June 21, 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, March 4, April 15, May 7, May 12, May 13, May 21, and May 26, 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,

/RA/

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

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REQUEST FOR ADDITIONAL INFORMATION

ENTERGY OPERATIONS, INC. (ENTERGY)

WATERFORD STEAM ELECTRIC STATION, UNIT 3 (WATERFORD 3)

DOCKET NO. 50-382

(The Section numbers in the following questions refer to the section numbers in Attachment 5 to the letter dated November 13, 2003, from Entergy.)

1. Please provide a quantified evaluation of the time needed for plant cooldown to achieve cold shutdown conditions per RSB BTP 5-1 (natural circulation cooldown using only safety grade equipment), and for plant cooldown per the requirements of Appendix R to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50 (regarding fire protection) for Waterford 3 at extended power uprate (EPU) power level and the current power level.

2. Clarify the statement in Section 2.6.4.4 of your EPU submittal, “the current limiting conditions for operations for time based reduced flow rate are acceptable at EPU condition.”

3. Section 2.6.4.3 states that the low temperature overpressure (LTOP) calculations have been revised. Please submit: (1) the mass and energy input transient assumptions and results, (2) the revised LTOP enable setpoint, and (3) the vent capacities for NRC staff review.

4. The small break loss-of-coolant accident (SBLOCA) methodology (S2M) which was used to perform the Waterford 3 analyses for the uprated power does not apply at the requested uprated power, since the sensitivity studies supporting the S2M methodology were performed at a lower power. The sensitivity studies to justify applicability of the S2M at a higher power are plant-specific and do not have generic applicability. Provide justification that the S2M applies to Waterford 3 at the requested uprated power. (Please see the NRC staff safety evaluation report for Palo Verde Nuclear Generating Station fuel transition)

5. Please confirm that the generically approved LOCA analysis methodologies used for the Waterford 3 uprate LOCA analyses continue to apply specifically to the Waterford 3 plant by: 1) showing that Waterford 3 operating at the uprated power is bounded by the assumptions used in analyses used to support the approval of the generic LOCA methodologies identified in the response to Question 4; and 2) providing a statement to confirm that Waterford 3 and its vendor continue to have **ongoing processes** which assure that LOCA **analysis input values bound the as-operated plant values** for those parameters. (The statement should be identical to the one in the question in order to avoid providing extraneous and/or irrelevant information which will not address the question.)

ENCLOSURE

6. What are the calculated large break LOCA and SBLOCA results per 10 CFR 50.46 (b) for Waterford 3 at the uprated power for both the new fuel and the resident fuel? Include in the evaluation of the local oxidation consideration of pre-event and post accident inside clad and outside clad oxidation.

7. Verify that the recently discovered error in the S2M methodology has been fixed in the Waterford 3 LOCA model.

8. Discuss the design of the Waterford 3 emergency core cooling system (ECCS) switch over from the injection mode to the ECCS sump recirculation mode. What was the decay heat source assumed in the design of the ECCS switch over from the injection mode to the ECCS sump recirculation mode for the present power? Does this assumed heat source change for the uprated power? Is the timing of the switch over affected? Please explain.

9. The Waterford 3 SBLOCA analyses take credit for the operation of the steam generator atmospheric dump valves (ADVs). Please show that the ADVs are fully safety grade for this use by identifying all ADV components and supporting systems needed to support the SBLOCA operation, and show that these components and supporting systems are safety grade. (e.g., if the ADVs rely on instrument air, show that the instrument air system is safety grade and has sufficient long term capacity to support repeated cycling of the valves). If the valves are only qualified to open and not re-seat, show how the reactor coolant system (RCS) pressure will be controlled to both keep the core covered and avoid cold overpressure for all break sizes. If stopping of ECCS pumps is involved, please show that the pumps can be re-started if and when needed.

10. a. Your February 5, 2004, Slide 5 stated that RCS flow rate would increase from 44,522 pound mass per second (lbm/sec) to 45,808 lbm/sec. Please provide the basis for this change.

    b. Table 2.6-2 of your submittal states the minimum rate is unchanged from 148.0 X 10^6 pound mass per hour (41,411 lbm/sec) but it also states that the core and reactor vessel differential pressures increase in each of the Table’s three columns and the nominal film coefficient is also shown to increase for the EPU. Please clarify this information with respect to ensuring that the 41,411 lbm/sec is bounding, that the film coefficient information is correct, and with respect to the Item 1.a values.

    c. Please also address the stated core flow rates with respect to the above.

    (We note similar information is provided elsewhere in your submittal, such as in Table 2.12-1.)

11. There are a number of references where the number of ADVs credited for the licensing bases is increased to two. Please address how this meets applicable single failure criteria.

12. What was the previously assumed numerical value of the volume in which boric acid accumulates for long term cooling analysis? What is the numerical value assumed for the EPU?

13. Please confirm that the Table 2.12-12 values for the boric acid makeup tanks, refueling water storage pool, and safety injection tanks are those used in the long term cooling analyses.

14. Please state the time by which the procedure(s) will reasonably ensure initiation of hot leg injection and provide a copy of the procedure(s). Provide the basis for the stated time.

15. In your evaluation of non-LOCA transient analyses, there are a few events for which the following conclusion has been drawn, "The analysis has been evaluated for EPU, the final safety analysis report (FSAR) results remain bounding, and a complete reanalysis was not required." Please provide a quantitative evaluation of these events to show that the consequences of these events at EPU conditions are bounded by the current analysis in FSAR.

16. CENPD-282-P-A, Technical Manual for the CENTS Code, is referenced in W3F1-2003-0074 (November 13, 2003). Since then, the NRC staff has reviewed and accepted WCAP-15996-P, Technical Description Manual for the CENTS Code (on December 1, 2003), which includes certain updates to the CENTS code. Does the version of the CENTS code that has been used in the non-LOCA analyses include any updates made since 1995? If so, then the updated CENTS technical manual should be cited.

17. Why is NUREG-75/087 (reference 2.13-14) cited, and not NUREG-0800?

18. Where is Figure 2.13-1 (moderator cooldown curve)?

19. Do the classes of moderate frequency incidents, infrequent incidents, and limiting faults correspond to Condition II, III, and IV events of American National Standards Institute N18.2? What are the acceptance criteria that are applied in the EPU analyses and evaluations for the classes of moderate frequency incidents, infrequent incidents, and limiting faults?

20. Please provide a tabulation to indicate that for each event, what specific acceptance criteria are satisfied, to demonstrate that the general acceptance criteria of the event's class are met?

21. For all events, in which a reactor trip is assumed to occur, does the negative reactivity insertion account for the most reactive control element assembly (CEA) being stuck in its fully withdrawn position?

22. In all analyses of post-trip thermal margin, especially in the steam line break (SLB) cases, is the minimum departure from nucleate boiling ratio (DNBR) calculated in the region of the assumed stuck CEA? How is that done?

23. In 2.13.0.2, Initial Conditions, it is noted that non-safety grade systems, that would act to mitigate a transient were not credited. Were any non-safety grade systems, that would act to aggravate a transient, credited?

24. How is decay heat determined and applied in the applicable analyses? What standard is used?

25. Are all the accident analyses and evaluations, presented in the application, cycle-independent? Are all the accident analyses and evaluations that bound certain events of this application also cycle-independent?

26. For event analyses that bound event analyses of different categories (e.g., reactor coolant pump shaft seizure, a Limiting Fault event, bounds a partial loss of forced reactor coolant flow, an Infrequent event), please identify the specific results and criteria that are compared in order to reach the bounding conclusion.

27. Table 2.13.0-2 indicates that the lower limit of the pressurizer safety valve (PSV) setpoint is 2425 pounds per square absolute (psia). Table 2.13.0-3 indicates that the reactor protection system (RPS) analytical setpoint for high pressurizer pressure is 2422 psia. This RPS setpoint could be much higher considering instrument uncertainties. Please discuss the consequences of a potential lifting of the PSV prior to RPS actuation, which would prevent a reactor trip from occurring.

28. The steam generator tube rupture (SGTR) analysis which assumed that a loss-of-offsite power (LOOP) occurs three seconds following reactor trip is non-conservative for the radiological consequences. This assumption is not consistent with the current licensing basis at Waterford. Please provide the results of a SGTR analysis assuming a LOOP occurs at the events initiation.

29. For a SGTR accident, the most limiting single failure is to assume a stuck open ADV on the failed steam generator after it is automatically open following the event. Please explain why this assumption is not reflected in the sequence of event provided for this event.

30. Table 2.13.0-1 indicates that the pressurizer safety valve lift transient is categorized as a "limiting fault" and bounded by the SBLOCA. Standard Review Plan 15.6.1 categorizes this event as a event of moderate frequency with the acceptance criteria associated with an event with moderate frequency occurrence. Please provide the results of an analysis for this event at EPU conditions to demonstrate that these acceptance criteria are met. Based upon its frequency of occurrence during "more than 260 pressurizer safety valve years of operation", and the observation that it could only be caused by a passive mechanical failure, does operating experience support the classification of this event as a faulted condition?

31. Consider the event where one or both pressurizer safety valves were to open during a moderate frequency event (e.g., loss of condenser vacuum), and then fail to reseat properly. If this failure rate were to be high enough, then the analysis of the moderate frequency event would have to account for the effects of an open pressurizer safety valve. What failure rate has been assumed for the proper reseating of pressurizer safety valves in the analyses of events

32. Table 2.13.0-1 indicates that the Increased Steam Flow event (2.13.1.1.3) is analyzed as a moderate frequency event. The acceptance criteria, inter alia, specify that the resulting radiological dose must be less than or equal to a small fraction of 10 CFR 100 limits. Please quantify "small fraction". How does this radiological dose limit compare with the requirements of paragraph 20.1 of 10 CFR 20?

33. Section 2.13.1.1.3.1 states that any one of the following events may cause an increase in steam flow:

    a) Inadvertent opening of the turbine admission valves. (approximately a 11% increase of the full power turbine flow rate)

    b) Failure in the Steam Bypass System that could result in an opening of one steam bypass valve. (approximately 12.3% of the full power turbine flow rate)

    c) Inadvertent opening of an ADV or SG safety valve. Each dump valve can release approximately 5.3% of the full-power steam flow, and the safety valve can pass approximately 9.3% of full power steam flow.

    Failure of a steam bypass valve is declared to be the most adverse event. How has this determination been made? Were the reactivity effects of asymmetric core cooling, caused by the opening of one SG safety valve, considered?

34. Define RTP (rated thermal power) SLB.

35. A major difference between the analysis results of the EPU SLB and the analysis results of current SLB is that the DNBR specified acceptable fuel design limit (SAFDL) is violated (Tables 2.13.1.3.1-3 and 2.13.1.3.1-4). How is the extent of fuel pin failure (e.g., 2%) determined?

36. Why is loss-of-normal feedwater flow (Section 2.13.2.2.5), considered to be an Infrequent Event, and not a Moderate Frequency Event?

37. Why is total loss-of-forced reactor coolant flow (Section 2.13.3.2.1), considered to be an Infrequent Event, and not a Moderate Frequency Event?

38. Why is Inadvertent loading of a fuel assembly into an improper position (Section 2.13.4.3.1), considered to be a Limiting Fault, and not an Infrequent Event?

39. Why is SGTR (Section 2.13.6.3.2), considered to be a Limiting Fault, and not an Infrequent or Moderate Frequency Event?

40. Why is small primary line break outside containment (Section 2.13.6.3.1), considered to be a Limiting Fault, and not an Infrequent Event?

41. Section 2.6.1.3.1.1, Thermal Margin Analysis, indicates that the Modified Statistical Combination of Uncertainties (MSCU) methodology is applied in the analyses. The minimum DNBR SAFDL would be 1.26, as listed in the current Technical Specifications. However, the FSAR still refers to the prior DNBR SAFDL of 1.19 (e.g., in Section

15.3.1.1, Partial Loss of Reactor Coolant Flow). When Amendment No. 183 was issued, on March 29, 2002, the Safety Evaluation Report noted that the FSAR had not been updated, and advised the applicant to update the FSAR, in accordance to the requirements of 10 CFR 50.71. Please make the necessary updates to the FSAR. Please indicate the minimum DNBR SAFDL and the calculated minimum DNBR for all applicable accident analyses. Please verify that all events that are bounded by FSAR analyses, with respect to thermal margin, are comparable to FSAR analyses that applied the MSCU method.

42. Please provide a quantified evaluation of the impacts of the EPU to a core power level of 3716 megawatts thermal (MWt) on the ability of Waterford 3 to cope with a Station Blackout (SBO) event. The evaluation should address the capacities of the condensate storage tank, turbine driven auxiliary feedwater pump, station batteries, and backup air supplies for air operated valves for decay heat removal and RCS cooldown during the time period of an SBO.

43. To support the results of the loss of normal feedwater transient, please provide the following information:

    1. Discuss the need for a time delay of emergency feedwater (EFW) flow to steam generators while the plant is operated below 15% rated power.
    2. The results of a loss of normal feedwater transient assuming that the EFW flow is delivered within one minute following the event to show the effect of overcooling at the beginning of the transient.
    3. Discuss the provisions made in plant emergency operating procedures (EOPs) for controlling EFW at the beginning of the event to prevent excess cooldown during this event.
    4. Discuss the phenomena involved that causes the RCS pressure to peak and then decrease prior to EFW flow being delivered to steam generators.

44. Please confirm that the event scenario of the SGTR thermal-hydraulic analysis is consistent with EOPs at Waterford 3.

45. Provide the results of a SGTR thermal-hydraulic analysis to demonstrate that the SG will not be overfilled by EFW flow during this event.

46. Please provide a tabulation of all computer codes and methodologies used in the re-analyses and indicate the staff approval status, any conditions and limitations on their use, and how the limitations are satisfied for application at Waterford 3.

47. Provide a tabulation of the thermal design parameters and compare them to the values assumed in safety analyses to demonstrate that the safety analyses assumptions are conservative.

48. Expand Table 2.13.0-2 to include all primary and secondary parameters used in the non-LOCA transients.

49. The reanalysis of the increased main steam flow transient assumes an initial pressurizer level at the upper limit of 67.5%. Please discuss the consequences if the lower limit of

21% is assumed in this analysis. Will the pressurizer be emptied much earlier in the sequence of event and cause loss of pressure control to RCS?

50. Tables 2.6-3 through 2.6-7 listed nuclear steam supply system design transients for Waterford 3. Please confirm that these design transients are applicable for the current core power level conditions and that they are unchanged for the EPU conditions.

51. Please confirm that only safety grade systems and components are credited in the reanalysis of all transients and accidents in your EPU report for Waterford 3.

52. Provide a more detailed rationale for your selection of initial plant conditions for each transient analyzed to achieve the most conservative results.

53. Please discuss the significance of assuming an initial power at 1 MWt for the analysis of an inadvertent opening of a steam generator ADV.

54. To support the results of the reanalysis for the increased main steam flow with LOOP, and sheared shaft with LOOP, please provide the calculated amount of fuel pins with their minimum DNBR (MDNBR) below the allowable MDNBR of 1.26 in each event analyzed. Compare the amount of fuel failure with the acceptance criteria for these events.

55. To support the reanalysis of the main SLB accident with LOOP, please provide the following: 1) the calculated amount of fuel pins with their MDNBR below the allowable MDNBR of 1.26 for the cases with a break inside containment; and 2) transient curves for the cases with a break outside the containment.

56. For the loss of condenser vacuum transient, please provide the following: 1) the sequence of events for the peak primary pressure case and the peak secondary pressure case; and 2) a separate set of transient curves for each case analyzed.

57. Discuss why the assumed break size for a main feedwater line break (MFLB) accident is different from that in the current licensing analyses. Provide a discussion of the break size assumed for a large MFLB relative to the double ended break of a main feedwater pipe.

58. The proposed TS 4.7.1.5.a (surveillance requirements (SR)) will change the full closure time of the main steam isolation valve from 4.0 seconds to the analysis value of 8.0 seconds which includes an assumed 1.0 second instrument response time. It is stated in your submittal that a closure time of 4.0 seconds, measured under static test conditions, demonstrates closure under plant operating conditions within the 8.0 seconds assumed in the safety analysis. Please provide the following information: 1) explain how this SR could be performed under plant operating conditions assumed in the safety analysis including the instrument response time for the required 8 seconds closure time; and 2) explain why a 4.0 seconds closure time under static test conditions demonstrates closure under plant operating conditions within the 8.0 seconds assumed in the safety analysis.

59. The proposed TS 4.7.1.6.a (SRs) will change the full closure time of the main feedwater isolation valve from 5.0 seconds to 6.0 seconds to include an instrument response time of 1.0 second. Please explain how this SR could be performed under plant operating conditions assumed in the safety analysis including the 1.0 second instrument response time for the total required 6.0 seconds closure time.

60. The proposed TS 3.7.1.1 specifies the maximum allowable power level with one or two main steam safety valves (MSSVs) inoperable. Please discuss why the maximum allowable power level with more than two inoperable MSSVs on any operating steam generator(s) are not specified in Table 3.7-2 of the proposed TS.

61. It is stated that the maximum allowable power level with inoperable MSSVs were determined by the results of the loss of condenser vacuum transients. Please provide the resulting peak primary and secondary pressures for the cases with the current power level compared to that for the uprated power level.

Waterford Steam Electric Station, Unit 3

cc:

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