May 30, 1986

Docket No.: 50-382

Mr. G. W. Muench
Acting Director - Nuclear Operations
Louisiana Power and Light Company
317 Baronne Street, Mail Unit 17
New Orleans, Louisiana 70160

Dear Mr. Muench:

Subject: Issuance of Amendment No. 5 to Facility Operating License No. NPF-38 for Waterford 3

The Commission has issued the enclosed Amendment No. 5 to Facility Operating License No. NPF-38 for the Waterford Steam Electric Station, Unit 3. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated December 2, 1985, as supplemented by letter dated April 3, 1986.

The amendment revises the Appendix A Technical Specifications by deleting the CPC Type I and Type II addressable constants; incorporating the results of the Waterford 3 COLSS out-of-service analysis; revising the requirements for low temperature overpressure protection to allow hydrostatic tests in Mode 4 and allowing the measurement of moderator temperature coefficient to be taken at any power level greater then 15% prior to reaching 40 EFPD core burnup rather than within 7 EFPD of reaching 40 EFPD core burnup.

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. 5 to NPF-38
2. Safety Evaluation

cc: See next page

PBD7	PBD7	OELD	~~PEICSB~~ RSB	AD:PWR-B	DIR:PBD7
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5/20/86	5/21/86	5/27/86	5/30/86	5/30/86	5/30/86

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Waterford 3

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ISSUANCE OF AMENDMENT NO. 5 TO FACILITY OPERATING
LICENSE NP. NPF-38 FOR WATERFORD 3

DISTRIBUTION
Docket File 50-382
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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. 5
License No. NPF-38

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

   A. The application for amendment, dated December 2, 1985, as supplemented by letter dated April 3, 1986, by Louisiana Power and Light Company (licensee), complies with standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of 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) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 5, 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: May 30, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 5

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
III	IV
2-2	2-1
3/4 1-4	3/4 1-3
3/4 2-1	-
3/4 2-2	-
3/4 2-6	3/4 2-5
3/4 2-8	3/4 2-7
3/4 2-9	3/4 2-10
3/4 2-12	3/4 2-11
3/4 3-6	3/4 3-5
3/4 3-12	3/4 3-11
3/4 4-34	3/4 4-33
B 3/4 2-1	-
B 3/4 2-1a	-
-	B 3/4 2-2
B 3/4 2-3	B 3/4 2-4
6-9	6-10
6-14	6-13

Delete the following pages:

2-5
2-6
2-7
2-8
B 2-7

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

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 1.20.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the DNBR of the reactor has decreased to less than 1.20, 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.

SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 2.0% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 2.0% delta k/k, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 2.0% delta k/k:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. At least once per 24 hours by consideration of the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration,and
6. Samarium concentration.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.2×10^{-4} delta k/k/°F whenever THERMAL POWER is $\leq$ 70% RATED THERMAL POWER, and

b. Less positive than 0.0×10^{-4} delta k/k/°F whenever THERMAL POWER is > 70% RATED THERMAL POWER, and

c. Less negative than -2.5×10^{-4} delta k/k/°F at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.3.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At greater than 15% of RATED THERMAL POWER, prior to reaching 40 EEPD core burnup.

c. At any THERMAL POWER, within 7 EFPD of reaching two-thirds of expected core burnup.

*With K_{eff} greater than or equal to 1.0.

#See Special Test Exception 3.10.2.

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 and either one or both CEACs is operable).

c. Automatically by CPC (when COLSS is out of service and neither CEAC is operable).

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 in 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 channels, is within the limits shown on Figure 3.2-1.

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.

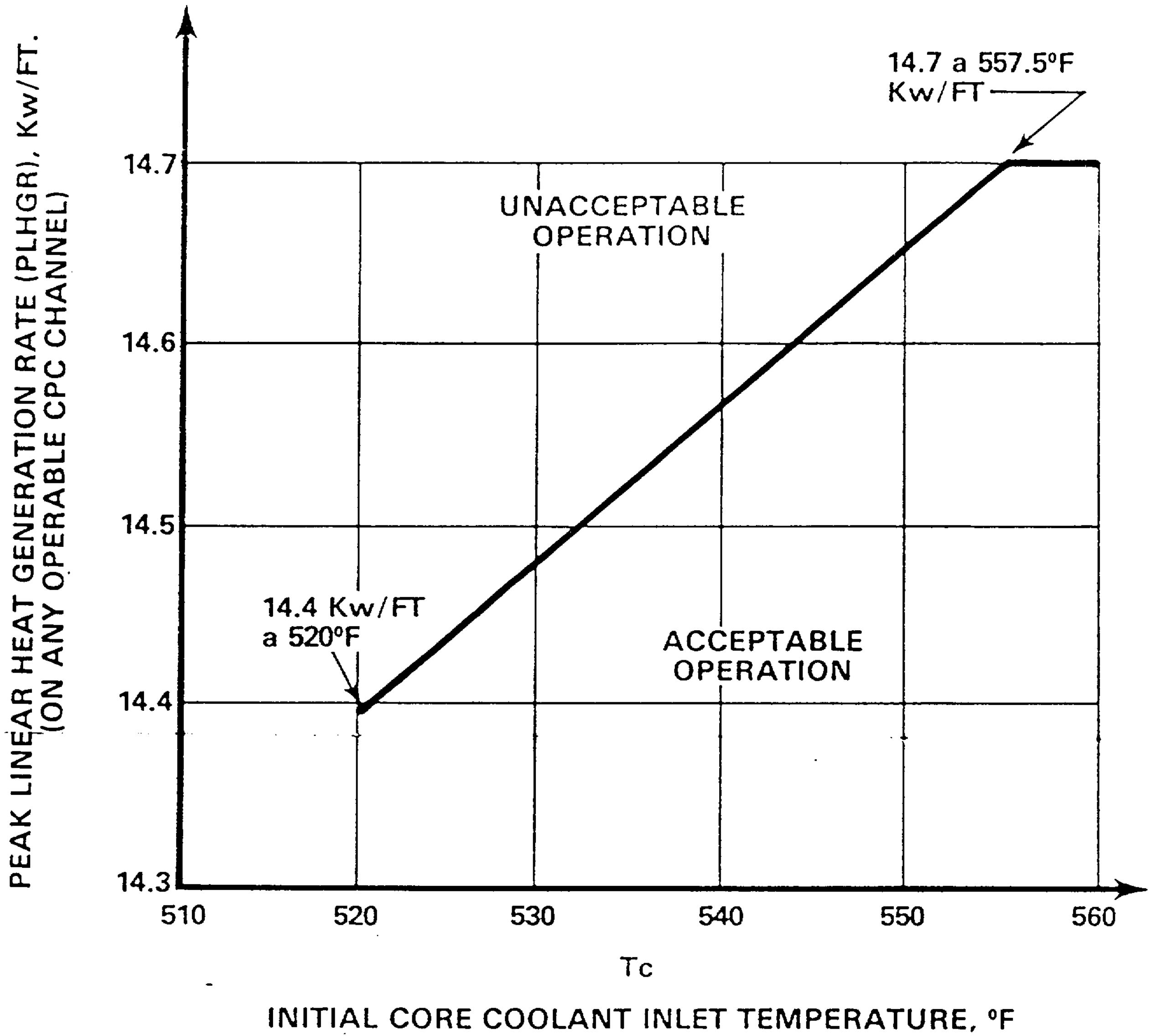

FIGURE 3.2-1a

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

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 19% 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 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

4.2.4.1 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 channels, is within the limit shown on 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)

4.2.4.4 The following DNBR penalty factors shall be verified to be included in the COLSS and CPC DNBR calculations at least once per 31 days:

BURNUP $\left(\frac{GWD}{MTU}\right)$	DNBR PENALTY (%)
0-10.0	0.50
10.0-20.0	1.00
20.0-30.0	2.00
30.0-40.0	3.50
40.0-50.0	5.50

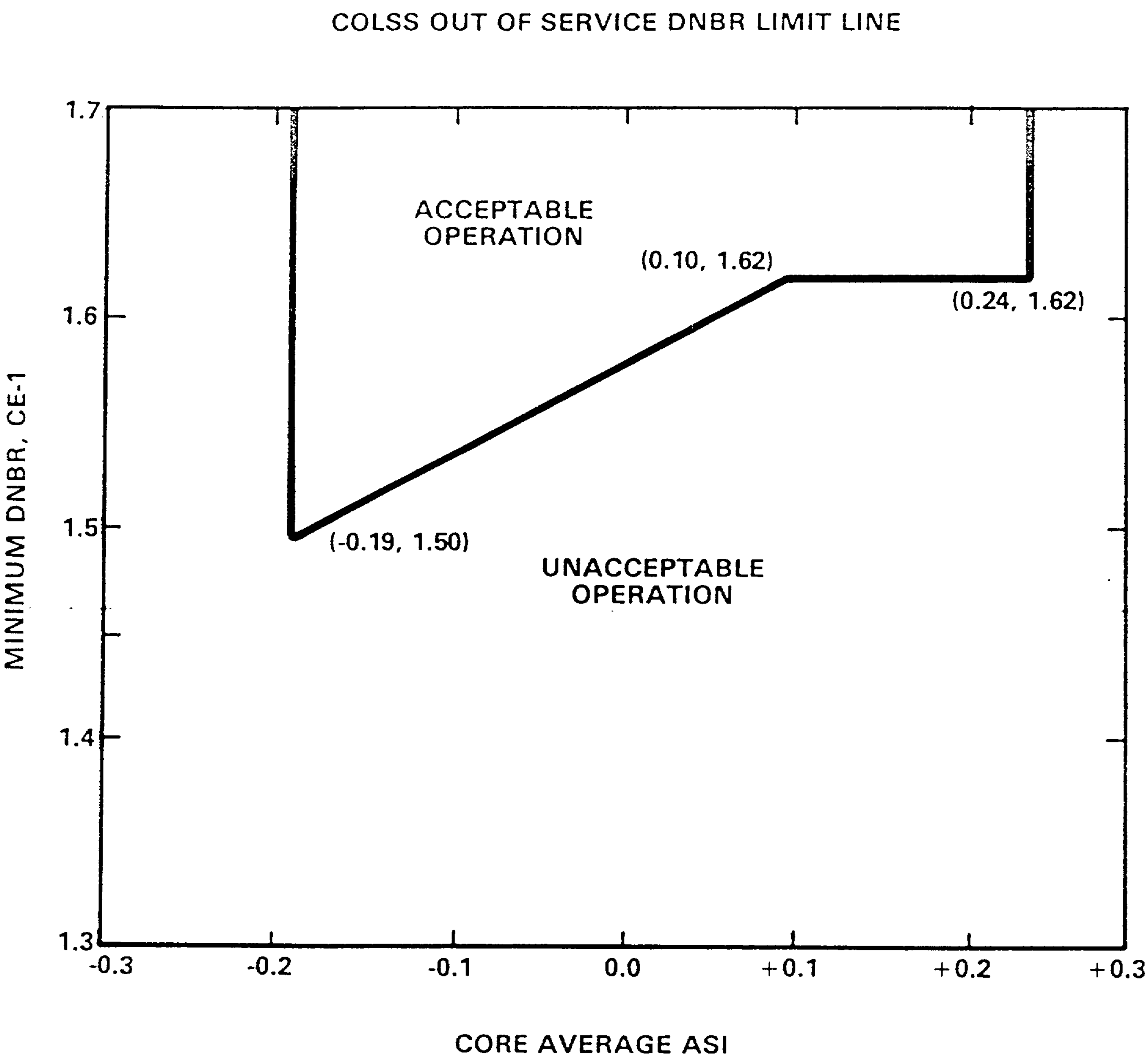

FIGURE 3.2-2

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

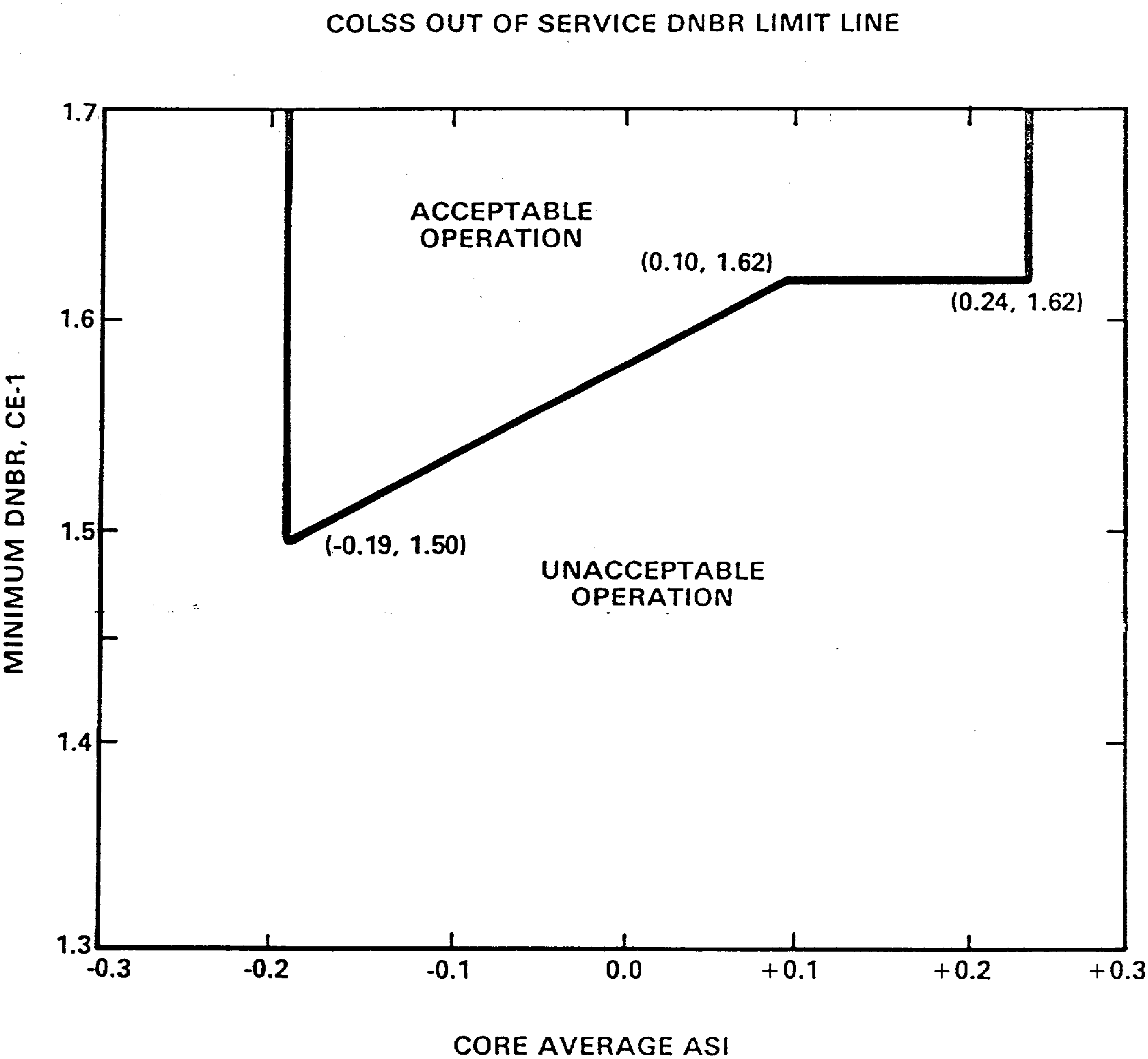

FIGURE 3.2-3

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

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.

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.2.6 The reactor coolant cold leg temperature (T_c) shall be maintained between 544°F and 558°F.*

APPLICABILITY: MODE 1 above 30% of RATED THERMAL POWER.

ACTION:

With the reactor coolant cold leg temperature exceeding its limit, restore the temperature to within its limit within 2 hours or reduce THERMAL POWER to less than 30% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.6 The reactor coolant cold leg temperature shall be determined to be within its limit at least once per 12 hours.

*Following a reactor power cutback in which (1) Regulating Groups 5 and/or 6 are dropped or (2) Regulating Groups 5 and/or 6 are dropped and the remaining Regulating Groups (Groups 1, 2, 3, and 4) are sequentially inserted, the upper limit on T_c may increase to 568°F for up to 30 minutes.

3/4.2.7 AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a. COLSS OPERABLE
$-0.23 \leq \text{ASI} \leq +0.50$

b. COLSS OUT OF SERVICE (CPC)
$-0.19 \leq \text{ASI} \leq +0.24$

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

ACTION:

With the AXIAL SHAPE INDEX outside its above limits, restore the AXIAL SHAPE INDEX to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.7 The AXIAL SHAPE INDEX shall be determined to be within its limit at least once per 12 hours using the COLSS or any OPERABLE Core Protection Calculator channel.

*See Special Test Exception 3.10.2.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Linear Power (Subchannel or Linear)	Linear Power Level - High Local Power Density - High DNBR - Low
2. Pressurizer Pressure - High	Pressurizer Pressure - High Local Power Density - High DNBR - Low
3. Containment Pressure - High	Containment Pressure - High (RPS) Containment Pressure - High (ESF)
4. Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS 1 and 2)
5. Steam Generator Level	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)
6. Core Protection Calculator	Local Power Density - High DNBR - Low

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other channel in the tripped condition within 1 hour, and

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Linear Power (Subchannel or Linear)	Linear Power Level - High Local Power Density - High DNBR - Low

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

2.	Pressurizer Pressure - High	Pressurizer Pressure - High Local Power Density - High DNBR - Low
3.	Containment Pressure - (RPS) High	Containment Pressure - High Containment Pressure - High (ESF)
4.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS 1 and 2)
5.	Steam Generator Level	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)
6.	Core Protection Calculator	Local Power Density - High DNBR - Low

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of channels OPERABLE one less those required by the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the reactor trip breakers of the inoperable channel are placed in the tripped condition within 1 hour; otherwise, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.

ACTION 6 - a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 7 inches (indicated position) of all other CEAs in its group.

b. With both CEACs inoperable, operation may continue provided that:

1. Within 1 hour the DNBR margin required by Specification 3.2.4b (COLSS in service) or 3.2.4d (COLSS out of service) is satisfied and the Reactor Power Cutback System is disabled, and

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Reactor Trip Breakers	N.A.	N.A.	M(10), S/U(1)	1, 2, 3*, 4*, 5*
14. Core Protection Calculators	S	D(2,4),R(4,5)	M(9),R(6)	1, 2
15. CEA Calculators	S	R	M,R(6)	1, 2
16. Reactor Coolant Flow - Low	S	R	M	1, 2

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

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

(1) Each startup or when required with the reactor trip breakers closed and the CEA drive system capable of rod withdrawal, if not performed in the previous 7 days.

(2) Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER: adjust the Linear Power Level signals and the CPC addressable constant multipliers to make the CPC ΔT power and CPC nuclear power calculations agree with the calorimetric calculation if absolute difference is greater than 2%. During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) Above 15% of RATED THERMAL POWER, verify that the linear power sub-channel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

(6) This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation or by calorimetric calculations and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty is included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by calorimetric calculations.

(9) The monthly CHANNEL FUNCTIONAL TEST shall include verification that the correct values of addressable constants are installed in each OPERABLE CPC.

(10) At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage trip function and the shunt trip function.

REACTOR COOLANT SYSTEM

PRESSURIZER HEATUP/COOLDOWN

LIMITING CONDITION FOR OPERATION

3.4.8.2 The pressurizer shall be limited to:

a. A maximum heatup rate of 200°F per hour,

b. A maximum cooldown rate of 200°F per hour, and

c. A maximum spray nozzle usage factor of 0.65.

APPLICABILITY: At all times.

ACTION:

a. With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

b. With the spray nozzle usage factor > 0.65, comply with requirements of Table 5.7-1.

SURVEILLANCE REQUIREMENTS

4.4.8.2.1 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

4.4.8.2.2 The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

4.4.8.2.3 Each spray cycle and the corresponding ΔT (water temperature differential) shall be recorded whenever main spray is initiated with a ΔT (water temperature differential) of > 130°F and whenever auxiliary spray is initiated with a ΔT (water temperature differential) of > 140°F.

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.8.3 Low temperature overpressure protection shall be provided by:

a. At least one of the following overpressure protection systems being OPERABLE:

   1. Both OPERABLE Shutdown Cooling (SDC) System suction line relief valves (SI-406A and SI-406B) each with a lift setting of less than or equal to 430 psia aligned to the Reactor Coolant System, or,

   2. The Reactor Coolant System depressurized with an RCS vent of greater than or equal to 5.6 square inches.

b. Establishing less than 100°F ΔT between RCS and steam generator temperature or ensuring the pressurizer water volume is less than 900 cubic feet (62.5%), prior to starting any reactor coolant pump.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 285°F#, MODE 5, and MODE 6 with the reactor vessel head on.

ACTION:

a. With one Shutdown Cooling System suction line relief valve inoperable, restore the inoperable valve to OPERABLE status within 7 days, or be in at least COLD SHUTDOWN and depressurize and vent the RCS within the next 8 hours.

b. With no Shutdown Cooling System suction line relief valves OPERABLE and capable of providing Reactor Coolant System overpressure protection, either:

   1. Restore at least one Shutdown Cooling System suction relief valve to OPERABLE status within 1 hour, or

   2. Be in at least COLD SHUTDOWN and depressurize and vent the RCS within the next 8 hours.

c. In the event either the Shutdown Cooling System suction relief valves(s) or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the Shutdown Cooling System suction relief valve(s) or RCS vent(s) on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

#260°F during inservice leak and hydrostatic testing with Reactor Coolant System temperature changes restricted in accordance with Specification 3.4.8.1g.

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 uncertainty and penalty factors for flux peaking augmentation and 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.

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.

In addition, when COLSS is out of service and both CEAC's are inoperable, the 57% penalty applied automatically in CPC can be credited in the CPC linear heat rate calculation since it is required only for transient protection. In this case, Figure 3.2-1 is automatically maintained by the CPC trip limit.

3/4.2.2 PLANAR RADIAL PEAKING FACTORS

Limiting the values of the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPCs to values equal to or greater than the measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) provides assurance that the limits calculated by COLSS and the CPCs remain valid. Data from the incore detectors are used for determining the measured PLANAR RADIAL PEAKING FACTORS. A minimum core power at 20% of RATED THERMAL POWER is assumed in determining the PLANAR RADIAL PEAKING FACTORS. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The periodic Surveillance Requirements for determining the measured PLANAR RADIAL PEAKING FACTORS provide assurance that the PLANAR RADIAL PEAKING FACTORS used in COLSS and the CPCs remain valid throughout the fuel cycle. Determining the measured PLANAR RADIAL PEAKING FACTORS after each fuel loading prior to exceeding 70% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.3 AZIMUTHAL POWER TILT - T_q

The limitations on the AZIMUTHAL POWER TILT are provided to ensure that design safety margins are maintained. An AZIMUTHAL POWER TILT greater than 0.10 is not expected and if it should occur, operation is restricted to only those conditions required to identify the cause of the tilt. The tilt is normally calculated by COLSS. A minimum core power of 20% of RATED THERMAL POWER is assumed by the CPCs in its input to COLSS for calculation of AZIMUTHAL POWER TILT. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The Surveillance Requirements specified when COLSS is out of service provide an acceptable means of detecting the presence of a steady-state tilt. It is necessary to explicitly account for power asymmetries because the radial peaking factors used in the core power distribution calculations are based on an untilted power distribution.

AZIMUTHAL POWER TILT is measured by assuming that the ratio of the power at any core location in the presence of a tilt to the untilted power at the location is of the form:

$$P_{tilt}/P_{untilt} = 1 + T_q\, g \cos(\Theta - \Theta_o)$$

where:

T_q is the peak fractional tilt amplitude at the core periphery

g is the radial normalizing factor

Θ is the azimuthal core location

Θ_o is the azimuthal core location of maximum tilt

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, of which the loss of flow transient is the most limiting. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

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, engineering heat flux, 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 3.2-3 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)

The DNBR penalty factors listed in Specification 4.2.4.4 are penalties used 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. 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.

RESPONSIBILITIES (Continued)

g. Review of unit operations to detect potential hazards to nuclear safety.

h. Performance of special reviews, investigations, or analyses and reports thereon as requested by the Plant Manager-Nuclear or the Safety Review Committee.

i. Review of the Security Plan and implementing procedures and submittal of recommended changes to the Safety Review Committee.

j. Review of the Emergency Plan and implementing procedures and submittal of recommended changes to the Safety Review Committee.

k. Review and documentation of judgment concerning prolonged operation in bypass, channel trip, and/or repair of defective protection channels of process variables placed in bypass since the last PORC meeting.

l. Review of proposed modifications to the CPC addressable constants based on information obtained through the Plant Computer-CPC data link.

m. Review of any accidental, unplanned or uncontrolled radioactive release including reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President, Nuclear Operations and to the Safety Review Committee.

n. Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL, and major changes to radwaste treatment systems.

AUTHORITY

6.5.1.7 The PORC shall:

a. Recommend in writing to the Plant Manager-Nuclear, prior to implementation except as provided in Specification 6.8.3, approval or disapproval of items considered under Specification 6.5.1.6a. through d. and l.

b. Render determinations in writing, prior to implementation except as provided in Specification 6.8.3, with regard to whether or not each item considered under Specification 6.5.1.6a. through e. constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the Senior Vice President-Nuclear Operations and the Safety Review Committee of disagreements between the PORC and the Plant Manager-Nuclear; however, the Plant Manager-Nuclear shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS

6.5.1.8 The PORC shall maintain written minutes of each PORC meeting that, at a minimum, document the results of all PORC activities performed under the responsibility and authority provisions of these technical specifications. Copies shall be provided to the Senior Vice President-Nuclear Operations and the Safety Review Committee.

6.5.2 SAFETY REVIEW COMMITTEE (SRC)

FUNCTION

6.5.2.1 The SRC shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,

b. Nuclear engineering,

c. Chemistry and radiochemistry,

d. Metallurgy,

e. Instrumentation and control,

f. Radiological safety,

g. Mechanical and electrical engineering and

h. Quality assurance practices.

COMPOSITION

6.5.2.2 The SRC shall be composed of at least five members, including the Chairman. Members of the SRC may be from within the LP&L organization or from organizations external to LP&L.

The qualifications of members selected for the SRC shall be in accordance with Section 4.7 of ANSI/ANS 3.1-1978.

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the SRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in SRC activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the SRC Chairman to provide expert advice to the SRC.

AUDITS (Continued)

p. The performance of activities required by the Quality Assurance Program to meet the provisions of Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975 at least once per 12 months.

AUTHORITY

6.5.2.9 The SRC shall report to and advise the Senior Vice President- Nuclear Operations on those areas of responsibility specified in Specifications 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of SRC activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each SRC meeting shall be prepared, approved, and forwarded to the Senior Vice President-Nuclear Operations within 14 days following each meeting.

b. Reports of reviews encompassed by Specification 6.5.2.7 shall be prepared, approved, and forwarded to the Senior Vice President-Nuclear Operations within 14 days following completion of the review.

c. Audit reports encompassed by Specification 6.5.2.8 shall be forwarded to the Senior Vice President-Nuclear Operations and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PORC and the results of this review shall be submitted to the SRC and the Senior Vice President-Nuclear Operations.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

SAFETY LIMIT VIOLATION (Continued)

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Senior Vice President-Nuclear Operations and the SRC shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the SRC and the Senior Vice President-Nuclear Operations within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978 and those required for implementing the requirements of NUREG-0737.

b. Refueling operations.

c. Surveillance and test activities of safety-related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. Modification of Core Protection Calculator (CPC) Addressable Constants, including independent verification of modified constants.

NOTES:

(1) Modification to the CPC addressable constants based on information obtained through the Plant Computer - CPC data link shall not be made without prior approval of the PORC.

(2) Modifications to the CPC software (including algorithm changes and changes in fuel cycle specific data) shall be performed in accordance with the most recent version of CEN-39(A)-P, "CPC Protection Algorithm Software Change Procedure," that has been determined to be applicable to the facility. Additions or deletions to CPC Addressable Constants or changes to Addressable Constant software limits values shall not be implemented without prior NRC approval.

h. Administrative procedures implementing the overtime guidelines of Specification 6.2.2f., including provisions for documentation of deviations.

i. PROCESS CONTROL PROGRAM implementation.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 5 TO FACILITY OPERATING LICENSE NO. NPF-38

LOUISIANA POWER AND LIGHT COMPANY

WATERFORD STEAM ELECTRIC STATION, UNIT 3

DOCKET NO. 50-382

1.0 INTRODUCTION

By letter dated December 2, 1985, as supplemented by letter dated April 3, 1986, Louisiana Power and Light Company (the licensee), requested changes to the Technical Specifications (Appendix A to Facility Operating License NPF-38) for the Waterford Steam Electric Station, Unit 3. The proposed changes would: (1) delete the Core Protection Calculator (CPC) Type I and Type II addressable constants; (2) incorporate the results of the Waterford Core Operating Limit Supervising System (COLSS) out-of-service analysis; (3) revise the requirements for low temperature overpressure protection (LTOP) to allow hydrostatic tests in Mode 4; and (4) allow the measurement of moderator temperature coefficient (MTC) following each refueling outage to be taken at any power level greater than 15% prior to reaching 40 effective full-power days (EFPD) core burnup rather than within 7 EFPD of reaching 40 EFPD core burnup.

2.0 DISCUSSION

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

Core Protection Calculator Addressable Constants

The proposed change removes the Type I and Type II Core Protection Calculator addressable constants from the Technical Specifications. The proposed change deletes Technical Specification 2.2.2, "Core Protection Calculator Addressable Constants"; deletes Table 2.2-2, which provided a listing of the CPC Type I and Type II addressable constants; and deletes Bases 2.2.2. The proposed change also revises the appropriate page of the Index, deletes the reference to Specification 2.2.2 from Notation (9) of Table 3.4-1 and deletes the Note in Administrative Control 6.8.1.g.

Core Operating Limits Supervisory System

The proposed change modifies the Technical Specifications to incorporate the results of the Waterford 3 COLSS out-of-service analysis. The change revises Technical Specifications 3.2.1, "Linear Heat Rate", 3.2.4, "DNBR Margin", 3.2.7, "Axial Shape Index", and the associated Bases. The change also adds Figure 3.2-1a "Allowable Peak Linear Heat Rate vs Tc for COLSS Out-of-Service"; deletes Figures 3.2-2 "DNBR Margin Operating Limit Based on COLSS" and replaces it with "DNBR Margin Operating Limit Based on CPC's (COLSS Out-of-Service, CEACs Operable)"; revises Figures 3.2-3 "DNBR Margin Operating Limit Based on CPC's (COLSS Out-of-Service, CEACs Inoperable)"; and revises Action 6.b.1 of Table 3.3-1.

Low Temperature Overpressure Protection

The proposed change revises the applicability of Technical Specification 3.4.8.3, "Overpressure Protection System", in Mode 4 to allow a lower reactor coolant system temperature during inservice leak and hydrostatic testing without imposing the requirements of low temperature overpressure protection.

Moderator Temperature Coefficient

The proposed change modifies Surveillance Requirement 4.1.1.3.2 of Technical Specification 3.1.1.3, Moderator Temperature Coefficient. The Requirement specifies the required MTC measurement frequency, including the requirement that MTC be measured within 7 effective full power days (EFPD) of reaching 40 EFPD core burnup. The proposed change allows this measurement to be taken at a thermal power level greater than 15% at any time prior to reaching 40 EFPD core burnup.

3.0 EVALUATION

Core Protection Calculator Addressable Constants

As a method to avoid gross errors in operator entry of an addressable constant, the Core Protection Calculators (CPC) software was designed with automatic acceptable input checks against range limits that are specified by the CPC functional design requirements. In addition, CPC surveillance requirements (Table 4.3-1) require that the monthly channel functional test shall include verification that the correct values of addressable constants are installed in each operable CPC. Modifications to the addressable constants are accomplished through strict administrative procedures as required by Technical Specification 6.8.1(g). CPC software changes involving additions or deletions to addressable constants as well as changes to software limit values are made and tested under NRC-approved software change procedures.

Based on these requirements, and on the fact that the NRC has previously approved the deletion of CPC addressable constants from the Palo Verde Unit 2, San Onofre Unit 2 and 3, and Arkansas Unit 2 Technical Specifications, the staff finds the proposed removal of the addressable constants and related administrative requirements from the Waterford 3 Technical Specifications acceptable provided that the following alternative measures are adopted:

(1) sufficient margin is maintained in CPC constants, such as azimuthal tilt allowance, to preclude excessive operator interaction with the CPCs during reactor operation, and

(2) no CPC software changes involving additions or deletions to addressable constants or changes to software limit values are made without prior approval of NRC.

Core Operating Limit Supervisory System

The Core Operating Limit Supervisory System (COLSS) is normally used to monitor linear heat rate, DNBR margin, and axial shape index. However, whenever COLSS is out of service, the CPCs are used to perform the safe monitoring function. Since the CPCs use excore neutron detectors and are required to provide protection during certain transients and accidents, the CPC uncertainties and margins are more limiting than those of COLSS, which uses incore neutron detectors. These extra conservatisms built into CPC for transient protection are not all required when the CPCs are being used for monitoring. The licensee has credited these conservatisms in the COLSS Out-of-Service limits of Technical Specifications 3.2.1, 3.2.4, and 3.2.7 when the CPC is being used for monitoring. These conservatisms provide approximately 10% LHR and 12% DNBR power margin credits which have been translated into COLSS Out-of-Service LCO adjustments in the proposed Technical Specification changes. Since these conservatisms are not taken from the CPCs but rather are credited in the COLSS Out-of-Service limits of the Technical Specifications, the CPC transient protection is not affected. However, these credits will have to be re-evaluated when the CPC changes associated with the CPC Improvement Program (CIP) described in CEN-308-P, "CPC/CEAC Software Modifications to the CPC Improvement Program" are implemented at Waterford Unit 3, since some CPC penalties were reduced as a part of the CIP. Therefore, the staff concludes that the licensee's proposal to take credit for the conservatisms built into the CPSs when they are being used in a monitoring function, is acceptable.

Low Temperature Overpressure Protection

Technical Specification 3.4.8.3 defines the requirements for low temperature overpressure protection (LTOP) provided by the shutdown cooling system (SCS) relief valves and the applicable modes of plant operation. The current restriction in 3.4.8.3 requires that LTOP be implemented during Mode 4 when the temperature of any reactor coolant (RCS) cold leg is less than 285°F. Also, Action Statement (a) of Technical Specification 3.4.9 requires that in the event that a code Class 1 component does not meet the integrity requirements of 3.4.9 (e.g., a weld repair), the structural integrity of the affected component must be restored prior to increasing the RCS temperature higher than 272°F (70°F above the minimum temperature required by NDT considerations, presently 202°F per Figure 3.4-2). With the SCS connected to the RCS, a hydrostatic test of the RCS, as required by the ASME code could be performed since the SCS relief valves would lift. The licensee's proposed change of Technical Specification 3.4.8.3 revises the applicability in Mode 4 to allow a lower RCS temperature of 260°F without LTOP to enable conducting the RCS hydrostatic test. Currently, Technical Specification 3.4.8.3 contains no provision for conducting the RCS hydrostatic test after a refueling outage. This provision is included in other plant Technical Specifications and the staff concludes that its omission from the Waterford 3 Technical Specifications was not due to any plant specific technical concerns but was due to an oversight.

Since the proposed change results in a minor and necessary relaxation of the LTOP implementation temperature specification, and is consistent with the existing Appendix G curve (Figure 3.4-2) as well as with other plant Technical Specifications, the proposed change is acceptable. However, to eliminate future changes to the technical specifications on temperature limitation for hydrostatic testing of the RCS due to future changes of Figure 3.4-2, the staff concludes that the note added to the applicability for Technical Specification 3.4.8.3 should be modified as follows:

> "During inservice leak and hydrostatic testing, use the temperature limitation indicated by the curve for inservice test on Figure 3.4-2, with Reactor Coolant System heatup rate restricted in accordance with Specification 3.4.8.1.g."

Moderator Temperature Coefficient

The limitation on Moderator Temperature Coefficient (MTC) specified in Technical Specification 3.1.1.3 is provided to ensure that the assumptions used in the accident and transient analyses remain valid throughout the fuel cycle. The MTC varies slowly as a function of core burnup, due principally to the reduction in reactor coolant system boron concentration with fuel burnup. It is most positive at the beginning of core life and becomes more negative with increasing burnup. The surveillance requirements for measurement of the MTC are intended to provide confirmation that the measured value is within its Technical Specification limit, thereby assuring that the calculational technique for calculating the MTC is valid. The proposed change does not alter the LCO criteria, nor is the measured MTC value used as input to any safety-related calculation. In addition, all standard criteria (e.g. xenon equilibrium) will be met when performing the MTC measurement during startup physics test to ensure an accurate measurement. Therefore, the proposed change allowing for an MTC measurement earlier in the core cycle has no effect on the existing safety margin and 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 occupation radiation exposure. The Commission has previously issued proposed findings that the amendment involves no significant hazards consideration, and there has been no public comment on such findings. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

6.0 CONCLUSION

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

Dated: May 30, 1986