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

August 23, 2012

Mr. Michael Perito
Vice President, Site
Grand Gulf Nuclear Station
Entergy Operations, Inc.
P.O. Box 756
Port Gibson, MS 39150

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION ON SEVERE ACCIDENT MITIGATION ALTERNATIVES FOR THE REVIEW OF THE GRAND GULF NUCLEAR STATION, UNIT 1, LICENSE RENEWAL APPLICATION

Dear Mr. Perito:

By letter dated October 28, 2011, Entergy Operations, Inc. (Entergy), submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54 (10 CFR Part 54), to renew Operating License NPF-29 for Grand Gulf Nuclear Station, Unit 1, for review by the U.S. Nuclear Regulatory Commission (NRC or staff). On May 21, 2012, the NRC issued a request for additional information (RAI) on the severe accident mitigation alternatives (SAMA) analysis. By letter dated July 19, 2012, Entergy provided responses to the RAI. The staff has identified, in the enclosure, areas where additional information is needed to complete the review. Additional requests for additional information may be issued in the future.

These requests for additional information were discussed with Mr. Rick Buckley of your staff and a mutually agreeable date for the response is within 45 days from the date of this letter. If you have any questions, please contact me at 301-415-6223 or by e-mail at david.drucker@nrc.gov.

Sincerely,

David Drucker, Sr. Project Manager
Projects Branch 2
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-416

Enclosure:
As stated

cc w/encl: Listserv

CLARIFICATION QUESTIONS ON GRAND GULF NUCLEAR STATION'S RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION ON THE ANALYSIS OF SEVERE ACCIDENT MITIGATION ALTERNATIVES

1. RAI 1.c

The response to this request for additional information (RAI) indicates that there are no unresolved equipment reliability or plant data issues that would impact the severe accident mitigation alternatives (SAMA) analysis. Clarify what is meant by unresolved and indicate if there are any resolved issues that could impact Grand Gulf Nuclear Station (GGNS) plant specific data involving risk significant components or systems that might significantly impact the SAMA analysis results.

2. RAI 1.e

The initial response to RAI 1.e does not provide adequate information to explain the approximate 40% difference between the core damage frequency (CDF) from the Level 1 and Level 2 quantifications, specifically the requested, "Describe specific contributions to the approximate 40% difference in CDF, such as some of the non-minimal cutsets or other reasons." Relative to the three reasons given for the differences in results, elaborate on the following:

a. While there are uncertainties in the minimal cut set upper bound technique for cutset quantification, particularly when it involves terms that are close to 1, to the best of our knowledge, this always results in an overestimate of the true result and is most significant for large early release frequency (LERF) or other Level 2 calculations that usually have a large number of events involving values close to 1. Therefore, the minimal cut set upper bound technique for cutset quantification would not appear to be a contributor to the Level 2 CDF result being less than the Level 1 CDF. Please explain.

b. While it is expected that the Level 1 sequence by sequence calculation would result in non-minimal cutsets, the third paragraph of the RAI response in correcting the footnote to Table E.1-7 states, "The total CDF from the level 1 model presented in Table E.1-7 is slightly higher than the single top solution in which non-minimal cutsets are subsumed." This indicates that the elimination of non-minimal cutsets is not a major factor in the Level 2 result being lower. Could quantification of the One-TOP Level 1 CDF model or alternatively combining all the Level 1 sequence cutsets and minimizing and then quantifying possibly justify what is termed "slightly" in the above quotation?

c. The response states "The level 2 LOSP recoveries in the cutsets are different than the level 1 recoveries, which is lowering the percent contribution of an SBO in the level 2 model." It is not clear what is meant by this. There should be a consistent evaluation of the recovery of the loss of offsite power (LOSP) in the two models. If it is meant that there is less credit for LOSP recovery in the Level 2 model because core damage has occurred, this should not impact the CDF. Please explain.

The difference in results from the two models leads to conflicting and inconsistent information in the SAMA submittal. For example, Table E.1-1 says that the CDF

contribution from LOSP is 14% of the CDF but the contribution from station blackout (SBO) is 36%. Except for the relatively small contribution to LOSP from a consequential LOSP, the SBO contribution is a subset of LOSP contribution. Further, the Case 1 result, which is based on the same Level 2 model as the LOSP contribution discussed previously, indicates that SBO contributes 13.6% of the CDF. Again, the SBO contribution should be something less than the LOSP contribution.

Provide further support for the Level 2 model giving a valid result for the total CDF and explain differences from the Level 1 results using specific examples of these reasons including requantification using different techniques or assumptions. Provide assurance that the Level 2 model result is not missing important sequences and/or cutsets.

3. RAI 1.f

The discussion of the disposition of Observation 85 should be clarified by providing specific information on the event tree sequences, the meaning of "provide insight into the Level 2 PRA core-damage binning process," and how these sequences are handled in this process.

The conclusion of each observation discussed is that the disposition remains applicable to the PRA used for the SAMA analysis. The assessment of timing of various events factors into many of the discussions (for example, Observations 87, 89, and 97). Clarify if these assessments were made for the extended power uprate operation. If not, review the assessments and address the power increase effect on the conclusions reached.

4. RAIs 2.c and 2.d

Although the response to RAI 2.c describes the representative sequence for each release category, it does not address the justification for choosing the sequence with the highest frequency versus the sequence with a higher source term and a lower but still important frequency. Unless the highest source term for important contributors to a release category is used for the base case and SAMA specific analysis, it is possible for the SAMA benefit to be underestimated. This could occur if a particular SAMA primarily affects the frequency of a sequence with a higher source term and lower frequency.

The response to RAI 2.d discusses a sensitivity study for the High/Early (H/E) Release Category (RC) using a lower frequency but higher source term alternate.

a. Was the highest source term used in this study for the important contributors to this RC? If not, justify the source term used for the sensitivity study.

b. Provide the results of this sensitivity study to support the statement that there is no change in the cost beneficial status of SAMAs. Include the MACCS2 results or Level 3 information similar to that provided in Table E.1-13 for the new H/E RC, the maximum averted dollar risk results for the revised base case as well as the results of the cost benefit analysis for each SAMA using the revised RC risk results.

c. Identify other RCs where the source term for the representative sequence is less than that for another important sequence and justify that the use of the selected source term does not underestimate SAMA benefits.

d. The third paragraph of the response states, "This increase to a high release was due to the failure of the drywell which had not been previously accounted for in the nodal analysis." Explain the statement that "drywell failure was not accounted for."

5. RAI 2.g

The RAI response indicates that the MAAP case chosen to represent no containment failure (NCF) is MAAP run GG10502D and is an intact accident scenario with radionuclide releases consistent with design leakage rates. The response to RAI 2.c indicates that this MAAP case was chosen for RC Low-Low/Early (LL/E), which is intended to represent containment failure end points. Describe this case and how its use for both release categories would affect the SAMA analysis results.

6. RAI 3.d

While the response to this RAI indicates that changes to the site since the individual plant examination of external event (IPEEE) impacts the site drainage characteristics and thus the IPEEE recommendations are no longer valid, it is not clear if the specific recommendations are impacted or not. Specifically address the current applicability of each of the five IPEEE recommendations. Note that the last recommendation addresses the adequacy of a flood barrier for the Standby Service Water A equipment hatch.

7. RAI 5.a

Review of the Phase I SAMA screening raises the following questions.

a. SAMA 9, reduce DC dependence between high-pressure injection and Automatic Depressurization System (ADS), is said to be addressed by SAMAs 27 and 28. These SAMAs make use of portable generator to supply DC power to buses or panels. Consider a SAMA that would provide a charging system (without a new generator) and battery that would make the high-pressure core spray (HPCS) independent of the other DC buses.

b. SAMA 36, enhance DC power availability by providing a direct connection from the diesel generator, the security diesel, or another source to the 250 V battery chargers or other required loads, is said to be addressed by SAMA 27. This SAMA makes use of a portable generator. Consider a SAMA that would provide the necessary connections but without the expense of a new portable generator, or explain why this is not feasible.

c. SAMA 42, install key-locked control switches to enable AC bus cross-ties and modify procedures to enhance the reliability of the AC power system, cites SAMA 12 as being similar. SAMA 12 addresses AC bus cross-ties but does not specifically address installing key-locked control switches or enhancing procedures. Consider these improvements to the current GGNS situation, or explain why this is not feasible.

d. SAMA 74, provide capability for alternate injection via the reactor water cleanup (RWCU), is dispositioned as already installed on the basis or procedures that direct use of the RWCU for alternate shutdown cooling. The purpose of this SAMA is improved injection capability not heat removal. Consider the use of the RWCU system for injection, or explain why this is not feasible.

e. SAMA 144, modify containment flooding procedure to restrict flooding to below the top of the active fuel, is dispositioned as already installed based on the Boiling Water Reactor Owners Group (BWROG) guidelines that directs flooding to above the top of the active fuel. Depending on the physical configuration, pressurization of the drywell as a result of flooding may require drywell venting. The stated purpose of this SAMA is to reduce the drywell pressurization and prevent the resulting venting from happening if the coolant level is restricted to below the top of the active fuel (but still adequate to cool the core debris). Evaluate if this is possible for the GGNS arrangement and if so consider such a SAMA.

f. SAMA 160, institute simulator training for severe accidents, is dispositioned as already installed with the statement that the technical support center and control room would be manned in a severe accident evolution to provide additional support by personnel familiar with SAGs. If the GGNS simulator does not include severe accident scenarios, provide a cost benefit analysis for this SAMA.

8. RAI 5.c

Describe whether it is feasible to manually open the HPCS minimum flow line isolation valve (1E22F012-C) in time to prevent HPCS failure. If the manual actions are determined feasible, consider the cost benefit of such a procedure.

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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION ON SEVERE ACCIDENT MITIGATION ALTERNATIVES FOR THE REVIEW OF THE GRAND GULF NUCLEAR STATION, UNIT 1, LICENSE RENEWAL APPLICATION ENVIRONMENTAL REVIEW

Dear Mr. Perito:

By letter dated October 28, 2011, Entergy Operations, Inc. (Entergy), submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54 (10 CFR Part 54), to renew Operating License NPF-29 for Grand Gulf Nuclear Station, Unit 1, for review by the U.S. Nuclear Regulatory Commission (NRC or staff). On May 21, 2012, the NRC issued a request for additional information (RAI) on the severe accident mitigation alternatives (SAMA) analysis. By letter dated July 19, 2012, Entergy provided responses to the RAI. The staff has identified, in the enclosure areas where additional information is needed to complete the review. Additional requests for additional information may be issued in the future.

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Sincerely,

/RA/
David Drucker, Sr. Project Manager
Projects Branch 2
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-416

Enclosure:
As stated

cc w/encl: Listserv

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DATE	8/20/12	8/21/12	8/22/12	8/23/12

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Letter to M. Perito from D. Drucker dated, August 23, 2012

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION ON SEVERE ACCIDENT MITIGATION ALTERNATIVES FOR THE REVIEW OF THE GRAND GULF NUCLEAR STATION, UNIT 1, LICENSE RENEWAL APPLICATION ENVIRONMENTAL REVIEW

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