April 20, 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,

/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. Section 2.11: Does the probabilistic risk assesment (PRA) model include equipment unavailability due to maintenance, or was a “zero maintenance” PRA model used? If the latter, justify that the PRA results can be meaningfully compared to the numerical risk acceptance guidelines contained in Regulatory Guide 1.174.

2. Page 2.11-2, Loss-of-Offsite Power (LOOP): Please provide the overall LOOP frequency, along with its constituent parts (plant-centered, grid-related, and weather-related frequencies). Describe the basis for estimating the LOOP frequencies and offsite power (OSP) recovery curves, identifying the methodology and data sources used. Justify that the data used is relevant to the Waterford 3 post-extended power uprate (EPU) grid environment.

   (Staff comment: NUREG-1784 indicates that since 1997 (when deregulation of the nation’s electrical grid commenced), the nationwide plant-centered LOOP frequency has decreased, the grid-related and weather-related LOOP frequencies have remained constant, and OSP recovery times have increased. Since mixing older data with newer data tends to smooth out (de-emphasize) these trends, the post-EPU PRA results may not reasonably portray the post-EPU plant risk.)

3. Page 2.11-2, LOOP: Does the PRA consider consequential LOOPs (a LOOP after reactor trip caused by grid collapse due to loss of the plant’s generation)? If so, provide the consequential LOOP probability, describe how it was developed, and perform a study to investigate the sensitivity of the overall PRA results to the consequential LOOP probability. If not, justify the omission.

   (Staff comment: NUREG-1784 indicates that the fraction of time when the nationwide electrical grid operates in a degraded condition has increased since when deregulation of nation’s electrical grid commenced in 1997, and that there an increased likelihood of suffering consequential LOOPs whenever the grid is degraded. It is not clear how the licensee’s PRA has considered the impact of the proposed EPU on grid stability and grid degradation, which in turn affects the likelihood of consequential LOOP events and overall plant risk.)

4. Pages 2.11-3 and 2.11-4, Component Failure Rates: As part of the plant modification needed to implement the EPU, new digital atmospheric dump valves (ADV) controllers will be installed. However, this section indicates that no component failure rates were revised. It is not clear that the ADV failure rate for the post-EPU plant will be the same as for the pre-EPU plant. The reliability of the ADVs is important in the post-EPU plant

since the licensing basis calculations indicate that the ADVs are needed to mitigate small break loss-of-coolant accidents (SBLOCAs).

5. Page 2.11-4, Success Criteria: A new Technical Specification is being added concerning ADV operability since the licensing basis thermal-hydraulic (T-H) calculations indicate that either 2 ADVs and 1 high pressure safety injection (HPSI) pump or 2 HPSI pumps are required to mitigate an SBLOCA. This section indicates that the post-EPU PRA success criteria for SBLOCAs do not include the ADVs, based on best-estimate T-H calculations performed specifically for the PRA. Why is there such a difference between the SBLOCA success criteria for the licensing basis and the PRA?

6. Page 2.11-4, Success Criteria: Please describe how the core damage frequency (CDF) contribution from anticipated transients without scram (ATWS) events is determined in the PRA by addressing the following questions:

   a. Does the PRA model of ATWS sequences consider only the reactor trip failure probability, or does it include the failure of other systems (turbine bypass, ADVs, feedwater, etc.) required to mitigate an ATWS?

   b. Were new T-H calculations of the post-EPU plant’s behavior during ATWS performed to specifically support the PRA? How do the primary and secondary pressure responses during an ATWS change as a result of the EPU?

   c. How does the increase the boron concentration in the boric acid makeup tank affect the plant’s behavior during ATWS? Is emergency boration considered in the PRA’s treatment of ATWS?

7. Page 2.11-4, Success Criteria: This section indicates that new T-H calculations were performed for the PRA using the CENTS code. In general, NRC has approved use of the CENTS code for transient analyses. It is not approved for demonstrating compliance with Section 50.46 of Title 10 of the *Code of Federal Regulations* criteria; however, it is acceptable for use in modeling SBLOCAs (including steam generator tube ruptures) for the purpose of demonstrating compliance to non-LOCA regulatory acceptance criteria. Were new T-H calculations made using CENTS to determine PRA success criteria and operator action timings for medium and large LOCAs? If so, please justify. Also, define the term “core-damage” as used in the PRA and explain how the results of T-H calculations were interpreted to determine whether or not core-damage occurred.

8. Pages 2.11-6 and 2.11-7, Table 2.11-1, Impact of EPU on human reliability analysis Time Available: Please update this table to include the human failure event probabilities in the pre- and post-EPU PRA models, and the Fussell-Vesely importance measures and the risk achievement worths for the post-EPU PRA model.

9. Page 2.11-7, LOOP Recovery: Please provide additional details about the convolution approach used to conduct the LOOP recovery analysis. Using the internal events post-EPU PRA model, conduct a sensitivity analysis to assess the impact of the convolution approach on the results by determining the CDF before any LOOP recoveries are considered.

10. Pages 2.11-8 through 2.11-10, Level 1 Internal Events Results: Do the results for the internal events PRAs (pre- and post-EPU) include the contribution from internal floods? If not, please provide them.

11. Page 2.11-16, Evaluation of Probabilistic Safety Analysis (PSA) Model Quality: This section states “A peer review of the individual plant examination results was performed.” Please confirm that the peer review cited in this statement refers to the peer review done in January 2000 using the Combustion Engineering Owner’s Group approach. Provide a list of A-level (4 of 19) and B-level (20 of 80) comments that have not yet been addressed.

12. Page 2.11-16, Evaluation of PSA Model Quality: This section states "In addition, the most recent update involved extensive revision ..." What quality process (internal reviews, peer reviews, etc.) was used to make these revisions? Also, what quality process was used to check that the modifications made to the PRA in order to assess the risk impact of the EPU were correctly performed?

13. Page 2.11-16, Evaluation of PSA Model Quality: This section states “In addition, the most recent update involved extensive revision ... in order to bring the PSA model up to current PSA standards, including the new ASME [American Society of Mechanical Engineers] PSA standard (Category II).” Please confirm that the Waterford 3 PRA meets Capability Category II defined in the ASME PSA standard by providing supporting evidence such as the results of any self-assessments or peer reviews.

cc:

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