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

April 1, 2013

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

SUBJECT: GRAND GULF NUCLEAR STATION, UNIT 1 – REQUEST FOR ADDITIONAL INFORMATION REGARDING ENTERGY'S CRITICALITY SAFETY ANALYSES (TAC NO. ME7111)

Dear Sir or Madam:

By letter dated September 9, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML112521287), Entergy Operations, Inc. (Entergy, the licensee), submitted a request for the Grand Gulf Nuclear Station, Unit No. 1 (GGNS) to 1) to revise the criticality safety analysis for the spent fuel and new fuel racks, 2) add additional TS requirements for the spent fuel and new fuel storage racks, and 3) remove the spent fuel pool loading license condition.

The U.S. Nuclear Regulatory Commission (NRC) staff has determined that additional information is needed for the staff to complete its review. This enclosed request for additional information (RAI) pertaining to the license amendment request was discussed with Mr. Guy Davant of your staff on March 8, 2013, and it was agreed that a response would be provided within 60 days after the audit on this proposed topic. The NRC staff will perform the audit on April 3 and 4, 2013, at the GEH offices in Wilmington, North Carolina. The NRC staff's proprietary version of the RAI is provided in Enclosure 1 and a non-proprietary version is provided in Enclosure 2.

NOTICE: Enclosure 1 to this letter contains Proprietary Information. Upon separation from Enclosure 1, this letter is DECONTROLLED.

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If circumstances result in the need to revise the requested response date, please contact me at (301) 415-1445 or via e-mail at Alan.Wang@nrc.gov.

Sincerely,

Alan Wang

Alan B. Wang, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-416

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Enclosures:
1. Request for Additional Information (proprietary)
2. Request for Additional Information (non-proprietary)

ENCLOSURE 2 (NON-PROPRIETARY)

REQUEST FOR ADDITIONAL INFORMATION

CRITICALITY SAFETY ANALYSES

GRAND GULF NUCLEAR STATION, UNIT NO. 1

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-416

REQUEST FOR ADDITIONAL INFORMATION

CRITICALITY SAFETY ANALYSES

GRAND GULF NUCLEAR STATION, UNIT 1

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-416

By letter dated September 9, 2011 (GNRO-2011/00076; Agencywide Documents Access and Management System (ADAMS) Accession No. ML112521287), as supplemented by letters dated November 21, 2011 (GNRO-2011/00104; ADAMS Accession No. ML113320260), April 18, 2012 (GNRO-2012/00027; ADAMS Accession No. ML12109A281), October 1, 2012 (GNRO-2012/00120; ADAMS Accession No. ML12276A152); and October 22, 2012 (GNRO-2012/00124; ADAMS Accession No. ML12296A417), Entergy Operations, Inc. (Entergy, the licensee), submitted a request for the Grand Gulf Nuclear Station, Unit No. 1 (GGNS) to 1) to revise the criticality safety analysis for the spent fuel and new fuel racks, 2) add additional TS requirements for the spent fuel and new fuel storage racks, and 3) remove the spent fuel pool loading license condition.

The U.S. Nuclear Regulatory Commission (NRC) staff has determined that the following additional information is needed for the NRC staff to complete our review.

1. Due to the axially dependent features in the rack introduced by Boraflex degradation, it is necessary to include review of the need for, and possibly incorporation of the use of, axial burnup distributions. Please revise the analysis to include conservative consideration of axial burnup distributions.

   Background:

   As fuel assemblies accumulate burnup, the fuel nearer the ends of each assembly accumulates significantly less burnup than the average. This has been extensively explored for pressurized-water reactor (PWR) burnup credit and is sometimes referred to as the "end effect". It has been shown that for PWR fuel it is conservative to use the average burnup value over the full length of the fuel assembly at low burnups, but at some point, as burnup accumulates, ignoring the axial burnup distribution becomes non-conservative.

   The peak reactivity method employed in the GGNS spent fuel criticality analysis does not currently include the consideration of the axial burnup distribution. Prior to the introduction of Boraflex degradation issues, it has been assumed that the reactivity peak occurred at low enough burnups that it was conservative to ignore the axial burnup distribution.

Boraflex degradation has introduced some axially dependent rack features. These include axially dependent degradation in the form of gaps, shrinkage, and dissolution. From NETCO report NET-287-01, the text in Section 4.3 and Figure 4-6 show that dissolution appears more frequently in the top 30" of the Boraflex panels.

> *Figure 4-6 shows that the axial distribution of local dissolution among the panels is more pronounced in the middle and upper regions of the panels. The panel area above approximately 110" of elevation seems to be more susceptible to panel dissolution effects. This is likely due to the increased flow of pool water in the panel enclosure of the upper sections of the panel as a result of shrinkage induced gaps.*

The text in Section 4.3 and Figures 4-7 and 4-8 of NET-287-01 indicate that a significant fraction of the gaps were detected in the top of the panels. Gaps in the Boraflex panels between assemblies permit increased neutronic communication between top end regions of the burned fuel assemblies.

The response provided in GNRO-2011/00025[1] to RAI-30 indicated that an analysis was performed to address Boraflex relocation during a seismic event with all Boraflex gaps migrated to the top of the panel. The response provided in GNRO-2012/00120 for RAI-14 also refers to the analysis described in GNRO-2011/00025. Migration of gaps to the top of the panel permits increased neutronic communication between top end regions of the burned fuel assemblies.

Boraflex dissolution, shrinkage, gap formation, and gap relocation during abnormal conditions all have the potential to increase the importance of the underburned fuel in the upper regions of the assembly. It is likely that these axially-dependent effects shift the transition point, where it becomes non-conservative to ignore axial burnup distribution, to lower assembly burnup values. It is not clear how much lower in burnup it will shift the transition. Consequently, it is necessary to review the impact of axial burnup distributions on both normal and abnormal conditions analysis. In the current Monte Carlo sampling analysis, the distribution of gap locations restricts the gap locations to the center of the assembly. It will likely be necessary to revise the gap location distribution.

2. With regard to the RAI-2 response provided in GNRO-2012/00120, please provide analysis demonstrating that, considering the full range of fuel assembly designs and lattice variations used at GGNS, the approach used is consistent with applicable requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.68, "Criticality accident requirements."

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[1] Krupa, M. A., Entergy Operations, Inc., letter to U.S. Nuclear Regulatory Commission, "Request for Additional Information Regarding Extended Power Uprate, Grand Gulf Nuclear Station, Unit 1," dated April 21, 2011 (GNRO-2011/00025) (ADAMS Accession No. ML111120329).

Background:

According to text in a section labeled "Legacy Fuel" in Attachment 1 of GNRO-2010/00073[2], GGNS has used GE14, GNF2, GE11, ATRIUM10, ANF 9x9, ANF 8x8 fuel assembly designs. A review of the RW-859(2002) data also shows that as of 2002 GGNS had used 800 assemblies identified in the RW-859 data as type G4608GP (GE 8x8 with two water holes). The G4608GP assemblies do not appear to be included in Table 2 of Attachment 1 to GNRO-2010/00073. It should be confirmed that these assemblies either were evaluated or that they have been moved out of the spent fuel pool (SFP) and into dry storage casks.

The text in GNRO-2010/00073 explains that the legacy fuel was determined to be bounded by the GE14 design basis assembly based solely on 2-D lattice peak reactivity calculations performed using CASMO-4. The analysis presented appears to ignore the "rack efficiency" factor, which is the ratio of in-rack k_{eff} divided by SCCG k_∞. Some lattices screened out based solely on SCCG k_∞ could yield higher in-rack k_{eff} values.

The response to RAI-2 provided in GNRO-2012/00120 states

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The RAI-2 response goes on to cite a paper titled "*Uncertainty Contribution to Final In-Rack K(95/95) from the In-Core k_∞ Criterion Methodology for Spent Fuel Storage Rack Criticality Safety Analyses,*" by J. C. Hannah that was presented at the PHYSOR 2010 meeting. This paper is based on an implicit assumption that the 12 lattices examined in the paper are bounding or are adequately representative of the population of fuel assemblies to be stored. It is not clear that such an assumption is appropriate for the GGNS criticality safety analysis. Logic based on parametric studies should have been provided to show that the limited set of arrays provides bounding examples for all arrays.

3. Several of the RAIs (i.e., 1, 4, 11, 14a, 18, and 20) sought information to provide a better understanding of the method used to credit the residual Boraflex in the Region I racks. The goal was to obtain information that would allow NRC staff to reach a conclusion that there was a reasonable assurance of safety. While the responses provided some additional information, the following information is needed to complete our review:

   a. Concerning development of sampling distributions:

      1) Please clarify if the issue of on-going Boraflex dissolution has been adequately incorporated. From the available literature, dissolution appears to have a significant effect on the edges of the panels, effectively

[2] Krupa, M. A., Entergy Operations, Inc., letter to U.S. Nuclear Regulatory Commission, "Supplemental License Amendment Request, Extended Power Uprate, Grand Gulf Nuclear Station, Unit 1," dated November 23, 2010 (GNRO-2010/00073) (ADAMS Accession No. ML103330146).

reducing their width and length beyond the radiation damage induced shrinkage. Boraflex dissolution does not appear to be limited to very high dose panels. The distributions used to describe the degraded Boraflex need to include consideration of the following:

- further reduction of panel width and length due to dissolution
- effects of scallop formation and any other forms of localized thinning
- future panel dissolution
- local (i.e. from panel to panel) dissolution rate variation that would occur due to manufacturing and environmental variations
- local dissolution rate variation due to manufacturing defects and fuel storage rack damage

Please describe how these considerations are included in the development of the distributions used to simulate the degraded Boraflex. Where the issues have not or cannot be covered, please provide a quantified justification to support the acceptability not considering these issues.

2) RAI-4.j requested justification for the limited amount of testing performed to quantify the Boraflex degradation. The response to RAI-4.j indicated that the selected panels were "representative". The response to RAI-4.e(ii) indicates that the gap size distribution was developed from blackness test measurements of the 208 panels in the 7th blackness test campaign conducted during March 24-27, 1999.

This limited amount of testing (208 panels in 1999 and 45 panels in 2007) may have missed some significant outliers with uncharacteristically higher Boraflex degradation. Such degradation might affect panels in racks that have damage or fabrication defects that allow higher flow rates through the panel gap than is typical.

- According to the RAI response, the gap size distribution was based on the 1999 blackness testing data. There are clearly some factors such as continuing dose accumulation and dissolution that would cause the distributions to evolve. Please provide justification for use of distributions, derived from the 13-year old data, in the safety analysis that will serve as the basis for current and future spent nuclear fuel (SNF) storage.
- A response to RAI-32 was provided in GNRO-2011/00025. The question from that original RAI was "How can we ascertain that the bounding panels in the Region 1 population have been

represented?" RACKLIFE can tell you what will happen to the average or typical panel and blackness or BADGER testing can tell something about the measured panels, but neither can tell you about the outliers. It is suspected that it will not take many outliers to have a significant local effect on k_{eff}. Since it is unlikely that all panels will be tested, it would probably be appropriate to include an abnormal condition that might be bounded by a model with one or a few completely dissolved panels. Therefore, the NRC staff has concluded that the original RAI-32 has not been adequately addressed and needs to be supplemented.

3) According to the text in NET-287-01, the GGNS spent fuel storage racks are constructed using three separately manufactured "shapes," which are referred to in Figure 3-2 as "cruciform, ell and tee shapes". It is not clear that the Boraflex degradation distributions and models adequately reflect the use of the three components.

The flow paths into and through the three shapes are probably significantly different. Also, the water-filled internal volumes likely vary between the shapes. These differences would lead to different dissolution patterns and rates for the three shapes. The Boraflex in some of the shapes may be dissolving faster than in others. Please examine the measured data to ensure a statistically significant number of measurements have been performed for all three shapes. Additionally, please examine the data for trends between the three shapes. Provide the results of these examinations. If there are differences, please describe how such differences are considered in the criticality analysis.

4) The distributions used to simulate the [[ ]] are based on past measurements. As is noted in the RAI responses, the distributions [[ ]] that existed at the time the measurements were made. It is not clear that the distributions will remain conservative throughout the remainder of the license period. Since dissolution continues at varying rates regardless of radiation dose, it is not clear that use of these distributions for Region 1 racks will remain conservative in the future. Please explain what measures, beyond simple screening of approximate Boraflex panel dose, have been or will be taken to ensure that the criticality safety analysis, which is based on the [[ ]], will remain valid.

b. Concerning gap simulation

1) RAI 4.a requested justification for ignoring potential correlations between distributions and between panels in a cell. The response [[

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2) RAI 4.d requested information about [[

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3) RAI-4.i requested information concerning the use of [[

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c. Concerning analysis of the results

1) The response to RAI 4.b indicates that the [[

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2) The first part of RAI 4.b questions the appropriateness of using the [[

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4. Some of the prior RAIs question how temperature variation is modeled. The effects of spent fuel storage rack moderator temperature/density were evaluated [[ ]]. No evidence was provided showing that rack k_{eff} values did not peak at temperatures [[ ]]. Please provide justification that performing the analysis at three points is sufficient to demonstrate that the rack k_{eff} values did not peak at temperatures [[ ]].

5. The following validation issues were not addressed or were addressed incorrectly in the RAI responses.

   a. The table provided in the response to RAI-19 erroneously claims that [[ ]] critical experiments include gadolinium in the fuel rods. There is no gadolinium in [[ ]]. Consequently, it is not clear that the gadolinium in the safety analysis models is adequately validated. Please describe how the remaining gadolinium bearing critical experiments compare to the in-rack safety analysis models. Include a comparison of the gadolinium worths from the safety models with the worths from the critical experiments. Use the critical experiment results to determine if there is a gadolinium related bias. This study should include a trend analysis for bias and bias uncertainty as a function of gadolinium worth. If there is a gadolinium related bias that would raise the maximum k_{eff} values, please incorporate the bias into the analysis.

   b. RAI-22 requested a list of nuclides credited in the analysis. The intent was to confirm that nuclides in the fuel compositions were either validated or an appropriate margin was adopted to cover poor validation of credited nuclides. The list of nuclides was not provided. Please confirm that all nuclides in the fuel compositions were validated or describe the margin to k_{eff} adopted to cover poor validation of credited nuclides.

      [[ ]].

6. In response to RAI-26, a misloaded fuel assembly abnormal condition was added. From the RAI response, it appears that the misload was evaluated only in the context of the Region II model. Please perform a calculation with the misload assembly at the Region I/II interface to confirm that the already modeled misload results in a larger k_{eff} increase.

7. With regard to response to RAI-14.a, please either (1) provide a quantified basis supporting the opinion that the Boraflex degradation model is adequately conservative to bound all credible abnormal conditions or (2) include in the analysis evaluation of abnormal conditions related to underestimation of Boraflex degradation and related to potential movement of degraded Boraflex.

Background

Underestimation of Boraflex degradation should be considered a credible abnormal condition because of the following issues:

a. The limited amount of blackness and BADGER testing performed.

b. The use of the RACKLIFE program to identify potentially severely degraded Boraflex panels.

c. The potential for wide spread degraded Boraflex movement during abnormal events such as earthquakes or more localized movement due to loads dropped on the fuel storage racks.

d. The local variation from RACKLIFE predictions due to manufacturing variation and defects.

e. The local variation from RACKLIFE predictions due to rack wear and damage.

f. The potential for severe underestimation of Boraflex degradation due to RACKLIFE input errors.

g. The continuing Boraflex degradation due to radiation damage and dissolution.

h. The implicit assumption that future Boraflex degradation will look like past Boraflex degradation.

The RAI response claims that the Boraflex degradation analysis approach used in NEDC-33621P is so conservative that no abnormal conditions related to Boraflex degradation need be considered. The RAI response appears to be based solely on unquantified engineering judgment.

Candidate abnormal events/initiators might include:

a. Underestimation of Boraflex degradation.

b. Non-conservative characterization or modeling of Boraflex shrinkage, gaps, or dissolution.

c. Local variation in Boraflex degradation larger than expected due to manufacturing variation, defects, rack wear, or rack damage.

d. Errors in RACKLIFE assembly-specific data, reactor operating data, pool water clean-up data, and "escape coefficient" data.

e. Relocation of degraded Boraflex due to seismic events, dropped loads, insertion/removal of bowed fuel assemblies, and local heating variation.

f. Anything else the applicant thinks should be added.

8. The NRC issued two Technical Letter Reports (ADAMS Accession Nos. ML12216A307 and ML12254A064) on September 30, 2012. The reports summarize the use of the RACKLIFE computational tool and the BADGER in-situ measurement technique and the uncertainties associated with them. As per the information discussed in the reports, the NRC staff has concerns regarding the uncertainties associated with a Boraflex management strategy employing RACKLIFE and BADGER methodologies.

   As degradation of the Boraflex material increases, the uncertainties associated with the methodologies may also increase. Increasing degradation also results in reduced margins in the criticality safety analysis (CSA), which may be relied upon to address uncertainty. Given the active degradation of Boraflex in the GGNS spent fuel pool, the increasing uncertainty of the predictive tools, and the decreasing margin to criticality, please describe your long-term plan to ensure subcriticality in the spent fuel pool while crediting this material in your CSA. Specifically:

   a. Do you plan to replace the Boraflex material or install additional neutron absorbing material in the future? If so, describe the proposed timeframe of this installation and the planned neutron absorbing material to be used.

   b. If the current management strategy (RACKLIFE and BADGER) will continue to be used, describe how you will account for increasing uncertainty in the RACKLIFE model as a function of increased degradation of the Boraflex material (and the rate of silica entering the pool water).

   c. Describe whether the inspection frequency and population of in-situ testing at GGNS will increase with increasing uncertainty in the RACKLIFE model.

   d. Discuss whether GGNS will continue the current management strategy until all of the Boraflex is no longer credited for criticality control and the SFP criticality is maintained by other means (i.e., geometric configuration).

If circumstances result in the need to revise the requested response date, please contact me at (301) 415-1445 or via e-mail at Alan.Wang@nrc.gov.

Sincerely,

/RA/

Alan B. Wang, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-416

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Enclosures:
1. Request for Additional Information (proprietary)
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