Entergy

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Attachment 1 contains PROPRIETARY information.

GNRO-2013/00050

July 2, 2013

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Criticality Safety Analysis License Amendment Request - Responses to NRC Requests for Additional Information

Grand Gulf Nuclear Station, Unit 1
Docket No. 50-416
License No. NPF-29

REFERENCES:

1. Entergy Operations, Inc. letter to the NRC (GNRO-2011/00076), *License Amendment Request - Criticality Safety Analysis and Technical Specification 4.3.1,* Criticality, September 9, 2011 (ADAMS Accession No. ML1125321287)

2. NRC letter to Entergy Operations, Inc., *Grand Gulf Nuclear Station, Unit 1 – Request for Additional Information Regarding Entergy's Criticality Safety Analyses (TAC No. ME7111)*, April 1, 2013

3. NRC letter to Entergy Operations, Inc., *Grand Gulf Nuclear Station, Unit 1 – Audit of the Criticality Safety Analysis for Spent Fuel Pool Storage License Amendment Request (TAC No. ME7111)*, May 17, 2013

Dear Sir or Madam:

In Reference 1, Entergy Operations, Inc. (Entergy) submitted to the NRC a license amendment request (LAR), which proposes to: 1) revise the criticality safety analysis (CSA) for the spent fuel and new fuel storage racks; 2) impose additional requirements for the spent fuel and new fuel storage racks in Technical Specification (TS) 4.3.1, *Criticality;* and 3) delete the spent fuel pool loading criteria Operating License Condition 2.C(45).

In References 2 and 3, the NRC transmitted to Entergy eight (8) requests for additional information (RAIs) pertaining to the CSA LAR. Responses to these RAIs are provided in Attachment 1 to this letter.

When Attachment 1 is removed from this letter, the entire document is NON-PROPRIETARY.

Global Nuclear Fuel – Americas, LLC (GNF) considers certain information contained in Attachment 1 to be proprietary and, therefore, exempt from public disclosure pursuant to 10 CFR 2.390. An affidavit for withholding this information, executed by GNF, is provided in Attachment 2. The subject information contained in Attachment 1 was provided to Entergy in a GNF transmittal letter that is referenced in the affidavit. Therefore, on behalf of GNF, Entergy requests Attachment 1 be withheld from public disclosure in accordance with 10 CFR 2.390(b)(1). A non-proprietary, redacted version of Attachment 1 is provided in Attachment 3.

This letter contains new commitments, which are identified in Attachment 4.

If you have any questions or require additional information, please contact Guy Davant at (601) 368-5756.

I declare under penalty of perjury that the foregoing is true and correct; executed on July 2, 2013.

Sincerely,

BSF/ghd

Attachments:
1. Responses to NRC Requests for Additional Information - Criticality Safety Analysis License Amendment Request (Proprietary Version)
2. Affidavit Supporting Request to Withhold Attachment 1 from Public Disclosure
3. Responses to NRC Requests for Additional Information - Criticality Safety Analysis License Amendment Request (Non-Proprietary Version)
4. List of Regulatory Commitments

cc: Mr. Arthur T. Howell
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U. S. Nuclear Regulatory Commission
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NRC Senior Resident Inspector
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ATTACHMENT 1

GRAND GULF NUCLEAR STATION

GNRO-2013/00050

RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION - CRITICALITY SAFETY ANALYSIS LICENSE AMENDMENT REQUEST

(Proprietary Version)

The header of each page in this attachment carries the notation "GNF Proprietary Information - Class III (Confidential)." GNF proprietary information is identified by a dotted underline inside double square brackets. [[This sentence is an example.{3}]] Figures and tables containing GNF proprietary information are identified with double square brackets before and after the object. In each case, the superscript notation {3} refers to Paragraph (3) of the affidavit provided in Attachment 2, which provides the basis for the proprietary determination. Specific information that is not so marked is not GNF proprietary.

ATTACHMENT 2

GRAND GULF NUCLEAR STATION

GNRO-2013/00050

AFFIDAVIT SUPPORTING REQUEST TO WITHHOLD ATTACHMENT 1 FROM PUBLIC DISCLOSURE

PROVIDED BY
GLOBAL NUCLEAR FUEL – AMERICAS, LLC
3901 CASTLE HAYNE ROAD
WILMINGTON, NC 28401

Global Nuclear Fuel – Americas

AFFIDAVIT

I, Lukas Trosman, state as follows:

(1) I am Engineering Manager, Reload Design and Analysis, Global Nuclear Fuel – Americas, LLC (GNF-A), and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in Enclosures 1 and 3 of GNF letter KGO-ENO-JB1-13-061, K. O'Connor (GNF-A) to F. Smith (Entergy), entitled "Grand Gulf Fuel Storage Criticality Safety Analysis of Spent Fuel Storage Racks Draft Local Dissolution RAI Support," dated June 26, 2013. GNF-A proprietary information within the text and tables in Enclosures 1 and 3 is identified by a dotted underline inside double square brackets. [[This sentence is an example.{3}]] In all cases, the superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GNF-A relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975 F2d 871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704 F2d 1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GNF-A's competitors without license from GNF-A constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals aspects of past, present, or future GNF-A customer-funded development plans and programs, resulting in potential products to GNF-A;

d. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. above.

(5) To address 10 CFR 2.390 (b) (4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GNF-A, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GNF-A, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or subject to the terms under which it was licensed to GNF-A. Access to such documents within GNF-A is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GNF-A are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains details of the nuclear fuel criticality licensing methodology for the GEH Boiling Water Reactor (BWR). Development of these methods, techniques, and information and their application for the design, modification, and analyses methodologies and processes was achieved at a significant cost GNF-A.

The development of the evaluation processes along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GNF-A asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GNF-A's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GNF-A's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical, and NRC review costs comprise a substantial investment of time and money by GNF-A.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GNF-A's competitive advantage will be lost if its competitors are able to use the results of the GNF-A experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GNF-A would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GNF-A of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 26th day of June 2013.

Lukas Trosman
Engineering Manager, Reload Design and Analysis
Global Nuclear Fuel – Americas, LLC

ATTACHMENT 3

GRAND GULF NUCLEAR STATION

GNRO-2013/00050

RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION -
GGNS CRITICALITY SAFETY ANALYSIS LICENSE AMENDMENT REQUEST

(Non-Proprietary Version)

RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION - CRITICALITY SAFETY ANALYSIS LICENSE AMENDMENT REQUEST

In a letter to the NRC[1], Entergy Operations, Inc. (Entergy) submitted a license amendment request (LAR), which proposes to: 1) revise the criticality safety analysis (CSA) for the spent fuel and new fuel storage racks; 2) impose additional requirements for the spent fuel and new fuel storage racks in TS 4.3.1, *Criticality;* and 3) delete the spent fuel pool loading criteria Operating License Condition 2.C(45).

In letters to Entergy[2], the NRC staff transmitted eight (8) requests for additional information (RAIs) pertaining to the CSA LAR. Responses to these RAIs are provided below.

RAI 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 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.

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[1] Entergy letter to the NRC (GNRO-2011/00076), *License Amendment Request - Criticality Safety Analysis and Technical Specification 4.3.1,* Criticality, September 9, 2011 (ADAMS Accession No. ML1125321287)

[2] NRC letter to Entergy, *Grand Gulf Nuclear Station, Unit 1 – Request for Additional Information Regarding Entergy's Criticality Safety Analyses (TAC No. ME7111)*, April 1, 2013; and NRC letter to Entergy Operations, Inc., *Grand Gulf Nuclear Station, Unit 1 – Audit of the Criticality Safety Analysis for Spent Fuel Pool Storage License Amendment Request (TAC No. ME7111)*, May 17, 2013

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[3], the text in Section 4.3 and Figure 4-6 show that dissolution appears more frequently in the top 30 inches 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[4] 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[5] 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.

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3 Provided in Entergy letter to the NRC (GNRO-2011/00025), *Request for Additional Information Regarding Extended Power Uprate,* April 21, 2011 (ADAMS Accession No. ML111120329)

4 Entergy letter to the NRC (GNRO-2011/00025), *Request for Additional Information Regarding Extended Power Uprate,* April 21, 2011 (ADAMS Accession No. ML111120329)

5 Entergy letter to the NRC (GNRO-2012/00120), *Request for Additional Information – GGNS Criticality Safety Analysis License Amendment Request,* October 1, 2012 (ADAMS Accession No. ML12276A152)

Response

As described in the third bullet of Section 3.6 of GE Hitachi Nuclear Energy Report NEDC-33621P, *Grand Gulf Nuclear Station Fuel Storage Criticality Safety Analysis of Spent and New Fuel Storage Racks*, Revision 0[6] (NEDC-33621P), the single lattice geometry and pin-specific isotopic definition is selected to conservatively represent the entire active fuel region of every bundle in storage. This methodology ensures that axial reactivity effects due to variations in burnup profiles are inherently bounded, as the lattice modeled to represent the entire bundle is at its peak reactivity burnup.

To further clarify this point, the impact of end effects becomes evident for bundles when the peak reactivity region of the bundle occurs at the axial ends of the active fuel length. As stated in the RAI, this occurs at some point as bundle average exposure is accumulated and the resulting exposure distribution moves the peak reactivity region of the bundle to the axial ends of the assembly. The important distinction in this case is the fact that no distribution in bundle exposure profile is assumed in the analysis, but rather the bundle is assumed to exist at a constant exposure for its entire fuel length. Because of this, as the bundle accumulates exposure it becomes more reactive along its entire length up to its peak reactivity burnup, and then becomes less reactive along its entire length past this peak reactivity point. In other words, assuming no distribution in exposure and peak reactivity burnup for the assembly inherently eliminates potential end effect concerns as well as other reactivity distribution effects that could result in higher rack reactivity.

To quantify the conservatism of the uniform, peak reactivity burnup assumption, several additional cases have been analyzed. The first case analyzes a 3-dimensional (3D) bundle in the storage rack system with a typical bundle exposure profile at discharge. In this burn-up profile, the top enriched node (Node 24) is near the VAN lattice geometry's peak reactivity exposure. The bundle exposure profile is provided in Figure 1-1 with the line "Approach A".

[6] Provided in Entergy letter to the NRC (GNRO-2010/00073), *Supplemental Information – License Amendment Request, Extended Power Uprate*, November 23, 2010 (ADAMS Accession No. ML103330093)

[[

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Figure 1-1: [[ ]]

[[

]]

The second approach, noted as "Approach B" in Figure 1-1, is similar to the approach in NEDC-33621P. This model assumes the peak reactivity node in this bundle exists for the entire length of the bundle. Note that the approach in NEDC-33621P is even more conservative, as it assumes a much higher reactivity lattice at its peak reactivity statepoint for the entire length of the bundle.

Both Approach A and Approach B bundle models are then integrated with the storage rack model in MCNP using the standard, centered gap model described in NEDC-33621P and assuming that the gaps have migrated to the top of the rack. Results are provided in Table 1-1, below. These results demonstrate that modeling the peak reactivity lattice for the entire length of the bundle is significantly more conservative than modeling the axially varying exposure profile explicitly in a 3D model in an attempt to capture "end effects".

Table 1-1: 3D Model Comparison Results

Gap Model	Exposure Model	
	Approach A	Approach B
Nominal Gaps (Center)	[[	
Seismic Gaps (Top)		]]

An additional sensitivity study was also performed to demonstrate the combination of different peak reactivity lattice geometries in a 3D bundle/rack model is bounded by the peak reactivity study provided in Section 6.3 of NEDC-33621P. In Section 6.3, Table 17 outlines the lattices explicitly studied in the rack system. These lattices were studied in a 3D geometry but assumed a single lattice design existed for the entire length of the bundle. In this study, a composite bundle was generated by modeling the geometry and isotopics from Case 10 in Table 17 from bundle height [[ ]], and Case 13 in Table 17 from bundle height [[ ]]. The in-rack k_{eff} for each case is provided in Table 1-2, below. The reactivity of the composite case falls between the reactivity values calculated for the individual lattices that make up the system. These results demonstrate that modeling the peak reactivity lattice from Table 17 for the entire length of the bundle is more conservative than explicitly modeling multiple peak reactivity lattices from Table 17 at their corresponding axial locations in the bundle.

Table 1-2: Design Basis Lattice Results

Lattice Design	In-Rack k_{eff}
Case 10	[[
Case 13	
Case10 + Case13 3D Bundle	]]

RAI 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."

Background

According to text in a section labeled "Legacy Fuel" in Attachment 1 of GNRO-2010/00073[7], GGNS has used GE14, GNF2, GE11, ATRIUM10, ANF 9x9, and 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:

> *[[*
>
> *]]*

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.

Response

The GE 6/7 (8x8) assemblies from the initial cycle were inadvertently omitted from Table 2 in Attachment 1 of GNRO-2010/00073. A previously performed criticality analysis of these assemblies determined them to be bounded significantly by the ANF 8x8 assemblies, which are evaluated in the new analysis. The results of this analysis are shown in Table 2-1, below. Therefore, since the GE 6/7 assemblies are bounded by the analysis performed for the ANF 8x8 assemblies, no further analysis is required.

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[7] Entergy letter to the NRC Commission (GNRO-2010/00073), *Supplemental License Amendment Request, Extended Power Uprate*, November 23, 2010 (ADAMS Accession No. ML103330146)

Table 2-1: Reactivity of Different Fuel Types

Assembly Type	Peak In-Rack k_{inf}
ANF 8x8	0.77253
GE 6/7	0.71929

The evaluation for the legacy fuel was performed using a CASMO-4 in-rack model both with and without Boraflex present. The fuel was depleted in CASMO-4 before being placed in rack geometry, consistent with the method in NEDC-33621P, which addresses Gd depletion and plutonium production. The depletion results were then transcribed into rack geometry to determine the in-rack k_{eff}. The evaluation is based on a comparison of the relative reactivity of the legacy fuel types to the design basis bundle (DBB) to ensure the DBB is more reactive. This comparison uses the most reactive lattice for each of the legacy fuel types.

CASMO-4 was previously compared to the KENO computer code for these bundles using similar lattices to develop a benchmark between the two. The 95/95 limit of the uncertainty in this benchmark was added to the peak reactivity calculated from CASMO-4 that includes Boraflex, in order to produce corrected peak reactivity. After adding this uncertainty, it can be seen in Table 2-2, below, that the GE14 DBB still bounds all legacy fuel, and the margin between the GE14 DBB and the next most limiting assembly (ATRIUM10) is significantly above the uncertainties associated with CASMO-4.

Table 2-2: Reactivity for all Legacy Fuel Compared to the GE14 DBB

Fuel Type	Peak Reactivity (w/ Gd)		Benchmark Uncertainty	Corrected Peak Reactivity
	No Boraflex	With Boraflex		
GE14 DBB	1.12436	0.87672	--	0.87672
ANF 8x8	1.05981	0.80906	0.00288	0.81194
ANF 9x9	1.08474	0.83748	0.00379	0.84127
ATRIUM10	1.10517	0.86197	0.00339	0.86536
GE11	1.07114	0.83047	0.00573	0.83620

To further demonstrate conformance of the design basis lattice selection process with applicable 10 CFR 50.68 requirements, [[ ]] additional lattices were studied in the fuel storage rack system along with the design basis lattice candidates described in Table 17 of NEDC-33621P. The additional lattices were selected based on a survey of the characteristics of GE14 and GNF2 lattices currently stored in the GGNS spent fuel pool to adequately represent the population of fuel assemblies to be stored. Table 2-3, below, provides the consolidated results of the survey and the bundles selected for further study, which are denoted in ***bold italics***. As noted in this table, the characteristics of fuel are generally consistent across fuel types and cycles loaded, providing confidence that the study of these lattices is appropriate to represent future cycles.

Table 2-3: [[ ]]

[[					

[[					
					]]

The lattices were studied over a range of exposures both in the in-core and in-rack geometries to quantify the 95/95 uncertainty in the linear fit of these values using a variation of the one-sided lower tolerance band method described in NUREG/CR-6698, *Guide for Validation of Nuclear Criticality Safety Calculational Methodology*. A graphical representation of the results is provided in Figure 2-1, below.

[[

]]

Figure 2-1: [[ ]]

Based on the information in Figure 2-1 and the in-core k_{inf} limit of 1.26 set for the analysis, the nominal reactivity ($k_{nominal}$) associated with the linear fit is defined as [[ ]]. The uncertainty of the fit was calculated to be [[ ]]. These values can be combined with all other biases, tolerances, and uncertainties to determine the final rack k(95/95) in two ways.

The first method (Design Basis Lattice Methodology), and the way that NEDC-33621P performs the k(95/95) statistical roll-up, assumes that the design basis lattice in-rack k_{eff} result is the appropriately conservative $k_{nominal}$ value to capture the potential uncertainty in the linear fit of the in-core k_{inf} methodology.

The second method (Fit Methodology) assumes $k_{nominal}$ is calculated based on the linear fit and the uncertainty of that fit must be included in the square root of the sum of the squares of all other uncertainty contributors, and, thus, in the final k(95/95) result. [[

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A comparison of the final k(95/95) values from these two approaches, along with their bias, tolerance, and uncertainty contributors, is provided in Table 2-4, below. Note the results presented related to the Design Basis Lattice Methodology are updated based on the previous RAI responses related to Revision 0 of NEDC-33621P.

Table 2-4: In-Core k_{inf} 95/95 Uncertainty vs. Design Basis Lattice k_{max} Results

	Fit Methodology	Design Basis Lattice Methodology	Difference
$k_{nominal}$	[[		
Δk_{Bias}			
$\Delta k_{Tolerance}$			
$\Delta k_{Uncertainty}$			
Total			]]

These results indicate that the design basis lattice methodology produces results that are nearly identical to results determined from calculating a $k_{nominal}$ value based on a fit and including the uncertainty of the linear fit in the final k(95/95) response. These results provide confidence that the design basis lattice selection process used in this evaluation is in conformance with applicable 10 CFR 50.68 requirements.

RAI 3

Several of the RAIs (i.e., 1, 4, 11, 14a, 18, and 20)[8] *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 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*

[8] Provided in Entergy letter to the NRC (GNRO-2012/00120), *Request for Additional Information – GGNS Criticality Safety Analysis License Amendment Request,* October 1, 2012 (ADAMS Accession No. ML12276A152)

- *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*

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.

Response

The GGNS Boraflex gap measurements are summarized in Figure 1 provided in GNRO-2010/00073. That figure shows the evolution of Boraflex panel losses based on Blackness test measurements over the first 13 years of rack in-service life and the BADGER measurement that occurred after 21 years of in-service life. Figure 1 presents the measurement database as a function of panel dose that has been shown to be related to panel shrinkage. For racks with an in-service life like GGNS, dissolution effects are dominantly impacted by the time a given panel has exceeded a relatively small dose. The GGNS measurement database reflects both shrinkage and dissolution effects. In order to separate the effects, sub-sets of the database were selected with a similar dose (i.e., narrow range of dose values). Since these panels achieved the target dose over different time periods, the panel loss attributable to dissolution effects over time can be evaluated. For this evaluation, the time since the initial irradiation is used (i.e., the end of Cycle 1 in 1986). Four separate sub-sets of the database were compiled, with different target dose values. The dose targets cover a wide range of dose values, with some extending above the Region I dose limit. The evaluation of this data is illustrated in Figure 3-1, below, which shows a strong relationship of panel loss with time that is not dependent on dose.

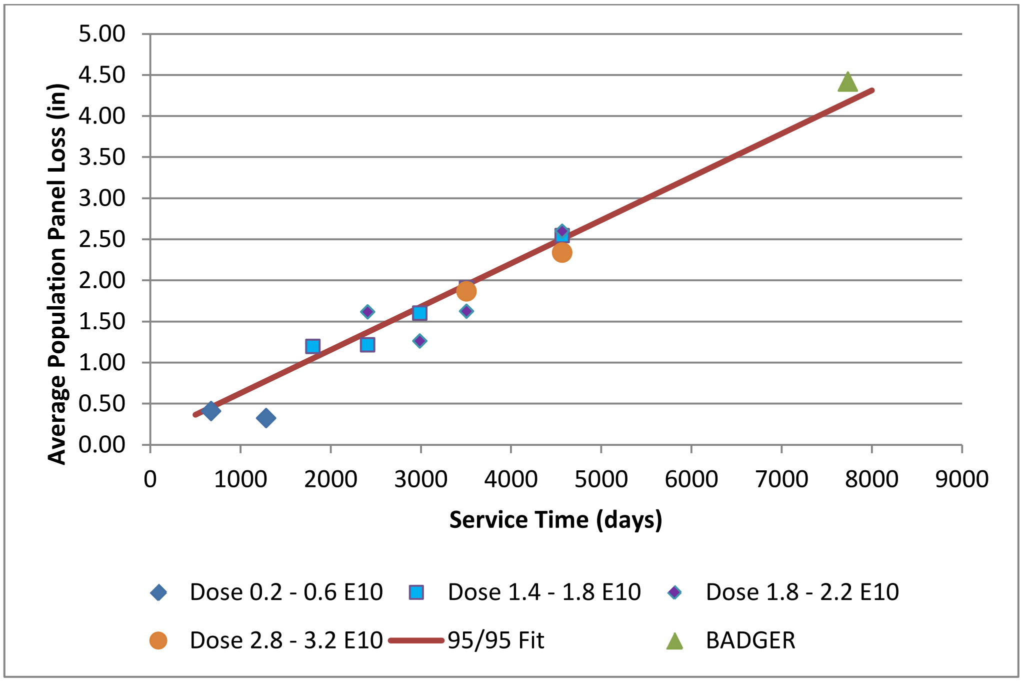

Figure 3-1: Subsets of Database with 95/95 Fit

A regression fit of the data resulted in an R-squared value of 0.906. The BADGER test data was not included in the development of the correlation since there are too few measurements with a common dose or irradiation history. However, subsets of low dose, long in-service life panels from the BADGER measurements demonstrate the correlation's applicability. The 95/95 upper limit of the fit is used to evaluate the dissolution effects on gap growth for the GGNS racks.

The Boraflex gap assumptions in NEDC-33621P are based on distributions that bound the measured gap performance from the 7th Blackness test campaign. These gap measurements include some dissolution effects. In order to determine the adequacy of the analysis assumptions relative to dissolution effects, margin between the Boraflex measurements and the analysis assumptions were evaluated using the dissolution correlation described above. This response revises the response to RAI 4.f provided in GNRO-2012/00120 to account for the effects of dissolution on Boraflex gaps. The 6th Blackness test campaign was used as the base condition since it is more representative of Region I gap performance. The vast majority of the Campaign 7 panels exceed the dose limit for Region I and dissolution is evident (see Figure 1 of GNRO-2010/00073). In addition to the 20% measurement uncertainty, the measurement data were randomly sampled 50 times in the same manner described in the response to RAI 4 provided in GNRO-2012/00120. Dissolution effects were included based on the 95/95 upper limit to the slope of the gap growth described above. Dissolution effects were evaluated for various in-service times to determine when the NEDC-33621P assumptions would no longer be bounding. The results for 24.5 years of dissolution are compared to the NEDC-33621P panel loss assumed distributions in Figure 3-2 and Figure 3-3, below.

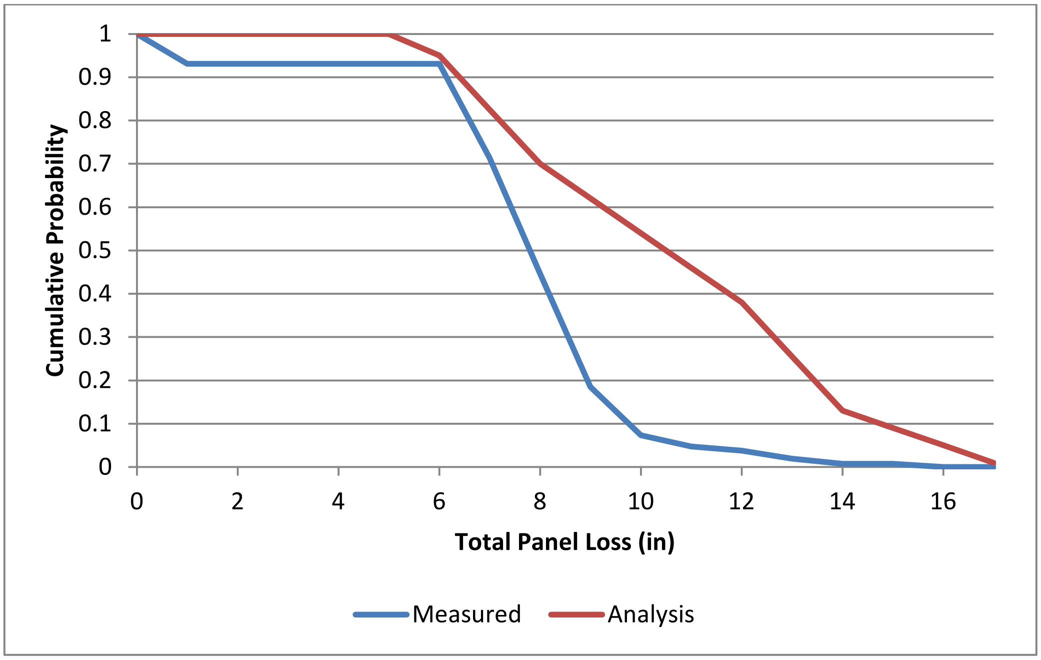

Figure 3-2: Cumulative Probability Distribution for Total Panel Loss

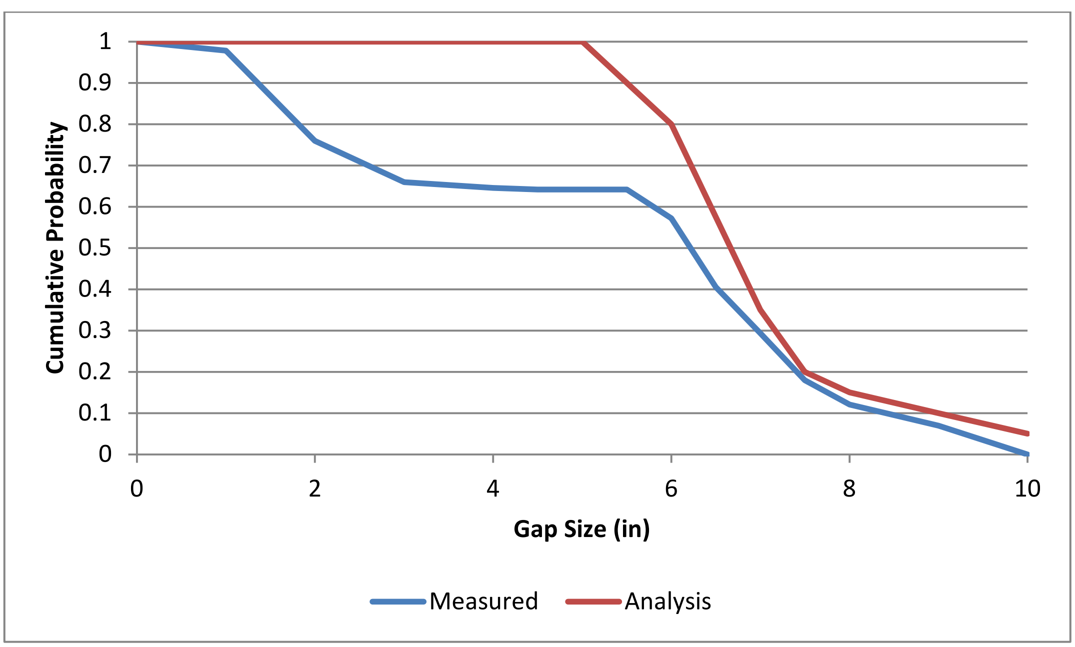

Figure 3-3: Cumulative Probability Distribution for Individual Gap Size

Figure 3-3 was generated by assigning the total panel loss due to dissolution to the largest gap in a given panel. As described in the response to RAI 2.4.b.2, below, the NEDC-33621P results are not very sensitive to the gap size distribution based on the sampling method used in the analysis.

The criticality analysis gap assumptions contain significant conservatisms to account for gap growth due to dissolution. Applying these results to the Campaign 6 completion date shows margin to the criticality analysis assumptions through October 2020. This margin more than adequately covers the period needed to complete the next GGNS BADGER test campaign, which is scheduled for the fall of 2013.

In order to adequately account for dissolution effects, an evaluation of the 2007 BADGER results was also performed. Gap growth attributable to 13 years of dissolution (December 2020) was added to the Region I BADGER results along with the 10% allowance for BADGER measurement uncertainty; see Figure 3-4, below.

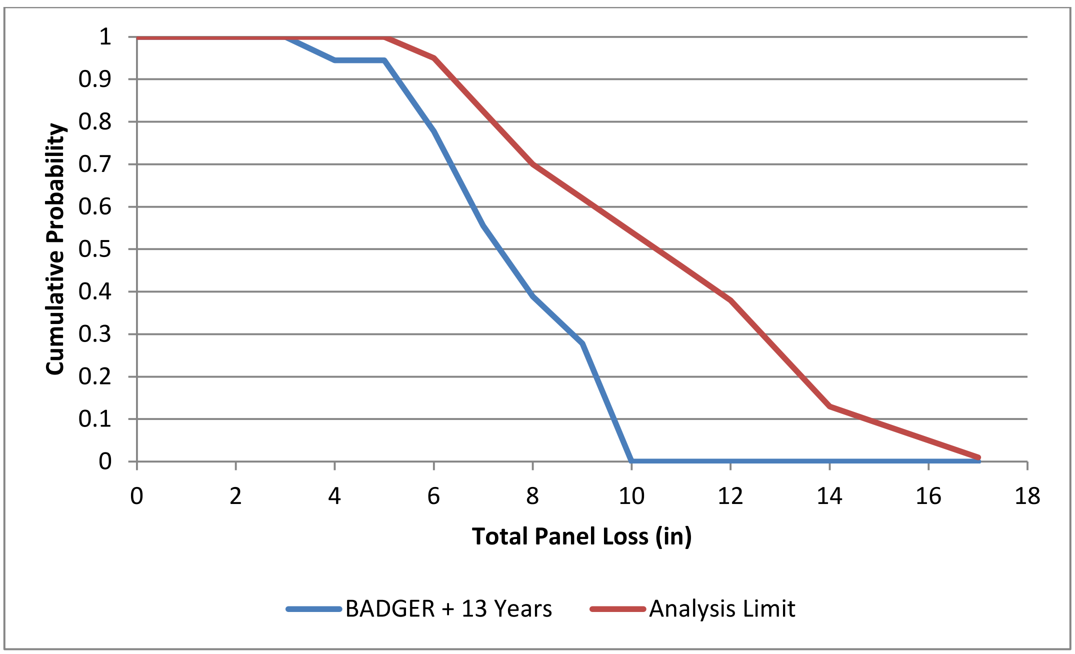

Figure 3-4: Cumulative Probability Distribution for Total Panel Loss of BADGER Data

This shows ample margin to the analysis assumptions through the next two BADGER test campaigns.

In order to ensure the criticality analysis remains applicable, the Boraflex monitoring program will be modified to incorporate updating the gap growth due to dissolution following each BADGER campaign. The updated dissolution rate will be applied to the most recent BADGER results through the end of the next BADGER test interval plus one year to confirm the continued applicability of the criticality analysis.

In addition to the gap growth due to dissolution discussed above, the reactivity effects of areas of local dissolution have been evaluated based on analysis of the BADGER test data. Detailed maps of the BADGER measurements for the Region I and Region II panels were analyzed to determine the location and geometry of areas where local dissolution is present. The evaluation considered dissolution features (e.g., scallops) associated with Boraflex gaps separately from dissolution features that are not associated with gaps. Features associated with gaps have the same location distribution as gaps. Features not associated with gaps are observed to be uniformly distributed axially, as shown in Figure 3-5, below.

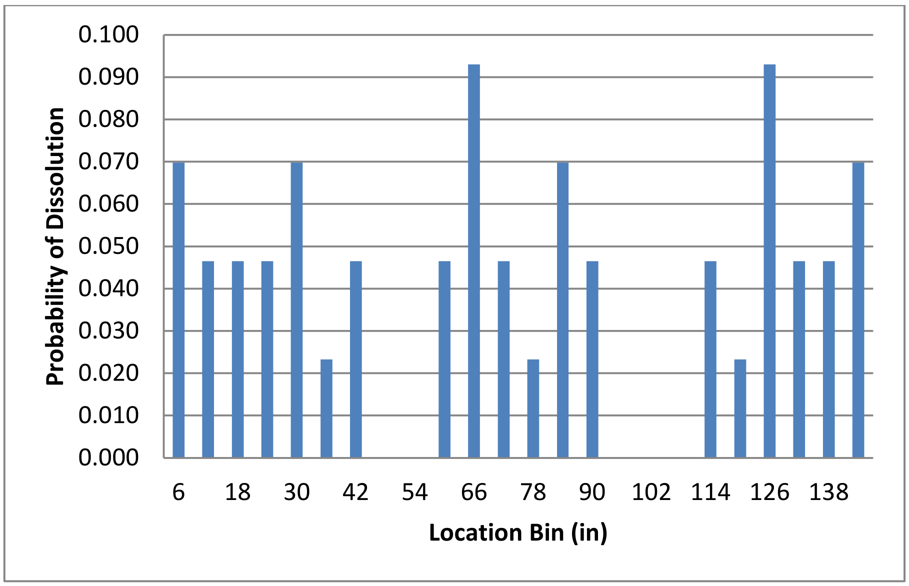

Figure 3-5: Axial Probability Distribution of Dissolution Features

For both sets of dissolution features, the dimensions of each feature were determined and a 95/95 upper limit was calculated, as summarized in Table 3-1, below.

Table 3-1: Local Dissolution Dimensions

	Non-Gap Local Dissolution			Region 2 Near Gap Local Dissolution		
	Fractional Diss Loss	Height (in)	Fractional Width	Fractional Diss Loss	Height (in)	Fractional Width
Mean	0.262	2.352	0.308	0.391	2.333	0.350
95/95 Limit	0.4331	4.9256	0.8058	0.685	4.392	0.618

The reactivity effects from including these features were determined by including dissolution features in the most reactive gap configuration case from the 200 gap configuration cases performed in response to RAI 3.c.2, below. Selecting the most reactive gap configuration case maximizes the reactivity impact of gap dissolution. Dissolution features not associated with gaps are modeled as a 50% reduction in panel thickness over an area that spans the entire panel width and extends 5 inches in length. Five such dissolution features are incorporated into each panel dispersed uniformly throughout the panel, beginning at least 12 inches from the top and bottom of the panel. The inclusion of five features in each panel is very conservative relative to the BADGER tests results provided in Figure 3-6, below.

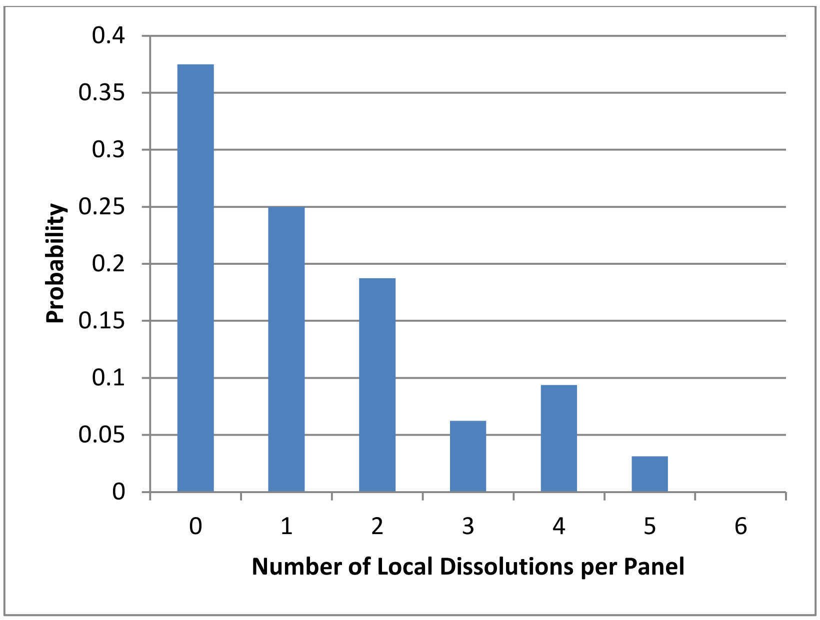

Figure 3-6: Probability of Having a Given Number of Local Dissolutions in a Panel

The dissolution features near gaps were modeled at the 95/95 analysis limits for the Region II panels from Table 3-1, above. These dimensions were selected from the Region II BADGER data since they result in a larger Boraflex volume loss and a slightly lower Boraflex areal density. The base model contains an equal probability of either one or two gaps occurring in each panel. Each panel includes a gap dissolution feature, described in Table 3-1, located adjacent to the edge of the largest gap on the side nearest the axial center of the panel. This location maximizes the reactivity impact of the dissolution. The modeling has additional conservatisms relative to the BADGER results, since the volume of Boraflex material removed in the model is based on the upper limit of each dimension. Actual measured individual dissolution features do not extend to the upper limit of each dimension. The density of the panel was increased to conserve the average panel Boron-10 (B-10) areal density for the base case, without any local dissolution effects. The resulting reactivity changes are attributable to the impact of dissolution effects on panels with the same average B-10 areal density, consistent with the BADGER and RACKLIFE results, which are reported as an average panel value. As shown in Table 3-2 below, the reactivity bias associated with these local dissolution effects is 0.00287 Δk, and is incorporated into the total bias in the criticality analysis.

Table 3-2: Spent Fuel Storage Rack Results Summary Region I

Term	Value
K_{Normal}	0.91645
$\Delta K_{LD\ Bias}$	0.00287
$\Delta K_{Total\ Bias}$	0.01126
$\Delta K_{Tolerances}$	0.00785
$\Delta K_{Uncertainty}$	0.01000
$K_{Max(95/95)}$	0.94556

Growth in the dissolution associated with gaps is addressed by the growth in gap size discussed above. As noted above, the model dissolution features have a significantly conservative total volume removal relative to the observed features. This provides significant margin to any small changes that might occur between BADGER tests when considering the panels used to develop the model had up to 21 years of in-service history. The Boraflex monitoring program will be modified to confirm the analyzed dissolution features remain bounding. The models for panel loss due to dissolution and the local features due to dissolution are based on GGNS-specific measurements. These were subject to the range of temperature and chemistry variations expected to occur in the future. Additionally, the panels are also subject to the expected range of manufacturing tolerances so the impacts of these tolerances are included when these models were developed. The treatment of discovered or new damage to the racks will be addressed a part of the normal corrective action process. As such, no additional allowance is merited to address these potential issues. Dissolution effects for the configuration associated with a potential seismic event are not specifically quantified due to the large margin to the acceptance criteria for that event.

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 208 panels in the 7th blackness test campaign conducted 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 Round 1 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.*

Response

Both BADGER and Blackness tests provide information on the local Boraflex performance. The BADGER test data provide more detailed, local information while the Blackness test provides data on the evolution of panel performance including performance at very high doses over a range of panel service life. The BADGER test extends the Blackness tests to higher service life and doses. The GGNS Boraflex performance associated with dose effects are shown in Figure 1 of GNRO-2010/00073. The response to RAI 3.a.1, above, provides information on the effects of service life. While uncertainties exist in the performance of the GGNS Boraflex, these uncertainties are addressed. The overall trends are consistent with the existing theory associated with Boraflex degradation. Credit for Boraflex performance is limited to moderate levels of dissolution and accumulated dose, with BADGER tests scheduled consistent with the expected rate of change. Assuming significantly more dire performance, such as the complete dissolution of panels, is not consistent with observations or existing theory, and would be primarily a matter of supposition. Such a set of assumptions are not merited.

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, examine the data for trends between the three shapes. Provide the results of these examinations. If there are differences, describe how such differences are considered in the criticality analysis.

   Response

   The vast majority of the racks in the GGNS spent fuel pool are made up of cruciform elements. "Tee" elements are only found on the edge of the racks, while "ell" elements are only found on some corners of the racks. Of the panels measured in the BADGER test campaign, only two are part of "tee" elements; none are part of "ell" elements. The available BADGER measurement data in Table 3-3, below, shows no bias toward "tee" elements having more loss than the cruciform elements.

Table 3-3: Comparison of Tee Element Panels to Nearby Cruciform Element Panels

Panel Location	Dose (rads)	Average Areal Density (g/cm^2)	Element Type
HH22N	3.02E+10	0.0213	Tee
HH24N	2.97E+10	0.0198	Tee
HH22S	3.02E+10	0.0173	Cruciform
HH24S	2.98E+10	0.0178	Cruciform
HH22E	3.06E+10	0.0184	Cruciform
HH24E	2.99E+10	0.0190	Cruciform

In addition to the review of the available measurements, an evaluation of the geometry of the various elements was performed to see if any bias would be expected. The ratio of the available flow area to cross sectional surface area of each element was determined. With larger available flow area, one would expect the dissolution of the Boraflex panel to tend to be greater since more water is available to interact with the Boraflex material. Since the elements are of different total size, the ratio of flow area to cross sectional surface area was used as a comparison between the elements. Table 3-4, below, shows the cruciform element has the largest flow area and flow area to cross sectional surface area ratio, which means it will tend to experience more interaction with water than the other elements, and potentially more dissolution. Note there are two variations in the “tee” element that have slightly different geometries. Since the cruciform is the dominant element used in the configuration of the racks, any effects from the larger flow area are reflected in the overall rack performance as measured by RACKLIFE. Therefore, RACKLIFE tends to overestimate the dissolution effects of the less frequently used “ell” and “tee” elements. Additionally, the bulk of the Boraflex measurements were performed on cruciform elements, since they are the most frequently used elements. To the extent that differences in flow area have an impact on dissolution, it is reflected in the measurement results.

Table 3-4: Boraflex Elements Flow Area Results

Element Type	Flow Area / Surface Area		Flow Area	
	Nominal	Min. Tolerance	Nominal	Min Tolerance
Ell	0.01625	0.13404	0.01365	0.10028
Tee A	0.01867	0.13276	0.02385	0.15160
Tee B	0.01695	0.13084	0.02166	0.14941
Cruciform	0.03150	0.15792	0.05323	0.23629

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 use of these 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.*

Response

See the response to RAI 3.a.1, above.

b. *Concerning gap simulation:*

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

]]

Response

There are several factors that affect the formation of gaps in a Boraflex panel. They include the accumulation of dose from the assembly in a specific cell and the adjoining cells. Since any single panel is equidistant to two storage cells in the GGNS design, the dose from both assemblies has the same potential impact. Other factors include variations in the Boraflex enclosure that would provide areas of resistance to Boraflex shrinkage. The dose from a given fuel assembly stored in a given cell would contribute to degradation patterns that would be correlated between the panels surrounding the assembly. However, the dose from adjacent assemblies and the Boraflex enclosure details would not contribute to the correlated degradation patterns.

In order to evaluate the presence of potential correlations between degradation patterns, the results from the 7th Blackness test campaign were analyzed. The location of gaps for each panel was binned using the same 6-inch nodalization used to develop the axial distributions in NEDC-33621P. For each fuel storage cell, the number of occurrences of two gaps occupying the same bin was tallied. A similar tally for three gaps occupying the same bin was made. The probability that these events would occur by chance was determined based on the number of opportunities (i.e., gap alignments) for a given storage configuration and the likelihood of any gap occupying a given axial location. Configurations with more gaps have more opportunities for alignment and, therefore, more gaps align by chance. The likelihood of any given gap occupying a given axial location was taken as a uniform value of 1/12, similar to the axial probability distribution in the criticality safety analysis. The total number of occurrences of two co-planar gaps and three co-planar gaps are presented in Table 3-5, below, along with the corresponding expected number of these configurations based on a random probability. While some gap alignment does occur, it is largely attributable to chance, consistent with the axial probability distribution in the criticality analysis.

Table 3-5: Total Number of Co-Planar Gaps

Occurrences	2 Co-Planar Gaps	3 Co-Planar Gaps
Expected	85.42	9.81
Measured	72	8

A potential correlation in total panel loss was evaluated based on panels above the 75 percentile of total panel loss. This covers panel losses from 3.2 inches to 13.5 inches. The number of storage locations with combinations of two panels in this range was tallied. These results were compared to the expected value based on the number of combinations with each panel having a 25% chance of occurrence. The results are shown in Table 3-6, below.

Table 3-6: Cells with 2 Panels Having a Total Panel Loss in Each Percentile Bin

Bin	4th	3rd	2nd	1st
	Probability			
Expected	0.375	0.375	0.375	0.375
Measured	0.3462	0.2885	0.3654	0.3077
	Number			
Expected	19.5	19.5	19.5	19.5
Measured	18	15	19	16

All four percentile bins are below the expected value based on a random probability of occurrence. The individual gap size distributions were not quantified since both the gap location and the total panel loss evaluations do not indicate a covariance performance and, as demonstrated in response to RAI 3.b.2, below, the criticality analysis results are not sensitive to the gap size distribution based on the method used to assign gap sizes.

2) *RAI 4.d requested information about [[*

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Response

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[[

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Figure 3-7: [[ ]]

[[

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Figure 3-8: [[ ]]

[[

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Table 3-7: [[ ]]

[[		
		]]

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

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Response

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c. Concerning analysis of the results:

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

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9 [[

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Response

See the response to RAI 3.c.2, below.

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

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Response

To further demonstrate the normality and conservatism of results from the Monte Carlo sampling process, [[ ]] were run. A histogram of the results is provided in Figure 3.9, below. This figure provides visual evidence that the distribution is normal in its characterization, and that a larger set of cases still passes the Anderson-Darling normality test.

[[

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Figure 3-9: [[ ]]

[[
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Table 3-8: [[ ]]

[[		
		]]

RAI 4

Some of the prior RAIs question how temperature variation is modeled[10]. 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 [[ ]].

Response

Additional moderator temperature cases were run at [[ ]] for both Region I and Region II, where the moderator density was modified according to the corresponding temperature. For Region I, additional runs between [[ ]] were run at [[ ]] intervals. These cases were run for more particles per cycle and more cycles than the original analysis cases to reduce the MCNP standard deviation to more clearly show the k_{eff} day with moderator temperature. As seen in Figure 4-1, below, the moderator temperature that produces the highest k_{eff} for Region I is [[ ]]. As shown in Figure 4-2, below, the additional Region II cases demonstrate [[ ]]. The k_{eff} values for [[ ]] from the new runs for both Regions I and II were within 1σ of the values previously reported.

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[10] Responses to RAIs 10, 12, and 19 provided in Entergy letter to the NRC (GNRO-2012/00120), *Request for Additional Information – GGNS Criticality Safety Analysis License Amendment Request,* October 1, 2012 (ADAMS Accession No. ML12276A152)

[[

]]

Figure 4-1: [[ ]]

[[

]]

Figure 4-2: [[ ]]

RAI 5

The following validation issues were not addressed or were addressed incorrectly in the RAI responses.[11]

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

Response

An updated summary of critical experiments with the corrected information is shown in Table 5-1, below. Of the critical benchmark experiments included, [[ ]] contain UO_2-Gd rods similar to those used in BWR fuel bundles and are, thus, adequate for validating Gd. [[ ]]. The Gd worth of the Region I and Region II lattices at peak reactivity are [[ ]], respectively.

---

[11] Responses to RAIs 12, 19, and 22 provided in Entergy letter to the NRC (GNRO-2012/00120), *Request for Additional Information – GGNS Criticality Safety Analysis License Amendment Request,* October 1, 2012 (ADAMS Accession No. ML12276A152)

Table 5-1: Summary of the Critical Benchmark Experiments

[[				
				]]

As shown in Figure 5.1, below, the Gd worth range of the critical benchmark experiments adequately covers the Gd worth of the bundles used in this analysis. Since the form of the Gd and the Gd worth of the critical benchmark experiments are similar to the Gd analyzed in the analysis, the validation of Gd is sufficiently covered by the critical benchmark validation.

To assess any Gd worth trend in the critical benchmark, the Gd worth was plotted against the k_{calc} results independently for each case that was analyzed. This plot is provided in Figure 5-1, below, and a trending results summary is provided in Table 5-2, below. [[

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[[

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Figure 5-1: [[ ]]

Table 5-2: Trending Results Summary

Trend Parameter	Intercept	Slope	r^2	$\tilde{\chi}^2$	Valid Trend
Gd Worth	[[				]]

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.*

[[

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Response

Table 5-3, below, lists the nuclides used in the analysis for the spent fuel isotopics. As stated in the response to RAI 22 provided in GNRO-2012/00120, [[

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Table 5-3: Nuclides Credited in Spent Fuel for the Analysis

[[			
			]]

As specified in Section 4.a.i of the NRC Interim Staff Guidance document DSS-ISG-2010-01, *Spent Fuel Pool Criticality Analysis*, the required additional margin to account for the validation extension for fission products can then be applied by increasing the validation uncertainty. [[

]], as assessed in the sensitivity study performed, is defined as the appropriate bias uncertainty adder for the validation which is included in the analysis.

[[

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RAI 6

In response to RAI 26[12], 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.

Response

The interface condition is shown below in Figure 6-1 and the interface condition with a misload is shown in Figure 6-2. The same assumptions and boundary conditions used in the interface evaluation in NEDC-33621P were used for this misload evaluation. The Δk worth for the misload condition at the interface ([[ ]]) is much less than the Δk worth for the misload condition in Region II included in NEDC-33621P ([[ ]]).

[[

]]

Figure 6-1: [[ ]]

[12] Provided in Entergy letter to the NRC (GNRO-2012/00120), *Request for Additional Information – GGNS Criticality Safety Analysis License Amendment Request,* October 1, 2012 (ADAMS Accession No. ML12276A152)

[[

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Figure 6-2: [[ ]]

RAI 7

Crediting degraded Boraflex to perform its safety function is problematic as it may react differently to normal and abnormal events than when it is in a pristine condition (i.e., it may settle during normal or seismic events, it may be susceptible to accelerated degradation during loss of cooling events). Describe the expected range of conditions considered in this analysis and how Boraflex is expected to respond under those conditions. Provide the rational for the Boraflex response. Describe how the analysis either bounds the expected Boraflex response or the licensee's controls to ensure the current analysis remains bounding.

Response

Boraflex performance during a seismic event is addressed in the response to RAI 14 provided in GNRO-2012/00120. The loss of cooling to the spent fuel pool results in a gradual increase in pool water temperature, up to the point of bulk boiling at the pool surface. As described in EPRI Report TR-107333, “The RACKLIFE Boraflex Rack Life Extensin Computer Code: Theory and Numerics,” silica release from Boraflex is dependent on pool water temperature. Extending the measured silica release rates to 212°F with a quadratic fit results in a silica release rate that is approximately 15 times higher than the rate for the normal operating temperature (~120°F). While this release rate is significantly higher than normal pool operation, the duration of such an event limits the impact on the overall performance of Boraflex. Even if such an event persisted for a full week, the release of which is equivalent to about 3.5 months of normal operation, it would have a small impact relative to the overall impact of over 20 years of operation.

In the event cooling to the spent fuel pool is lost, site procedures direct operators to inject water from available cooling water sources to ensure the level in the spent fuel pool remains adequate for cooling. More severe events that postulate the loss of a large fraction of pool inventory are beyond the design basis of the system and are being address as part of the GGNS FLEX mitigation strategy, which has been submitted to the NRC in accordance with NRC Order EA-12-051.[13]

RAI 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.*

   Response

   There is currently no specific plan or schedule to replace or install additional neutron-absorbing material at this time.

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).*

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13 Entergy letter to the NRC (GNRO-2013/00015), *Overall Integrated Plan in Response to March 12, 2012, Commission Order to Modify Licenses With Regard To Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)*, February 28, 2013 (ADAMS Accession No. ML13059A316)

Response

GGNS has scheduled a BADGER test for the fall 2013 using the recently developed BADGER system to reduce the uncertainty between RACKLIFE and BADGER. This uncertainly is confirmed as part of the Boraflex monitoring program following each BADGER test. Additionally, the potential impact of degradation on RACKLIFE and BADGER uncertainties is limited by the amount of Boraflex degradation assumed in the Region I analysis. Storage cells with degradation beyond those limits are converted to Region II checkerboard configurations where no Boraflex is credited.

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

Response

As noted in the response to RAI 3.a.1, above, the results of in-situ tests are used to project performance through the next test interval. There are no predefined plans to increase the frequency of testing beyond the current 5-year frequency.

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).*

Response

A number of options exist maintain sufficient spent fuel storage capacity if the racks are largely converted to Region II configuration. These include additional dry cask storage, rack replacement and neutron absorber inserts. The most desirable option has not yet been determined.

ATTACHMENT 4

GRAND GULF NUCLEAR STATION

GNRO-2013/00050

<u>LIST OF REGULATORY COMMITMENTS</u>

LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
In order to ensure the criticality analysis remains applicable, the Boraflex monitoring program will be modified to incorporate updating the gap growth due to dissolution following each BADGER campaign. The updated dissolution rate will be applied to the most recent BADGER results through the end of the next BADGER test interval plus one year to confirm the continued applicability of the criticality analysis.		✓	
The Boraflex monitoring program will be modified to confirm that the analyzed dissolution features remain bounding.		✓	