December 21, 2001

Mr. John T. Herron
Vice President Operations
Entergy Operations, Inc.
17265 River Road
Killona, LA 70066-0751

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - REQUEST FOR ADDITIONAL INFORMATION RELATED TO TECHNICAL SPECIFICATION CHANGE REGARDING APPENDIX K MARGIN RECOVERY - POWER UPRATE REQUEST (TAC NO. MB2971)

Dear Mr. Herron:

By letter dated September 21, 2001, Entergy Operations, Inc. proposed changes to the Waterford Steam Electric Station, Unit 3 (Waterford 3) Technical Specifications, which would allow an increase in the rated thermal power of Waterford 3 from 3,390 megawatts thermal (MWt) to 3,441 MWt.

After reviewing your request, the Nuclear Regulatory Commission staff has determined that additional information is required to complete the review. On several occasions, in the time period between November 15, 2001, and December 19, 2001, 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: See next page

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REQUEST FOR ADDITIONAL INFORMATION

ENTERGY OPERATIONS, INC.

WATERFORD STEAM ELECTRIC STATION, UNIT 3

Entergy Operations, Inc., the licensee for Waterford Steam Electric Station, Unit 3 (Waterford 3), submitted on September 21, 2001, an application requesting approval of changes to the Waterford 3 Operating License and Technical Specifications associated with an increase in the licensed power level, from 3,390 megawatt thermal (MWt) to 3,441 MWt. These changes result from the increased feedwater flow measurement accuracy achieved by utilizing high accuracy ultrasonic flow measurement instrumentation.

The staff, on reviewing the application, has determined the need for additional information as detailed below.

Materials and Chemical Engineering Branch

1. The Waterford 3 submittal for the 1.5 percent power uprate (PU) does not provide any discussion concerning the effects of the proposed PU on the integrity of the reactor vessel with respect to:

    a. Pressurized thermal shock
    b. Heatup and cooldown pressure-temperature (PT) limits
    c. Upper shelf energy

    Please address the effects of the PU on the above listed topics.

2. In Section 3.6.2.6.1 to Attachment 2 of its application, the licensee discussed structural integrity of the steam generators (SGs) under PU conditions. It appears that the structural integrity evaluation of SG tube degradation was focused on satisfying the stress and fatigue specifications of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code).

    a. The Nuclear Regulatory Commission (NRC) staff requests that the licensee evaluate the structural integrity of the SG tubes in terms of Regulatory Guide 1.121, “Bases for Plugging Degraded PWR [Pressurized Water Reactor] Steam Generator Tubes.”
    b. The licensee also needs to evaluate and discuss the acceptability of the leakage integrity of the SG tubes under PU conditions.

3. NRC has issued the following generic communications regarding SG tube plugs:

    - NRC Information Notice 89-65, “Potential for Stress Corrosion Cracking in Steam Generator Tube Plugs Supplied by Babcock and Wilcox”
    - NRC Information Notice 89-33, “Potential Failure of Westinghouse [Electric Company] Steam Generator Tube Mechanical Plugs"

- NRC Bulletin No. 89-01, “Failure of Westinghouse Steam Generator Tube Mechanical Plugs,” and Supplements 1 and 2, and,
- NRC Information Notice 94-87, “Unanticipated Crack in A Particular Heat of Alloy 600 Used for Westinghouse Mechanical Plugs for Steam Generator Tubes”

The application discusses SG tube plugs in Section 3.6.2.6.3 of Attachment 2.

a. Discuss if any of the above NRC generic communications are applicable to the tube plugs used in Waterford 3 SGs and the steps that have been taken to meet the NRC staff’s recommendations in the above generic communications.
b. Discuss any degradation detected in tube plugs and the associated repair method other than those discussed in Item 3.a.

4. Discuss the impact of the PU on each of the degradation mechanisms of the SG tubes and on the inspection intervals for SG tubes.

5. In Section 4.1.2, Flow Accelerated Corrosion (FAC), the application states the following:

   "... CHECKWORKS models will be revised, as appropriate, to incorporate flow and thermodynamic states that are projected for uprated conditions. The results of these models will be factored into future inspection/pipe replacement plans consistent with the current FAC Program requirements."

   The staff requests additional information on the revisions to the current CHECKWORKS models. Specifically, the staff requests details on the revisions to the models and details on how the impact of these changes will be factored into future inspections or pipe replacements. These details should include a comprehensive list of changes to the models, the means by which the new results will be captured into future inspections or pipe replacements, and the basis for the scheduling of the pipe replacements.

Mechanical and Civil Engineering Branch

6. In Section 3.6.1, with regard to loss-of-coolant accident (LOCA) hydraulic loads produced by the tributary lines, confirm that the current design basis LOCA loads produced by the as-built tributary lines are bounded by the design basis LOCA resulting from the mechanistic failure of main coolant loop piping.

7. In Section 3.6.2.2.2, with regard to flow and pump induced vibration, you state that the current analysis uses a mechanical flow that changes by less than 1 percent for the revised operating condition. Provide the basis for your conclusion. You also state that the revised operating conditions alter the T_{hot} fluid density, but did not provide the magnitude of the change in T_{hot} fluid density. Provide the basis and magnitude of the change in T_{hot} fluid density and confirm that there is no increase in the potential for flow induced vibration.

8. In Section 3.6.2.2.3, with regard to the structural integrity of reactor internals for the 1.5 percent PU condition, you based your conclusions on results of previous analyses either performed by you or by others. However, details of such analyses were not provided. Please provide a justification for the applicability of these analyses to the

1.5 percent PU condition. Provide a summary of evaluation results, including the maximum calculated stresses and cumulative fatigue usage factors (CUFs), for the critical reactor internal components including the baffle/barrel region components, core barrel, baffle plate, baffle/former bolts, and lower core plate for the 1.5 percent uprated power conditions. Also provide the ASME Code and Code Edition used for the evaluation of the reactor internal components, and if different from the Code of record, please justify and reconcile the differences.

9. In Section 3.6.2.9, with regard to Nuclear Steam Supply System (NSSS) piping, provide the calculated stresses, CUFs, and allowable stress for the most critical locations in the piping system.

10. In Section 3.6.3.2, with regard to primary piping thermal expansion loads, you stated that ΔT values associated with current and uprated conditions are both less than the ΔT value used in the analysis of record (AOR). Provide the ΔT values associated with current and uprated conditions, and that used in the AOR.

Reactor Systems Branch

11. The NSSS Operating Point parameters for PU conditions were calculated for a PU of 1.7 percent (3,448 MWt) in order to bound the requested PU of 1.5 percent.

    a. Provide a table comparing the NSSS Operating Points at the current 100 percent power (3390 MWt) to the recalculated uprate NSSS Operating Points in Table 3.3.1-1.
    b. Provide a listing of the Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident/transient safety analyses which incorporate these uprate Operating Point parameters. For those that do not, please provide justification that the current values used in the analyses are bounding.

12. Please provide a quantitative discussion confirming that the Low Temperature Overpressure Protection relief valves have adequate relief capacity to remove the additional decay heat generated by the 1.5 percent PU such that there is no increase in peak pressure for this transient. Include a discussion of the NRC-approved methodology used to perform this analysis.

13. The application states in Sections 3.5.9 and 3.5.10 that Core Protection Calculator System (CPCS) and Core Operating Limit Supervisory System (COLSS), respectively, will require changes to "constants" to account for the 1.5 percent PU.

    a. Please discuss the constants requiring revision and the NRC-approved methodology to be used to calculate the updated constants for a 1.5 percent PU.
    b. For a situation where the Caldon Leading Edge Flow Meter (LEFM) CheckPlus™ meter becomes inoperable, please discuss the impact of continued use of the 1.5 percent PU constants for the lower rated thermal power (RTP) level. If the constants for these systems need to be converted back to the current 100 percent RTP values, please discuss how this will be accomplished, including the time

needed to revise these constants and any impacts on the ability of CPCS and COLSS to continue to perform their design basis functions.

14. The application refers to a previously proposed 8 percent PU and associated analyses in certain sections of the submittal to justify the 1.5 percent PU. The 8 percent PU analyses are being used to justify that the 1.5 percent is bounded (by the 8 percent PU analyses) for the following topics:

    - Shutdown Cooling System
    - Emergency Feedwater System
    - Condensate Storage Pool/Wet Cooling Tower Basin Requirements

    Please provide the following information:

    a. References to submitted analyses or NRC Safety Evaluation Report that documents staff review of the 8 percent PU.
    b. If the documents requested above do not exist, please provide quantitative results demonstrating that these systems continue to meet their functional design requirements and acceptance criteria at the 1.5 percent PU conditions.

15. Please discuss the impacts of the changes in SG Thermal-Hydraulic performance (circulation ratio/bundle liquid flow, damping factor, SG pressure drop, and moisture carryover) and the increase in primary to secondary system pressure differential on the UFSAR Chapter 15 accident and transient safety analyses.

16. The main steam isolation valve (MSIV) design is based on the full design pressure differential across the valves at a RTP of 102 percent of 3,390 MWt.

    a. Please confirm that the pressure differential across the valves assuming the 1.5 percent PU operating conditions remains bounded by the assumptions in the original design analyses and operating conditions.
    b. Please discuss the plant operating or accident conditions which result in the maximum expected differential pressure for MSIV closure.

17. In Section 3.10.3 - Non-LOCA/Transient Analyses, the third paragraph includes the following statement: "...there are adverse changes in the docketed results of the Non-LOCA transient analyses." Based on the discussion which preceded this statement, it appears that the licensee meant to say that "..., there are no adverse changes in the docketed results of the Non LOCA transient analysis." Please clarify this statement.

18. UFSAR Section 15.4.1.1, "Uncontrolled CEA [Control Element Assembly] Withdrawal from Subcritical Conditions," states that a 5.41 percent analytical limit is used for the Logarithmic Power Level - High Trip setpoint. Section 3.10.1 of the PU submittal states that the analytical limit for this setpoint is 4.4 percent of RTP. Please clarify this discrepancy and discuss any impact resulting from a change in reactor trip timing.

19. In Section 3.6.4, it is stated that the fluence value for the projected 20 effective full power years (EFPYs) of operation was derived from the results of capsule W-97 which was removed about 1991. The evaluation report (BAW-2177) states that the cross sections

used were from an early version of the BUGLE set. Those cross sections result in non-conservative flux and fluence evaluations because the non-conservative ENDF/B-IV cross section data was used. This fact combined with the PU raise a question about the validity of the extrapolation fluence value to 20 EFPYs. It is also stated that another surveillance capsule will be removed at the end of the current fuel cycle (11) when the reactor will have accumulated 14 EFPYs. In this context please consider the following:

a. Describe how the projected fluence values used for the 20 EFPY PT curves and cold overpressure protection limits provide sufficient margin when the known nonconservatisms with the cross sections are considered?
b. If a new capsule is to be removed in the 2002 outage (the results of which will be available in 2003), why do you need to extrapolate to 20 EFPYs while you have the facility to update the fluence for the cycle 13 refueling (which will occur much earlier than 20 EFPYs)?

20. The recent experience from Calvert Cliffs Nuclear Power Plant has shown that the cladding corrosion is worse in high burnup regime and is consistently underestimated by the CENP corrosion model. Additionally, the PU may increase cladding corrosion levels. Please provide cladding corrosion predictions for PU conditions and assess the potential impact in fuel operation for Waterford 3.

21. With respect to the impacts of the proposed PU on the nuclear fuel core design, thermal-hydraulic design and fuel rod design analyses, please:

a. Provide a listing of the NRC-approved codes and methodologies to be used for the fuel core design process discussed in Section 3.13.1 of the submittal.
b. Confirm that all parameters and assumptions to be used for analyses described in Sections 3.13.1 through 3.13.3 remain within any code limitations or restrictions.

22. The licensee reported that the existing UFSAR Chapter 15 Non-LOCA/transient analyses-of-record bound plant operation at the proposed PU level, and therefore, reanalysis was not required. For the UFSAR Chapter 15 accident and transient analyses:

a. Confirm that the analyses-of-record either have been previously approved by the NRC or were conducted using methods or processes that were previously approved by the NRC. Provide a reference to the NRC's previous approvals.
b. Confirm that the analyses as described in the UFSAR, Revision 11, dated May 2001, are the current analyses-of-record. For those analyses which are not, please provide the following:

    i. Major assumptions used in the re-analyses. Provide justification for any assumptions which deviate from that used in the UFSAR analyses.
    ii. Describe methods and computer codes used for the re-analyses and confirm that they have been previously approved by the NRC staff. Provide justification for any changes in methodology from the existing analyses.
    iii. Provide the results of the re-analyses including primary and secondary system peak pressure, minimum departure from nucleate boiling ratio

(MDNBR), Peak Linear Heat Generation Rate, and/or amount of fuel failed.

c. Confirm that bounding event determinations continue to be valid.

23. Certain transients as described in the UFSAR show MDNBR results of 1.19, which does not meet the acceptance criteria listed in Table 3.10.3-1 of the licensee submittal for MDNBR $\geq$ 1.26. Please discuss this discrepancy. If these are not the current analyses of record, then please provide the information requested in item 22.b.i-iii above.

24. Please provide a detailed discussion regarding the impact of the PU conditions on the Uncontrolled Control Element Assembly (CEA) Withdrawal (CEAW) at Subcritical analysis. Section 15.4.1.1 of the UFSAR states that the Linear Heat Generation Rate limit is exceeded and relies on a detailed deposited energy calculation to demonstrate that centerline fuel temperature remains below melt temperature. Please provide the analysis results and the fuel melt temperature for the fuel burnup assumed in the analysis.

25. For the Uncontrolled CEAW at Power transient, the application states that the reactor trip credited for this event is the Variable Over Power Trip (VOPT). Section 15.4.1.3 of the UFSAR states that a Low DNBR Trip is credited. Please discuss this discrepancy and its impacts.

26. Please provide additional discussion and detail regarding the impact of the PU conditions on the CEA Misoperation event (UFSAR Section 15.4.1.4), specifically regarding the ratio of the available thermal margin at the start of the event to the available thermal margin at the termination of the event.

27. Please provide the results for the analysis of record for the Uncontrolled CEAW from Subcritical - Modes 3, 4 and 5, All Full Length CEAs on the Bottom (UFSAR Section 15.4.1.7) event. The results for this analysis are not shown in the UFSAR.

28. For the Inadvertent Loading of a Fuel Assembly into an Improper Position event (UFSAR Section 15.4.3.1), please discuss in greater detail the statement that the consequences of these misloads are limited by the initial DNBR margin. Also, please address the acceptance criteria for this event as listed in NUREG-0800, Standard Review Plan, Section 15.4.7, "Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position."

29. In Section 3.6.2.2.1, with respect to CEA Drop Time Analyses, the licensee states that, "Uprate to 3441 MWt will slightly increase the power level in leading rodded fuel assemblies, but will not change the burnup levels of those fuel assemblies, since the excess reactivity will be depleted faster." Please clarify this statement.

30. The CPCS within the reactor protection system initiates reactor trips based on low DNBR and high local power density.

    a. Describe how the CPCS DNBR and VOPT trip functions are modeled in the Chapter 15 safety analyses of the design basis transients and accidents.

b. Describe how the proposed PU with the reduced power measurement uncertainty affect the CPCS and the safety analyses.

31. Pursuant to 10 CFR 50.46(b)(5), long-term cooling requirements following a LOCA are established. One aspect of long-term cooling following a LOCA is to ensure boric acid accumulation will not prevent core cooling by applying an acceptable Evaluation Model (EM) to analysis of boric acid accumulation and to determination of the time available for switchover to hot leg injection. If these topics have not been reanalyzed in support of the PU request and Waterford 3 has documented application of a staff-approved EM to these topics, then please provide references to this documentation. If these topics have been reanalyzed in support of the PU request or Waterford 3 does not have a staff-approved EM, then please supply a complete description of the methodology. If you will be referencing CENPD-254, please describe the volume in which boric acid is assumed to accumulate and provide the bases for using those volumes.

32. To show that the referenced, generically-approved LOCA analysis methodologies apply specifically to the Waterford 3 plant, provide a statement that Waterford 3 and its vendor have ongoing processes which assure that LOCA analysis input values for parameters sensitive to peak cladding temperature bound the as-operated plant values for those parameters.

33. The Waterford 3 PU submittal references CENPD-137, Supplement 2-P-A, April 1998, as the generically-approved Small Break LOCA (SBLOCA) methodology, and will become the methodology included in licensing documentation and used to perform the Waterford 3 SBLOCA licensing analyses for the uprated power. The NRC approved CENPD-137, Supplement 2-P-A invoking unique criteria for that specific methodology and the then-existing or then-proposed plant conditions. Provide documentation that demonstrate how all the terms and conditions for use of that methodology have been satisfied and explain how this methodology continues to be applicable to Waterford 3 at the uprated power.