Entergy

Entergy Operations, Inc.
P.O. Box 756
Port Gibson, MS 39150

Kevin Mulligan
Site Vice President
Grand Gulf Nuclear Station
Tel. (601) 437-7500

GNRO-2015/00046

August 10, 2015

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Clarification to Fluence Methodology License Amendment Request letter GNRO-2014/00080 dated November 21, 2014 to include DRAFT marked-up UFSAR pages and a Regulatory Commitment to revise the GGNS UFSAR upon NRC approval of the Fluence Methodology LAR
Grand Gulf Nuclear Station, Unit 1
Docket No. 50-416
License No. 29

REFERENCES:

1. Severity Level IV non-cited violation of 10 CFR 50.59, "Changes, Tests, and Experiments" involving the licensee's 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; Grand Gulf Nuclear Station - NRC Integrated Inspection Report 05000416/2013004, dated November 27, 2013

2. U.S. Nuclear Regulatory Commission Letter, "Requests for Additional Information for the Review of the Grand Gulf Nuclear Station, License Renewal Application," dated August 28, 2013 (Accession No. ML13227A394)

3. Grand Gulf Nuclear Station Letter, "Response to Requests for Additional Information (RAI) set 47," dated September 23, 2013 (Accession No. ML13266A368)

4. U.S. Nuclear Regulatory Commission Regulatory Guide, Regulatory Guide 1.190, dated March 2001 (Accession No. ML010890301)

5. Grand Gulf Nuclear Station Letter GNRO-2014/00080, "License Amendment Request - Application to Revise Grand Gulf Nuclear Station Unit 1's Current Fluence Methodology from 0 EFPY Through the End of Extended Operations to a Single Fluence Method," dated November 21, 2014

Dear Sir or Madam:

In accordance with the provisions of Section 50.90 of Title 10 Code of Federal Regulations (10 CFR), Entergy Operations, Inc. (Entergy) submitted a request for an amendment to revise the existing license basis for Grand Gulf Nuclear Station (GGNS), Unit 1 in letter GNRO-2014/00080. This letter clarifies that the license basis to be revised upon approval of the requested amendment is the GGNS Updated Final Safety Analysis Report (UFSAR).

The proposed amendment is to revise Grand Gulf Nuclear Station, Unit 1's UFSAR to adopt a single fluence calculation method. This change is needed to address a legacy issue in which the current method was determined to be utilized without receiving prior NRC approval (reference 1).

Attachment 1 provides DRAFT marked-up pages of applicable sections of the GGNS UFSAR for the proposed change. Attachment 2 provides the DRAFT clean pages. Attachment 3 provides a Regulatory Commitment to revise the affected sections of the GGNS UFSAR upon approval of the Fluence Calculation Methodology LAR. This report also applies to the Maximum Extended Load Line Limit Plus (MELLLA+) License Amendment Request (LAR) in letter GNRO-2013/00012 (Accession No. ML 13269A140). Although this request is neither exigent nor emergency, your prompt review is requested.

This letter contains one new commitment found in Attachment 3. If you have any questions or require additional information, please contact Mr. James Nadeau at (601) 437-2103.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 10, 2015.

Sincerely,

KJM/ras

Attachments:

1. DRAFT Marked-Up Pages of Affected GGNS UFSAR Chapter 4
2. DRAFT Clean Pages
3. LIST OF REGULATORY COMMITMENTS

cc: with Attachments

U. S. Nuclear Regulatory Commission
ATTN: Mr. Alan Wang, NRR/DORL (w/2)
Mail Stop OWFN 8 B1
Washington, DC 20555-0001

U.S. Nuclear Regulatory Commission
ATTN: Ms. Rebecca Richardson, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation
Mail Stop O-11 F1
Washington, DC 20555

cc: without Attachments

U.S. Nuclear Regulatory Commission
ATTN: Mr. Mark Dapas, (w/2)
Regional Administrator, Region IV
1600 East Lamar Boulevard
Arlington, TX 76011-4511

NRC Senior Resident Inspector
Grand Gulf Nuclear Station
Port Gibson, MS 39150

Dr. Mary Currier, M.D., M.P.H
State Health Officer
Mississippi Department of Health
P. O. Box 1700
Jackson, MS 39215-1700

Attachment 1

GNRO-2015/00046

DRAFT Marked-Up Pages of Affected GGNS UFSAR Chapter 4

Computer Code	Function Lattice Physics
BWR Reactor Simulator	Calculates 3-dimensional nodal power distributions, exposures and thermal hydraulic characteristics as burnup progresses.

4.1.4.5 Neutron Fluence Calculations

~~Neutron vessel fluence calculations were carried out using a one-dimensional discrete ordinates Sn transport code with general anisotropic scattering.~~

Insert 'A'

~~This code is a modification of a widely used discrete ordinates code which will solve a wide variety of radiation transport problems. The program will solve both fixed source and multiplication problems. Slab, cylinder, and spherical geometry are allowed with various boundary conditions. The fluence calculations incorporated as an initial starting point neutron fission distributions are prepared from core physics data as a distributed source. Anisotropic scattering was considered for all regions. The cross sections were prepared with a 1/E flux weighting, P_L matrices for anisotropic scattering but did not include resonance self-shielding factors. Fast neutron fluxes at locations other than the core mid-plane were calculated using a two-dimensional discrete ordinate code. The two-dimension code is an extension of the one-dimensional code.~~

Additional vessel fluence calculations, which comply with the requirements of Regulatory Guide 1.190, are described in Section 4.3.2.8.

4.1.4.6 Thermal-Hydraulic Calculations

The digital computer program is a parallel flow path used to perform the steady-state BWR reactor core thermal-hydraulic analysis. Program input includes the core geometry, operating power, pressure, coolant flow rate and inlet enthalpy, and power distribution within the core. Output from the program includes core pressure drop, coolant flow distribution, critical power ratio, and axial variations of quality, density, and enthalpy for each channel type.

4.1.5 Deleted

4.1.6 References

1. Crowther, R. L., "Xenon Considerations in Design of Boiling Water Reactors," APED-5640, June 1968

2. Beitch, L., "Shell Structures Solved Numerically by Using a Network of Partial Panels," AIAA Journal, Volume 5, No. 3, March 1967

3. Wilson E. L., "A Digital Computer Program For the Finite Element Analysis of Solids With Non Linear Material Properties," Aerojet General Technical Memo No. 23, Aerojet General, July 1965

4. Farhoomand, I., and Wilson E. L., "Non-Linear Heat Transfer Analysis of Axisymmetric Solids," SESM Report SESM71-6, University of California at Berkeley, Berkeley, California, 1971

5. McConnelee J. E., "Finite-Users Manual," General Electric TIS Report DF 69SL206, March 1969

6. Clough R. W. and Johnson C. P., "A Finite Element Approximation For the Analysis of Thin Shells," International Journal Solid Structures, Vol. 4, 1968

7. "A Computer Program For the Structural Analysis of Arbitrary Three-Dimensional Thin Shells," Report No. GA-9952, Gulf General Atomic

8. Burgess, A. B., "User Guide and Engineering Description of HEATER Computer Program," March 1974

9. Young, L. J., "FAP-71 (Fatigue Analysis Program) Computer Code," GE/NED Design Analysis Unit R. A. Report No. 49, January 1972

10. Deleted

11. Rashid Y. R., "Theory Report for CREEP-PLAST Computer Program," GEAP-10546, AEC Research and Development Report, January 1972

12. Deleted

13. Deleted

14. Deleted

15. Deleted

Insert "B"

16. MPM-614993, Rev 5, Benchmarking of MPM Methods For Nuclear Plant Neutron Transport Calculations

17. MPM-814779, Rev. 5, Neutron Transport Analysis For Grand Gulf Nuclear Station.

4.3.2.7 Stability

4.3.2.7.1 Xenon Transients

Boiling water reactors do not have instability problems due to xenon. This has been demonstrated by operating BWRs for which xenon instabilities have never been observed, (such instabilities would readily be detected by the LPRMs), by special tests which have been conducted on operating BWRs in an attempt to force the reactor into xenon instability, and by calculations. All of these indicators have proven that xenon transients are highly damped in a BWR due to the large negative power coefficient.

Analysis and experiments conducted in this area are reported in Reference 12.

4.3.2.7.2 Thermal-hydraulic Stability

This subject is covered in subsection 4.4.4.6.

4.3.2.8 Vessel Irradiations

three The neutron fluxes at the vessel have been calculated using the ~~one~~ dimensional ~~discrete ordinates~~ transport code described in subsection 4.1.4.5. ~~The discrete ordinates code was used in a distributed source mode with cylindrical geometry. The geometry described six regions from the center of the core to a point beyond the vessel. The core region was modeled as a single homogenized cylindrical region.~~ The coolant water region between the fuel channel and the shroud was described containing saturated water at 550°F and 1050 psi. The material compositions for the stainless steel in the shroud and the carbon steel in the vessel contain the mixtures by weight as specified in the ASME material specifications for ASME SA 240, 304L, and ASME SA 533 grade B. In the region between the shroud and the vessel, the presence of the jet pumps was included ~~ignored~~. A simple diagram showing the regions, dimensions and weight fractions are shown in Figure 4.3-29.

Insert 'c' New Paragraph

The distributed source used for this analysis was obtained from the gross radial power description. The distributed source at any point in the core is the product of the power from the power description and the neutron yield from fission. By using the neutron energy spectrum, the distributed source is obtained for position and energy. The integral over position and energy is normalized to the total number of neutrons in the core region. The core region is defined as a 1-centimeter thick disc with no transverse leakage. The power in this core region is set equal to the maximum power in the axial direction. The radial and axial power distributions are shown in Figures 4.3-30 and 4.3-22.

28. EMF-2493(P), "MICROBURN-B2 Based Impact of Failed/Bypassed LPRMs and TIPs, Extended LPRM Calibration Interval, and Single Loop Operation on Measured Radial Bundle Power Uncertainty," Framatome-ANP (Proprietary) December 2000.

29. Kelley, R. H., et al., Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, November 1981 (XN-NF-79-71(P) Rev. 2)

30. Gitnick, B. J., et al., Grand Gulf Unit 1 Cycle 2 Plant Transient Analysis, August 1986 (XN-NF-86-36 Rev. 3)

31. ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," Advanced Nuclear Fuels Corporation, August 1990.

32. EMF-2481(P) Revision 0, "Fuel Design Report for Grand Gulf Unit 1 Cycle 12 Atrium-10 Fuel Assemblies," Siemens Power Corporation, November 2000.

33. EMF-1997(P)(A) Revision 0, "ANFB-10 Critical Power Correlation," July 1998.

34. EMF-2245(P)(A), "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel," Siemens Power Corporation, August 2000.

35. EMF-2209(P)(A) Revision 2, "SPCB Critical Power Correlation," Framatome-ANP, September 2003.

36. MPM-104772, "Neutron Transport Analysis for Grand Gulf Nuclear Station," February 2004.

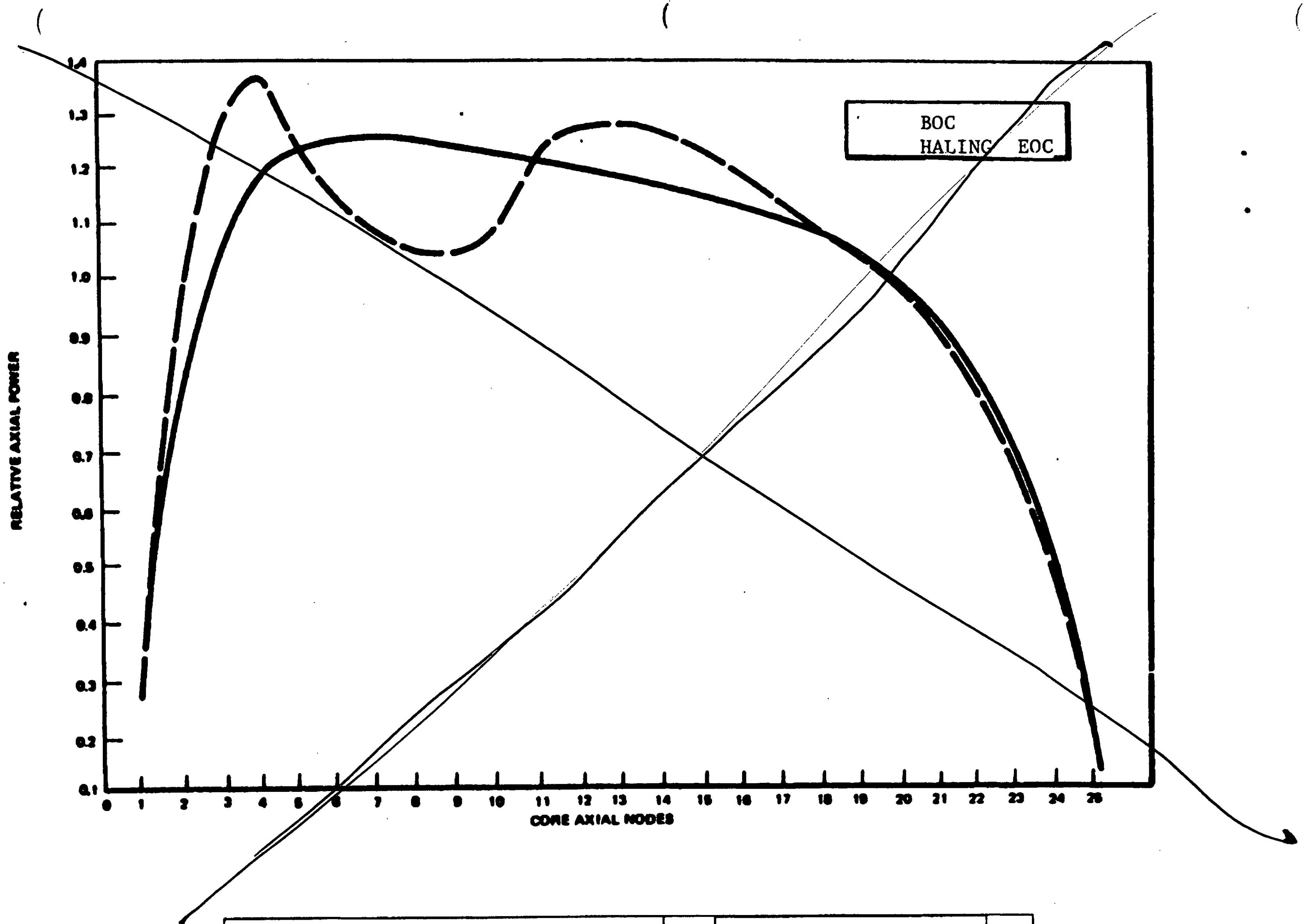
37. TAC No. MB6687, "Nine Mile Point Nuclear Station, Unit No. 1 - Issuance of Amendment RE: Pressure-Temperature Limit Curves and Tables," October 27 2003.

38. "TORT-DORT-PC, Two and Three Dimensional Discrete Ordinates Transport", Version 2.7.3, CCC-543, RCISS Computer Code Collection.

39. "BUGLE-96, Coupled 47 Neutron, 20 Gamma Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications", RSICC Data Library Collection.

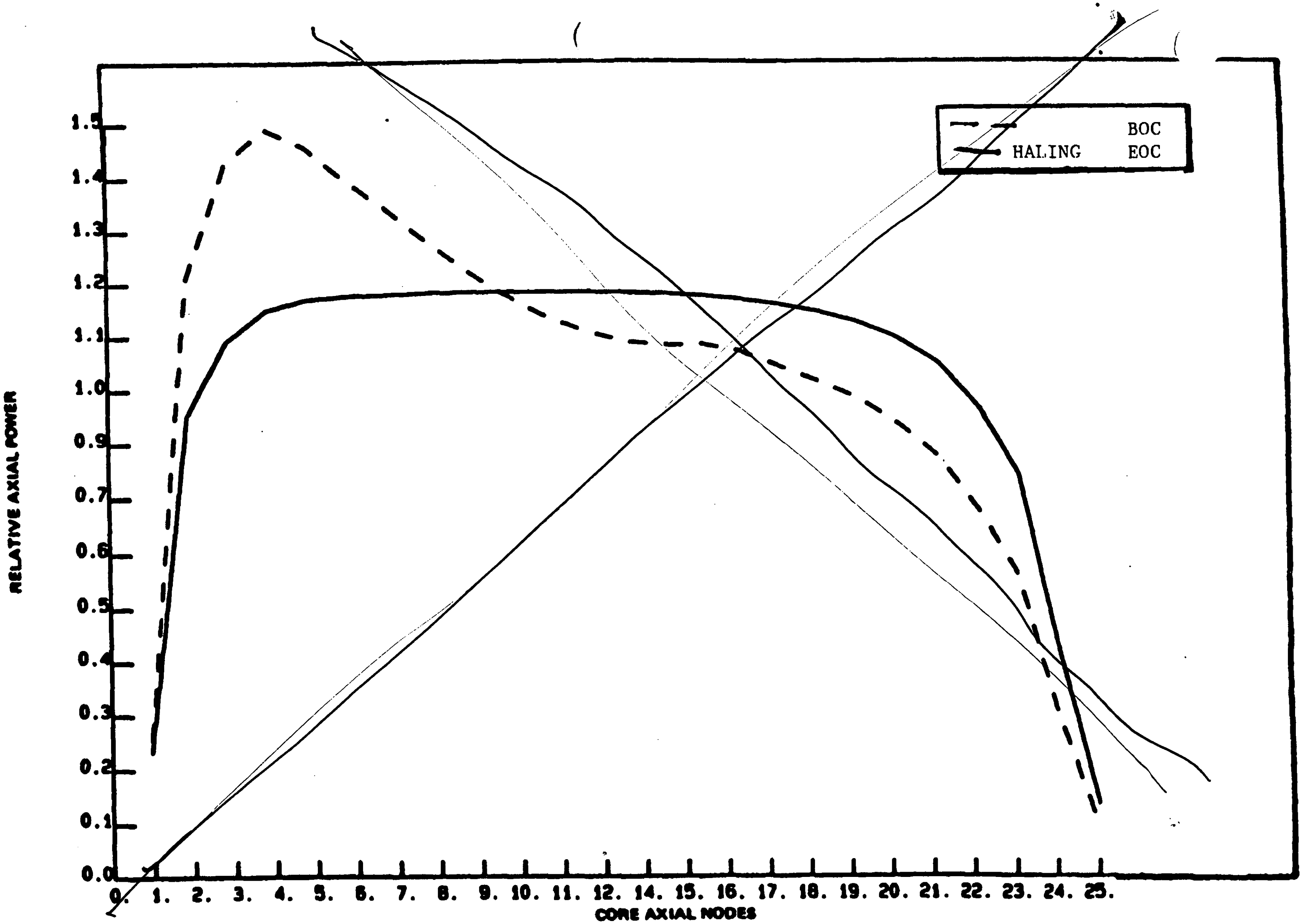
40. "Casmo-4/Simulate-3, Studsvik's Advanced Three-Dimensional Two Group Reactor Analysis Code."

Insert 'D'



GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	TYPICAL INITIAL CYCLE BEGINNING OF CYCLE AND END OF CYCLE CORE AVERAGE AXIAL POWER FIGURE 4.3-22

REV. 6



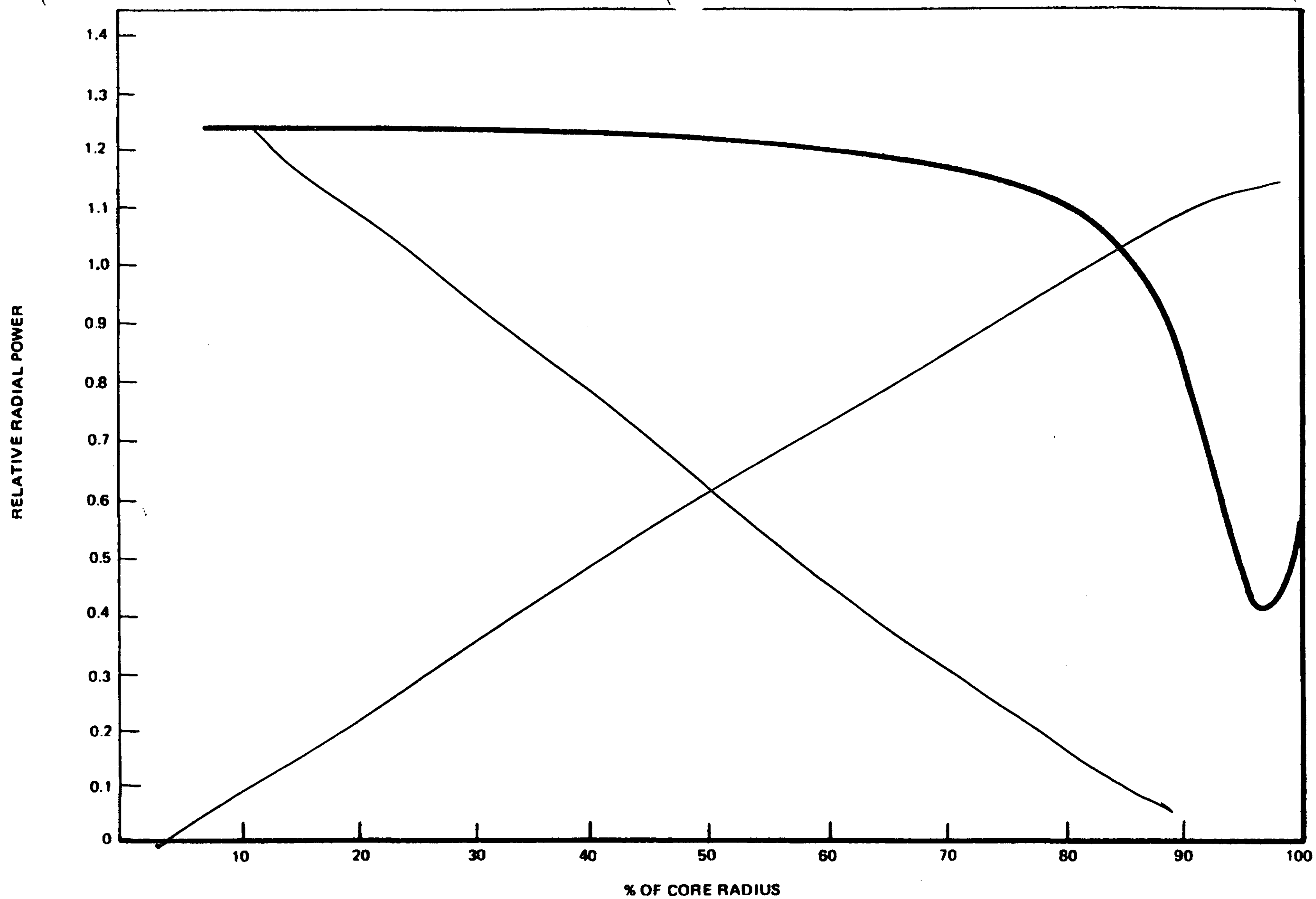
GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	TYPICAL RELOAD CYCLE BEGINNING OF CYCLE AND END OF CYCLE CORE AVERAGE AXIAL POWER FIGURE 4.3-22A

REV. 6

MATERIAL		RADIUS (in.)	MATERIAL	MATERIAL DENSITY
NO.	NAME			
1	REACTOR CORE	95.75	WATER UO_2 ZIRCONIUM 304L STAINLESS STEEL	0.318 g/cm^3 2.334 g/cm^3 0.978 g/cm^3 From ASME SA240
2	WATER	105.94	WATER	0.74 g/cm^3
3	SHROUD	107.94	304L STAINLESS STEEL	FROM ASME SA 240
4	WATER	125.5	WATER	0.74 g/cm^3
5	VESSEL	131.68	CARBON STEEL	FROM ASME SA 240
6	AIR		AIR	1.3×10^{-3} g/cc

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	MODEL FOR ONE-DIMENSIONAL TRANSPORT ANALYSIS OF VESSEL FLUENCE FIGURE 4.3-29

Rev. 10



GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	RADIAL POWER DISTRIBUTIONS USED IN THE VESSEL FLUENCE CALCULATION FIGURE 4.3-30

Rev. 10

Attachment 2

GNRO-2015/00046

DRAFT Clean Pages

Insert 'A' for Section 4.1.4.5

Calculations of the best estimate neutron fluence, and its uncertainty, to the Grand Gulf Nuclear Station (GGNS) reactor pressure vessel (RPV), core shroud and top guide horizontal and vertical welds, as well as to several beltline vessel nozzles have been performed. The fluence calculations were carried out using a three dimensional (3D) neutron transport model for each fuel cycle starting from cycle 1 through cycle 21. The 3D neutron transport calculations were benchmarked on a plant- specific basis by comparing calculated results against previously performed core region 2D synthesis data as well as by calculation of the calculated-to-measured (C/M) ratios for GGNS dosimetry. In addition, a comprehensive benchmarking report, Reference 16, of MPM methods has been prepared.

The neutron transport calculation procedures and dosimetry analysis methods meet standards specified by the NRC and ASTM as appropriate. In particular, the neutron transport analysis meets the requirements of Regulatory Guide 1.190 (RG 1.190). Since RG 1.190 is focused on 2D synthesis methods, it is strictly applicable to analyses in the active fuel region. Nevertheless, the guidance provided in RG 1.190 was followed to the extent practical for modeling work in the regions above and below the active fuel region. The 3D neutron transport calculations were used to determine detailed fluence profiles at the end of cycle 21 (28.088 Effective Full Power Years - EFPY), and projected to exposures of 35 EFPY and 54 EFPY, Reference 17.

Insert 'B' for Section 4.1.6

16. MPM-614993, Rev. 5, Benchmarking of MPM Methods For Nuclear Plant Neutron Transport Calculations, May 2015

17. MPM-814779, Rev. 5, Neutron Transport Analysis For Grand Gulf Nuclear Station, May 2015

Insert 'C' for Section 4.3.2.8

Calculations of the best estimate neutron fluence, and its uncertainty, to the Grand Gulf Nuclear Station (GGNS) reactor pressure vessel (RPV), core shroud and top guide horizontal and vertical welds, as well as to several beltline vessel nozzles have been performed. The fluence calculations were carried out using a three dimensional (3D) neutron transport model for each fuel cycle starting from cycle 1 through cycle 21. The 3D neutron transport calculations were benchmarked on a plant-specific basis by comparing calculated results against previously performed core region 2D synthesis data as well as by calculation of the calculated-to-measured (C/M) ratios for GGNS dosimetry. In addition, a comprehensive benchmarking report, Reference 41, of MPM methods has been submitted.

The neutron transport calculation procedures and dosimetry analysis methods meet standards specified by the NRC and ASTM as appropriate. In particular, the neutron transport analysis meets the requirements of Regulatory Guide 1.190 (RG 1.190). Since RG 1.190 is focused on 2D synthesis methods, it is strictly applicable to analyses in the active fuel region. Nevertheless, the guidance provided in RG 1.190 was followed to the extent practical for modeling work in the regions above and below the active fuel region. The 3D neutron transport calculations were used to determine detailed fluence profiles at the end of cycle 21 (28.088 EFPY), and projected to exposures of 35 EFPY and 54 EFPY, Reference 42.

Summary of Shroud and Top Guide Fluence Results

The fluences reported, Reference 42, were calculated at the inner diameter (ID) surface of the welds. With the exception of horizontal welds H4 and H5, all horizontal shroud weld fluences are below 5E+20 n/cm2 through at least 54 EFPY. At the end of cycle 21, the maximum fluence to shroud welds H4 and H5 are 1.22E+21 n/cm2 and 4.88E+20 n/cm2, respectively.
With the exception of the top guide vertical welds V7 and V8, all vertical shroud weld fluences are below 5E+20 n/cm2 at the end of cycle 21 (28.088 EFPY). When extrapolated to 54 EFPY exposure, the vertical shroud weld fluences are still below 5E+20 n/cm2 except for the vertical welds V7 and V8, and welds V13 through V16. Welds V7 and V8 are located in the plate at the bottom of the top guide. These welds extend across the entire diameter of the plate and thus lie, in part, directly above the reactor core. Weld V7 is defined from the core centerline to the top guide OD at 90 degrees, and V8 extends from the core centerline to the top guide OD at 270 degrees. As a result of the high void fraction of the water-steam mixture above the core, there is relatively little water shielding for this plate, and the fast neutron flux is therefore very high. The maximum exposure to this weld is calculated to be about 2.32E+21 n/cm2 at the end of cycle 21. For this weld, the fluence at various radial points from 0 to the outer edge of the top guide is calculated and included in the appendices of the report.

Summary of Vessel, Vessel Internals, and Cycle 1 Dosimetry Results

The transport calculations were also performed to evaluate fluence for the surveillance capsule and for the reactor vessel. Comparisons with dosimetry measurements at the GGNS surveillance capsule location at the end of cycle 1 were made and excellent agreement was found. The Calculated/Measured (C/M) ratio averaged over all of the dosimeters is 0.98.

Maximum fluence to the reactor vessel wetted surface was calculated to be 1.74E+18 n/cm2 (E > 1 MeV) at the end of cycle 21, and 4.03E+18 n/cm2 (E > 1 MeV) after 54 EFPY. Included is the calculated dpa attenuation through the vessel as well as the dpa attenuation determined using the RG 1.99 (Rev 2) equation. The dpa attenuation for locations above and below the active fuel region was calculated for the shell 1, 2 and 3 plates and welds and also for the N1, N2, N6, and N12 nozzles.

The NRC defines the beltline region in 10CFR50, Appendix G as "the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage." The beltline region is defined as any area that exceeds a fast fluence of 1.0E+17 n/cm2.

At EOC 20 (26.220 EFPY), the vessel fluence will exceed 1.0E+17 n/cm2 at locations about 9.3 inches below the BAF in shell 1 region up to about 11.4 inches above the TAF in shell 2 region. The axial fluence profile at EOC 21 (28.088 EFPY) shows that the vessel maximum fast fluence exceeds 1.0E+17 n/cm2 outside of the 150 inch active fuel region, and extends above horizontal weld AC and below weld AB. In particular, at EOC 21, the vessel fluence will exceed 1.0E+17 n/cm2 at locations about 10.3 inches below BAF, and about 41.4 inches above the TAF. At 54 EFPY, the vessel fluence will exceed 1.0E+17 n/cm2 at locations about 17.1 inches below BAF and about 56.1 inches above TAF. The N12 (water level instrumentation) nozzles have exceeded a fluence level of 1.0E+17 n/cm2 at the end of cycle 21.

Fluence values for the capsule, vessel, and shroud in the beltline region (except for the very top and bottom of the core) are estimated to have uncertainties of 14.7%, 15.8%, and 13.3%, respectively. These uncertainties are within the value of ± 20% specified by RG 1.190. Moreover, the 3D calculations, Reference 41, have been benchmarked against GGNS cycle 1 capsule dosimetry measurements which are in excellent agreement (C/M = 0.98). The calculations of shroud, vessel, and capsule fluence meet all of the requirements of RG 1.190. Similarly, the results of calculations performed above and below the core meet the requirements of RG 1.190 except for the ± 20% criterion.

For the upper shroud and top guide welds, and for the N6 nozzle, the uncertainties are greater than 30%. Based on guidance provided in Equation 6 of Regulatory Guide 1.190, it is reasonable to multiply the calculated fluences by 1 plus the 1 sigma uncertainty for the cases where the uncertainty is over 30%.

<u>Insert 'D' for Section 4.3.6</u>

41. MPM-614993, Rev. 5, Benchmarking of MPM Methods For Nuclear Plant Neutron Transport Calculations, May 2015

42. MPM-814779, Rev. 5, Neutron Transport Analysis For Grand Gulf Nuclear Station, May 2015

43. GGNS-NE-15-00003, Rev. 0, Grand Gulf Nuclear Station Fluence Effect on RPV Internal Components at EPU Operating Conditions.

Attachment 3

GNRO-2015/00046

LIST OF REGULATORY COMMITMENTS

LIST OF REGULATORY COMMITMENTS

This table identifies actions discussed in this letter for which Entergy commits to perform. Any other actions discussed in this submittal are described for the NRC's information and are **not** Commitments.

COMMITMENT	ONE-TIME ACTION	CONTINUING COMPLIANCE	SCHEDULED COMPLETION DATE (if required)
Entergy will revise the affected sections of Chapter 4 of the GGNS UFSAR upon approval of the Fluence Calculation Methodology LAR	✓		October 30, 2015