September 18, 1992

Docket No. 50-382

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

Dear Mr. Barkhurst:

SUBJECT: ISSUANCE OF AMENDMENT NO. 76 TO FACILITY OPERATING LICENSE NPF-38 - WATERFORD STEAM ELECTRIC STATION, UNIT 3 (TAC NO. M83314)

The Commission has issued the enclosed Amendment No. 76 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 April 24, 1992, as supplemented by letter dated August 27, 1992.

The amendment changes the Appendix A Technical Specifications by increasing the time for closure of the main steam isolation valves.

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:
David L. Wigginton, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 76 to NPF-38
2. Safety Evaluation

cc w/enclosures:
See next page

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

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David L. Wigginton, Senior Project Manager
Project Directorate IV-1
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

WATERFORD STEAM ELECTRIC STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 76
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 April 24, 1992, as supplemented by letter dated August 27, 1992, 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. 76 , 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

John T. Larkins, Director
Project Directorate IV-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 18, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 76

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 7-9 | 3/4 7-9 |
| 3/4 3-23 | 3/4 3-23 |

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

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

ACTION:

MODE 1

With one main steam line isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, be in at least HOT STANDBY within the next 6 hours.

MODES 2, 3, and 4

With one main steam line isolation valve inoperable, subsequent operation in MODE 2, 3, or 4 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 4.0 seconds when tested pursuant to Specification 4.0.5.

## 3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

### LIMITING CONDITION FOR OPERATION

3.7.2 The temperature of the secondary coolant in the steam generators shall be greater than 115°F when the pressure of the secondary coolant is greater than 210 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure to less than or equal to 210 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

### SURVEILLANCE REQUIREMENTS

4.7.2 The pressure of the steam generators shall be determined to be less than 210 psig at least once per hour when the temperature of the secondary coolant is less than 115°F.

## 3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

### LIMITING CONDITION FOR OPERATION

3.7.2 The temperature of the secondary coolant in the steam generators shall be greater than 115°F when the pressure of the secondary coolant is greater than 210 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure to less than or equal to 210 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

### SURVEILLANCE REQUIREMENTS

4.7.2 The pressure of the steam generators shall be determined to be less than 210 psig at least once per hour when the temperature of the secondary coolant is less than 115°F.

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

| INITIATING SIGNAL AND FUNCTION | RESPONSE TIME IN SECONDS |
|---|---|
| 2. Pressurizer Pressure-Low | |
| a. Safety Injection (ECCS) | |
| (1) High Pressure Safety Injection | ≤ 30.0*/18.5** |
| (2) Low Pressure Safety Injection | ≤ 45.5*/34.0** |
| b. Containment Isolation | ≤ 23.5*/12.0** |
| c. Containment Cooling | ≤ 31.0*/19.5** |
| 3. Containment Pressure-High | |
| a. Safety Injection (ECCS) | |
| (1) High Pressure Safety Injection | ≤ 30.0*/18.5** |
| (2) Low Pressure Safety Injection | ≤ 45.5*/34.0** |
| b. Containment Isolation | ≤ 23.5*/12.0** |
| c. Main Steam Isolation | ≤ 5.0*/5.0** |
| d. Main Feedwater Isolation | ≤ 6.0*/6.0** |
| e. Containment Cooling | ≤ 31.0*/19.5** |
| 4. Containment Pressure--High-High | |
| a. Containment Spray Pump | ≤ 15.2*/2.7** |
| b. Containment Spray Valves | ≤ 11.0*/11.0** |
| c. CCW to RCP Valves | ≤ 23.5*/12.0** |
| 5. Containment Area Radiation-High# | |
| Containment Purge Valves Isolation | ≤ 6.2*/6.2** |
| 6. Steam Generator Pressure-Low | |
| a. Main Steam Isolation | ≤ 5.0*/5.0** |
| b. Main Feedwater Isolation | ≤ 6.0*/6.0** |
| 7. Refueling Water Storage Pool-Low | |
| Containment Sump Recirculation | ≤ 120.0*/108.5** |
| 8. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage) | |
| Loss of Power (0 volts) | ≤ 2*** |
| 9. 480V Emergency Bus Undervoltage (Loss of Voltage) | |
| Loss of Power (0 volts) | N.A. |
| 10. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage) | |
| Loss of Power | ≤ 11*** |

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

| INITIATING SIGNAL AND FUNCTION | RESPONSE TIME IN SECONDS |
|---|---|
| 11. Steam Generator Level-Low | |
| Emergency Feedwater Pump and Block Valves | ≤ 54.0*/42.0** |
| 12. Wide Range Steam Generator Level-Low | |
| Emergency Feedwater Control Valves | ≤ 25.0*/25.0** |
| NOTE: Response time for all Motor-Driven and Steam-Driven Emergency Feedwater Pumps on all ESF signal starts. | ≤ 54.0 |

TABLE NOTATIONS

*Diesel generator starting and sequence loading delays included. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

**Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

***Response time measured from the sensing relay to the channel output only.

#Response time does not include the detector.

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

| INITIATING SIGNAL AND FUNCTION | RESPONSE TIME IN SECONDS |
|---|---|
| 11. Steam Generator Level-Low | |
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| NOTE: Response time for all Motor-Driven and Steam-Driven Emergency Feedwater Pumps on all ESF signal starts. | ≤ 54.0 |

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**Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

***Response time measured from the sensing relay to the channel output only.

#Response time does not include the detector.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 76 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 April 24, 1992, as supplemented by letter dated August 27, 1992, Entergy Operations, Inc. (the licensee) submitted a proposal to modify Waterford 3 Technical Specification (TS) Surveillance Requirement 4.7.1.5, "Main Steam Isolation Valves," and Table 3.3-5, "Engineered Safety Features Response Time."

This modification proposes to increase the allowed Main Steam Isolation Valve (MSIV) closure time from 3 seconds to 4 seconds during surveillance testing.

In response to questions to the licensee, Entergy Operations, Inc., submitted a letter dated August 27, 1992, to clarify the reasons for the increase and further state the basis for the codes/methods used in the analysis. This information was for clarification and does not change or affect the staff's no significant hazards consideration published on June 24, 1992 (57 FR 28200).

## 2.0 EVALUATION

MSIV closure time is significant for several events analyzed in the Final Safety Analysis Report (FSAR) Chapter 6.2 and Chapter 15. These analyses can be categorized into two groups based on the assumed MSIV closure time being greater than or less than 4 seconds. The events analyzed with a MSIV closure time greater than 4 seconds are not affected by this proposed change. Those events are:

- Full Power Double Ended Steam Line Break (SLB) Inside Containment with Concurrent Loss of Offsite Power
- Double Ended Steam Line Break Inside and Outside Containment During Mode 3 Operation with Concurrent Loss of Offsite Power
- Steam System Piping Failure: Pre-Trip Power Excursion Analysis Outside Containment with Loss of Offsite Power.

The analyses that use less than a 4-second MSIV closure time were evaluated in detail by the licensee and are discussed in the sections that follow.

2.1 Main Steam Line Break (FSAR Section 6.2)

The existing main steam line break (MSLB) analyses, which are the limiting analyses for the MSIV closure time, are located in Section 6.2 of the FSAR. They assume that the MSIV remains fully open and then closes instantaneously 3.0 seconds after receiving a signal to close. The resulting mass and energy release cause an increase in containment temperature and pressure.

The licensee reanalyzed the mass and energy released to containment for the three most limiting, in terms of peak containment temperature and pressure, MSLB cases using a 4.0 second, instead of a 3.0 second, MSIV closure time. These reanalyses use a more detailed representation of the steam line piping and pressure drop from the steam generator to the MSIV while using the same code/method as used in the original analysis. In addition, the licensee took credit for the decrease in the MSIV flow area during the 4.0 seconds the valve is closing instead of assuming instantaneous closure at the end of 3.0 seconds. Consequently, less mass and energy was calculated to be released to the containment for all three cases when a 4.0 second MSIV closure time was used.

An analysis of containment peak temperature and pressure using the revised mass and energy releases would result in lower peak values. Therefore, the current MSLB analyses in the FSAR bounds the peak containment temperature and pressure that would be calculated with the revised 4.0 second MSIV closure time.

The three most limiting MSLB cases used by the licensee were 75% power with a containment cooling train failure, 102% power with a containment cooling train failure, and 75% power with the failure of one MSIV to close.

NRC Information Notice 91-069, "Errors In Main Steam Line Break Analyses For Determining Containment Parameters," addressed the consequences that may result from not considering the possibility that feedwater could continue to flow following a MSLB inside containment. This could lead to a licensee choosing a MSLB case that does not maximize the post-accident containment temperature and pressure.

Section 6.2.1.4 of the licensee's Final Safety Analysis Report states in part that "A single failure of one main feedwater isolation valve during a postulated main steam line break accident is accommodated by closure of the main feedwater regulating valve upon receipt of a main steam isolation signal (MSIS). Thus, assuming a single failure, feedwater flow to the affected steam generator is terminated by the closure of the main feedwater regulating valves within 5 seconds after receipt of signal." Therefore, Waterford 3 is not susceptible to the condition described in Information Notice 91-069.

2.2 Increased Main Steam Flow Due to One Turbine Bypass Valve Failing in the Open Position (FSAR Section 15.1.1.3)

During this transient the steam flow increases, causing excess heat removal from the reactor coolant system (RCS). The excess heat removal reduces RCS temperature and pressure and increases core power due to a negative moderator temperature coefficient (MTC). The increase in core power decreases the departure from nucleate boiling ratio (DNBR) such that at 18.2 seconds after the initiating event, a low DNBR reactor trip signal is generated. At 255.2 seconds the low steam generator pressure generates a MSIS which closes the MSIVs 2 seconds later.

The times of interest in this event are the time of minimum DNBR and the time of reactor trip. The MSIVs close long after (255.2 seconds) those events, therefore, increasing the closure time from 3 seconds to 4 seconds will not affect plant response.

2.3 A Steam Line Break at Hot Zero Power Outside Containment with Concurrent Loss of Offsite Power (FSAR Section 15.1.3)

In this event, the main steam line is assumed to rupture upstream of the MSIV shortly after a shutdown from full-power operation. This transient causes a large steam generator mass release and radiological consequences that are bounded by the full-power steam piping failure event outside containment: Pre-Trip Power Excursion with Loss of Offsite Power (FSAR Section 15.1.3.3). The latter event is analyzed with an MSIV closure time of almost 11 seconds, bounding the 4 second closure time.

2.4 Feedwater System Pipe Break with Loss of Offsite Power (FSAR 15.2.3.1)

During this transient, saturated liquid flows from the affected steam generator through the break causing an instantaneous loss of feedwater to the steam generator. This causes a gradual heat-up of the primary and secondary systems. The ruptured steam generator empties causing a rapid increase in RCS temperature and pressure. A reactor trip on high pressurizer pressure occurs at 15.4 seconds.

With the concurrent loss of normal AC power, the plant will undergo a simultaneous loss of forced reactor coolant flow and a turbine trip. Losing the heat sink and reactor coolant pump flow generates a concern regarding the RCS peak pressure. An increase in MSIV closure time enhances cooling, causing a lower peak pressure. Consequently, the increase in the MSIV closure time to 4 seconds will not adversely affect the consequences of this transient.

2.5 Loss of Normal Feedwater Flow with an Active Failure in the Steam Bypass System (FSAR Section 15.2.3.2)

The loss of feedwater flow increases steam generator pressure and temperature causing the turbine bypass valves to open, increasing steam flow and RCS cooldown rate. When the bypass control valves fail to close in the presence

of a closure signal, the cooldown of the primary system causes core power to increase due to a negative moderator temperature coefficient. The maximum heat flux will cause the reactor to trip at 42.6 seconds on low steam generator water level. At 77.3 seconds a main steam isolation signal (MSIS) is generated due to low steam generator pressure causing the MSIVs to close 3 seconds later.

Changing the MSIV closure time from 3 seconds to 4 seconds will not affect the transient results significantly with respect to DNB or radiological consequences since (1) the MSIV closure occurs after the maximum heat flux and reactor trip and (2) the secondary mass release through the MSIV in the one additional second is not significant. However, the increase in MSIV closure time can cause a slight increase in RCS temperature which will have a negligible effect on the course of the transient.

2.6 Single Reactor Coolant Pump Shaft Seizure with a Stuck Open Secondary Safety Valve (FSAR Section 15.3.3.2)

Approximately 1 second into this transient, a reactor trip signal is generated due to a low DNBR. The main turbine trip and reduction in feedwater flow causes steam generator pressure to increase, which opens the steam generator safety valves. Assuming one safety valve remains open throughout the transient, the steam flow through the valve reduces the pressure in the steam generator. The low steam generator pressure generates a MSIS - at about 700 seconds into the transient - causing the MSIV to close 3 seconds later.

Changing the closure time will not have an effect on thermal margin since the minimum DNBR occurs before the MSIV closure.

3.0 SAFETY CONCLUSION

Based on the review of the FSAR Chapter 6.2 MSLB event, the staff concludes that the proposal to increase the time for MSIV full closure by 1.0 second, as discussed in TS 4.7.1.5 and Table 3.3-5 of the TSs, is acceptable because the mass and energy calculated to be released inside containment with the new closure time is bounded by the original FSAR analysis when credit is taken for flow choking during MSIV closure. This more realistic portrayal of MSIV valve closure is also acceptable.

Based on the review of the Chapter 15 events that have an MSIV closure time of less than four seconds, the staff concludes that the increase of the MSIV closure time from 3 seconds to 4 seconds either (1) has no effect, (2) has a negligible effect, or (3) is bounded by a previously analyzed event. The staff, therefore, finds the licensee's proposal to modify TS Surveillance 4.7.1.5 and Table 3.3-5 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 in 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 (57 FR 28200). 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 Contributors: M. Snodderly
S. Brewer

Date: September 18, 1992