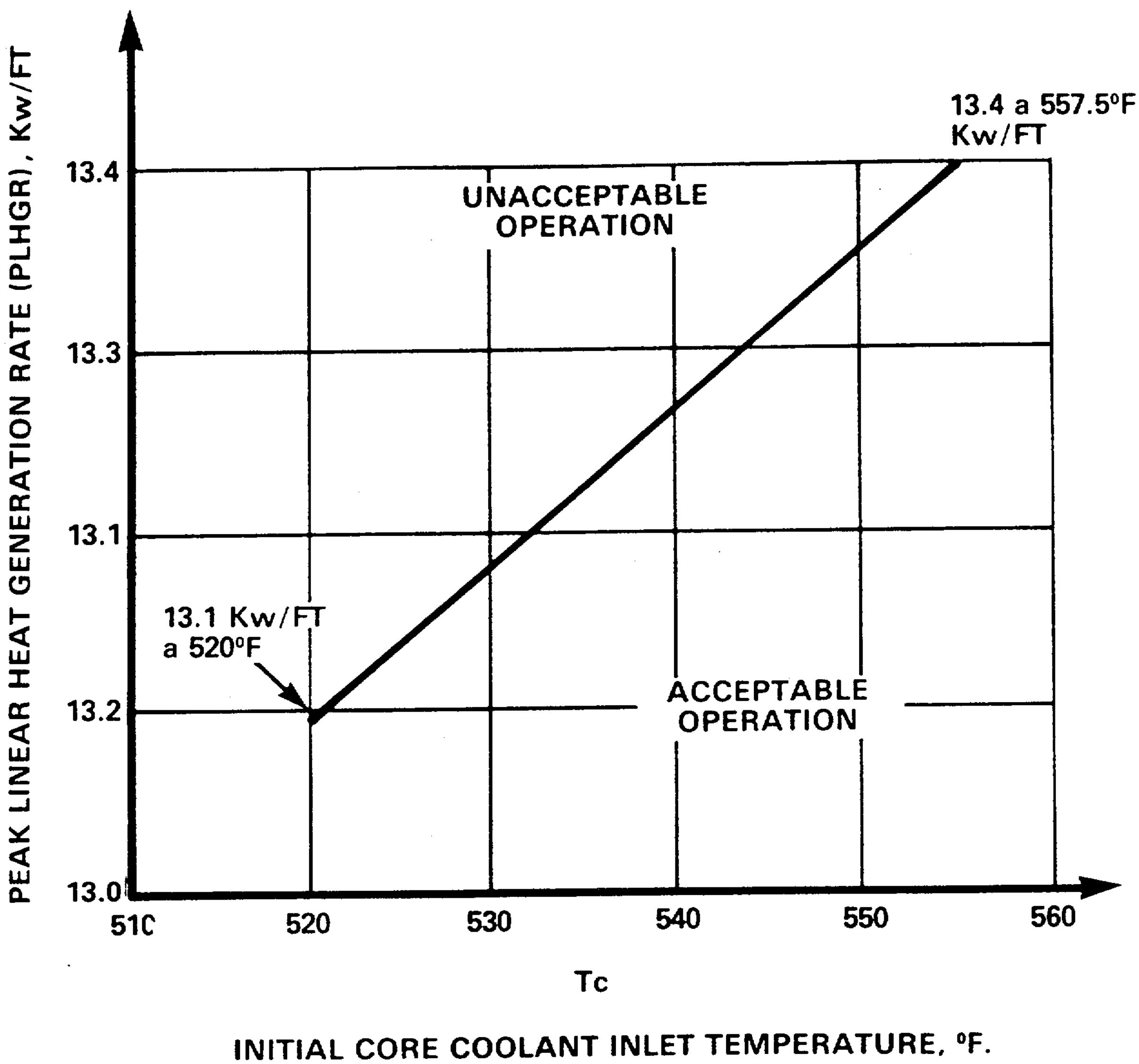

FIGURE 3.2-1

ALLOWABLE PEAK LINEAR HEAT RATE VS Tc

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The AZIMUTHAL POWER TILT shall be determined to be within the limit above 20% of RATED THERMAL POWER by:

a. Continuously monitoring the tilt with COLSS when the COLSS is OPERABLE.

b. Calculating the tilt at least once per 12 hours when the COLSS is inoperable.

c. Verifying at least once per 31 days, that the COLSS Azimuthal Tilt Alarm is actuated at an AZIMUTHAL POWER TILT greater than the AZIMUTHAL POWER TILT Allowance used in the CPCs.

d. Using the incore detectors at least once per 31 days to independently confirm the validity of the COLSS calculated AZIMUTHAL POWER TILT.

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by one of the following methods:

a. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR (when COLSS is in service, and either one or both CEACs are operable); or

b. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR decreased by 13% RATED THERMAL POWER (when COLSS is in service and neither CEAC is operable); or

c. Operating within the region of acceptable operation of Figure 3.2-2 using any operable CPC channel (when COLSS is out of service and either one or both CEACs are operable); or

d. Operating within the region of acceptable operation of Figure 3.2-3 using any operable CPC channel (when COLSS is out of service and neither CEAC is operable).

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

ACTION:

With the DNBR not being maintained:

1. As indicated by COLSS calculated core power exceeding the appropriate COLSS calculated power operating limit; or

2. With COLSS out of service, operation outside the region of acceptable operation of Figure 3.2-2 or 3.2-3, as applicable;

within 15 minutes inititate corrective action to increase the DNBR to within the limits and either:

a. Restore the DNBR to within its limits within 1 hour, or

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

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR 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 DNBR, as indicated on any OPERABLE DNBR channel, is within the limit shown on Figure 3.2-2 or Figure 3.2-3.

4.2.4.3 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 DNBR.

SURVEILLANCE REQUIREMENTS (Continued)

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COLSS OUT OF SERVICE DNBR LIMIT LINE

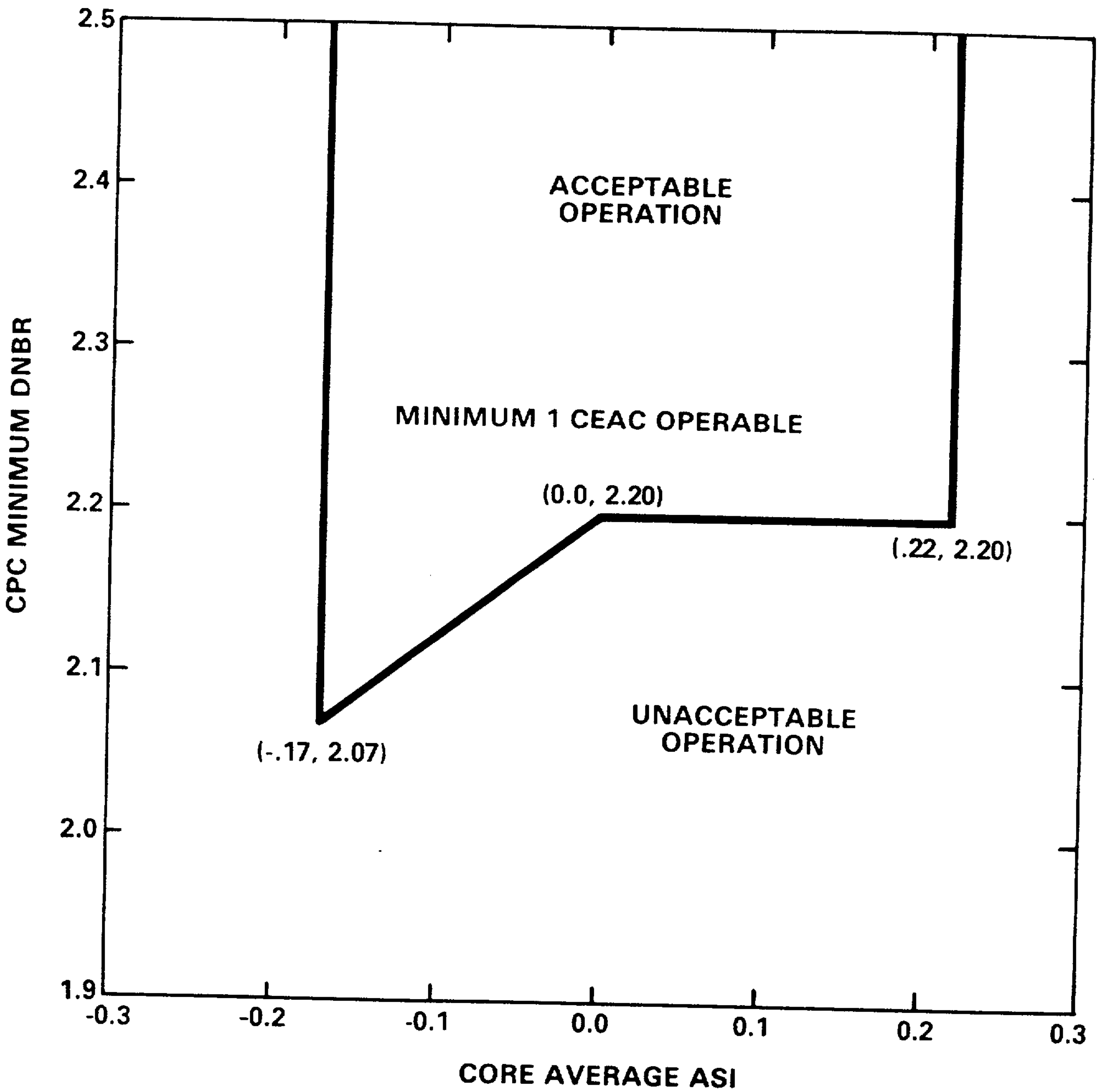

FIGURE 3.2-2

DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS
(COLSS OUT OF SERVICE, CEACs OPERABLE)

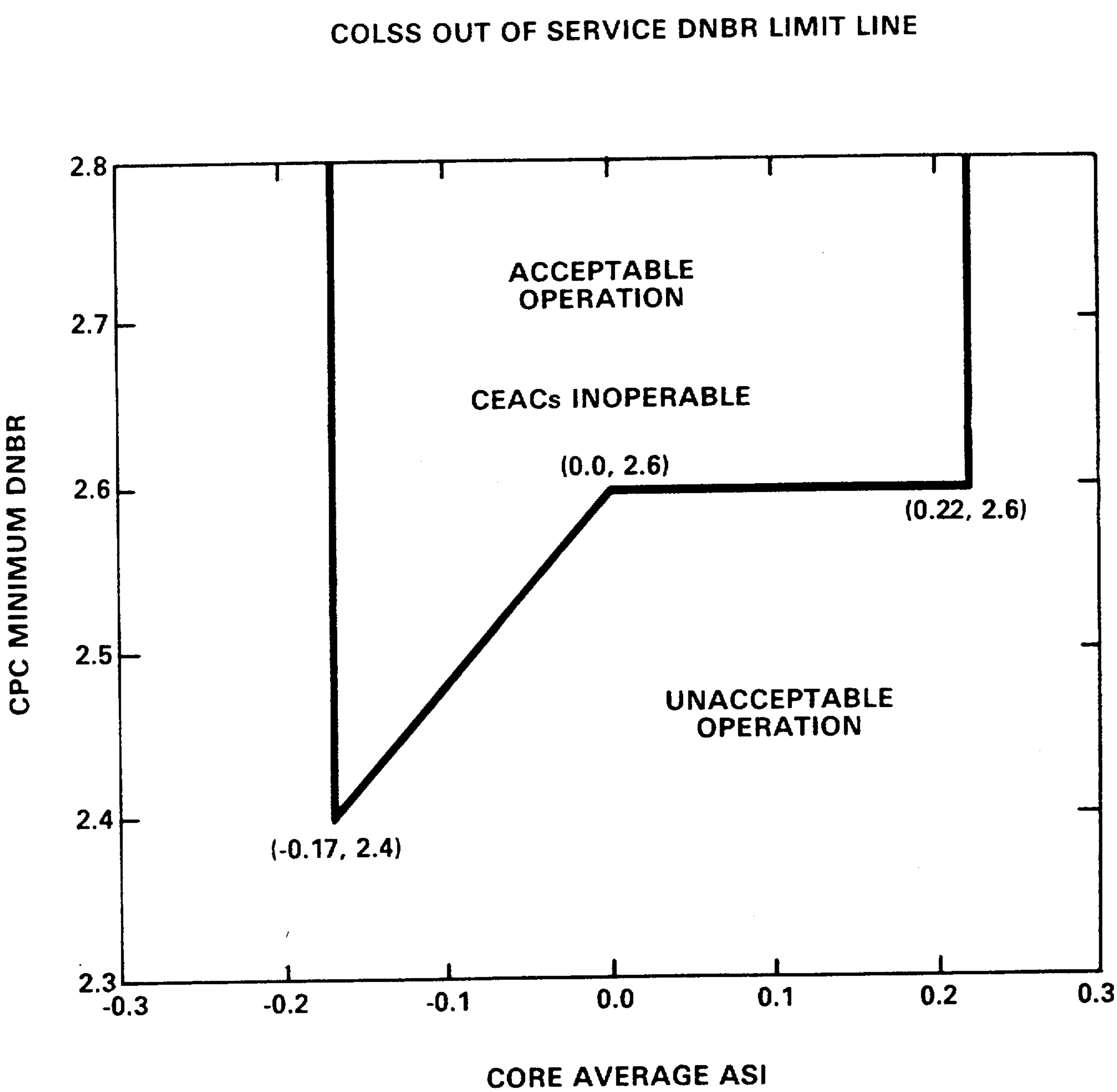

FIGURE 3.2-3

DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS
(COLSS OUT OF SERVICE, CEACs INOPERABLE)

POWER DISTRIBUTION LIMITS

3/4.2.5 RCS FLOW RATE

LIMITING CONDITION FOR OPERATION

3.2.5 The actual Reactor Coolant System total flow rate shall be greater than or equal to 148.0×10^6 lbm/h.

APPLICABILITY: MODE 1.

ACTION:

With the actual Reactor Coolant System total flow rate determined to be less than the above limit, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 The actual Reactor Coolant System total flow rate shall be determined to be greater than or equal to the above limit at least once per 12 hours.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

2. Within 4 hours:

   a) All full-length and part-length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to the inoperable status.

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA group 6 motion permitted by a) above, when the CEDMCS may be operated in either the Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full-length and part-length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 6 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 7 inches (indicated position) of all other CEAs in its group.

ACTION 7 - With three or more auto restarts of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	Not Applicable
2. Linear Power Level - High	≤ 0.40 second*
3. Logarithmic Power Level - High	≤ 0.40 second*
4. Pressurizer Pressure - High	≤ 0.90 second
5. Pressurizer Pressure - Low	≤ 0.90 second
6. Containment Pressure - High	≤ 1.70 seconds
7. Steam Generator Pressure - Low	≤ 0.90 second
8. Steam Generator Level - Low	≤ 0.90 second
9. Local Power Density - High	
a. Neutron Flux Power from Excore Neutron Detectors	≤ 0.429 second*
b. CEA Positions	≤ 0.424 second**
c. CEA Positions: CEAC Penalty Factor	≤ 0.379 second
10. DNBR - Low	
a. Neutron Flux Power from Excore Neutron Detectors	≤ 0.429 second*
b. CEA Positions	≤ 0.424 second**
c. Cold Leg Temperature	≤ 0.258 second#
d. Hot Leg Temperature	≤ 0.429 second#
e. Primary Coolant Pump Shaft Speed	≤ 0.237 second**
f. Reactor Coolant Pressure from Pressurizer	≤ 0.429 second##
g. CEA Positions: CEAC Penalty Factor	≤ 0.379 second

TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
11. Steam Generator Level - High	Not Applicable
12. Reactor Protection System Logic	Not Applicable
13. Reactor Trip Breakers	Not Applicable
14. Core Protection Calculators	Not Applicable
15. CEA Calculators	Not Applicable
16. Reactor Coolant Flow - Low	0.70 second

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

**Response time shall be measured from the time the CPC/CEAC receives an input signal until the electrical power is interrupted to the CEA drive mechanism.

#Response time shall be measured from the output of the sensor. RTD response time for all the RTDs shall be measured at least once per 18 months. The measured P_τ of the slowest RTD shall be less than or equal to 8 seconds (P_τ assumed in the safety analysis).

##Response time shall be measured from the output of the pressure transmitter. The transmitter response time shall be less than or equal to 0.70 second.

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	R and S/U(1)	1, 2, 3*, 4*, 5*
2. Linear Power Level - High	S	D(2,4),M(3,4), Q(4)	M	1, 2
3. Logarithmic Power Level - High	S	R(4)	M and S/U(1)	1, 2, 3, 4, 5
4. Pressurizer Pressure - High	S	R	M	1, 2
5. Pressurizer Pressure - Low	S	R	M	1, 2
6. Containment Pressure - High	S	R	M	1, 2
7. Steam Generator Pressure - Low	S	R	M	1, 2
8. Steam Generator Level - Low	S	R	M	1, 2
9. Local Power Density - High	S	D(2,4), R(4,5)	M, R(6)	1, 2
10. DNBR - Low	S	S(7), D(2,4), M(8), R(4,5)	M, R(6)	1, 2
11. Steam Generator Level - High	S	R	M	1, 2
12. Reactor Protection System Logic	N.A.	N.A.	M and S/U(1)	1, 2, 3*, 4*, 5*

MOVABLE CONTROL ASSEMBLIES (Continued)

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The maximum CEA drop time restriction is consistent with the assumed CEA drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 520°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

The establishment of LSSS and LCOs requires that the expected long and short-term behavior of the radial peaking factors be determined. The long term behavior relates to the variation of the steady-state radial peaking factors with core burnup and is affected by the amount of CEA insertion assumed, the portion of a burnup cycle over which such insertion is assumed, and the expected power level variation throughout the cycle. The short term behavior relates to transient perturbations to the steady-state radial peaks due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the CEAs during anticipated power reductions and load maneuvering. Analyses are performed based on the expected mode of operation of the NSSS (base loaded, or load maneuvering) and from these analyses CEA insertions are determined and a consistent set of radial peaking factors defined. The Long Term Steady State and Short Term Insertion Limits are determined based upon the assumed mode of operation used in the analyses and provide a means of preserving the assumptions on CEA insertions used. The limits specified serve to limit the behavior of the radial peaking factors within the bounds determined from analysis. The actions specified serve to limit the extent of radial xenon redistribution effects to those accommodated in the analyses. The Long and Short Term Insertion Limits of Specification 3.1.3.6 are specified for the plant which has been designed for primarily base loaded operation but which has the ability to accommodate a limited amount of load maneuvering.

The Transient Insertion Limits of Specification 3.1.3.6 and the Shutdown CEA Insertion Limits of Specification 3.1.3.5 ensure that (1) the minimum SHUTDOWN MARGIN is maintained, and (2) the potential effects of a CEA ejection accident are limited to acceptable levels. Long-term operation at the Transient Insertion Limits is not permitted since such operation could have effects on the core power distribution which could invalidate assumptions used to determine the behavior of the radial peaking factors. Insertion of Reg. Groups 5 and 6 is permitted to be essentially tip-to-tip within the limits imposed by the Transient Insertion Limit Line. This method of insertion is protected from sequence errors by the Core Protection Calculators.

The Part Length CEA Insertion Limits of Specification 3.1.3.7 ensure that adverse power shapes and rapid local power changes which affect radial peaking factors and DNB considerations do not occur as a result of a part-length CEA group covering the same axial segment of the fuel assemblies for an extended period of time during operation.

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the Local Power Density channels in the Core Protection Calculators (CPCs), provides adequate monitoring of the core power distribution and is capable of verifying that the linear heat rate does not exceed its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core power operating limit corresponding to the allowable peak linear heat rate. Reactor operation at or below this calculated power level assures that the limit of Figure 3.2-1 is not exceeded.

The COLSS calculated core power and the COLSS calculated core power operating limits based on linear heat rate are continuously monitored and displayed to the operator. A COLSS alarm is annunciated in the event that the core power exceeds the core power operating limit. This provides adequate margin to the linear heat rate operating limit for normal steady-state operation. Normal reactor power transients or equipment failures which do not require a reactor trip may result in this core power operating limit being exceeded. In the event this occurs, COLSS alarms will be annunciated. If the event which causes the COLSS limit to be exceeded results in conditions which approach the core safety limits, a reactor trip will be initiated by the Reactor Protective Instrumentation. The COLSS calculation of the linear heat rate limit includes appropriate uncertainty and penalty factors necessary to provide a 95/95 confidence level that the maximum linear heat rate calculated by COLSS is greater than or equal to that existing in the core. To ensure that the design margin to safety is maintained, the COLSS computer program includes an F_{xy} measurement uncertainty factor of 1.053, an engineering uncertainty factor of 1.03, a THERMAL POWER measurement uncertainty factor of 1.02 and appropriate penalty factors for rod bow.

Parameters required to maintain the operating limit power level based on linear heat rate, margin to DNB and total core power are also monitored by the CPCs (assuming minimum core power of 20% of RATED THERMAL POWER). The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being less accurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-1a can be maintained by utilizing a predetermined local power density margin and a total core power limit in the CPC trip channels. The above listed uncertainty and penalty factors are also included in the CPCs.

These penalty factors are determined from uncertainties associated with planar radial peaking measurements, engineering heat flux uncertainty, axial densification, software algorithm modelling, computer processing, rod bow, and core power measurement.

BASES

The additional uncertainty terms included in the CPC's for transient protection are credited in Figure 3.2-1a since this curve is intended to monitor the LCO only during steady state operation.

AZIMUTHAL POWER TILT - T_q (Continued)

P_{tilt}/P_{untilt} is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.

3/4.2.4 DNBR MARGIN

The limitation on DNBR as a function of AXIAL SHAPE INDEX represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provides adequate monitoring of the core power distribution and is capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. The COLSS calculation of core power operating limit based on the minimum DNBR limit includes appropriate penalty factors which provide a 95/95 probability/confidence level that the core power calculated by COLSS, based on the minimum DNBR limit, is conservative with respect to the actual core power limit. These penalty factors are determined from the uncertainties associated with planar radial peaking measurements, state parameter measurement, software algorithm modelling, computer processing, rod bow, and core power measurement.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2.2 or Figure 3.2-3 can be maintained by utilizing a predetermined DNBR as a function of AXIAL SHAPE INDEX and by monitoring the CPC trip channels. The above listed uncertainty and penalty factors plus those associated with startup test acceptance criteria are also included in the CPCs which assume a minimum core power of 20% of RATED THERMAL POWER. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being less accurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings.

DNBR MARGIN (Continued)

A DNBR penalty factor has been included in the COLSS and CPC DNBR calculations to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher average burnup will experience a greater magnitude of rod bow. Conversely, lower burnup assemblies will experience less rod bow. In design calculations, the penalty for each batch required to compensate for rod bow is determined from a batch's maximum average assembly burnup applied to the batch's maximum integrated planar-radial power peak. A single net penalty for COLSS and CPC is then determined from the penalties associated with each batch, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

3/4.2.5 RCS FLOW RATE

This specification is provided to ensure that the actual RCS total flow rate is maintained at or above the minimum value used in the LOCA safety analyses, and that the DNBR is maintained within the safety limit for Anticipated Operational Occurrences (AOO).

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

This specification is provided to ensure that the actual value of reactor coolant cold leg temperature is maintained within the range of values used in the safety analyses, with adjustment for instrument accuracy of ±2°F, and that the peak linear heat generation rate and the moderator temperature coefficient effects are validated.

3/4.2.7 AXIAL SHAPE INDEX

This specification is provided to ensure that the actual value of AXIAL SHAPE INDEX is maintained within the range of values used in the safety analyses, to ensure that the peak linear heat rate and DNBR remain within the safety limits for Anticipated Operational Occurrences (AOO).

3/4.2.8 PRESSURIZER PRESSURE

This specification is provided to ensure that the actual value of pressurizer pressure is maintained within the range of values used in the safety analyses. The inputs to CPCs and COLSS are the most limiting. The values are adjusted for an instrument accuracy of ± 25 psi. The sensitive events are SGTR, LOCA, FWLB and loss of condenser vacuum to initial high pressure, and MSLB to initial low pressure.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 12 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 October 1, 1986, as supplemented by letters dated October 29 and November 19, 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) revise the Core Protection Calculator (CPC) Departure from Nucleate Boiling Ratio (DNBR) setpoint; (2) revise the Core Operating Limit Supervisory System (COLSS) out-of-service DNBR limits; (3) revise the peak linear heat rate; (4) revise the reactor protective system instrumentation response times; and (5) revise the control element assembly insertion limits.

2.0 DISCUSSION

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

2.1 Core Protection Calculator DNBR Setpoint (NPF-38-43)

The proposed change would revise Technical Specification 2.1.1.1, "Safety Limits, Reactor Core-DNBR"; 2.2.1, "Safety Limits, Reactor Trip Setpoints"; and the associated Bases to these Specifications. Specifically, the DNBR reactor trip setpoint would be increased from 1.205 to 1.260. In addition, the pressurizer pressure range over which the DNBR algorithm used by the core protection calculators is valid would be changed from 1845-2355 psia to 1860-2375 psia.

2.2 Core Operating Limit Supervisory System Out-of-Service DNBR Limits (NPF-38-44)

The proposed change would revise the core operating limit supervisory system out-of-service departure from nucleate boiling ratio limit lines in Technical Specification 3.2.4, "Power Distribution Limits, DNBR Margin" and the associated Bases. The proposed change would also delete the rod bow penalty given as a function of burnup in Surveillance Requirement 4.2.4.4 and currently imposed on the core protection calculator calculated DNBR.

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2.3 Peak Linear Heat Rate (NPF-38-46)

The proposed change would modify Technical Specification 3.2.1, "Power Distribution Limits, Linear Heat Rate" (LHR) and the associated Bases. Adherence to the LHR limits required by this Specification ensures that the clad surface temperature will remain below 2200°F in the event of a Loss of Coolant Accident (LOCA) and also ensures that the consequences of any Anticipated Operational Occurrence (AOO) will be acceptable.

2.4 Reactor Protective Systems Instrumentation Response Times (NPF-38-47)

The proposed change would revise the protective instrument response times given in Table 3.3-2 of Technical Specification 3.3.1, "Reactor Protective Instrumentation Response Times".

2.5 Control Element Assembly Insertion Limits (NPF-38-48)

The proposed change would revise Figure 3.1-2 of Technical Specification 3.1.3.6, "Regulating CEA Insertion Limits," and Bases B 3/4.1.3.

3.0 EVALUATION

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

3.1 Core Protection Calculator DNBR Setpoint (NPF-38-43)

For Cycle 2, the uncertainties associated with fuel manufacturing variations, as well as certain thermal-hydraulic uncertainties, will be combined using the methodology associated with the Statistical Combination of Uncertainties (SCU). These methods have been previously used on other Combustion Engineering (CE) plants and have been approved by the NRC.

Use of the SCU methodology results in an increase in the minimum allowable value of the DNBR safety limit and the reactor trip setpoint for DNBR from a value of 1.205 to a Cycle 2 value of 1.260. This new value ensures, with a 95% confidence level, that if the hot channel in the core reaches the DNBR safety limit, there is still a 95% probability that DNBR has not occurred. This is the same probability/confidence level that applies to the 1.205 DNBR value that was used during Cycle 1. Thus, although some uncertainties have been removed from the actual CPC calculation of core heat flux (and corresponding DNBR), they have been factored into the determination of the reactor trip setpoint for low DNBR as well as the associated DNBR Safety Limit. The proposed change is, therefore, acceptable.

The change request would also modify the pressurizer pressure range over which the DNBR algorithm used by the CPCs is valid. The new range would be from 1860 to 2375 psia and is only slightly different from the range used during Cycle 1 (1845 to 2355 psia). The high pressurizer pressure limit in the CPCs is not credited in any of the transient analyses but is merely the limit of applicability for the DNBR algorithm, including appropriate uncertainties.

The change is being requested in order to make the Waterford 3 parameter range consistent with ranges for other CE plants that utilize the CPC digital protection system. The staff, therefore, finds the proposed change acceptable.

In response to the staff's recommendations, the licensee has revised Bases 2.1.1 to reference the use of SCU and has updated the CPC-related references.

3.2 Core Operating Limit Supervisory System Out-of-Service DNBR Limits (NPF-38-44)

COLSS is normally used to monitor DNBR margin. When at least one control element assembly calculator (CEAC) is operable, Specification 3.2.4a provides enough margin to DNB to accommodate the limiting AOO without failing the fuel. This has been reverified by the Cycle 2 safety analyses submitted by letter dated October 1, 1986. When neither CEAC is operable, the CPCs lack the CEA position information necessary to ensure a reactor trip when necessary. In this case, the COLSS calculated core power must be reduced to ensure that the limiting AOO will not result in fuel failure. Currently, Specification 3.2.4b requires that the COLSS calculated core power be maintained at 19% below the COLSS calculated power operating limit to compensate for this potential error in the CPC calculation. This was based on the Cycle 1 safety analyses. As a result of the reevaluation of the limiting AOOs for Cycle 2, the proposed revision would decrease this requirement adjustment to 13%. This new value was derived using approved methods (see Cycle 2 Safety Analyses referenced above) and merely reflects the changes in core parameters in Cycle 2 compared to Cycle 1. It is, therefore, acceptable.

Whenever COLSS is out-of-service, the CPCs are used to perform the same monitoring function. However, the extra conservatisms built into the CPC for transient protection are not all required when the CPCs are being used for monitoring. In order not to affect the CPC transient protection, these conservatisms are not taken from the CPC but are credited in the COLSS out-of-service limits given in Figures 3.2-2 and 3.2-3. A reevaluation of the limiting AOOs performed as part of the Cycle 2 safety analyses has verified that by maintaining the margin shown in the proposed revised Figures 3.2-2 and 3.2-3, sufficient margin exists to ensure that the limiting Cycle 2 AOO will not result in fuel failure. The proposed revisions to these figures are, therefore, acceptable.

Currently, Surveillance Requirement 4.2.4.4 imposes a rod bow penalty as a function of fuel burnup on the CPC-calculated DNBR. However, for Cycle 2, the effects of fuel rod bowing on DNBR margin has been incorporated directly into the DNBR safety limit by use of SCU. Therefore, deletion of the rod bow penalty table in Specification 4.2.4.4 is acceptable.

3.3 Peak Linear Heat Rate (NPF-38-46)

When the core protection calculators are used to monitor the core power distribution, the LHR must be maintained below the maximum allowable value shown in Figure 3.2-1a. This Figure takes credit for the large conservatisms built into the CPCs for transient protection which are not all required when the CPCs are being used for monitoring. Figure 3.2-1a would be modified as a result of changes in the Cycle 2 core parameters. When the CEACs are inoperable, a penalty is applied automatically in the CPC calculation of linear power density which assures a conservative LHR calculation for monitoring as well as protection. However, this penalty is no longer large enough to maintain adequate LHR margin for Cycle 2. Therefore, Specification 3.2.1c, which allows the CPCs to automatically monitor the LHR, has been deleted. This is acceptable. Also, there is no additional penalty when the CEACs are inoperable since the LOCA analyses, which are used to determine the LHR limit, are not impacted by the operability of the CEACs. Therefore, the same LHR limit applies for both operable and inoperable CEACs. The modification of Specification 3.2.1b and the associated Figure 3.2-1a, such that they now apply whether or not any CEACs are operable, is, therefore, acceptable.

3.4 Reactor Protective System Instrumentation Response Times (NPF-38-47)

The current values of the response times associated with the CPC low DNBR trip and high Local Power Density (LPD) trip are conservatively long. These long delay times were assumed in the Cycle 1 safety analyses since plant specific measurements were not available at the time. Because of less favorable core parameters that result from an 18-month fuel cycle, several Cycle 2 anticipated operational occurrences and accidents require the CPCs to respond with shorter (faster) response times than those given in the current Technical Specification Table 3.3-2. Specifically, these events are: (1) the loss of load to one steam generator, which is dependent upon the maximum response time of the cold leg temperature instrumentation; (2) the control element assembly withdrawal and steam line break, which are dependent upon the maximum response times of the neutron flux power input from the ex-core detectors; and (3) the loss of flow, which is dependent upon the maximum response time of the reactor coolant pump shaft speed signals in the CPCs. These events were reanalyzed using response times consistent with the proposed change to Table 3.3-2 for the Cycle 2 safety analyses. Since the results show that these events are either bounded by the Cycle 1 safety analysis or they are within the acceptance criteria specified in Section 15 of the NRC Standard Review Plan, the proposed changes to the neutron flux power, cold leg temperature, and coolant pump shaft speed response times, which have been justified by actual measured values, are acceptable.

Other instrument response times associated with the CPC reactor trips are proposed to be shortened in Table 3.3-2. Although these shorter response times are not currently required in order to meet NRC acceptance criteria in the event of an AOO or accident, they are being proposed in anticipation of potential future cycle requirements. These values are acceptable since they have also been justified by the Cycle 2 safety analyses and are consistent with the actual response times measured prior to and during Cycle 1.

The increase in the hot and cold leg resistance temperature detector (RTD) response times from 6 seconds to 8 seconds has been proposed in order to account for the potential degradation of the RTD response times that has been observed at other plants. Since this change is consistent with the value used in the Cycle 2 safety analyses, which resulted in all applicable NRC acceptance criteria being satisfied, the proposal is acceptable.

3.5 Control Element Assembly Insertion Limits (NPF-38-48)

Technical Specification 3.1.3.6 imposes limits on the allowable position of the regulating CEA groups and on the allowable time interval within a given CEA position range. The CEA insertion limits are given by Figure 3.1-2. The greater the CEA insertion, the larger the positive reactivity addition is if the CEA were to be withdrawn. Therefore, these limits provide protection against events such as the CEA ejection accident by limiting how far CEAs may be inserted into the core. Figure 3.1-2 also limits CEA insertion to a level that results in acceptable pin power peaking factors and maintains adequate shutdown margin.

The proposed change would increase the maximum allowable insertion of Group 6 by 7.5 inches during long term steady state operation. In addition, it would reduce the allowable insertion during transient operation for Groups 5 and 6 above 20% power. Between 0 and 20% power, insertion of Group 4 would be limited to 6 inches. These revised limits were used to calculate radial power distributions for rodded configurations and net available scram worths as well as other safety related data such as ejected CEA and dropped CEA worths for Cycle 2. Since the calculations used to obtain the revised limits were performed with accepted methods and the appropriate values were used in the safety analyses, the proposed changes are acceptable.

During low power operation, Axial Shape Index control is proposed by maneuvering Groups 5 and 6 essentially tip-to-tip within the limits established by the Figure 3.1-2. Tip-to-tip operation of Groups 5 and 6 means that these groups can be inserted into the core by essentially the same amount. An additional statement is included in the Bases to clarify that this operation is acceptable. Since the CEA groups will always be limited to the transient insertion limits shown in Figure 3.1-2 and, in practice, Group 5 will be slightly more withdrawn than Group 6 to avoid a potential core protection calculator CEA sequence error reactor trip, the proposed tip-to-tip operation is acceptable.

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 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.

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 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 Contributor: L. Kopp

Dated: January 16, 1987