UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 24, 1998

50-382

Mr. Charles M. Dugger
Vice President Operations
Entergy Operations, Inc.
P. O. Box B
Killona, LA 70066

SUBJECT: ISSUANCE OF AMENDMENT NO. 145 TO FACILITY OPERATING LICENSE NPF-38 - WATERFORD STEAM ELECTRIC STATION, UNIT 3 (TAC NO. MA3538)

Dear Mr. Dugger:

The Commission has issued the enclosed Amendment No. 145 to Facility Operating License No. NPF-38 for the Waterford Steam Electric Station, Unit 3. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 18, 1998, as superseded by letter dated September 23,1998.

The amendment changes the Appendix A TSs by revising Note "1" in Table 2.2-1, "Reactor Protective Instrumentation Trip Setpoint Limits" and Note "a" in Table 3.3-1, "Reactor Protective Instrumentation," both applicable to high logarithmic power reactor trip instrumentation. Additionally, the requested changes clarify the terms RATED THERMAL POWER and THERMAL POWER used in Tables 2.2-1, 3.3-1 and 4.3-1. A Bases change is made to support these changes.

It should be noted that the NRC published a Federal Register Notice based on your first submittal dated September 11, 1998. Unfortunately, we determined that this submittal lacked quality and attention to detail. Your proposed changes to resolve the identified issues were determined to be inadequate and your discussion in the application was inconsistent with a licensee event report submitted by you on August 27, 1998, on same issue. This submittal was superseded by your letter dated September 18, 1998. However, the September 18, 1998, letter was superseded again by letter dated September 23, 1998, because you did not evaluate the issue throughly.

This amendment is being issued on an emergency basis without prior notice regarding the September 23, 1998, request.

1/1
DF01

A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,
ORIGINAL SIGNED BY:
Chandu P. Patel, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosures: 1. Amendment No. 145 to NPF-38
2. Safety Evaluation

cc w/encls: See next page

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A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Chandu P. Patel

Chandu P. Patel, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosures: 1. Amendment No. 145 to NPF-38
2. Safety Evaluation

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Waterford 3

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

WATERFORD STEAM ELECTRIC STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 145
License No. NPF-38

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated September 18, 1998, as superseded by letter dated September 23, 1998, complies with the standards and requirements of the Atomic Energy Act of l954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. NPF-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. 145, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Chandu P. Patel

Chandu P. Patel, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 24, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 145

TO FACILITY OPERATING LICENSE NO. NPF-38

DOCKET NO. 50-382

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

REMOVE PAGES	INSERT PAGES
2-3	2-3
2-4	2-4
3/4 3-4	3/4 3-4
-	3/4 3-4a
3/4 3-12	3/4 3-12
B 2-3	B 2-3

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	≤ 108% of RATED THERMAL POWER	≤ 108.76% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	≤ 0.257% of RATED THERMAL POWER (6)	≤ 0.280% of RATED THERMAL POWER (6)
4. Pressurizer Pressure - High	≤ 2350 psia	≤ 2359 psia
5. Pressurizer Pressure - Low	≥ 1684 psia (2)	≥ 1649.7 psia (2)
6. Containment Pressure - High	≤ 17.1 psia	≤ 17.4 psia
7. Steam Generator Pressure - Low	≥ 764 psia (3)	≥ 749.9 psia (3)
8. Steam Generator Level - Low	≥ 27.4% (4)	≥ 26.48% (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)	≤ 88.62% (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	≥ 19.00 psid (7)	≥ 18.47 psid (7)

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 the reset point of the bistable. The reset point shall be within 3.0×10^{-5} % of RATED THERMAL POWER* below the bistable setpoint which is nominally 10^{-4}% of RATED THERMAL POWER*. This accounts for the deadband of the bistable.

(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) As measured by the Logarithmic Power Channels.

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

*As measured by the Logarithmic Power Channels.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

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

TABLE 3.3-1 (Continued)

TABLE NOTATION

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

#The provisions of Specification 3.0.4 are not applicable.

**Not applicable above 10^{-4}% RATED THERMAL POWER.[1]

(a) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER[1]; bypass shall be automatically removed when THERMAL POWER[1] is less than or equal to the reset point of the bistable. The reset point shall be within 3.0×10^{-5}% of RATED THERMAL POWER[1] below the bistable setpoint which is nominally 10^{-4}% of RATED THERMAL POWER[1]. This accounts for the deadband of the bistable.

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

(c) Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER[1]; bypass shall be automatically removed when THERMAL POWER[1] is greater than or equal to 10^{-4}% of RATED THERMAL POWER[1]. During testing pursuant to Special Test Exception 3.10.3, trip may be manually bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 5% of RATED THERMAL POWER.

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

(e) See Special Test Exception 3.10.2.

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

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

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[1] As measured by the Logarithmic Power Channels.

TABLE 3.3-1 (Continued)

TABLE NOTATION

ACTION STATEMENTS

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

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

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)	Q(9),R(6)	1, 2
15. CEA Calculators	S	R	Q,R(6)	1, 2
16. Reactor Coolant Flow - Low	S	R	Q	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.

#The provisions of Specification 4.0.4 are not applicable when reducing reactor power to less than 10^{-4}% of RATED THERMAL POWER[(a)] from a reactor power level greater than 10^{-4}% of RATED THERMAL POWER[(a)]. Upon reducing power below 10^{-4}% of RATED THERMAL POWER[(a)], a CHANNEL FUNCTIONAL TEST shall be performed within 2 hours if not performed during the previous 31 days. This requirement does not apply with the reactor trip breakers open.

(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):

a. Between 15% and 80% of RATED THERMAL POWER, compare the Linear Power Level, the CPC at ΔT power, and CPC nuclear power signals to the calorimetric calculation.

If any signal is within -0.5% to +10% of the calorimetric calculation, then do not calibrate except as required during initial power ascension following refueling.

If any signal is less than the calorimetric calculation by more than 0.5%, then adjust the affected signal(s) to within 0.0% to +10.0% of the calorimetric caluclation.

If any signal is greater than the calorimetric calculation by more than 10%, then adjust the affected signal(s) to within 0.0% to 10% of the calorimetric.

b. At or above 80% of RATED THERMAL POWER, compare the Linear Power Level, the CPC Δ T power, and CPC nuclear power signals to the calorimetric calculation. If any signal differs from the calorimetric calculation by an absolute difference of more than 2%, then adjust the affected signal(s) to agree with the calorimetric calculation within -2% to +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 subchannel 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 or verify acceptable values for the shape annealing matrix elements used in the Core Protection Calculators.

[(a)]As measured by the Logarithmic Power Channels.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Linear Power Level - High

The Linear Power Level - High trip provides reactor core protection against rapid reactivity excursions which might occur as the result of an ejected CEA. This trip initiates a reactor trip at a linear power level of less than or equal to 108% of RATED THERMAL POWER.

Logarithmic Power Level - High

The Logarithmic Power Level - High trip is provided to protect the integrity of fuel cladding and the Reactor Coolant System pressure boundary in the event of an unplanned criticality from a shutdown condition. A reactor trip is initiated by the Logarithmic Power Level - High trip at a THERMAL POWER* level of less than or equal to 0.257% of RATED THERMAL POWER* unless this trip is manually bypassed by the operator. The operator may manually bypass this trip when the THERMAL POWER* level is above 10-4% of RATED THERMAL POWER*; this bypass is automatically removed when the THERMAL POWER* level decreases to 10-4% of RATED THERMAL POWER*.

Pressurizer Pressure - High

The Pressurizer Pressure - High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides Reactor Coolant System protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is at less than or equal to 2350 psia which is below the nominal lift setting of 2500 psia for the pressurizer safety valves and its operation avoids the undesirable operation of the pressurizer safety valves.

Pressurizer Pressure - Low

The Pressurizer Pressure - Low trip is provided to trip the reactor and to assist the Engineered Safety Features System in the event of a Loss of Coolant Accident. During normal operation, this trip's setpoint is set at greater than or equal to 1684 psia. This trip's setpoint may be manually decreased, to a minimum value of 100 psia, as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at less than or equal to 400 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached.

*As measured by the Logarithmic Power Channels.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 145 TO

FACILITY OPERATING LICENSE NO. NPF-38

ENTERGY OPERATIONS, INC.

WATERFORD STEAM ELECTRIC STATION, UNIT 3

DOCKET NO. 50-382

1.0 INTRODUCTION

By application dated September 18, 1998, as superseded by letter dated September 23, 1998, Entergy Operations, Inc. (the licensee), submitted a request for changes to the Waterford Steam Electric Station, Unit 3 (Waterford 3), Technical Specifications (TSs). The requested changes would revise Note "1" in Table 2.2-1, "Reactor Protective Instrumentation Trip Setpoint Limits" and Note "a" in Table 3.3-1, "Reactor Protective Instrumentation (RPI)," both applicable to high logarithmic power (HLP) reactor trip instrumentation. Additionally, the requested changes clarify the terms RATED THERMAL POWER and THERMAL POWER used in Tables 2.2-1, 3.3-1 and 4.3-1. A Bases change is being proposed to support these changes.

2.0 EVALUATION

The HLP trip is provided to trip the reactor when indicated neutron flux power from the excore detectors reaches a preset value in two out of the four instrumentation channels. This trip provides protection against inadvertent withdrawal of control element assembly (CEA) initiated from low power or subcritical conditions. The nominal trip set point of this instrumentation is $\leq$ 0.257% of rated thermal power (RTP). Waterford 3 RPI design provides a permissive to manually bypass the HLP reactor trip on increasing power (start-up) when reactor thermal power reaches a specific value and automatically remove the bypass on decreasing power (shutdown) when the reactor thermal power reaches another specified value. This automatic removal of the trip bypass ensures that the trip will be available in the event of a CEA withdrawal from low power or subcritical conditions. The HLP trip manual bypass is provided to allow reactor power increase into mode 1 during controlled reactor start-up. Without the bypass in place, a reactor trip is generated when the trip setpoint is reached, preventing further power increase. This manual bypass will occur only during controlled power increase and not if the increase is due to an inadvertent CEA withdrawal.

In the present TSs for Waterford 3, Note "1" of Table 2.2-1 and Note "a" of Table 3.3-1 state that the HLP trip may be manually bypassed above 10^{-4}% of RTP and the bypass shall be automatically removed when reactor thermal power is less than or equal to 10^{-4}% of RTP. To accomplish this TS requirement the associated instrumentation channel bistable provides a permissive for manual bypass of HLP reactor trip when the reactor power is greater than 10^{-4}% of RTP. When reactor power is equal to 10^{-4}% of RTP and decreasing, the manual bypass is automatically removed and the bistable is armed to provide a reactor trip signal on HLP.

At Waterford 3, the same bistable provides two automatic functions. One is to remove the HLP trip bypass at or below 10^{-4}% of RTP and the second is to remove Core Protection Calculator (CPC) trip bypass (low DNBR and low power density) when reactor thermal power is equal to or greater than 10^{-4}% of RTP. A single bistable can not both energize and de-energize at the same value as required by the TS. The CPC trip bypass automatic removal occurs at the bistable set point which is 10^{-4}% of RTP, but the HLP trip bypass automatic removal occurs at the reset value of the bistable which is slightly below 10^{-4}% of RTP. This is due to an inherent deadband, which causes arming of the bistable to take place at a lower power than 10^{-4}% of RTP. As such, the actual bistable operation is found to be in noncompliance with the TS requirement because the bistable does not satisfy both the CPC and HLP provisions of the TS at the same value. To accommodate the inherent deadband in the bistable, the proposed changes to the TS notes revise the less than or equal to 10^{-4}% of RTP requirement for automatic removal of HLP reactor trip bypass. The changes state that the bypass shall be automatically removed when thermal power is less than or equal to the reset point of the bistable. The reset point is specified to be within 3.0×10^{-5}% of RTP below the bistable set point which is nominally 10^{-4}% of RTP.

The licensee's evaluation of these changes indicates: since the difference between the bistable reset value and its nominal trip set point is small, accounting for the slightly different bypass removal power level caused by the bistable deadband would result in negligible change to the calculated peak power and heat flux for the pertinent CEA withdrawal events. Therefore, the consequences of any accident previously evaluated will not significantly change. Additionally, the safety analysis described in FSAR section 15.4.1.2 assumed that a CEA withdrawal from critical conditions can be initiated from the lowest power level (the most limiting initial condition) at which the HLP trip is not available. In this case, a reactor trip is generated by the CPC variable overpower trip function. The licensee considered a case where initial power level could theoretically be just above the bistable reset value during a shutdown when the power decrease was stopped such that the manual bypass for HLP trip was not yet removed and the trip function would not be armed. The licensee concluded that this condition would be highly improbable. Furthermore, the licensee's standard practice is to trip the reactor between 5% and 10% power during a shutdown, and the power level would decrease logarithmetically through the 10^{-4}% setpoint and not hang up above the bistable reset value. The staff agrees with the licensee's conclusions and therefore, finds this change acceptable.

In addition to the above, the licensee proposed clarification to specify that the setpoints for the bistable are based upon output from the logarithmic power channels of the excore nuclear instrumentation. TSs 2.2.1 and 3.3.1 use the terms THERMAL POWER and RATED THERMAL POWER to specify the setpoints of the bistable. THERMAL POWER and RATED THERMAL POWER are defined in TSs 1.34 and 1.24 in terms of the total amount of heat transferred from the core to the reactor coolant system. This includes a contribution from decay heat produced by the core.

Contrary to these definitions, the logarithmic power channels that provide input into the bistable do not include contributions from decay heat. Decay heat contributions would prevent the actual thermal power of the core from reaching the level of the setpoints for the bistable for the duration of a normal shutdown. This contradicts the purpose of the High Local Power Density and Low Departure from Nucleate Boiling trips bypass capability in the bistable. The intent of the original wording is to describe the power level as indicated by the logarithmic power channel. The proposed revisions annotate the references to RATED THERMAL POWER and THERMAL POWER to clarify that the parameter of interest is the power level as indicated by the logarithmic power channel and does not include contribution from decay heat. This change is provided to eliminate any confusion regarding the impact of decay heat on these parameters.

The staff review of the proposed changes to the logarithmic power reactor trip tables notes found that the changes do not significantly affect the safety function of this instrumentation which is to trip the reactor on a rod withdrawal incident at low reactor power or subcritical conditions. It is also determined that the proposed changes will not affect the other functions of the bistable which is to enable the CPC reactor trip. In addition, the proposed clarifications regarding RATED THERMAL POWER and THERMAL POWER provide better definitions of the terms as applied in applicable tables. Therefore, the staff finds the proposed change to be acceptable.

3.0 EMERGENCY CIRCUMSTANCES

On September 11, 1998, a TS Change Request, NPF-38-210, was submitted to amend TS Table 2.2-1 and Table 3.3-1. At that time Waterford 3 was operating in Mode 1 and the proposed changes were not affecting the operation. The licensee concluded that criteria for exigency or emergency were not justified. However, the licensee indicated that if plant status were to change significantly, the emergency criteria would be applicable as this change would be required for plant startup. After the September 11, 1998 submittal, the plant conditions changed such that Waterford 3 was placed in Mode 4 on September 18, 1998. This forced outage was the result of unexpected leakage from a pressurizer safety relief valve. As degradation of this valve could not be anticipated, this was an unplanned outage. The approval of the proposed changes is required for plant restart from this forced outage. Based on this need and the criteria provided in 10 CFR 50.91a, by letter dated September 18, 1998, the licensee superseded the previous request and requested that the proposed request be processed on an emergency basis.

Waterford 3 is preparing to restart following the forced outage. Requiring literal compliance with the present TS will preclude resumption of power operations until the TS change request is approved. This would require several days to allow for the normal public notice period to expire. Accordingly, the Commission finds that an emergency situation exists pursuant to 10 CFR 50.91(a)(5) because failure to act timely would result in prevention of resumption of operation. The staff finds that the licensee did not create the emergency situation and acted promptly once it became aware of the need to act promptly.

4.0 FINAL NO SIGNIFICANT HAZARD CONSIDERATION DETERMINATION

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

(1) Involve a significant increase in the probability or consequences of any accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin or safety.

This amendment has been evaluated against the standards in 10 CFR 50.92. The NRC has made a final no significant hazards consideration regarding this amendment because of the following:

The proposed changes modify the table notations for the bistable in TS 2.2.1 and 3.3.1. The proposed changes to these trip bypass removal functions do not adversely impact any system, structure, or component design or operation in a manner that would result in a change in the frequency or occurrence of accident initiation. The reactor trip bypass removal functions are not accident initiators. System connections and the trip setpoints themselves are not affected by trip bypass removal setpoint variations. Since the deadband for the bistable is small, there is a negligible impact on the CEA withdrawal analyses. Revised analyses, accounting for slightly different bypass removal power levels caused by the bistable deadband, would result in negligible changes to the calculated peak power and heat flux for the pertinent CEA withdrawal events. Therefore, the consequences of any accident previously evaluated will not significantly change.

With respect to the clarification proposed for the THERMAL POWER input to the bypass capability of the affected reactor trips for the bistable, the proposed change does not alter the manner of operation of the operating bypasses and automatic bypass removals. This change corrects a discrepancy between the formal definition of this terminology and its use in the context of the applicable TSs.

Therefore, the proposed change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

The trip bypass removal functions in question protect against possible reactivity events. The power, criticality levels, and possible bank withdrawals associated with these trip functions have already been evaluated. Therefore, all pertinent reactivity events have previously been considered. Slight differences in the power level at which the automatic trip bypass removal occurs will not cause a different kind of accident.

The proposed changes to Note "1" of Table 2.2-1 and Note "a" of Table 3.3-1 do not alter any plant system, structure, or component. Furthermore, these changes do not reduce the capability of any safety-related equipment to mitigate anticipated operational occurrences.

In addition, no new or different accidents result from clarifying the THERMAL POWER input to the operating bypasses and automatic bypass removals of the affected reactor trips. The results of previously performed accident analyses remain valid.

Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The safety function associated with the CPC and HLP trip functions are maintained. Since the deadband for the bistable is small, there is a negligible impact on the CEA withdrawal analyses. Calculated peak power and heat flux are not significantly changed as a result of the bistable deadband. All acceptance criteria are still met for these events.

Clarification of the THERMAL POWER input to the operating bypasses and automatic bypass removals of the bistable does not alter the operation of the operating bypasses and automatic bypass removals of the affected reactor trips. This change corrects a discrepancy between the formal definition of this terminology and its use in the context of the applicable TSs.

Therefore, the proposed change will not involve a significant reduction in a margin of safety.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Louisiana State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has made a final no significant hazards consideration finding with respect to this amendment. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the

Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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