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

March 26, 2004

Mr. Joseph E. Venable
Vice President Operations
Entergy Operations, Inc.
17265 River Road
Killona, LA 70066-0751

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 (WATERFORD 3) - REQUEST FOR ADDITIONAL INFORMATION RELATED TO REVISION TO FACILITY OPERATING LICENSE AND TECHNICAL SPECIFICATIONS - EXTENDED POWER UPRATE REQUEST (TAC NO. MC1355)

Dear Mr. Venable:

By letter dated November 13, 2003, and supplemented by letters dated January 29 and March 4, 2004, Entergy Operations, Inc. proposed revisions to the Waterford 3 operating license and Technical Specifications which would allow an increase in the rated thermal power from 3,441 megawatts thermal (MWt) to 3,716 MWt.

After reviewing your request, the Nuclear Regulatory Commission staff has determined that additional information is required to complete the review. We discussed this information with your staff by telephone and they agreed to provide the additional information requested in the enclosure within 30 days of receipt of this letter.

If you have any questions, please call me at (301) 415-1480.

Sincerely,

N. Kalyanam, Project Manager, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosure: Request for Additional Information

cc w/encl: See next page

Waterford Steam Electric Station, Unit 3

cc:

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REQUEST FOR ADDITIONAL INFORMATION

ENTERGY OPERATIONS, INC.

WATERFORD STEAM ELECTRIC STATION, UNIT 3 (WATERFORD 3)

DOCKET NO. 50-382

Instrumentation and Controls

1. Discuss the instrument setpoint methodology used to calculate trip setpoints and allowable values of the plant parameters affected by the extended power uprate (EPU). If your methodology has not been previously reviewed by the U. S. Nuclear Regulatory Commission (NRC) staff, then submit a copy of the plant instrument setpoint methodology for the staff's review and approval. If you use "method 3" specified in Independent Safety Analysis S67.04.02, then confirm that a check calculation is performed to account for all loop uncertainties not measured during the channel operational test/channel functional test. Please assure that adequate margin exists between the analytical limit and the allowable value that equals or exceeds the value of uncertainties not measured during the channel operational test. Please provide the documentation of the calculation which demonstrates the existence of an adequate margin. Discuss how the channel operability is determined for each of the plant parameters affected by the power uprate.

2. The Waterford 3 License Amendment Request NPF-38-249, EPU, dated November 13, 2003 (the application) states on page 2.13-10 that, as part of the power uprate, the response times for core protection calculator system low departure from nucleate boiling ratio and high low-power density trips were reviewed and enhancements to clarify the time requirements were identified, which included reductions in some of the times required to be assumed in safety analyses. Please identify these response time reduction items and the related safety analysis sections.

3. The application states on page 2.4-1 that the EPU also affects the atmospheric dump valve (ADV) controllers. The existing ADV analog controllers are being replaced with more accurate digital controllers. The ADV controllers perform safety-related functions, and therefore, covered by Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR), Appendix A, General Design Criterion 1, "Quality Standards and Records," which requires, in part, that structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Additionally, 10 CFR 50.55a(h) requires, in part, that protection systems satisfy the criteria of the Institute of Electrical and Electronics Engineers (IEEE) Standard 603, "Criteria for Safety Systems for Nuclear Power Generating Stations." Discuss the modification package of the ADV controllers to be installed at the Waterford 3 and address the following design requirements:

   a. Compliance with IEEE-603 (or IEEE-279) requirements
   b. Software life cycle process planning
   c. Design verification and validation process
   d. Configuration management process

e. Maintenance, testing, and calibration process
f. Environmental considerations, such as electro-magnetic interference, radio-frequency interference, effects of temperature and humidity, etc.

4. If ADV controllers use the commercial software-based devices, the digital components to be used in safety systems must be qualified for their intended application. Address the Waterford 3 plant-specific dedication of commercial grade digital equipment for nuclear safety application with respect to the guidance provided in NRC Standard Review Plan (SRP), Appendix 7.0-A, Section C.3.8, "Review of the Acceptance of Commercial-Grade Digital Equipment," and SRP Section 7.1, "Acceptance Criteria on Supplemental Guidance for Digital Computer-Based Safety Systems."

5. The ADVs are credited for small-break loss-of-coolant accident (SBLOCA) mitigation at greater than 70 percent rated power. A new Technical Specification (TS) was proposed. Because the ADV controllers are digital devices, the common mode failure due to software error should be considered. Additional surveillance may be required after detecting one ADV inoperable. Discuss the adequacy of the proposed TS with respect to the common mode failure concern.

6. The EPU application, in Section 2.13.1.1.4.1, "General Description of the Event," states that one ADV may be inadvertently opened due to operator error or due to a failure in the ADV control system. Analyze the consequence for inadvertently open all ADVs due to a common mode failure at ADV digital controllers.

7. In Attachment 1, "Analysis of Proposed Technical Specification Changes," Section 4.0, states that in an effort to improve clarity for the operators, the word "indicated" or phrase "an indicated" is being added to identify those values in TS that can be compared directly to plant instrument readings to ensure TS compliance. The staff considers that the plant instrument readings from indicators can only be used for channel check to detect a gross failure of an instrument channel. It is not acceptable to be used for TS compliance. There is no assurance that the "indicated" reading is reliable and conservative. The number in the TS should be based on safety analysis, and should meet the requirements of 10 CFR 50.36.(c)(2) to establish the lowest functional capability or performance level of equipment required for safe operation of the facility. Therefore, the word "indicated" or "an indicated" cannot be used in the TS.

8. Section 7.8.3.2 of the final safety analyses report (FSAR), "Diverse Emergency Feedwater Actuation System (DEFAS)," states that the DEFAS actuation signals are interlocked with steam generator pressure. The power uprate requires a setpoint change on "steam generator pressure-low." Verify that the proposed setpoint change does not affect the DEFAS operation or cause inadvertent actuation of the DEFAS.

9. Verify the inconsistency between TS and BASES. For example, TS 3/4.2.6, "Reactor Coolant Cold Leg Temperature," Limiting Condition for Operation 3.2.6 states "The reactor coolant cold leg temperature shall be maintained between 536°F and 549°F." while Bases 3/4.2.6 insert states "The safety analysis assumes that cold leg temperature is maintained between 553°F and 552°F or indicated temperatures of 556°F and 549°F."

Quality and Maintenance

10. Attachment 3 of your letter dated January 29, 2004 (Supplement 1), states that test, "Load Changes," will be performed only at 95 percent power. This test occurred various times between 50 percent and 100 percent power during initial power ascension tests.

    Given the planned modifications on the secondary plant (ADV controller replacement, high pressure main turbine rotor blade replacement, moisture separator reheater relief valve capacity change), please describe how one test will be sufficient to adequately ascertain plant response as specified in FSAR Section 14.2.12.3.31, "Control Systems Checkout."

11. Attachment 4 of Supplement 1 states that test SIT-TP-707, "Steam Bypass Control System (SBCS) Capacity Checks," will not be performed. Given the planned modifications on the ADV controllers and SBCS, provide details on the testing method(s) planned to verify and validate that the ADVs will not actuate prior to the SBCS in the event of a load reject.

12. Attachment 4 of Supplement 1 states that test SIT-TP-724, "Temperature Decalibration Verification," will not be performed. The evaluation/justification for not performing the test states in the first sentence, "Algorithms contained within the core protection calculator systems (CPCS) are unchanged." The next sentence begins, "The update algorithm within CPCS accommodates for a change in cold leg temperature ..." Based on the lack of clarity in the justification, please explain whether the algorithms within the CPCS are changed or unchanged, and if changed, what test(s) will be performed for validation and verification.

13. Attachment 1 of Supplement 1 describes that a modification of the reactor coolant system $T_{average}$ versus pressurizer level program will occur to accommodate a lower reactor coolant system operating temperature. Please provide additional details regarding testing to verify proper operation of affected control systems (i.e. pressurizer level and pressure control).

19. Attachment 1 of Supplement 1 briefly describes the impact of individual modifications on dynamic plant response. The information currently provided in the application is not sufficient for the staff to evaluate how the scope of EPU related modifications, known system interactions, transient behavior of systems important to safety, functional system requirements in response to anticipated operational occurrences (AOOs), and other factors were considered in the aggregate. Please describe the process/methodology used in considering how, in the aggregate, the planned EPU modifications could affect expected system interactions, transient behavior of systems important to safety, functional system requirements in response to AOOs, and other factors which could affect the dynamic response of the plant.

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