Mr. Ross P. Barkhurst
Vice President Operations
Entergy Operations, Inc.
Post Office Box B
Killona, LA 70066

May 8, 1

SUBJECT: ISSUANCE OF AMENDMENT NO. 106 TO FACILITY OPERATING LICENSE NPF-38 - WATERFORD STEAM ELECTRIC STATION, UNIT 3 (TAC NO. M88434)

Dear Mr. Barkhurst:

The Commission has issued the enclosed Amendment No.106 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 December 14, 1993, as supplemented by letter dated March 3, 1995.

The amendment changes the Appendix A TSs by removing the reactor vessel material specimen withdrawal schedule and by updating the reactor coolant system pressure-temperature (P-T) curves.

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. 106 to NPF-38
2. Safety Evaluation

cc w/encls: See next page

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

May 8, 1995

Mr. Ross P. Barkhurst
Vice President Operations
Entergy Operations, Inc.
Post Office Box B
Killona, LA 70066

SUBJECT: ISSUANCE OF AMENDMENT NO. 106 TO FACILITY OPERATING LICENSE NPF-38 - WATERFORD STEAM ELECTRIC STATION, UNIT 3 (TAC NO. M88434)

Dear Mr. Barkhurst:

The Commission has issued the enclosed Amendment No. 106 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 December 14, 1993, as supplemented by letter dated March 3, 1995.

The amendment changes the Appendix A TSs by removing the reactor vessel material specimen withdrawal schedule and by updating the reactor coolant system pressure-temperature (P-T) curves.

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. 106 to NPF-38
2. Safety Evaluation

cc w/encls: See next page

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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. 106
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 December 14, 1993, as supplemented by letter dated March 3, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, 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.

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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. 106 , 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: May 8, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 106

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
XXI	XXI
3/4 4-3	3/4 4-3
3/4 4-5	3/4 4-5
3/4 4-29	3/4 4-29
3/4 4-30	3/4 4-30
3/4 4-31	3/4 4-31
3/4 4-32	3/4 4-32
3/4 4-34	3/4 4-34
B 3/4 4-7	B 3/4 4-7
B 3/4 4-10	B 3/4 4-10

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4.3-6	REMOTE SHUTDOWN INSTRUMENTATION SURVEILLANCE REQUIREMENTS	3/4 3-43
3.3-10	ACCIDENT MONITORING INSTRUMENTATION	3/4 3-45
4.3-7	ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS	3/4 3-46
3.3-12	DELETED	3/4 3-56
4.3-8	DELETED	3/4 3-58
3.3-13	EXPLOSIVE GAS MONITORING INSTRUMENTATION	3/4 3-61
4.3-9	EXPLOSIVE GAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS	3/4 3-65
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4.7-1	SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM	3/4 7-8
3.7-3	ULTIMATE HEAT SINK MINIMUM FAN REQUIREMENTS	3/4 7-14
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LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loop(s)/train(s) listed below shall be OPERABLE and at least one reactor coolant and/or shutdown cooling loops shall be in operation.*

a. Reactor Coolant Loop 1 and its associated steam generator and at least one associated reactor coolant pump,**

b. Reactor Coolant Loop 2 and its associated steam generator and at least one associated reactor coolant pump,**

c. Shutdown Cooling Train A,

d. Shutdown Cooling Train B.

APPLICABILITY: MODE 4

ACTION:

a. With less than the above required reactor coolant and/or shutdown cooling loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is a shutdown cooling loop, be in COLD SHUTDOWN within 24 hours.

b. With no reactor coolant or shutdown cooling loop in operation, suspend
all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

*All reactor coolant pumps and shutdown cooling pumps (LPSI pumps) may be deenergized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 272°F unless (1) the pressurizer water volume is less than 900 cubic feet or (2) the secondary water temperature of each steam generator is less than 100°F above each of the Reactor Coolant System cold leg temperatures.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be $\geq$ 50% of wide range indication at least once per 12 hours.

4.4.1.3.3 At least one reactor coolant or shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

LIMITING CONDITION FOR OPERATION

3.4.1.4 At least two of the loop(s)/trains listed below shall be OPERABLE and at least one reactor coolant and/or shutdown cooling loop shall be in operation.*

a. Reactor Coolant Loop 1 and its associated steam generator and at least one associated reactor coolant pump**,

b. Reactor Coolant Loop 2 and its associated steam generator and at least one associated reactor coolant pump**,

c. Shutdown Cooling Train A,

d. Shutdown Cooling Train B.

APPLICABILITY: MODE 5 with reactor coolant loops filled**.

ACTION:

a. With less than the above required reactor coolant and/or shutdown cooling loops OPERABLE or with less than the required steam generator level, immediately initiate corrective action to return the required loops to OPERABLE status or to restore the required level as soon as possible.

b. With no reactor coolant or shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1 The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.4.2 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be ≥50% of wide range indication at least once per 12 hours.

4.4.1.4.3 At least one reactor coolant loop or shutdown cooling train shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

---

*All reactor coolant pumps and shutdown cooling pumps (LPSI pumps) may be deenergized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 272°F unless (1) the pressurizer water volume is less than 900 cubic feet or (2) the secondary water temperature of each steam generator is less than 100°F above each of the Reactor Coolant System cold leg temperatures.

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.5 Two shutdown cooling loops shall be OPERABLE# and at least one shutdown cooling loop shall be in operation.*

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible.

b. With no shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.5 At least one shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

#One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling loop is OPERABLE and in operation.

*The shutdown cooling pump (LPSI pump) may be deenergized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

SURVEILLANCE REQUIREMENTS (Continued)

Reactor Vessel material surveillance program - withdrawal schedule in FSAR Table 5.3-10. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

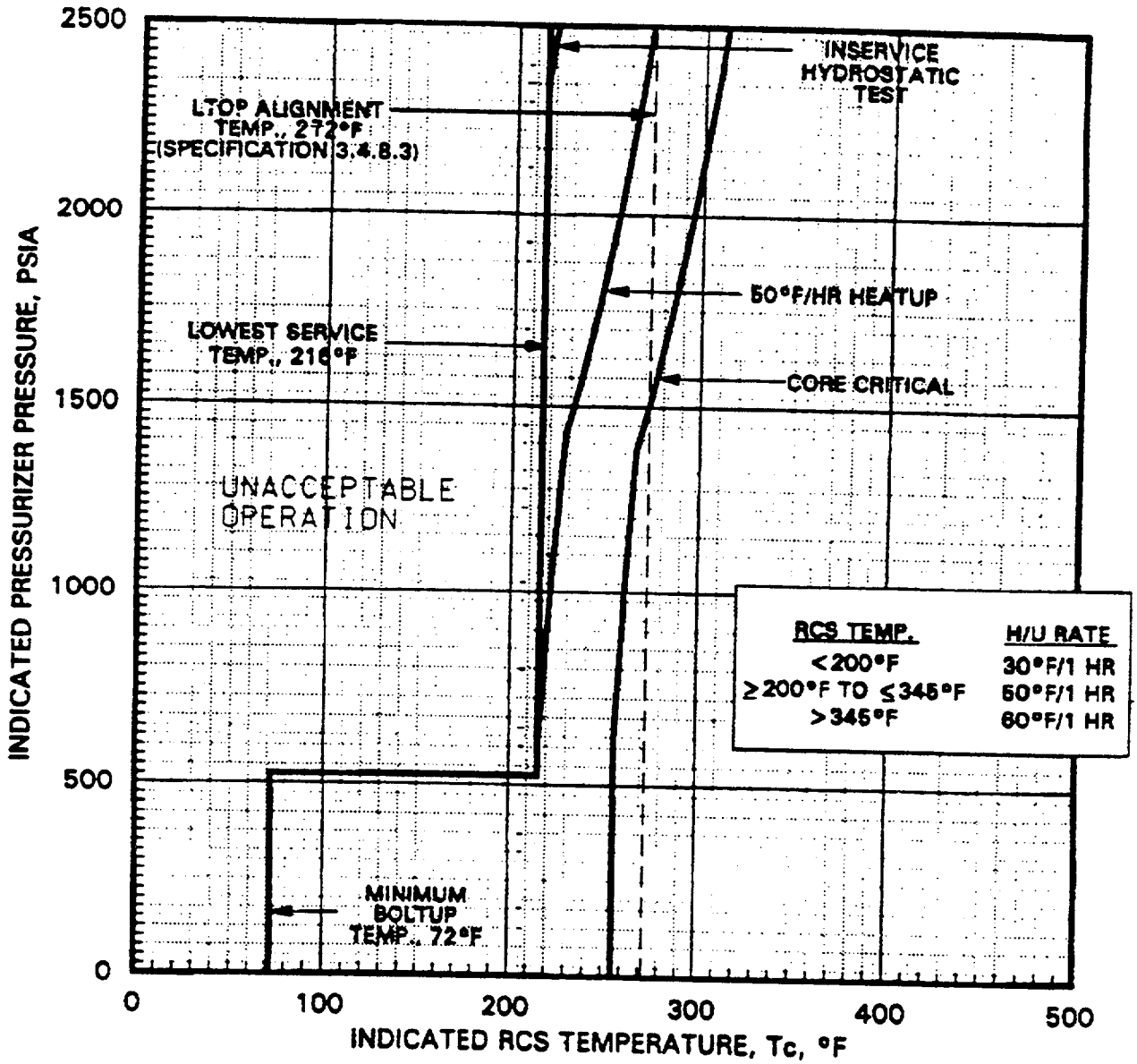

FIGURE 3.4-2

WATERFORD UNIT 3 HEATUP CURVE

REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS

0-15 EFPY

CURVE BASIS: PEAK SURFACE FLUENCE = 2.29×10^{19} n/cm^2 @ 20 EFPY

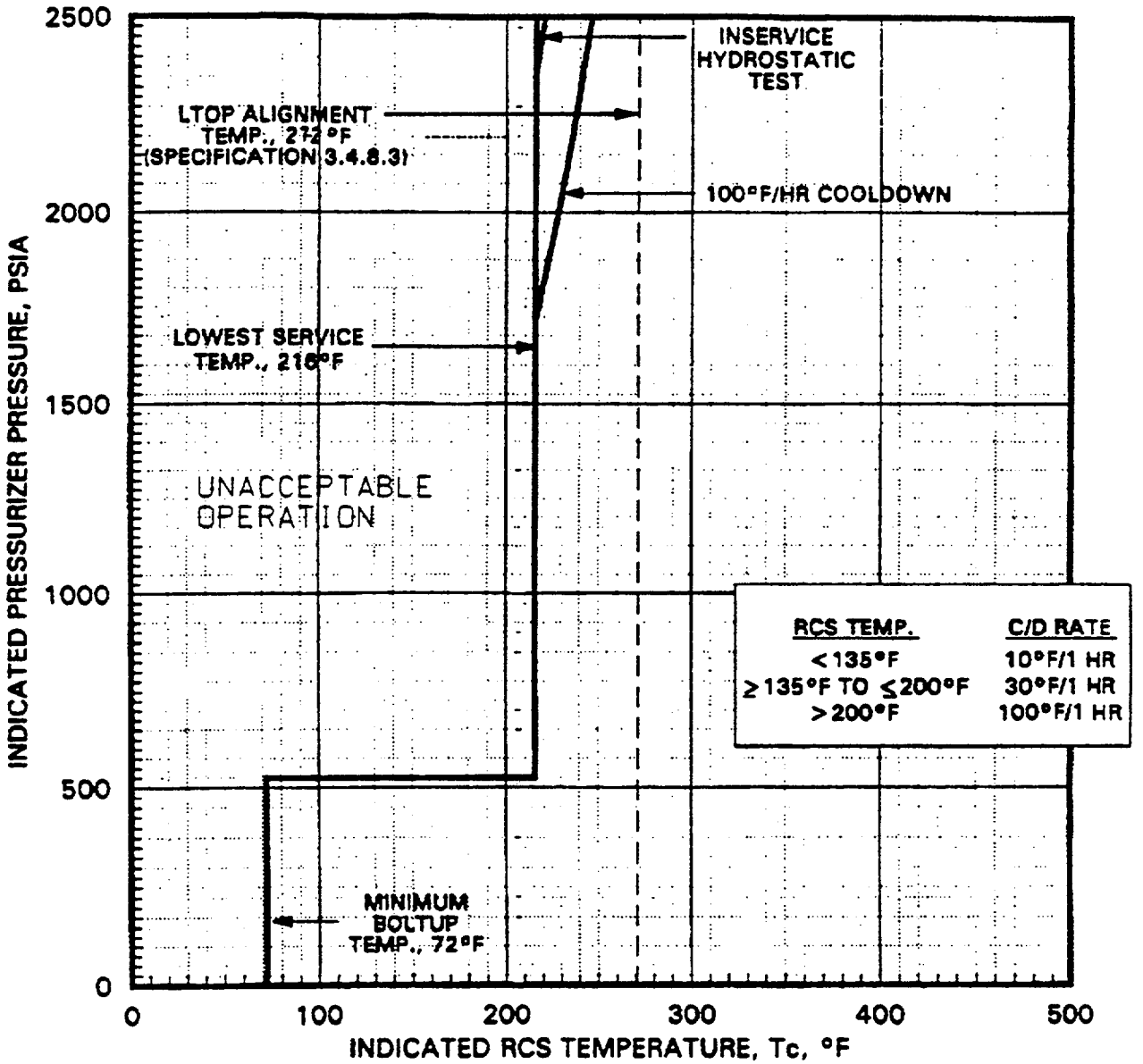

FIGURE 3.4-3

WATERFORD UNIT 3 COOLDOWN CURVE

REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS

0-15 EFPY

CURVE BASIS: PEAK SURFACE FLUENCE = 2.29×10^{19} n/cm^2 @ 20 EFPY

TABLE 4.4.5

This TABLE has been deleted.

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.

LIMITING CONDITION FOR OPERATION

3.4.8.3 Two Shutdown Cooling (SDC) System suction line relief valves (SI-406A and SI-406B) shall be OPERABLE with a lift setting of less than or equal to 430 psia.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 272°F#, MODE 5, and MODE 6 when the head is on the reactor vessel and the RCS is not vented through a 5.6 square inch or larger vent.

ACTION:

a. With one SDC System suction line relief valve inoperable in MODE 4, restore the inoperable valve to OPERABLE status within 7 days, or depressurize and vent the RCS through at least a 5.6 square inch vent within the next 8 hours.

b. With one SDC System suction line relief valve inoperable in MODES 5, or 6, either (1) restore the inoperable valve to OPERABLE status within 24 hours, or (2) complete depressurization and venting of the RCS through at least a 5.6 square inch vent within a total of 32 hours.

c. With both SDC System suction line relief valves inoperable, complete depressurization and venting of the RCS through at least a 5.6 square inch vent within 8 hours.

d. In the event either the SDC System suction line relief valve(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 SDC System suction line relief valve(s) or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

e. 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.

The heatup and cooldown limit curves Figures 3.4-2 and 3.4-3 are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate of up to 60°F per hour or cooldown rate of up to 100°F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period indicated on Figures 3.4-2 and 3.4-3. The limitations on the Reactor Coolant System heatup and cooldown rates are further restricted due to stress limitations in the Reactor Coolant Pump. As part of the LOCA support scheme, the Reactor Coolant Pump has a ring around the suction nozzle of the pump. The support skirt is welded to the ring. Due to this design, the heatup and cooldown rates must be limited to maintain acceptable thermal stresses.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these test are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper and nickel content of the material in question, can be predicted using FSAR Table 5.3-1 and the recommendations of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The heatup and cooldown limit curves Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-82 and 10 CFR Part 50 Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. The surveillance specimen withdrawal schedule is shown in FSAR Table 5.3-10. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the delta RT_{NDT} determined from the surveillance capsule is different from the calculated delta RT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figures 3.4-2 and 3.4-3 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50.

TABLE B 3/4.4-1

REACTOR VESSEL FRACTURE TOUGHNESS

Piece Number	Drawing Number	Code Number	Material	Vessel Location	Drop Weight NDTT(F)	RT(A) NDT(F)	Charpy 30 ft-lb. Fix Temp. (F) Long.	Charpy 50 ft-lb. Fix Temp. (F) Long.	35 Mils Lateral Expansion Temp. (F) Long.	Charpy Upper Shelf Energy (ft-lb.) Long.
106-101	741701 6103	M-603-1	SA508 CL-II	Closure Head Flange	20	20	-15	2	-5	159
126-101	741701 6103	M-1001-1	SA508 CL-II	Vessel Flange	20	20	-20	-5	2	154
131-102A	741701 6103	M-1013-1	SA508 CL-I	Safe End	0	0	-35	0	-15	148
131-102D	741701 6103	M-1013-2	SA508 CL-I	Safe End	0	0	-35	0	-15	148
131-102C	741701 6103	M-1013-3	SA508 CL-I	Safe End	0	0	-27	0	-32	149
131-102B	741701 6103	M-1013-4	SA508 CL-I	Safe End	0	0	-27	0	-30	149
131-101A	741701 6103	M-1012-1	SA508 CL-I	Safe End	0	0	0	25	-5	146
131-101B	741701 6103	M-1012-2	SA508 CL-I	Safe End	0	0	0	25	-5	146
128-301	741701 6103	M-1011-1	SA508 CL-II	Outlet Nozzle	-20	-20	-37	0	5	99
128-101	741701 6103	M-1010-1	SA508 CL-II	Inlet Nozzle	20	20	-37	10	0	135
128-101	741701 6103	M-1010-2	SA508 CL-II	Inlet Nozzle	20	20	-50	-35	-40	140
128-101	741701 6103	M-1010-3	SA508 CL-II	Inlet Nozzle	10	10	-70	-47	-42	133
128-101	741701 6103	M-1010-4	SA508 CL-II	Inlet Nozzle	30	30	-40	-20	-30	140
128-301	741701 6103	M-1011-2	SA508 CL-II	Outlet Nozzle	0	0	-30	-10	-12	188
124-102	741701 6103	M-1003-1	SA533-B CL-I	Intermediate Shell Plate	-30	-30	-30	-10	-10	144
124-102	741701 6103	M-1003-2	SA533-B CL-I	Intermediate Shell Plate	-50	-50	-55	-12	-15	149

TABLE B 3/4.4-1 (Continued)

Piece Number	Drawing Number	Code Number	Material	Vessel Location	Drop Weight NDTT(F)	RT[A] NDT(F)	Charpy 30 ft-lb. Fix Temp. (F) Long.	Charpy 50 ft-lb. Fix Temp. (F) Long.	35 Mils Lateral Expansion Temp. (F) Long.	Charpy Upper Shelf Energy (ft-lb.) Long.
124-102	741701 6103	M-1003-3	SA533-B CL-I	Intermediate Shell Plate	-50	-42	-22	-2	-10	138
122-102	741701 6103	M-1002-1	SA533-B CL-I	Upper Shell Plate	-40	-8	13	32	23	151
122-102	741701 6103	M-1002-2	SA533-B CL-I	Upper Shell Plate	-20	-20	-20	12	15	128
122-102	741701 6103	M-1002-3	SA533-B CL-I	Upper Shell Plate	-40	-40	-20	0	0	153
154-102	741701 6103	M-1007-1	SA533-B CL-I	Bottom Head Torus	-80	-80	-72	-62	-60	174
152-101	741701 6103	M-1008-1	SA533-B CL-I	Bottom Head Dome	-40	-40	-35	-10	-15	141
104-102	741701 6103	M-1005-1	SA533-B CL-I	Closure Head Torus	-30	-30	-25	0	-2	160
142-101	741701 6103	M-1004-1	SA533-B CL-I	Lower Shell Plate	-50	-15	10	25	20	163
142-101	741701 6103	M-1004-2	SA533-B CL-I	Lower Shell Plate	-20	22	37	62	55	144
142-101	741701 6103	M-1004-3	SA533-B CL-I	Lower Shell Plate	-50	-10	12	30	25	145
102-101	741701 6103	M-1006-1	SA533-B	Closure Head Dome	-50	-25	-5	15	10	138

[A] MTEB Position 5-2 "Fracture Toughness Requirements," Paragraph 1.1(3)(b).

PRESSURE/TEMPERATURE LIMITS (Continued)

The maximum RT_{NDT} for all Reactor Coolant System pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 90°F. The Lowest Service Temperature limit line shown on Figures 3.4-2 and 3.4-3 is based upon this RT_{NDT} since Article NB-2332 of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps, and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia (as corrected for elevation and instrument error).

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of the shutdown cooling system relief valve or an RCS vent opening of greater than 5.6 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 272°F. Each shutdown cooling system relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 100°F above the RCS cold leg temperatures or (2) inadvertent safety injection actuation with injection into a water-solid RCS. The limiting transient includes simultaneous, inadvertent operation of three HPSI pumps, three charging pumps, and all pressurizer backup heaters in operation. Since SIAS starts only two HPSI pumps, a 20% margin is realized.

The restrictions on starting a reactor coolant pump in MODE 4 and with the reactor coolant loops filled in MODE 5, with one or more RCS cold legs less than or equal to 272°F, are provided in Specification 3.4.1.3 and 3.4.1.4 to prevent RCS pressure transients caused by energy additions from the secondary system which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 100°F above each of the RCS cold leg temperatures. Maintaining the steam generator less than 100°F above each of the Reactor Coolant System cold leg temperatures (even with the RCS filled solid) or maintaining a large surge volume in the pressurizer ensures that this transient is less severe than the limiting transient considered above.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 106 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 December 14, 1993, as supplemented by letter dated March 3, 1995, Entergy Operations, Inc. (the licensee), submitted a request for changes to the Waterford Steam Electric Station, Unit 3, Technical Specifications (TSs). The requested changes would revise pressure-temperature (P-T) limits in the Waterford Unit 3 TSs. The licensee revised the P-T limits for the unit and requested to extend the applicable period of the P-T limits from a current eight effective full-power years (EFPY) to 15 EFPY. The requested changes will also remove the reactor vessel material specimen withdrawal schedule from the TSs. In addition, the licensee has requested to revise the low temperature overpressure protection enable temperature value and the related TSs.

2.0 BACKGROUND

Section 182a of the Atomic Energy Act (the "Act") requires applicants for nuclear power plant operating licenses to state TSs to be included as part of the license. The Commission's regulatory requirements related to the content of TSs are set forth in 10 CFR 50.36. That regulation requires that the TSs include items in five specific categories, including (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. However, the regulation does not specify the particular requirements to be included in a plant's TSs.

The Commission has provided guidance for the contents of TSs in its "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" ("Final Policy Statement"), 58 FR 39132 (July 22, 1993), in which the Commission indicated that compliance with the Final Policy Statement satisfies Section 182a of the Act. In particular, the Commission indicated that certain items could be relocated from the TSs to licensee-controlled documents, consistent with the standard enunciated in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). In that

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case, the Atomic Safety and Licensing Appeal Board indicated that "technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety."

Consistent with this approach, the Final Policy Statement identified four criteria to be used in determining whether a particular matter is required to be included in the TSs, as follows: (1) installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary; (2) a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (3) a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (4) a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.[1] As a result, existing TSs requirements which fall within or satisfy any of the criteria in the Final Policy Statement must be retained in the TSs, while those TS requirements which do not fall within or satisfy these criteria may be relocated to other, licensee-controlled documents.

3.0 EVALUATION

A. Reactor Coolant System P-T Limits

The staff evaluates the P-T limits based on the following NRC regulations and guidance: Appendix G to 10 CFR Part 50; Generic Letters (GLs) 88-11 and 92-01; Regulatory Guide (RG) 1.99, Rev. 2; and Standard Review Plan (SRP) Section 5.3.2. Appendix G to 10 CFR Part 50 requires that P-T limits for the reactor vessel must be at least as conservative as those obtained by Appendix G to Section III of the ASME Code. GL 88-11 requires that licensees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation by calculating adjusted reference temperature (ART) of reactor vessel materials. The ART is defined as the sum of initial nil-ductility transition reference temperature (RT_{NDT}) of the material, the increase in RT_{NDT} caused by neutron irradiation, and a margin to account for uncertainties in

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[1] The Commission recently promulgated a proposed change to §50.36, pursuant to which the rule would be amended to codify and incorporate these criteria (59 FR 48180, September 20, 1994). The Commission's Final Policy Statement specified that LCOs for Reactor Core Isolation Cooling, Isolation Condenser, Residual Heat Removal, Standby Liquid Control, and Recirculation Pump Trip are included in the TS under Criterion 4 (58 FR 39132). The Commission has solicited public comments on the scope of Criterion 4, in the pending rulemaking.

the prediction method. The increase in RT_{NDT} is calculated from the product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the vessel material. GL 92-01 requires licensees to submit reactor vessel materials data, which the staff will use in the review of the P-T limits submittals. SRP 5.3.2 provides guidance on calculation of the P-T limits using linear elastic fracture mechanics methodology specified in Appendix G to Section III of the ASME Code.

For the Waterford Unit 3 reactor vessel, the licensee determined that lower shell plate M-1004-2, is the limiting material at the 1/4T location and 3/4T locations (T = the thickness of the reactor vessel beltline). The licensee calculated an ART of 65.4°F at the 1/4T location and 54.0°F at the 3/4T location. The calculation variables are shown in the attached table.

The staff verified the copper and nickel contents and initial RT_{NDT} of plate M-1004-2 with respect to the NRC reactor vessel material submitted by the licensee in response to GL 92-01. The staff also reviewed the fluence value used by the licensee in the revision of the P/T curves which is discussed below.

The proposed revisions were based on a ABB Combustion Engineering report C-MECH-ER-021, Rev.00 which was attachment C to December 14, 1993 letter. This report used the fluence values from surveillance capsule report BAW-2177, "Analysis of Capsule W-97 Entergy Operations, Inc. Waterford Generating Station , Unit No. 3," for the evaluation of capsule W-97. The W-97 capsule report was submitted to the NRC by licensee's letter dated November 25, 1992.

Capsule W-97 was irradiated in Cycles 1 through 4 for 4.44 EFPY of operation, at a location 7 degrees off a major horizontal axis. The analytical methodology utilized the DOT-IV two dimensional code in (r,θ) geometry with ENDF/B-IV based cross sections. A P_3 angular scattering and a S_8 angular quadrature approximations were used. Pin wise neutron sources were used which were provided by the licensee. However, the capsule report normalized the calculated values to the measured dosimeter activity, but no attempt was made to determine the associated uncertainty. In addition, the initial requested extension was about 16 EFPY, (which is much longer than the average extension request) based on one capsule measurement for which the uncertainty was not estimated. The staff requested the licensee to justify the extension in terms of the potential uncertainty by letter dated January 11, 1995. However, in lieu of estimating the uncertainty the licensee elected to lower the period of applicability to 15 EFPY, while the fluence estimate remained unchanged i.e., for 20 EFPY. The estimated fluence at 20 EFPY is about 25 percent higher than the estimated fluence at 15 EFPY. The 25 percent compensate for the potential error due to the normalization of the estimate to the measured dosimetry. We find the proposed period of performance conservative and thus, acceptable.

The staff performed an independent calculation of the ART for the reactor vessel materials using RG 1.99, Revision 2. Based on the staff's calculation of the ARTs values, the staff is in agreement with the licensee's calculations.

Substituting the limiting ARTs in the attached table into equations in SRP 5.3.2, the staff verified that the proposed P-T limits for heatup, cooldown, criticality, and inservice hydrostatic test satisfy the requirements in Paragraphs IV.A.2 & IV.A.3 of Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange RT_{NDT} of 20°F, the staff has determined that the proposed P-T limits have satisfied the requirement for the closure flange region during normal operation, hydrostatic pressure test and leak test.

Based on above discussion, the staff concludes that the proposed P-T limits for heatup, cooldown, inservice hydrostatic test, and criticality are valid for 15 EFPY because the limits conform to the requirements of Appendix G of 10 CFR Part 50 and GL 88-11. Hence, the proposed P-T limits may be incorporated in the Waterford Unit 3 TSs.

Changes to the text of the TSs which related to P-T limits were editorial in nature, and found to be acceptable alterations.

B. Removal of Reactor Vessel Material Specimen Withdrawal Schedule from TSs

As a part of the amendment, the licensee proposed to remove the surveillance capsule withdrawal schedule from the Waterford Unit 3 TSs. In 1991, the staff issued Generic Letter 91-01 to allow the removal of the withdrawal schedule from the TSs provided that the withdrawal schedule is relocated in a controlled document such as UFSAR and the schedule has satisfied Appendix H to 10 CFR Part 50. The licensee stated that the withdrawal schedule is currently in the Waterford UFSAR. The staff has verified that the withdrawal schedule in the UFSAR satisfies Appendix H to 10 CFR Part 50; therefore, the staff concludes that the removal of the surveillance capsule withdrawal schedule from TSs is acceptable.

On this basis, the staff concludes that these requirements do not need to be controlled by TSs, and changes to the surveillance capsule withdrawal schedule, which will be described in the UFSAR, will adequately be controlled by Appendix H to 10 CFR Part 50. The staff has concluded, therefore, that relocation of the surveillance capsule withdrawal schedule described above is acceptable because (1) their inclusion in TSs is not specifically required by 10 CFR 50.36 or other regulations, (2) the withdrawal schedule requirements are not required to avert an immediate threat to the public health and safety, and (3) changes to these withdrawal schedule requirements will require prior NRC approval in accordance with Appendix H to Part 50.

C. Change In Low Temperature Overpressure Protection Enable Temperature

The licensee has elected to change the low temperature overpressure protection (LTOP) enable temperature from 285° F to 272° F. The calculations were performed such that the new temperature will protect the Appendix G limits. In addition, it complies with Reactor Systems Branch Technical Position 5-2, Rev. 3, in that the enable temperature is defined as the water temperature corresponding to a metal temperature of at least RT_{NDT} + 90° F at the beltline location (1/4t or 3/4t) that is controlling the Appendix G limit. Therefore, we find the new enable temperature of 272° F to be acceptable.

The changes in TSs 3.4.1.3, 3.4.1.4, 3.4.8.3 and Bases, B 3/4 4-10 paragraph 4, involve the change of the LTOP enable temperature from 285° F to 272° F. Because the enable temperature change has been found acceptable we find the proposed TS changes both necessary and acceptable.

3.0 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.

4.0 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 previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (59 FR 2867). 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.

5.0 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.

Attachment: Table

Principal Contributors: L. Lois
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Date: May 8, 1995

Table

ADJUSTED REFERENCE TEMPERATURE FOR

REACTOR VESSEL LIMITING MATERIALS

FOR PRESSURE-TEMPERATURE LIMITS

15 EFFECTIVE FULL POWER YEARS

WATERFORD UNIT 3

Limiting Material: Lower Shell Plate M-1004-2	Cu	Ni	Fluence	ΔRT_{NDT}	Initial RT_{NDT}	Margin	ART
Waterford Unit 3							
(1/4)T	.03	.58	1.36E19	21.7	22	21.7	65.4
(3/4)T	.03	.58	0.48E19	16.0	22	16.0	54.0

ATTACHMENT