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

August 18, 2015

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 - ISSUANCE OF AMENDMENT RE: ADOPTION OF SINGLE FLUENCE METHODOLOGY (TAC NO. MF5303)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 204 to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1 (GGNS). This amendment revises the Updated Final Safety Analysis Report (UFSAR) in response to your application dated November 21, 2014, as supplemented by letters dated February 18, March 30, May 8, June 11, and August 10, 2015.

The amendment revises the GGNS UFSAR from the use of two different fluence calculational methods to the use of a single 3D fluence methodology for 0 to 54 effective full power years, the end of extended operations.

Enclosure 2 to this letter contains Proprietary Information. Upon separation from Enclosure 2, this letter is DECONTROLLED.

The NRC had determined that the related safety evaluation (SE) contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations*, Section 2.390, "Public inspections, exemptions, requests for withholding." Accordingly, the NRC staff has also prepared a non-proprietary version of the SE, which is provided in Enclosure 3. The proprietary version of the SE is provided in Enclosure 2. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

Alan Wang

Alan B. Wang, Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-416

Enclosures:
1. Amendment No. 204 to NPF-29
2. Safety Evaluation (proprietary)
3. Safety Evaluation (non-proprietary)

cc w/encls 1 and 3: Distribution via Listserv

ENCLOSURE 1

AMENDMENT NO. 204

TO FACILITY OPERATING LICENSE NO. NPF-29

ENTERGY OPERATIONS, INC.

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

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

ENTERGY OPERATIONS, INC.

SYSTEM ENERGY RESOURCES, INC.

SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

ENTERGY MISSISSIPPI, INC.

DOCKET NO. 50-416

GRAND GULF NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 204
License No. NPF-29

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Operations, Inc. (the licensee), November 21, 2014, as supplemented by letters dated February 18, March 30, May 8, June 11, and August 10, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, by Amendment No. 204, the license is amended to authorize revision to the Updated Final Safety Analysis Report (UFSAR) as set forth in the application for amendment by Entergy Operations, Inc., dated November 21, 2014, as supplemented by letters dated February 18, March 30, May 8, June 11, and August 10, 2015. Entergy Operations, Inc., shall update the UFSAR to reflect the revised licensing basis authorized by this amendment in accordance with 10 CFR 50.71(e).

3. The license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance. The licensee will perform the following as described in the licensee's letters dated June 11, and August 10, 2015, and the NRC staff's safety evaluation for this amendment:

Entergy will identify the outside of the beltline region dosimetry sample locations	October 30, 2015
Entergy will revise the affected sections of Chapter 4 of the GGNS UFSAR upon approval of the Fluence Calculation Methodology LAR	October 30, 2015
Entergy will schedule collection of samples from outside the beltline region	December 30, 2015
Entergy will confirm that future C/M [calculated-to-measure] fluence values at the dosimetry sample locations are reasonably close to one	November 30, 2016
Entergy will include the definition of "reasonably close to one" regarding C/M fluence values at the dosimetry sample locations	November 30, 2016
Entergy will provide plans to address if future C/M fluence values at the dosimetry sample locations are not reasonably close to one	December 30, 2016

In addition, the licensee shall include the revised information in the Grand Gulf Nuclear Station, Unit 1 Updated Final Safety Analysis Report in the next periodic update in accordance with 10 CFR 50.71(e), as described in the licensee's application dated November 21, 2014, as supplemented by letters dated February 18, March 30, May 8, June 11, and August 10, 2015, and the NRC staff's safety evaluation for this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Meena K. Khanna, Chief
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: August 18, 2015

<u>ENCLOSURE 3</u>
(NON-PROPRIETARY)

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR

RELATED TO AMENDMENT NO. 204

TO FACILITY OPERATING LICENSE NO. NPF-29

ENTERGY OPERATIONS, INC.

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

Proprietary information pursuant to Section 2.390 of Title 10 of the *Code of Federal Regulations* has been redacted from this document.

Redacted information is identified by blank space enclosed within double brackets

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 204 TO

FACILITY OPERATING LICENSE NO. NPF-29

ENTERGY OPERATIONS, INC., ET AL.

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By application dated November 21, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14325A752), as supplemented by letters dated February 18, March 30, May 8, June 11, and August 10, 2015 (ADAMS Accession Nos. ML15049A536, ML15089A524, ML15128A552, ML15162B088, and ML15222B264, respectively), Entergy Operations, Inc. (the licensee), requested to revise the licensing basis to adopt a new fluence methodology for the Grand Gulf Nuclear Station, Unit 1 (GGNS). The supplemental letters dated March 30, May 8, June 11, and August 10, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 31, 2015 (80 FR 17087). The NRC's staff's notice considered the November 21, 2014, application and the supplemental letter dated February 18, 2015.

Specifically, the new Fluence Calculational Methodology will provide an analysis for a single fluence methodology from 0 effective full power years through the end of extended operations. The new methodology will be incorporated into GGNS Updated Final Safety Analyses Report (UFSAR).

2.0 REGULATORY EVALUATION

Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001 (ADAMS Accession No. ML010890301), describes methods and assumptions acceptable to the NRC staff for determining the reactor pressure vessel (RPV) neutron fluence with respect to the General Design Criteria (GDC) contained in Appendix A of 10 CFR 50, "General Design Criteria for Nuclear Power Plants." In consideration of the guidance set forth in RG 1.190, GDC 14, 30, and 31 are applicable. GDC 14, "Reactor coolant pressure boundary [RCPB]," requires that the RCPB "shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture." GDC 30, "Quality of reactor coolant pressure boundary," requires among other things, that components comprising the RCPB "be designed, fabricated, erected, and tested to the highest quality standards practical." GDC 31, "Fracture prevention of reactor coolant pressure boundary," pertains to the design of the RCPB, stating:

> The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

3.0 TECHNICAL EVALUATION

3.1 Background

In the license amendment request dated November 21, 2014, the licensee states, in part:

> During the GGNS License Renewal process, it was determined that the current fluence methodology should have received NRC approval prior to being utilized. This resulted in a Severity Level IV Non-Cited Violation (NCV) of 10 CFR 50.59, "Changes, Tests, and Experiments" involving failure to obtain a license amendment pursuant to 10 CFR 50.90 prior to implementing a new method of evaluation for determining reactor vessel neutron fluence, as documented in the Grand Gulf Nuclear Station - NRC Integrated Inspection Report 05000416/2013004, dated November 27, 2013 [ADAMS Accession No. ML13331B343].

Consequently, the licensee submitted this application as the corrective action to address the NCV.

3.2 Summary of Technical Information Provided by the Licensee

The November 21, 2014, letter included three attachments. Attachment 1 provides a description of the proposed change. Attachment 2 provides a topical report from MP Machinery and Testing, LLC (MPM) covering benchmarking of the single fluence method being adopted. Attachment 3 provides a report describing how the single fluence method is applied to GGNS over the entire period of extended operation (i.e., from 0 to 54 effective full power years).

By letter dated February 18, 2015 (ADAMS Accession No. ML15036A564), the NRC staff informed the licensee that the application was unacceptable, as it did not provide sufficient detail regarding the proposed fluence methodology, and provided the licensee an opportunity to supplement the application. On February 18, 2015, the licensee submitted a letter, which included three attachments responding to the NRC staff's acceptance review findings. Attachment 1 provided a partial response to the requests for additional information (RAIs) from the NRC staff's acceptance review. Attachment 2 provided a revised single fluence method benchmarking topical report from MPM. Attachment 3 provided a revised report describing how the single fluence method is applied to GGNS.

By letter dated March 30, 2015, the licensee provided the information needed to address the remaining acceptance review RAI in Attachment 1, as a revised single fluence method benchmarking topical report from MPM.

By email dated March 30, 2015 (ADAMS Accession No. ML15090A139), the NRC sent an RAI with four questions to the licensee. By letter dated May 8, 2015, the licensee provided two attachments, responding to three of the four RAI questions. Attachment 1 contained the non-proprietary RAIs responses and Attachment 2 contained a proprietary version of the RAI responses. In a letter dated June 11, 2015, the licensee submitted the response to RAI 4.

In summary, the scope of the NRC staff evaluation covered the following:

- November 21, 2014, letter, Attachment 1 describing the proposed changes associated with the single fluence method;
- February 18, 2015, letter, Attachment 1 containing a partial RAI response to the NRC letter dated February 18, 2015, regarding the license amendment request (LAR) acceptance review;
- February 18, 2015, letter, Attachment 3 containing a revised report describing how the single fluence method is applied to GGNS;
- March 30, 2015, letter, Attachment 1 containing a revised single fluence method benchmarking report;
- May 8, 2015, letter, Attachments 1 and 2 containing RAI 1, 2, and 3 responses to the March 30, 2015, RAI request; and
- June 11, 2015, letter, Attachment 1 containing the RAI 4 response to the March 30, 2015, RAI request.

By letter dated August 10, 2015, the license clarified that the licensing basis document that will be revised is the UFSAR. This letter provided a draft of the revised UFSAR pages for information.

3.3 NRC Staff Evaluation

In general, the guidance provided in RG 1.190 indicates that the following attributes comprise an acceptable fluence calculation:

- A fluence calculation performed using an acceptable methodology.
- Plant-specific qualification by comparison to measured fluence values.
- Benchmark comparison to approved results of a test facility.
- Analytic uncertainty analysis identifying possible sources of uncertainty.

Reviewing MPM's single fluence method benchmarking report, the NRC staff found that fluence calculations are performed in a manner consistent with the guidance set forth in RG 1.190. MPM summarizes the fluence methodology consistency with RG 1.190 in Section 2.2 of the benchmarking report.

A solution to the Boltzmann transport equation is approximated using the three-dimensional (3D) discrete ordinates code known as Three-dimensional Discrete Ordinates Transport, Version 2.7.3 (TORT), which is available from the Radiation Safety Information Computational Center maintained by Oak Ridge National Laboratory (ORNL). Three-dimensional flux solutions are directly calculated rather than constructed using a synthesis of azimuthal, axial, and radial flux. The licensee uses an appropriate cross-section library based on ENDF/B-VI nuclear data, and intended for use in light-water reactor shielding and RPV dosimetry applications. Numeric approximations include a P3 Legendre expansion to represent anisotropic scattering and S16 angular quadrature for angular flux discretization. These cross-section data and modeling approximations are consistent with the modeling guidance contained in RG 1.190.

Space and energy dependent core power distributions and associated core parameters are treated on a fuel cycle and plant specific basis in order to obtain neutron source distributions that are appropriately averaged over each fuel cycle. Fuel cycle and plant specific treatment includes explicit accounting of initial enrichment, burnup, and axial power distributions. Neutron source energy spectral effects, neutrons per fission, and energy per fission are accounted for by using appropriate fission fractions for the fissionable uranium and plutonium isotopes based on the initial enrichment and burnup history of the fuel assemblies that are the major contributors to the RPV fluence. The staff confirmed that the licensee's neutron source and transport calculations, as described above, were performed consistent with the modeling guidance set forth in RG 1.190.

Methods Qualification

Regulatory Position 1.4.1, 1.4.2, and 1.4.3 from Table 1, "Summary of Regulatory Positions on Calculation and Dosimetry," of RG 1.190 states the following:

> Methods Qualification. The calculational methodology must be qualified by both (1) comparisons to measurement and calculational benchmarks and (2) an analytic uncertainty analysis. The methods used to calculate the benchmarks

must be consistent (to the extent possible) with the methods used to calculate the vessel fluence. The overall calculational bias and uncertainty must be determined by an appropriate combination of the analytic uncertainty analysis and the uncertainty analysis based on the comparisons to the benchmarks.

The MPM benchmark report discusses methods qualification of both two-dimensional (2D) and 3D RG 1.190 methods. However, the method used for GGNS is the 3D method. Therefore, the NRC staff evaluation focused on the benchmarking of the 3D method as discussed below.

<u>Operating Reactor Measurement Benchmarking</u>

Operating reactor measurement benchmarking was performed specific to GGNS as described in RG 1.190. A single GGNS Cycle 1 dosimetry benchmark calculation was compared with measurement with excellent agreement demonstrating the ability to select and implement appropriate: transport method options, nuclear data libraries, material specification, geometry, etc., specific to GGNS. The staff concluded that the licensee's approach is acceptable, since it is consistent with the guidelines provided in RE 1.190.

<u>Pressure Vessel Simulator Benchmarking</u>

Reactor pressure vessel simulator benchmarking was performed in accordance with RG 1.190. Calculations were compared with the benchmark measurements from the Poolside Critical Assembly (PCA) simulator at ORNL as documented in the licensee's letter dated March 30, 2015, Attachment 1. The NRC staff determined this to be an acceptable test facility as it is specifically referenced in RG 1.190. The PCA benchmark calculations (with seven data points) were compared with the benchmark measurements with excellent agreement demonstrating the ability to model pressure vessel geometry and dosimetry at various locations. The staff concluded that the licensee's approach is acceptable, since it is consistent with the guidelines provided in RE 1.190.

<u>Boiling-Water Reactor Calculational Benchmarking</u>

A boiling-water reactor (BWR) calculational benchmark was not performed for the 3D method being qualified. However, extensive validation of past methods (i.e., using 2D synthesis methods) demonstrates excellent agreement between the MPM calculations and the benchmark reference calculations. Limited comparison of the current method (based on state-of-the-art 3D methods) to past methods 2D-based methods shows that similar results can be expected with the 3D method for similar types of fluence calculations. Additionally, it has been demonstrated with the operating reactor measurement benchmarking that the 3D method is fully capable of producing fluence estimates for GGNS. Therefore, the staff concludes that the licensee's use of the BWR calculational benchmark results from previous methods (i.e., using 2D synthesis methods), which have shown to provide results comparable to the 3D method being qualified for similar types of fluence calculations, is acceptable to satisfy the BWR calculational benchmarking requirement as discussed in RG 1.190.

Fluence Calculational Uncertainty

Regulatory Position 1 and 1.4.3 from Table 1, "Summary of Regulatory Positions on Calculation and Dosimetry," of RG 1.190 states the following:

> **Fluence Calculational Uncertainty**. The vessel fluence (1 sigma) calculational uncertainty must be demonstrated to be ≤ [less than or equal to] 20% for RT_{PTS} [reference temperature, RT_{NDT}, evaluated for the end of life fluence] and RT_{NDT} [reference temperature for a reactor vessel material, under any conditions] determination. In these applications, if the benchmark comparisons indicate differences greater than 20%, the calculational model must be adjusted or a correction must be applied to reduce the difference between the fluence prediction and the upper 1-sigma limit to within 20%. For other applications, the accuracy should be determined using the approach described in Regulatory Position 1.4, and an uncertainty allowance should be included in the fluence estimate as appropriate in the specific application.

Within the beltline, all uncertainties are calculated to be below 20 percent, therefore, no bias correction is required as indicated in RG 1.190. However, for some limited locations outside of the original beltline, uncertainties are estimated to exceed 20 percent.

For all uncertainties greater than 20 percent, bias and uncertainty correction is applied conservatively with respect to Equation 6 of RG 1.190, as discussed in the RAI 1 response provided by letter dated May 8, 2015.

RG 1.190 states that for uncertainties greater than 30 percent, "the methodology [of RG 1.190] is not applicable and the application will be reviewed on an individual basis." In the response to RAI 1b provided in the licensee's letter dated February 18, 2015, it states, in part that **[[**

]] This uncertainty treatment is more conservative than what Equation 6 of RG 1.1.90 implies is appropriate since only the uncertainty that is greater than 20 percent is applied as part of the calculated fluence correction factor in Equation 6. The NRC staff agrees that it is acceptable to apply bias and uncertainty correction, based on a conservatively modified RG 1.190, Equation 6, which applies uncertainty correction based on the full magnitude of the greater than 20 percent uncertainty.
The NRC staff understands that the uncertainty analysis for the greater than 30 percent uncertainty RPV location – **[[**

]] – is being dominated by uncertainty in the steam density above the core since fluence estimates at these above-core RPV locations are highly sensitive to changes in above-core steam density. **[[**

]] The discussion is expanded further in the report documenting how the single fluence method is applied to GGNS, where it is stated that

[[

]]

Furthermore, in the response to RAI 2 in the letter dated May 8, 2015, the licensee describes various steady-state thermal-hydraulic analyses performed for Cycle 20 and Cycle 21 core designs during Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operation to demonstrate the appropriateness of the above-core steam density uncertainty analysis. The thermal-hydraulic analyses determined that [[

]] The corresponding above-core void fraction based on the expected cycle average core flow was determined to be [[

]]. The response further states that although the MELLLA+ operating domain allows operation with core flow as low as 80 percent, core operating design limits would not allow for operation at core flows this low for extended periods of time, maintaining that the multi-cycle averaged core flow is the more appropriate flow rate for ensuring that the fluence analysis assumptions remain applicable. However, the thermal-hydraulic analyses demonstrated that cycle average core flows as low as 81 percent would still meet the fluence analysis upper limit void fraction assumption of 0.82. Finally, the integrated multi-cycle average core flow will be confirmed to exceed 81 percent for core designs using the MELLLA+ operating domain. It is also noted that there is no credit for void fraction distribution effects near the core edge where lower void fractions will occur.

In order to better understand the safety significance of the LPCI nozzle N6 with the high fluence uncertainty at the edge of the new beltline in relation to other RPV beltline components with relatively low fluence uncertainties, the NRC staff requested that the licensee address what the potential is for RPV components with high fluence uncertainties that are outside of the original beltline to be limiting with respect to pressure/temperature (P/T) curve generation over the entire period of extended operation. In the response to RAI 3, in the licensee's letter dated May 8, 2015, it provided a detailed analysis showing that the P/T curves are limited by LPCI nozzle N6 for some pressure ranges, which highlights the importance of ensuring that the above-core void fraction is appropriately treated.

Recognizing the importance of ensuring that the above-core void fraction is appropriately treated, the licensee stated, in part, the following in the response to RAI 1a in its letter dated February 18, 2015, Attachment 1:

> Installing dosimetry capsules and/or taking scrapings in specified areas outside of the beltline region would provide dosimetry data for a future benchmark analysis outside of the beltline region. The uncertainty in the fluence calculations at locations above the top of the core is dominated by uncertainty in the water density. Taking scrapings and/or inserting dosimetry in these locations (during future refueling outages) would not only provide benchmarking data, but it would also provide the data needed to check the output from thermal hydraulics codes

that can be used in future improvements of the upper region water density modeling.

To confirm that the above-core void fraction is being appropriately treated, the NRC staff asked the licensee to provide a formal regulatory commitment in order to qualify the 3D fluence method for fluence calculations outside of the original beltline region and to provide more detail regarding the plans for installing dosimetry capsules and/or scrapings including the proposed locations for dosimetry capsule installation and/or scrapings. In its response to RAI 4, the licensee provided formal regulatory commitments to plan, schedule, and collect samples from outside the original beltline region during the refueling outage, which is scheduled at the end of 2016 for subsequent dosimetry analysis and verification of the above-core water density distribution modeling assumptions. The licensee noted that if technical problems should arise in the collection of the samples, plans would be made to assure that the additional actions in order to prepare for sample collection will be completed no later than the 2018 refueling outage. The licensee has made a regulatory commitment to confirm that the calculated-to-measured (C/M) fluence at the various dosimetry locations are reasonably close to one. The NRC staff reviewed the licensee's plan to take select RPV scrapings for performing dosimetry analysis and finds it acceptable to allow for future qualification of the 3D fluence method for fluence calculations outside of the original beltline region.

After reviewing the various methods for the qualification of benchmarking activities and uncertainty analyses supporting the calculation of the RPV neutron fluence at GGNS, the NRC staff has reasonable assurance that the neutron fluence can be appropriately estimated for all of the RPV components over the entire period of extended operation.

Conclusion

The NRC staff has reviewed the proposed 3D fluence method for use in calculating the RPV neutron fluence at GGNS for 0 effective full power years through the end of extended operations and concludes that the method has been applied consistent with the applicable requirements discussed in RG 1.190, and therefore, is acceptable. This safety evaluation applies only to the 3D method described in the benchmarking report of the 3D fluence method being adopted and the report describing how the 3D fluence method is applied to GGNS. No other methods (i.e., 2D based methods) were reviewed by the NRC as part of this request. The NRC notes that 2D based methods have shown to be less accurate and have higher uncertainties associated with the above core RPV components. In addition, the NRC staff has reviewed the revised UFSAR pages provided in the August 10, 2015, letter and has determined they reflect the approval of the 3D fluence methodology, as provided in this license amendment request.

4.0 REGULATORY COMMITMENTS

In its letters dated June 11, and August 10, 2015, the licensee proposed the following Regulatory Commitments:

COMMITMENT	SCHEDULED COMPLETION DATE
Entergy will identify the outside of the beltline region dosimetry sample locations	October 30, 2015
Entergy will revise the affected sections of Chapter 4 of the GGNS UFSAR upon approval of the Fluence Calculation Methodology LAR	October 30, 2015
Entergy will schedule collection of samples from outside the beltline region	December 30, 2015
Entergy will confirm that future C/M fluence values at the dosimetry sample locations are reasonably close to one	November 30, 2016
Entergy will include the definition of "reasonably close to one" regarding C/M fluence values at the dosimetry sample locations	November 30, 2016
Entergy will provide plans to address if future C/M fluence values at the dosimetry sample locations are not reasonably close to one	December 30, 2016

The licensee has proposed these regulatory commitments as "One-Time Actions." The licensee also noted in its letter dated June 11, 2015, that:

> If technical evaluation of collecting samples (i.e. drilling holes) from the shroud and/or top guide prohibits sample collection or determines additional actions are required which cannot be performed prior to the 2016 refueling outage, plans will continue to perform those additional actions in order to prepare for sample collection during the 2018 refueling outage.

The NRC staff used the requirements discussed in RG 1.190 to evaluate the licensee's 3D fluence methodology. The NRC staff has determined that these proposed regulatory commitments are required for the approval of the UFSAR change and is part of the basis for NRC staff approval of this license amendment. As such, the NRC staff has determined that these regulatory commitments are needed to provide a basis to qualify the 3D fluence method for fluence calculations outside of the original beltline region and also, to provide more detail regarding the plans for installing dosimetry capsules and/or scrapings including the proposed locations for dosimetry capsule installation and/or scrapings. The NRC staff also agrees that if unforeseen situations occur, sample collection may be delayed but no later than the 2018 refueling outage. As such, these regulatory commitments must be incorporated into the licensing basis documents (in this case the UFSAR) and any future changes to this action must be evaluated under the criteria of 10 CFR 50.59. The NRC staff has elevated these actions to implementation requirements as described in the amendment issuance pages. Per the implementation requirements, these actions will be incorporated into the licensee's UFSAR upon implementation of this amendment. Therefore, the actions, originally proposed as regulatory commitments, are no longer regulatory commitments and cannot be modified or deleted by the licensee under their commitment management program.

5.0 PUBLIC COMMENT

On March 31, 2015, the NRC staff published a "Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed Significant Hazards Consideration Determination, and Opportunity for Hearing," in the *Federal Register* associated with the proposed amendment request (80 CFR 17083). In accordance with the requirements in 10 CFR 50.91, "Notice for public comment: State consultation," the notice provided a 30-day period for public comment on the proposed no significant hazards consideration (NSHC) determination. In addition, in the Notice Section A. "Opportunity to Request a Hearing and Petition for Leave to Intervene," provides the public with the process for which a hearing may be requested. Public comments were received on April 30, 2015 (ADAMS Accession No. ML15138A095) regarding this fluence calculational methodology license amendment request . Some of the issues discussed in the public comments do not specifically pertain to the proposed NSHC determination.

For example, the commenter raises concerns regarding certain materials-related topics. The commenter notes that:

> Old nuclear reactors, such as Grand Gulf, are more subject to embrittlement failure due to neutron & hydrogen attack. Failure could also be induced by corrosion. This problem worsened by uprates, as at Grand Gulf. They are further stress old RPVs. Sudden failure of the RPV would lead to a catastrophic nuclear disaster.

By letter dated July 18, 2012, the NRC staff issued Amendment No. 191, "Extended Power Uprate [EPU]." As part of that review, the NRC staff did an extensive review of the effects of the EPU on the reactor pressure vessel and internals (RPV and RVI). The NRC staff reviewed the effects of EPU on the reactor vessel material surveillance program, the upper-shelf energy (USE) requirements, the Pressure-Temperature Limit requirements (PTLR), RPV circumferential weld properties and irradiation-assisted stress-corrosion cracking (IASCC). The NRC staff concluded that the licensee had performed an acceptable assessment of the effects of operating at EPU conditions on the RPV and RVI components and that the licensee has programs that will continue to maintain an acceptable course of action for managing the susceptibilities to degradation of RVI and RPV components. The EPU analyses bounds the previous analyses of these components.

In its response, the NRC staff has addressed the following statements from the public comment that it interprets as related to the fluence calculational methodology:

1. "Plus or minus 20% uncertainty, i.e. 40% total uncertainty, for anything, but especially for the reactor pressure vessel beltline embrittlement is unacceptable and constitutes premeditated homicide."

2. "Furthermore, there seems to be a more general side-stepping of statistical method and all logic in your document."

3. "Plus-minus 20% error, as you are allowing, which is 40% uncertainty (error-variation) is unacceptable by any scientific standard. For something so dangerous there should be 98 to 99% certainty with a 50 to 100% contingency of protection. Instead: 'An extensive benchmarking program has been carried out to qualify the MPM neutron transport methodology. All of the requirements of RG 1.190 have been met. In particular, all C/M results fall within allowable limits (+/- 20%), and it was determined that no bias need be applied to MPM fluence results. The uncertainty analysis indicates that all fluence results in the beltline region have uncertainty of less than 20%. The results of this analysis are documented in References I and 2. This meets the requirement of RP 1.4.1, 1.4.2, and 1.4.3. This is wrong. It is dangerous. It is unacceptable."

<u>NRC Response</u>

The comments refer to RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Fluence." Specifically, the comments question the appropriateness of the RG 1.190 assumption of a 20 percent fluence calculational uncertainty allowance. Therefore the NRC response is with respect to the fluence calculational method guidance in RG 1.190.

In RG 1.190, Section C., "Regulatory Position," Item 1. "Neutron Fluence Calculational Methods," the second paragraph states:

> Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 requires that this methodology be properly qualified. Qualification includes determination of the uncertainty in the reactor vessel fluence as described in Regulatory Position 1.4. The uncertainty of

> the fluence must be 20% (1-σ) or less when the fluence is used to determine RT-PTS and RT-NDT for complying with 10 CFR 50.61 and Revision 2 of Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," respectively. **It should be recognized that this 20% uncertainty value has been included in the margin term for the RT-PTS** [emphasis added].

The quoted paragraph mentions RG 1.99, which "describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels." As explained in RG 1.99:

> Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," [to 10 CFR Part 50], which implement in part, [General Design] Criterion 31 [of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50], necessitate the calculation of changes in fracture toughness of reactor vessel materials caused by neutron radiation throughout the service life.

Thus, following RG 1.190 ensures that fluence calculational methods are appropriately qualified so that fluence values from these appropriately qualified methods can be used in the determination of 10 CFR 50, Appendix G required RT-NDT[1] values. In general, this qualification process includes verification that fluence calculations will be accurate to within 20%, which ensures that the margin term for the RT-NDT, as indicated by the bolded text above, is not exceeded. Note that the RT-PTS[2] is specific terminology that applies only to pressurized water reactors (PWRs). However, since RT-PTS uses the same formulation as the more general RT-NDT which is applicable to both PWRs and boiling water reactors such as GGNS, the bolded text above also applies to RT-NDT.

In addition to the rigorous calculational methodology described in RG 1.190, dosimetry monitoring is required as part of the 10 CFR 50, Appendix H surveillance program. Appendix H requires licensees to maintain reactor vessel material surveillance programs. Appendix H states: "The purpose of the material surveillance program required by this appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors which result from exposure of these materials to neutron irradiation and the thermal environment." Surveillance capsules are withdrawn periodically from the reactor vessel providing measurements from dosimetry wires in the surveillance capsules. These measurements are used to further qualify the calculational methodology of RG 1.190. As RG 1.190 states: "Because of the importance and the difficulty of

[1] 10 CFR 50.61 defines RT-NDT as the reference temperature for a reactor vessel material, under any conditions. For the reactor vessel beltline materials, RT-NDT must account for the effects of neutron radiation. This definition is consistent with the 10 CFR 50, Appendix G definition.
[2] 10 CFR 50.61 defines RT-PTS as the reference temperature, RT-NDT, evaluated for the EOL Fluence for each of the vessel beltline materials, using the procedures of paragraph (c) of the section. Paragraph (c) states that RT-PTS must be calculated for each vessel beltline material using a fluence value, which is the EOL fluence for the material. RT-PTS must be evaluated using the same procedures used to calculate RT-NDT.

these calculations, the methods must be qualified by comparison to measurements to ensure a reliable and accurate vessel fluence determination."

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Mississippi State official was notified of the proposed issuance of the amendment. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been one public comment on such finding published in the *Federal Register* on March 31, 2015 (80 FR 17087). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: A. Patel

Date: August 18, 2015

The NRC had determined that the related safety evaluation (SE) contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations*, Section 2.390, "Public inspections, exemptions, requests for withholding." Accordingly, the NRC staff has also prepared a non-proprietary version of the SE, which is provided in Enclosure 3. The proprietary version of the SE is provided in Enclosure 2. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Alan B. Wang, Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-416

Enclosures:
1. Amendment No. 204 to NPF-29
2. Safety Evaluation (proprietary)
3. Safety Evaluation (non-proprietary)

cc w/encls 1 and 3: Distribution via Listserv

DISTRIBUTION:
PUBLIC
LPL4-2 r/f
RidsNrrDorlDpr Resource
RidsRgn4MailCenter Resource
APatel, NRR
RidsAcrsAcnw_MailCTR Resource
RidsNrrDorlLpl4-2 Resource
RidsNrrDssSrxb Resource
RidsNrrDssStsb Resource
RidsNrrPMGrandGulf Resource
RidsNrrLAPBlechman Resource

ADAMS Accession Nos.
Proprietary: ML15203A291, Non-Proprietary: ML15229A218 *concurrence via memorandum

OFFICE	NRR/DORL/LPL4-2/PM	NRR/DORL/LPL4-2/LA	NRR/DSS/SXRB/BC*
NAME	AWang	PBlechman	CJackson
DATE	8/11/15	8/4/15	7/13/15
OFFICE	OGC - NLO	NRR/DORL/LPL4-2/BC	NRR/DORL/LPL/4-2/PM
NAME	CKanatas	MKhanna	AWang
DATE	8/13/15	8/14/15	8/18/15

OFFICIAL RECORD COPY