September 1, 1995

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

SUBJECT: ISSUANCE OF AMENDMENT NO. 111 TO FACILITY OPERATING LICENSE NPF-38 - WATERFORD STEAM ELECTRIC STATION, UNIT 3 (TAC NO. M91168)

Dear Mr. Barkhurst:

The Commission has issued the enclosed Amendment No. 111 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 9, 1994, as supplemented by letter dated July 25, 1995.

The amendment changes the Appendix A TSs by revising the allowable opening tolerances on the pressurizer safety valves (PSVs) and the main steam line safety valves (MSSVs) from ±1% to ±3%. However, following testing, the as-left lift setting of the PSVs and MSSVs will be within ±1% of the pressure specified in the TSs.

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. 111 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

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Chandu P. Patel

Chandu P. Patel, Project Manager
Project Directorate IV-1
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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. 111
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 9, 1994, as supplemented by letter dated July 25, 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.

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. 111 , 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 to be implemented within 60 days.

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 1, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 111

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
3/4 4-7	3/4 4-7
3/4 4-8	3/4 4-8
3/4 7-1	3/4 7-1
3/4 7-2	3/4 7-2
3/4 7-3	3/4 7-3
3/4 7-4*	3/4 7-4*

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 psia ± 3%.*

APPLICABILITY: MODE 4.

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes (except cooldown in shutdown cooling) and place an OPERABLE shutdown cooling loop into operation.

SURVEILLANCE REQUIREMENTS

4.4.2.1 Verify each required pressurizer code safety valve is OPERABLE in accordance with Specification 4.0.5. Following testing, lift settings shall be within ± 1%.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2500 psia ± 3%.*

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 Verify each required pressurizer code safety valve is OPERABLE in accordance with Specification 4.0.5. Following testing, lift settings shall be within ± 1%.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves shall be OPERABLE with lift settings as specified in Table 3.7-1.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With both reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Linear Power Level-High trip setpoint is reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 Verify each required main steam line code safety valve lift setpoint per Table 3.7-1 in accordance with Specification 4.0.5. Following testing, lift settings shall be within ± 1%.

TABLE 3.7-1

STEAM LINE SAFETY VALVES PER LOOP

	VALVE NUMBER		LIFT SETTING (± 3%)*	ORIFICE SIZE
	Line No. 1	Line No. 2		
a.	2MS-R613A (MS-106A)	2MS-R619B (MS-106B)	1070 psig	26 in^2
b.	2MS-R614A (MS-108A)	2MS-R620B (MS-108B)	1085 psig	26 in^2
c.	2MS-R615A (MS-110A)	2MS-R621B (MS-110B)	1100 psig	26 in^2
d.	2MS-R616A (MS-112A)	2MS-R622B (MS-112B)	1115 psig	26 in^2
e.	2MS-R617A (MS-113A)	2MS-R623B (MS-113B)	1125 psig	26 in^2
f.	2MS-R618A (MS-114A)	2MS-R624B (MS-114B)	1135 psig	26 in^2

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

TABLE 3.7-2

MAXIMUM ALLOWABLE LINEAR POWER LEVEL-HIGH TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING OPERATION WITH BOTH STEAM GENERATORS

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE LINEAR POWER LEVEL-HIGH TRIP SETPOINT (PERCENT OF RATED THERMAL POWER)
1	86.8
2	69.4
3	52.1

EMERGENCY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator emergency feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two feedwater pumps, each capable of being powered from separate OPERABLE emergency busses, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one emergency feedwater pump inoperable, restore the required emergency feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two emergency feedwater pumps inoperable be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three emergency feedwater pumps inoperable, immediately initiate corrective action to restore at least one emergency feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2 The emergency feedwater system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 92 days on a STAGGERED TEST BASIS by:

1. Verifying that each motor-driven pump develops a discharge pressure of greater than or equal to 1298 psig on recirculation flow.

2. Verifying that the turbine-driven pump develops a discharge pressure of greater than or equal to 1342 psig on recirculation flow when the steam generator pressure is greater than 750 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 111 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 9, 1994, as supplemented by letter dated July 25, 1995, 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 the allowable opening tolerances on the pressurizer safety valves (PSVs) and the main steam line safety valves (MSSVs) from ±1% to ±3%. However, following testing, the as-left lift setting of the PSVs and MSSVs will be within ±1% of the pressure specified in the TSs. At Waterford 3, there are a total of 12 MSSVs (i.e., six per main steam line), each set at increments which range from 1070 psig to 1135 psig, and there are two PSVs with a lift setting of 2500 psia.

The July 25, 1995, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the original Federal Register notice.

2.0 EVALUATION

TSs 3.4.2.1 and 3.4.2.2 contain requirements for PSVs operability with lift setting of 2500 psia ±1%; TS 3.7.1.1 contains the MSSVs operability requirements with reference to the lift settings specified in Table 3.7-1, which allows a ±1% tolerance. The 1% allowed tolerance on the PSVs and MSSVs has been occasionally exceeded during past surveillance testing. To accommodate setpoint drift that may occur with these valves during plant operation, Waterford 3 requested to increase the setpoint tolerance from ±1% to ±3%.

All of the transient and accident analyses documented in the updated final safety analysis report (UFSAR) were evaluated by licensee to determine the impact of the proposed changes to the TSs. For the cases where the TSs changes had an adverse impact on event consequences, a detailed evaluation or reanalysis of the limiting events has been performed by the licensee. The licensee indicated that the setpoint tolerance change impacts UFSAR analyses with respect to reactor coolant system (RCS) overpressurization, steam generator overpressurization, required overpower margin and peak clad temperature criteria.

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A. Consequences of increasing PSV and MSSV setpoint tolerance from +1% to +3%

The licensee's submittal included the results of the impact of the simultaneous increase in PSVs and MSSVs tolerance from ±1% to ±3% for the loss of condenser vacuum (LOCV) with a single failure, and the feedwater line break (large and small) events. The licensee concluded that the peak RCS and peak secondary pressures remain within the acceptable limits (i.e., less than 110% of the RCS design pressure and less than 110% of the steam generator design pressure) with simultaneous +3% tolerance for PSV and MSSVs. The evaluation demonstrated that the acceptance criteria continued to be met.

The increase in MSSVs lift pressure also adversely impacts the required overpower margin (ROPM) for some control element assembly (CEA) misoperation events. The increase in secondary pressure and temperature results in a lower primary to secondary heat transfer and in turn higher primary temperature. The higher primary temperature has an adverse impact on the CEA misoperation events in the presence of a positive moderator temperature coefficient (MTC). The MTC is a major contributor to the severity of these events. The licensee stated that the impact of the tolerance change on the CEA misoperation events are factored into the core operating limit supervisory system (COLSS) and core protection calculators (CPCs) setpoints at Waterford 3.

The increase in MSSVs lift pressure also adversely impacts the peak clad temperature during a small break LOCA (SBLOCA) event. The increase in MSSV lift pressure results in a higher steam generator (SG) pressure and in turn higher RCS pressure during the limiting SBLOCA event. The higher RCS pressure decreases the safety injection flow and increases break flow, resulting in a higher peak clad temperature. The limiting small break LOCA was analyzed by ABB-CE. The analysis resulted in a peak clad temperature higher than the result in the current analysis in UFSAR for Waterford 3, but within the acceptable limit and lower than the peak clad temperature for the large break LOCA event.

The limiting event for the peak secondary pressure (LOCV) was analyzed by licensee with a MSSV opening setpoint tolerance of +3%. This event was analyzed with 1, 2, 3, and 4 MSSVs inoperable respectively, to confirm the validity of the TS Table 3.7-2, "Maximum Allowable Linear Power Level High Trip Setpoint With Inoperable Steam Line Safety Valves During Operation With Both Steam Generators." The analysis for the cases with 1, 2, and 3 MSSVs inoperable per operable SG, resulted in acceptable peak SG pressure, however, for the case with four inoperable MSSVs per operable SG, the secondary peak pressure slightly exceeded (by 1 psi) the peak SG pressure acceptance criteria (110% of the design pressure, 1210 psia). Based on the above, the licensee requested to modify TS Table 3.7-2 to remove the option for the four inoperable MSSVs from the TSs.

B. PSV setpoint tolerance change from -1% to -3%

The licensee indicated that, this change does not adversely impact any of the previously analyzed events. Therefore, no event had to be reevaluated for this change. The concern with the PSV opening at -3% of the nominal setpoint (2425 psia) is that the PSV may open prior to, and interfere, with the

Pressurizer Pressure-High Reactor Trip, resulting in more severe consequences. The Pressurizer Pressure-High Reactor Trip Setpoint assumed in the analyses is the current TS limit plus a conservative instrument uncertainty based on the limiting accident conditions. The current TS limit for the Pressurizer Pressure-High Reactor Trip Setpoint is 2365 psia with an Allowable Value of 2372 psia. Additionally, by letter dated June 22, 1994, licensee has proposed a Pressurizer Pressure-High Reactor Trip Setpoint of 2350 psia and an Allowable Value of 2359 psia. When that request is approved by the staff under separate correspondence, it will provide additional separation between the PSV opening and the Pressurizer Pressure-High Reactor Trip. Thus, the licensee believes that sufficient separation exists between the minimum allowed PSV opening setpoint and the Pressurizer Pressure-High Reactor Trip Setpoint.

C. MSSV setpoint tolerance change from -1% to -3%

This change primarily impacts the UFSAR reported secondary steam release through the MSSVs due to the earlier opening of the MSSVs and the corresponding dose results. The impact of this change on all of the UFSAR analyses were evaluated by the licensee and found to be insignificant. The event that was impacted the most is the steam generator tube rupture (SGTR) concurrent with loss of offsite power. The total increase in offsite dose for this event is found to be about 0.22 Rem. The licensee stated that, this small increase in dose does not exceed the acceptance criteria of 10 CFR Part 100.

The proposed changes in the TSs include the provision that the PSVs and MSSVs will be tested in accordance with the requirements of Section XI of the ASME Code. In the event an MSSV or PSV lifts outside the setpoint tolerance values, the Section XI provisions for adjusting the setpoint and testing additional valves will apply.

As discussed above, the licensee has determined that the proposed TS changes do not result in a significant reduction in the margin of safety. The limiting transient in each accident category has been analyzed to determine the effect of the change in the setpoint tolerances. Further, in order to prevent the setpoints from drifting outside the ±3% range, the licensee will continue to require MSSV and PSV setpoint tolerances to be restored to ±1% following the testing. This will prevent excessive setpoint drift which would cause the peak system pressures to exceed the allowable limits.

The staff has reviewed the licensee's submittals and agrees with their conclusion that the analysis demonstrates the acceptability of the proposed TS changes. The proposed increase in the setpoint tolerances for the PSVs and the MSSVs has been shown to be acceptable for meeting the plant design basis. Also, for those occurrences where the as-found setpoints of PSVs and MSSVs are in excess of ± 1%, resetting to within ±1% of the nominal setpoint will be required following testing. In addition, the proposed changes to the TSs are consistent with the requirements of the Improved Standard Technical Specifications found in NUREG-1432. Therefore, these proposed TS changes have no significant safety impact on the operation of Waterford 3, and are 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 (60 FR 6300). 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.

Principal Contributor: C. Patel

Date: September 1, 1995