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GNRO-2016/00015

March 23, 2016

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

SUBJECT: Non-Proprietary Version of MP Machinery and Testing, LLC (MPM) Proprietary Report 814779, Neutron Transport Analysis for Grand Gulf Nuclear Station, previously included in letters GNRO-2014/00080 and GNRO-2015/00011
Grand Gulf Nuclear Station, Unit 1
Docket No. 50-416
License No. NPF-29

REFERENCES:

1. 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)

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

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

4. Grand Gulf Nuclear Station Letter GNRO-2014/00080, "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.

5. Grand Gulf Nuclear Station Letter GNRO-2015/00011, "Supplement to 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 February 18, 2015.

6. Grand Gulf Nuclear Station Letter GNRO-2015/00053, "Affidavits for the Proprietary MP Machinery and Testing, LLC (MPM) reports attached in letters GNRO-2014/00080 and GNRO-2015/00011," dated August 18, 2015.

Dear Sir or Madam:

The Single Method Neutron Fluence Calculation Methodology License Amendment Request (LAR), reference 4, submitted November 21, 2014, and the Supplement to that LAR, reference 5, submitted February 18, 2015, included the Proprietary MP Machinery and Testing, LLC (MPM) Neutron Transport Analysis for Grand Gulf Nuclear Station, report MPM-814779, Revisions 1 and 2. However, Non-Proprietary versions of report MPM-814779 were not provided in those letters.

Attachment 1 provides the Non-Proprietary version of report MPM-814779, Revision 1, previously provided in Attachment 3 to letter GNRO-2014/00080. Attachment 2 provides the Non-Proprietary version of report MPM-814779, Revision 2, previously provided in Attachment 3 to letter GNRO-2015/00011.

Attachments 3 and 4 provide the Affidavits for the Proprietary versions of report MPM-814779, Revisions 1 and 2, previously provided in letters GNRO-2014/00080 and GNRO-2015/00011.

This letter contains no new commitments. 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 March 23, 2016.

Sincerely,

KJM/ras

Attachments:

1. MP Machinery and Testing, LLC (MPM) Neutron Transport Analysis for Grand Gulf Nuclear Station, report MPM-814779, Revision 1 (Non-Proprietary Version)
2. MP Machinery and Testing, LLC (MPM) Neutron Transport Analysis for Grand Gulf Nuclear Station, report MPM-814779, Revision 2 (Non-Proprietary Version)
3. MP Machinery and Testing, LLC Affidavit for Proprietary Version of report MPM-814779, Revision 1
4. MP Machinery and Testing, LLC Affidavit for Proprietary Version of report MPM-814779, Revision 2

cc: with Attachment and Enclosures

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Attachment 1

GNRO-2016/00015

MPM Machinery and Testing, LLC (MPM) Neutron Transport Analysis for Grand Gulf Nuclear Station, report MPM-814779, Revision 1

(Non- Proprietary Version)

This is a non-proprietary version of Attachment 3 of letter GNRO-2014/00080 which has the proprietary information removed. Redacted information is identified by blank space enclosed with double brackets [[ ]].

NON-PROPRIETARY VERSION

Report Number MPM-814779
Revision 1

Neutron Transport Analysis For Grand Gulf Nuclear Station

MP Machinery & TESTING, LLC

November, 2014

MPM Technical Report

Neutron Transport Analysis for
Grand Gulf Nuclear Station

MPM Report Number MPM-814779
Revision 1

Final Report, November, 2014

MP Machinery and Testing, LLC

2161 Sandy Drive

State College, PA 16803

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[[

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M P Manahan, Sr.

M. P. Manahan, Sr.
President

11/17/2014

Date

J. Nemet
QA Manager

11/17/2014

Date

EXECUTIVE SUMMARY

This work was undertaken to calculate the best estimate neutron fluence, and its uncertainty, to the Grand Gulf Nuclear Station (GGNS) reactor pressure vessel, core shroud and top guide horizontal and vertical welds, as well as to several beltline vessel nozzles. Figures ES-1 and ES-2 show the shroud and top guide weld designations, and Figure ES-3 shows the weld seam and plate locations for the reactor vessel. 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 of MPM methods has been prepared under separate cover.

The neutron transport calculational procedures and dosimetry analysis methods meet standards specified by the NRC and ASTM as appropriate. In particular, the 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. [[

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Summary of Shroud and Top Guide Fluence Results

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Summary of Vessel 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. [[

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Benchmarking and Uncertainty Analysis

Fluence values for the capsule, vessel, and shroud in the beltline region (except for the very top and bottom of the core) are estimatedto have uncertainties of [[
]] These uncertainties are within the value of ± 20% specified by RG 1.190.

Moreover, the 3D calculations have been benchmarked [[

]].

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Table ES-2 Maximum Fluence to GGNS Shroud and Top Guide Vertical Welds.

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Table ES-3 GGNS Maximum Calculated Vessel Fluence and Fluence with dpa Attenuation.

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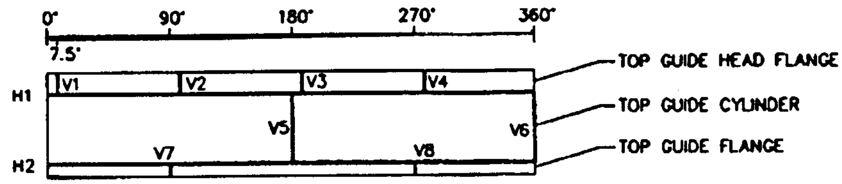

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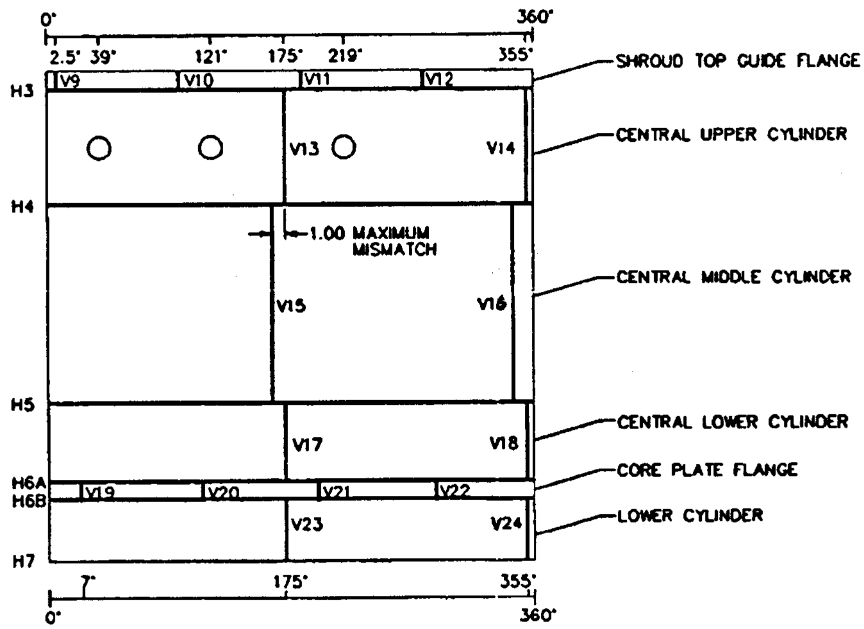

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Figure ES-3 GGNS Vessel Rollout Drawing Showing Beltline Weld Seam and Plate Locations. The Increase in the Axial Extent of the Beltline Requires that Radiation Damage Effects for Shell 1, 2 and 3 Materials be Included in the P-T Curve Analysis.

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1 INTRODUCTION

This work was undertaken to calculate the best estimate neutron fluence, and its uncertainty, to the Grand Gulf Nuclear Station (GGNS) top guide, reactor pressure vessel, vessel nozzles, and core shroud horizontal and vertical welds. The neutron exposure at the top guide, shroud, and, in particular, the shroud welds is an important concern for many Boiling Water Reactors (BWRs) since cracks have been observed in several plants. Figures 1-1 and 1-2 show the GGNS top guide and shroud weld designations, and Figure 1-3 shows the GGNS weld seam and plate locations for the reactor vessel. These designations are used throughout the report to identify the vertical and horizontal welds of interest. It is important to note that, with the exception of welds V7 and V8, all of the shroud and top guide welds are located within the cylindrical portion of these components. [[

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The neutron fluence calculations were carried out using a three dimensional (3D) TORT model for each fuel cycle through the end of cycle 21. At the time of the calculations, the plant had not completed cycle 20, and therefore the fuels data has been extrapolated to the estimated end of cycle 20. It will be necessary to update the cycle 20 transport analysis sometime in the future to represent the as-burned cycle. [[

]] The neutron transport models are fully described in Section 2. The calculational procedures meet standards specified by the Nuclear Regulatory Commission (NRC) and American Society for Testing and Materials (ASTM) as appropriate. In particular, the beltline analysis meets the requirements of Regulatory Guide 1.190 (RG 1.190) [1-2]. An important requirement of RG 1.190 is methodology benchmarking. In addition to the plant-specific benchmarking for GGNS, other MP Machinery and Testing, LLC (MPM) benchmark work is summarized in Section 3. A full discussion of the benchmarking of MPM methods is given in Reference [1-3].

The shroud fluence results are presented in Section4 and the Appendices. The vessel fluence calculations are given in Section5. RG 1.190 requires that a detailed uncertainty analysis be performed to identify each source of uncertainty and the impact on the overall accuracy of the calculation. The uncertainty analysis results are reported in Section6. Summary and conclusions are given in Section 7.

1.1 Section 1 References

[1-1] [[

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[1-2] Regulatory Guide 1.190, Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence, U. S. Nuclear Regulatory Commission, March 2001.

[1-3] [[

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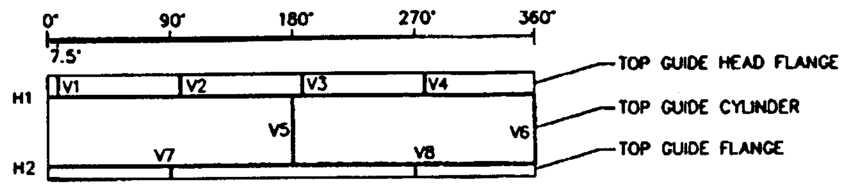

Figure 1-1 Weld Designations for GGNS Top Guide.

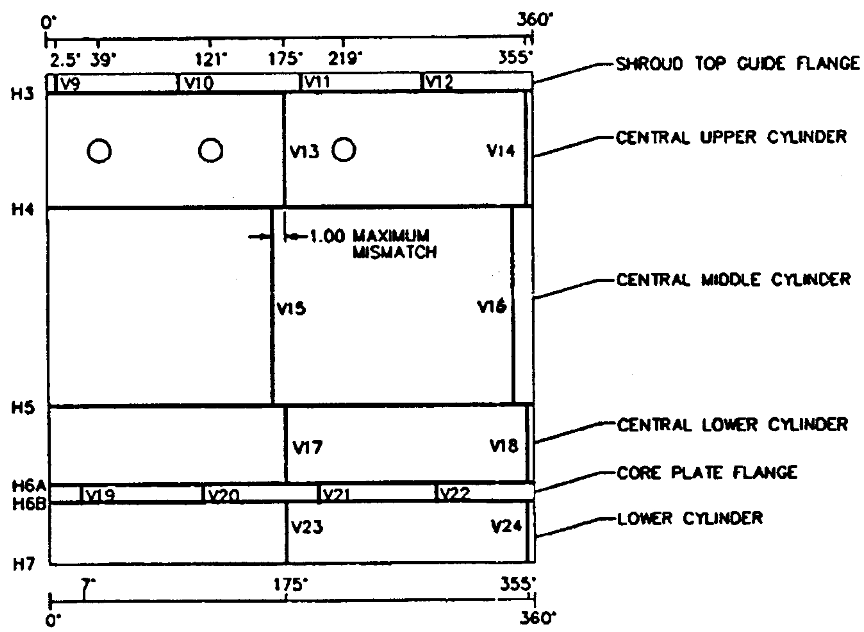

Figure 1-2 Weld Designations for GGNS Shroud.

Non-Proprietary

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Figure 1-3GGNS Vessel Rollout Drawing Showing Beltline Weld Seam and Plate Locations. The Increase in the Axial Extent of the Beltline Requires that Radiation Damage Effects for Shell 1, 2 and 3 Materials be Included in the P-T Curve Analysis.

2 NEUTRON TRANSPORT MODEL DESCRIPTION

The neutron exposure of reactor structures is determined by a neutron transport calculation, or a combination of neutron transport calculations, to represent the distribution of neutron flux in three dimensions. The calculation determines the distribution of neutrons of all energies from their source from fission in the core region to their eventual absorption or leakage from the system. The calculation uses a model of the reactor geometry that includes the significant structures and geometrical details necessary to define the neutron environment at locations of interest. This chapter provides a summary of the neutron source representation used in the GGNS model along with a description of the key model features.

2.1 Source Representation

During reactor operation, the neutron flux level at any point in the shroud or vessel will vary due to changes in fuel composition, power distributions within the core, and water void fraction. These changes occur between fuel cycles due to changes in fuel loading and fuel design, and within a fuel cycle due to fuel and poison burnup and resultant changes in power shape, control rod position, fission contributions by nuclide, and void fraction vs. axial height in each fuel bundle. Power shape throughout a typical cycle's worth of operation has similar characteristics from cycle-to-cycle. Power starts out being preferentially produced in the bottom half of the core, and, as the fuel cycle progresses, the power peak shifts to higher core locations. The fuel burnup decreases the fraction of fissions coming from U235 and increases the fraction of fissions from plutonium isotopes. This results in slight changes in the fission spectrum and in the number of fissions per unit power. The control rod patterns are altered at several discreet time intervals throughout the cycle. These sequence exchanges (SE) result in step changes in the power distribution and in the distribution of reactor water densities, especially in local areas near the control rods. [[

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2.2 Neutron Transport Model

The transport calculations for GGNS were carried out in 2Dand 3D geometries using the DORT two-dimensional and the TORT three-dimensional discrete ordinates codes [2-13], respectively. [[ ]] The DORT code is an update of the DOT code which has been in use for this type of problem for many years. The TORT code has the same calculational methodology as DORT, except that it extends the calculation into the third dimension. [[ ]] The energy group boundaries for the 47 groups are given in Table 2-1. [[

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The computer codes were obtained from the Radiation Safety Information Computational Center (RSICC) at Oak Ridge National Laboratory. Each code was then compiled on the computer used by MPM for the calculations and a series of test cases were run to verify the code performance. The test cases all agreed within allowable tolerance with established results. This verification was conducted under the MPM Nuclear Quality Assurance Program. The calculational procedures meet standards specified by the NRC and ASTM as appropriate. In particular, the analysis (including all modeling details and cross-sections) is consistent with RG 1.190 [2-16], and the 3D calculational methodology has been benchmarked to measured plant-specific BWR data as described in Section 3. [[ ]]

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2D R-θ Calculations

The R-θ layout at core axial midplane is shown schematically in Figure 2-1. Some of the structures in this figure (the surveillance capsule and jet pumps) are not to scale. A more detailed listing of dimensions for the various structures is given in Tables 2-2 through 2-4. Dimensions were obtained from plant drawings and the GGNS design inputs which are referenced in the tables that were transmitted to MPM by Reference [2-19]. As shown in Figure 2-1, all structures outside the core were modeled with a cylindrical symmetry except for the inclusion of a surveillance capsule centered at 3 degrees and jet pump structures located in the downcomer region. [[

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2D R-Z Calculations

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Flux Synthesis

[[ ]] In order to estimate the fluence rate in the three dimensional geometry, the following standard synthesis equation was used to evaluate the flux (φ) [[ ]]

$$\varphi(R,\theta,Z) = \varphi(R,\theta) * \varphi(R,Z) / \varphi(R)$$

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3D TORT Calculations

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2.3 Section 2 References

[2-1] [[

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[2-2] RSICC Peripheral Shielding Routine Code Collection, PSR-277, LEPRICON, PWR Pressure Vessel Surveillance Dosimetry Analysis System, available from the Radiation Safety Information Computational Center, Oak Ridge National Laboratory, Oak Ridge, TN, June 1995.

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[2-13] RSICC Computer Code Collection, CCC-543, TORT-DORT-PC, Two- and Three-Dimensional Discrete Ordinates Transport Version 2.7.3, available from the Radiation Safety Information Computational Center, Oak Ridge National Laboratory, Oak Ridge, TN, June 1996.

[2-14] RSICC Data Library Collection, DLC-185, 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, available from the Radiation Safety Information Computational Center, Oak Ridge National Laboratory, Oak Ridge, TN, March 1996.

[2-15] ASTM Designation E482-89, Standard Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance, in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1997.

[2-16] Regulatory Guide 1.190, Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence, U. S. Nuclear Regulatory Commission, March 2001.

[2-17] [[

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[2-18] [[

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[2-19] [[

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Table 2-1Neutron Energy Group Structure used in the GGNS Transport Calculations – 47 Groups.

Energy Group	Upper Energy (MeV)	Energy Group	Upper Energy (MeV)
1	1.733E+01	25	2.972E-01
2	1.419E+01	26	1.832E-01
3	1.221E+01	27	1.111E-01
4	1.000E+01	28	6.738E-02
5	8.607E+00	29	4.087E-02
6	7.408E+00	30	3.183E-02
7	6.065E+00	31	2.606E-02
8	4.966E+00	32	2.418E-02
9	3.679E+00	33	2.188E-02
10	3.012E+00	34	1.503E-02
11	2.725E+00	35	7.102E-03
12	2.466E+00	36	3.355E-03
13	2.365E+00	37	1.585E-03
14	2.346E+00	38	4.540E-04
15	2.231E+00	39	2.145E-04
16	1.920E+00	40	1.013E-04
17	1.653E+00	41	3.727E-05
18	1.353E+00	42	1.068E-05
19	1.003E+00	43	5.044E-06
20	8.208E-01	44	1.855E-06
21	7.427E-01	45	8.764E-07
22	6.081E-01	46	4.140E-07
23	4.979E-01	47	1.000E-07
24	3.688E-01		

Table 2-2 GGNS Reactor Component Radial Dimensions.

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Table 2-2 GGNS Reactor Component Radial Dimensions (continued).

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Table 2-2GGNSReactor Component Radial Dimensions (continued).

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Table 2-3GGNS Reactor Component Azimuthal Locations.

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Table 2-4GGNS Reactor Component Axial Dimensions.

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Figure 2-1GGNS R-θ Geometry used in the TORT Calculations.

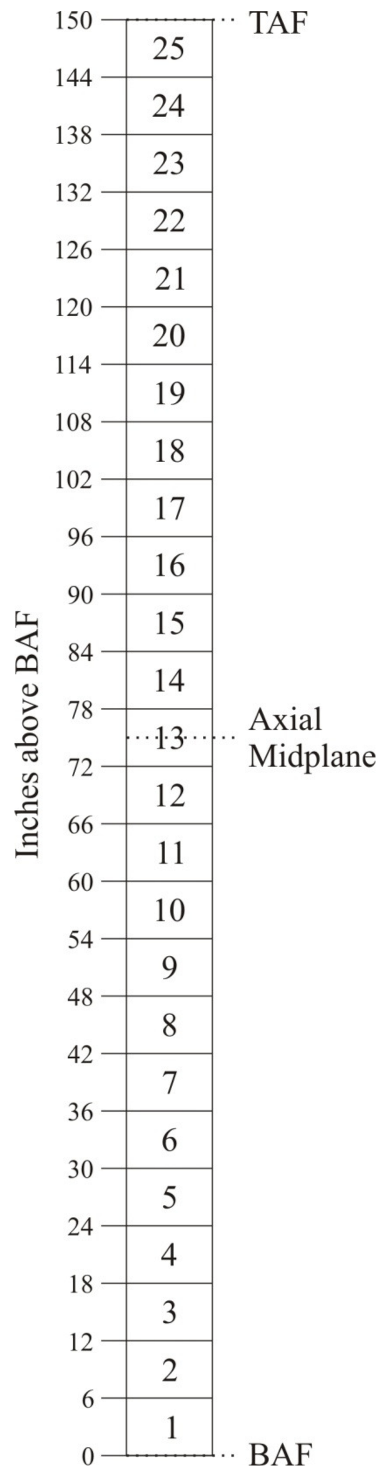

Figure 2-2 Diagram Showing Locations of Fuel Nodes versus Axial Height Relative to BAF.

3
BENCHMARK ANALYSES

NRC Regulatory Guide 1.190 requires that neutron transport methods satisfy compliance requirements and they must also demonstrate specified accuracy through benchmarking. This section of the report summarizes the work done to demonstrate that the MPM methods meet the NRC's requirements.

3.1 Compliance with RG 1.190

The United States Nuclear Regulatory Commission has issued RG 1.190 entitled, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [3-1]. This guide covers recommended practices for neutron transport calculations and applies to other reactor components in addition to the primary emphasis on the pressure vessel. The regulatory positions in the guide that pertain to calculational methodology are summarized in Table 3-1 which is taken directly from RG 1.190. The table references paragraphs in the guide that give more detailed information on each position. The compliance of the GGNS fluence calculations with the guide is summarized below.

<u>Fluence Calculational Methods</u>

Fluence Determination: This calculation was performed using an absolute fluence calculation. Meets guide requirement.

Modeling Data: The MPM methodology is based on documented and verified plant-specific data. Further, the calculations use as-built data for plant structures and material compositions whenever these data are available. The fuel data is specific for each fuel cycle and includes results for power distributions and water densities taken from the fuel depletion analysis.
Meets guide requirement.

Nuclear Data: The calculations used the BUGLE-96 cross section set that is based on the latest version (VI) of the Evaluated Nuclear Data File (ENDF/B). The BUGLE-96 set has undergone extensive testing and benchmarking to ensure its validity for light water reactor (LWR) calculations.
Meets guide requirement.

Cross-Section Angular Representation: The calculations use a P3 angular decompostion in accordance with the guide.
Meets guide requirement.

Cross-Section Group Collapsing: All calculations are performed with the BUGLE-96 library which is collapsed to 47 neutron groups. Benchmarking has shown that the 47-group structure is adequate for LWR neutron transport calculations.
Meets guide requirement.

Neutron Source: The neutron source is calculated taking into account changes in neutrons per fission, energy per fission, and the average fission spectrum which develops as the U235 is burned and other isotopes, such as Pu239, increase in fission fraction.
Meets guide requirement.

End-of-Life Predictions: The fluence has been calculated for every cycle up to the present. [[

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Meets guide requirement.

Spatial Representation: The present methodology meets or exceeds these requirements. [[

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Meets guide requirement.

Multiple Transport Calculations: It was not necessary to use bootstrapping for these calculations so this requirement does not apply.

Point Estimates: This requirement only applies to Monte Carlo calculations which are not used here.

Statistical Tests: This requirement only applies to Monte Carlo calculations which are not used here.

Variance Reduction: This requirement only applies to Monte Carlo calculations which are not used here.

Capsule Modeling: The capsule geometry is modeled using as-built drawings. [[

]].
Meets guide requirement.

Spectral Effects on RT_{NDT}: This requirement only applies to extrapolation through the vessel and does not affect the benchmark calculations. However, when fluence within the vessel is required, the displacement per atom (dpa) extrapolation methodology is applied to vessel calculations as specified in RG 1.99, Revision 2. Data are supplied to enable extrapolation using dpa calculated extrapolation or using RG 1.99 Rev. 2 extrapolation to account for the spectral shift.
Meets guide requirement.

Cavity Calculations: Cavity calculations are not important at GGNS, and no cavity dosimetry work has been performed at GGNS.

Methods Qualification: Methods qualification for these calculations is discussed in detail in Sections 3.2 and 3.3, which deals with benchmarking of the methodology. A complete analytical uncertainty analysis is described in Section 6, and this analysis was carried out in accordance with the guide. In summary, 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%.
Meets guide requirement.

Fluence Calculational Uncertainty: An extensive evaluation of all contributors to the uncertainty in the calculated fluence was made for the BWR plant calculations performed to date. This evaluation indicated that the uncertainty in calculated fluences in the reactor beltline region is below 20% as specified in the guide. In addition, the comparisons with measurements indicate agreement well within the 20% limit. The agreement of calculations with measurements to within ±20% uncertainty indicates that the MPM calculations can be applied for fluence determination with no bias.
Meets guide requirement.

Fluence Measurement Methods

The regulatory positions in the guide that pertain to fluence measurement methods are summarized in Table 3-2.The compliance of the GGNS fluence measurements with the guide is summarized below.

Spectrum Coverage: GGNS does not have dosimetry sets installed to provide spectrum definition. This is in common with GE BWRs which have only limited dosimetry installed in surveillance capsules. The GGNS dosimetry analyzed to date is from iron wires attached to a surveillance capsule and removed after the first cycle of operation. Calculated neutron spectra are validated using the detailed measurements for test reactors and the calculational benchmark included in RG 1.190. Results are documented in Reference [3-2].

Dosimeter Nuclear and Material Properties: The GGNS dosimetry wire material characterization was performed by GE (Reference [3-3]). All nuclear constants and parameters used in the dosimetry counting and analysis follow ASTM standard procedures and use validated nuclear constants.
Meets guide requirement.

Corrections: [[

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Meets guide requirement.

Response Uncertainty: Uncertainty analyses for each dosimeter have been reported in all of the surveillance capsule reports performed by MPM. [[

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Meets guide requirement.

Validation: [[

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*Fast-Neutron Fluence :*M/C ratios are determined for each dosimeter measurement using calculated detector responses (Reference [3-2]).

Meets guide requirement.

Measurement-to-Calculated Ratios: The M/C ratios, standard deviations, and comparisons between calculation and measurement have been done and have been reported as discussed earlier (References [3-2]).

Meets guide requirement.

<u>Reporting Provisions</u>

The regulatory positions in the guide that pertain to neutron fluence and uncertainty reporting are summarized in Table 3-3.The compliance of the GGNS reporting with the guide is summarized below.

Neutron Fluence and Uncertainties: This report on the GGNS calculations, as well as the benchmark report (Reference [3-2]), meets all RG 1.190 reporting requirements.
Meets guide requirement.

Multigroup Fluences: Because of the extensive amount of data, only the fluence distributions for neutrons with energy greater than 1 MeV are reported, except for the vessel where neutrons

with energy greater than 0.1 MeV and dpa are also reported. The multigroup fluence rates are permanently stored for future access if required. These are not seen as having any application except for possible future dosimetry analysis. Multigroup flux values are readily available for all dosimetry locations.
Meets guide requirement.

Bias Reporting: No bias is observed and none is applied.
Meets guide requirement.

Integral Fluences and Uncertainties: These are all reported.
Meets guide requirement.

Comparisons of Calculation and Measurement: These are reported for all dosimetry locations.
Meets guide requirement.

Standard Neutron Field Validation: Not applicable.

Specific Activities and Average Reaction Rates: This is contained in all reports with measured dosimetry data.
Meets guide requirement.

Corrections and Adjustments to Measured Quantities: No corrections made.
Meets guide requirement.

3.2 Summary of MPM Methods

The neutron exposure of reactor structures to determine the spatial distribution of neutrons in the reactor components is determined by either a 3D neutron transport calculation, or by a combination of R-θ, R-Z, and R neutron transport calculations in support of the 2D synthesis method. In particular, the calculation determines the distribution of neutrons of all energies from their source from fission in the core region to their eventual absorption or leakage from the system. The transport calculations use a model of the reactor geometry that includes the significant structures and geometrical details necessary to accurately define the neutron environment at all locations of interest. A brief summary of the MPM methods is included here.

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Whenever possible, the plant geometry is modeled using as-built drawings. In cases where as-built drawings are not available, design drawings are used. During reactor operation, the neutron flux level at any point in the shroud or vessel will vary due to changes in fuel composition, power distributions within the core, and water void fraction. These changes occur between fuel cycles due to changes in fuel loading and fuel design, and within a fuel cycle due to fuel and poison burnup and resultant changes in power shape, control rod position, fission contributions by nuclide, and void fraction vs. axial height in each fuel bundle. To achieve the highest accuracy, changes in the three-dimensional bundle power distributions and water density distributions must be appropriately modeled. [[

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3.3 Benchmarking MPM Methodology

Although RG 1.190 was specifically developed to address calculation of fluence to the vessel, the guide can be considered to apply to other reactor components such as the shroud, surveillance capsules, and internal structures. As mentioned, MPM methods include the standard 2D synthesis, [[ ]] and 3D TORT. In order to meet the methods qualification requirement of RG 1.190, the MPM calculational methodology has been validated by comparison with measurement and calculational benchmarks. Comparisons of calculations with measurements have been made for the Pool Critical Assembly (PCA) pressure vessel simulator benchmark, [[ ]] Comparisons with a BWR calculational benchmark have also been completed.

The qualification of the methods used for the reactor transport calculations can be divided into two parts. The first part is the qualification of the cross-section library and calculational methods that are used to calculate the neutron transport. The second part is the validation of the method by comparison of calculations with dosimetry measurements for both representative plants and also for the current plant being analyzed, which, in the current case, is the GGNS plant. Details of the extensive benchmarking efforts performed by MPM are contained in a separate document [3-2].

Measurement and Calculational Benchmarks

The qualification of the cross-section library and calculational methods is particularly important for vessel fluence calculations because of the large amount of neutron attenuation between the source in the core and the vessel. The cross-sections are first developed as an evaluated file that details all of the reactions as a continuous function of energy. The ENDF/B evaluators take into account various measurements, including integral measurements (such as criticality and dosimetry measurements) that provide a test of the adequacy of the evaluation. For transport calculations using discrete ordinates, the ENDF/B cross-section files must be collapsed into multigroup files. This is done in two steps. First, fine-group cross-sections are calculated (Vitamin B6 library). [[ ]]

contains cross-sections collapsed using a BWR core spectrum, a PWR core spectrum, a PWR downcomer spectrum, a PWR vessel spectrum, and a concrete shield spectrum. These various cross-section libraries are then tested against various benchmarks and compared with measured results and results calculated using the fine-group cross-sections [3-7]. [[

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]] A particularly appropriate benchmark for geometry outside the core which includes the vessel, is the PCA benchmark [3-9]. This benchmark provides validation of the transport through typical reactor structures and a simulated reactor vessel in a simple geometry. The PCA has high-accuracy measurement results extending from inside a simulated thermal shield through to the outside of a simulated vessel. The PCA benchmark was calculated using the MPM methodology and detailed results are reported in [3-2]. [[

]]

Another benchmarking requirement of RG 1.190 is to compare with a suitable calculational benchmark. The calculational benchmarks used to satisfy this requirement are documented in Reference [3-11]. The benchmark problems include 3 different PWR geometries and a single BWR problem. It is intended that the analyst select the benchmark problem or problems appropriate to the plant being analyzed. [[

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Benchmarking MPM Calculations against Measured Data in Operating BWRs

The second part of benchmarking is to compare calculations with measurements in a geometry as close as possible to that which is being analyzed. Comparisons were made with BWR surveillance dosimetry measurements [[

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Calculated activation results from each dosimeter type all fell within ±20% of the measurement. The dosimetry results have been averaged for each set and the results are shown in Table 3-6. If the average results from the nine dosimetry sets are averaged, the average calculated-to-measured (C/M) ratio [[ ]] which indicates that the MPM methodology does not exhibit any consistent bias for BWR calculations. This meets the criterion set by RG 1.190 for acceptability of the calculations.

Benchmarking 3D TORT Methods

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]] Further, TORT was also benchmarked directly to the GGNS cycle 1 dosimetry data and

[[

]] very good agreement between the 3D calculation and the measurement, [[ ]]

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3.4 Conclusions

In summary, it is concluded that the RG 1.190 requirement for qualification of the MPM methods for BWR neutron transport analyses by comparisons to measurement and calculational benchmarks has been fully satisfied. Moreover, the agreement of calculations with measurements to within ±20% uncertainty indicates that the MPM calculations can be applied for fluence determination with no bias.

3.5 Section 3 References

[3-1] Regulatory Guide 1.190, Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence, U. S. Nuclear Regulatory Commission, March 2001.

[3-2] [[

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[3-3] [[

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[3-4] W. N. McElroy, “Data Development and Testing for Fast Reactor Dosimetry “, Nucl. Tech. 25, 177 (1975).

[3-5] R. Gold and W. N. McElroy, "The Light Water Reactor Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP): Past Accomplishments, Recent Developments, and Future Directions," Reactor Dosimetry: Methods, Applications, and Standardization, ASTM STP 1001, Harry Farrar and E. P. Lippincott, eds., American Society for Testing and Materials, Philadelphia, 1989, pp 44-61.

[3-6] RSICC Data Library Collection, DLC-185, 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, available from the Radiation Safety Information Computational Center, Oak Ridge National Laboratory, Oak Ridge, TN, March 1996.

[3-7] Ingersoll, D. T., et. al., "Production and Testing of the VITAMIN-B6 Fine-Group and the BUGLE-93 Broad-Group Neutron/Photon Cross-Section Libraries Derived from ENDF/B-VI Nuclear Data," ORNL-6795, NUREG/CR-6214, January 1995.

[3-8] [[

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[3-9] Remec, I. and Kam, F.B.K., Pool Critical Assembly Pressure Vessel Facility Benchmark, NUREG/CR-6454, (ORNL/TM-13205), USNRC, July 1997.

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[3-11] Carew, J. F., Hu, K., Aronson, A., Prince, A., and Zamonsky, G., PWR and BWR Pressure Vessel Fluence Calculational Benchmark Problems and Solutions, NUREG/CR-6115 (BNL-NUREG-52395), September, 2001.

Table 3-1 Summary of Regulatory Guide 1.190 Positions on Fluence Calculation Methods.

Regulatory Position – Fluence Calculation Methods	Regulatory Section
Fluence Determination. Absolute fluence calculations, rather than extrapolated fluence measurements, must be used for the fluence determination.	1.3
Modeling Data. The calculation modeling (geometry, materials, etc.) should be based on documented and verified plant-specific data.	1.1.1
Nuclear Data. The latest version of the Evaluated Nuclear Data File (ENDF/B) should be used for determining nuclear cross- sections. Cross-section sets based on earlier or equivalent nuclear-data sets that have been thoroughly benchmarked are also acceptable. When the recommended cross-section data change, the effect of these changes on the licensee-specific methodology must be evaluated and the fluence estimates updated when the effects are significant.	1.1.2
Cross-Section Angular Representation. In discrete ordinates transport calculations, a P_3 angular decomposition of the scattering cross-sections (at a minimum) must be employed.	1.1.2
Cross-Section Group Collapsing. The adequacy of the collapsed job library must be demonstrated by comparing calculations for a representative configuration performed with both the master library and the job library.	1.1.2
Neutron Source. The core neutron source should account for local fuel isotopics and, where appropriate, moderator density. The neutron source normalization and energy dependence must account for the fuel exposure dependence of the fission spectra, the number of neutrons produced per fission, and the energy released per fission.	1.2
End-of-Life Predictions. Predictions of the vessel end-of-life fluence should be made with a best-estimate or conservative generic power distribution. If a best estimate is used, the power distribution must be updated if changes in core loadings, surveillance measurements, or other information indicate a significant change in projected fluence values.	1.2
Spatial Representation. Discrete ordinates neutron transport calculations should incorporate a detailed radial- and azimuthal-spatial mesh of ~2 intervals per inch radially. The discrete ordinates calculations must employ (at a minimum) an S_8 quadrature and (at least) 40-80 intervals per octant.	1.3.1
Multiple Transport Calculations. If the calculation is performed using two or more "bootstrap" calculations, the adequacy of the overlap regions must be demonstrated.	1.3.1

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Table 3-1 Summary of Regulatory Guide 1.190 Positions on Fluence Calculation Methods (continued).

Regulatory Position – Fluence Calculation Methods	Regulatory Section
Point Estimates. If the dimensions of the tally region or the definition of the average-flux region introduce a bias in the talley edit, the Monte Carlo prediction should be adjusted to eliminate the calculational bias. The average-flux region surrounding the point location should not include material boundaries or be located near reflecting, periodic or white boundaries.	1.3.2
Statistical Tests. The Monte Carlo estimated mean and relative error should be tested and satisfy all statistical criteria.	1.3.2
Variance Reduction. All variance reduction methods should be qualified by comparison with calculations performed without variance reduction.	1.3.2
Capsule Modeling. The capsule fluence is extremely sensitive to the geometrical representation of the capsule geometry and internal water region, and the adequacy of the capsule representation and mesh must be demonstrated.	1.3.3
Spectral Effects on RT_{NDT}. In order to account for the neutron spectrum dependence of RT_{NDT}, when it is extrapolated from the inside surface of the pressure vessel to the T/4 and 3T/4 vessel locations using the > 1-MeV fluence, a spectral lead factor must be applied to the fluence for the calculation of ΔRT_{NDT}.	1.3.3
Cavity Calculations. In discrete ordinates transport-calculations, the adequacy of the S_8 angular quadrature used in cavity transport calculations must be demonstrated.	1.3.5
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.	1.4.1, 1.4.2, 1.4.3
Fluence Calculational Uncertainty. The vessel fluence (1 sigma) calculational uncertainty must be demonstrated to be 20% for RT_{PTS} and RT_{NDT} 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.	1, 1.4.3

Table 3-2 Summary of Regulatory Guide 1.190 Positions on Fluence Measurement Methods.

Regulatory Position - Fluence Measurement Methods	Regulatory Section
Spectrum Coverage. The set of dosimeters should provide adequate spectrum coverage.	2.1.1
Dosimeter Nuclear and Material Properties. Use of dosimeter materials should address melting, oxidation, material purity, total and isotopic mass assay, perturbations by encapsulations and thermal shields, and accurate dosimeter positioning. Dosimeter half-life and photon yield and interference should also be evaluated.	2.1.1
Corrections. Dosimeter-response measurements should account for fluence rate variations, isotopic burnup effects, detector perturbations, self-shielding, reaction interferences, and photofission.	2.1.2
Response Uncertainty. An uncertainty analysis must be performed for the response of each dosimeter.	2.1.3
Validation. Detector-response calibrations must be carried out periodically in a standard neutron field.	2.2
Fast-Neutron Fluence. The E > 1 MeV fast-neutron fluence for each measurement location must be determined using calculated spectrum-averaged cross-sections and individual detector measurements. As an alternative, the detector responses may be used to determine reaction probabilities or average reaction rates.	2.3
Measurement-to-Calculation Ratios. The M/C ratios, the standard deviation and bias between calculation and measurement, must be determined.	2.3

Table 3-3 Summary of Regulatory Guide 1.190 Positions on Fluence and Uncertainty Reporting.

Regulatory Position - Reporting Provisions	Regulatory Section
Neutron Fluence and Uncertainties. Details of the absolute fluence calculations, associated methods qualification and fluence adjustments (if any) should be reported. Justification and a description of any deviations from the provisions of this guide should be provided.	3.1
Multigroup Fluences. Calculated multigroup neutron fluences and fluence rates should be reported.	3.2
Bias Reporting. The value and basis of any bias or model adjustment made to improve the measurement-to-calculation agreement must be reported.	3.2
Integral Fluences and Uncertainties. Calculated integral fluences and fluence rates for E > 1 MeV and their uncertainties should be reported.	3.3
Comparisons of Calculation and Measurement. Measured and calculated integral E > 1 MeV fluences or reaction rates and uncertainties for each measurement location should be reported. The M/C ratios and spectrum averaged cross-section should also be reported.	3.4
Standard Neutron Field Validation. The results of the standard field validation of the measurement method should be reported.	3.5
Specific Activities and Average Reaction Rates. The specific activities at the end of irradiation and measured average reaction rates with uncertainties should be reported.	3.5
Corrections and Adjustments to Measured Quantities. All corrections and adjustments to the measured quantities and their justification should be reported.	3.5

Table 3-4 Comparison of Calculated and Measured Results for PCA.

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Table 3-5 Comparison of Calculated and Benchmark Results for the Reactor Vessel Calculated at Reactor Axial Peak.

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Table 3-6 Tabulation of Dosimetry Results for Operating BWR Plants.

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4
SHROUD AND TOP GUIDE FLUENCE RESULTS

An important concern for many BWR reactors is the neutron exposure of the shroud and top guide welds in these components. Because the shroud is located close to the fuel region, the exposure is relatively high, and this must be taken into account to evaluate the possibility of stress corrosion cracks and crack growth. Of special importance is determining weld locations that exceed the irradiation assisted stress corrosion cracking (IASCC) fast (E>1.0 MeV) neutron threshold fluence. A fluence of 5E+20 n/cm^2 has been proposed as a screening threshold, but more recent data suggest a lower threshold for some materials. In any event, radiation damage effects on the SCC crack growth model must be considered in setting future inspection intervals.

4.1 Shroud Weld Fluence Results

Locations of all the shroud vertical and horizontal welds are shown schematically in Figures 1-1 and 1-2, and the weld locations are listed in Tables4-1 and 4-2. Evaluations of the fluence for all of these welds have been performed using the 3D TORT method described in Section 2. Tables 4-3 and 4-4 summarize the calculated maximum fluences to the shroud and top guide welds. These fluences were calculated at the inner diameter (ID) surface of the welds at the point along the welds where the fluence is a maximum. [[

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4.2 Comparison with Past Results

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4.3 Section4 References

[4-1] [[
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[4-2] [[
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Non-Proprietary

Table 4-1 Locations for Fluence Evaluation in the Shroud and Top Guide Horizontal Welds.

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Table 4-2 Locations for Fluence Evaluation in the Shroud and Top Guide Vertical Welds.

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Non-Proprietary

Table 4-3 Estimated Maximum Fluence to GGNS Shroud and Top Guide Horizontal Welds.

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Table 4-4 Estimated Maximum Fluence to GGNS Shroud and Top Guide Vertical Welds.

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Non-Proprietary

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Figure 4-1Fast Flux (E>1.0 MeV) at the Shroud IR to Weld H4 for Cycles 1 through 21.

Non-Proprietary

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Figure 4-2 Fast Fluence (E>1.0 MeV) to Shroud Weld H4 at the End of Cycle 21 (28.088 EFPY).

Non-Proprietary

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Figure 4-3 Fast Fluence (E>1.0 MeV) to Shroud Weld H4 at 35 EFPY.

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Figure 4-4 Fast Fluence (E>1.0 MeV) to Shroud Weld H4 at 54 EFPY.

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Figure 4-5 Fast Fluence (E>1.0 MeV) to Shroud Vertical Welds V13-V16 at the End of Cycle 21 (28.088 EFPY).

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Figure 4-6 Fast Fluence (E>1.0 MeV) to Shroud Vertical Welds V13-V16 at 35 EFPY.

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Figure 4-7 Fast Fluence (E>1.0 MeV) to Shroud Vertical Welds V13-V16 at 54 EFPY.

5
VESSEL AND CAPSULE FLUENCE RESULTS

5.1 Cycle 1 Dosimetry Analysis

The GGNS cycle 1 dosimetry was originally analyzed using 2D synthesis [5-1]. The past work has been reanalyzed using 3D TORT calculations. [[

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The detailed power history for GGNS cycle 1 operation is presented in Table 5-3 [5-2]. [[

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The removable dosimetry packet consisted of 3 iron wires. The dosimetry packet was located on the side of the capsule. The location of the dosimetry was provided in the plant documentation as indicated in Table 2-3. [[

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5.2 Pressure Vessel Analysis

The analysis of the vessel was carried out using the 3D methodology described in Section 2. The maximum flux location at the vessel shell course plates varies slightly from cycle-to-cycle as a result of the different fuel loadings. Therefore, the maximum fluence for the shell course plates was conservatively calculated [[

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Plots of the azimuthal and axial traverses of fluence at the vessel wetted surface at the end of cycle 21, along with the results at the vessel 1/4T and 3/4T positions, are shown in Figures 5-1 and 5-2, respectively. [[

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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." [[

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As shown in Figure 5-3, the reactor vessel is made up of plates that are welded together. For the purpose of PT curve calculations, fluences have been evaluated for shell 1, shell 2, and shell 3 materials. [[

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Radiation embrittlement effects are often correlated with fast fluence (E > 1 MeV). However, it is generally thought that dpa might be a better correlation parameter since it accounts for spectral effects and, if this is correct, the use of the fast fluence (E > 1 MeV) values within the vessel might under predict the radiation damage at locations within the vessel. Therefore, the NRC requires the use of a dpa attenuation in the vessel for preparation of pressure-temperature operating curves. The fluence attenuation factors within the vessel can be evaluated using calculated dpa attenuation from Table 5-9 or using the dpa formulation specified in the RG 1.99 (Rev 2) [5-6]. The fluence values using both these attenuation methods are given in Table 5-18 [[

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This evaluation of the dpa cross section is based on the ENDF-IV cross-section file. [[

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The vessel has nozzle penetrations at several locations, and neutron exposure at the nozzles is of concern for neutron damage analysis. Four sets of nozzles were evaluated, and the locations of these nozzles along with the fluence evaluation points are summarized in Table 5-19. As shown in the table, the nozzle fluences are determined at the maximum point in the nozzle weld as well as at the conservative locations specified in Reference [5-9]. For nozzles below the core, the maximum fluence point occurs at the top of the nozzle. The reverse is true for nozzles above the core. The maximum fluence values at the nozzles are summarized in Table 5-20. Fluences at conservative locations near the nozzles specified by GGNS in Reference [5-9] are given in Tables 5-21 and 5-22. The data in Table 5-21 is at the elevation specified, but at the maximum azimuthally, whereas the data in Table 5-22 is at the locations specified in [5-9]. [[

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5.3 Comparison with Past Results

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5.4 Section 5 References

[5-1] [[

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[5-2] [[

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[5-3] ASTM Designation E263-00, Standard Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Iron, in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 2003.

[5-4] ASTM Designation E1005-03, Standard Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance, E706(IIIA), in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 2003.

[5-5] [[

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[5-6] Regulatory Guide 1.99, Revision 2, Radiation Embrittlement of Reactor Vessel Materials, U. S. Nuclear Regulatory Commission, May 1988.

[5-7] ASTM Designation E693-94, Standard Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA), E706(ID), in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 2000.

[5-8] ASTM Designation E693-01, Standard Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA), E706(ID), in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 2003.

[5-9] [[

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Table 5-1Calculated Fe(n,p) Reaction Rates at the Cycle 1 Dosimetry Location.

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Table 5-2Flux Spectrum at Dosimetry Location.

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Table 5-3 Grand Gulf Cycle 1 Power History.

Date	Days In Period	Cycle Cumulative (MWd/MTU)	Period Effective Full Power Days	Period Average Fraction Of Full Power
30-Sep-84	Begin	0		
3-Nov-84	35	178	6.789	0.1940
14-Nov-84	11	235	2.174	0.1976
21-Nov-84	7	295	2.288	0.3269
4-Feb-85	75	618	12.320	0.1643
12-Apr-85	67	906	10.985	0.1640
25-Apr-85	13	1071	6.293	0.4841
17-May-85	22	1384	11.938	0.5427
4-Jun-85	18	1666	10.756	0.5976
3-Jul-85	29	2140	18.079	0.6234
15-Jul-85	12	2351	8.048	0.6707
26-Jul-85	11	2596	9.345	0.8495
7-Aug-85	12	2863	10.184	0.8487
21-Aug-85	14	3158	11.252	0.8037
23-Aug-85	2	3187	1.106	0.5531
28-Aug-85	5	3269	3.128	0.6255
5-Sep-85	8	3466	7.514	0.9392
10-Sep-85	5	3591	4.768	0.9535
19-Sep-85	9	3808	8.277	0.9196
22-Sep-85	3	3865	2.174	0.7247
26-Sep-85	4	3914	1.869	0.4672
30-Sep-85	4	4026	4.272	1.0680
12-Oct-85	12	4334	11.748	0.9790
15-Dec-85	64	4360	0.992	0.0155
18-Dec-85	3	4459	3.776	1.2587
28-Dec-85	10	4586	4.844	0.4844
9-Jan-86	12	4861	10.489	0.8741
11-Jan-86	2	4891	1.144	0.5721
31-Jan-86	20	5177	10.909	0.5454
12-Feb-86	12	5376	7.590	0.6325
25-Feb-86	13	5498	4.653	0.3579
28-Feb-86	3	5551	2.022	0.6738
15-Mar-86	15	5812	9.955	0.6637
18-Apr-86	34	6250	16.706	0.4914
28-Apr-86	10	6419	6.446	0.6446

Table 5-3 Grand Gulf Cycle 1 Power History (continued).

Date	Days in Period	Cycle Cumulative (MWd/MTU)	Period Effective Full Power Days	Period Average Fraction of Full Power
9-May-86	11	6612	7.361	0.6692
25-May-86	16	6907	11.252	0.7032
28-May-86	3	6964	2.174	0.7247
13-Jun-86	16	7286	12.282	0.7676
27-Jun-86	14	7615	12.549	0.8963
3-Jul-86	6	7738	4.691	0.7819
18-Jul-86	15	8038	11.442	0.7628
30-Jul-86	12	8229	7.285	0.6071
7-Aug-86	8	8389	6.103	0.7628
18-Aug-86	11	8586	7.514	0.6831
23-Aug-86	5	8701	4.386	0.8773
3-Sep-86	11	8782	3.089	0.2809
5-Sep-86	2	8823	1.564	0.7819

Table 5-4Nuclear Parameters Used in the Evaluation of Neutron Sensors.

Monitor Material	Reaction of Interest	Isotopic Fraction	Approximate Response Threshold	Product Half-Life
Iron	$Fe^{54}(n,p)Mn^{54}$	0.05845	2 MeV	312.3 days

Table 5-5Tabulation of Dosimetry Results.

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Non-Proprietary

Table 5-6 Azimuthal Variation of Maximum Fluence at the Vessel Wetted Surface.

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Table 5-6Azimuthal Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Table 5-6 Azimuthal Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Non-Proprietary

Table 5-6 Azimuthal Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Non-Proprietary

Table 5-6 Azimuthal Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Table 5-6 Azimuthal Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Table 5-6 Azimuthal Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Non-Proprietary

Table 5-7 Axial Variation of Maximum Fluence at the Vessel Wetted Surface.

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Table 5-7 Axial Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Table 5-7 Axial Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Table 5-7 Axial Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Non-Proprietary

Table 5-7 Axial Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Table 5-7 Axial Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Non-Proprietary

Table 5-7 Axial Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Table 5-7 Axial Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Table 5-7Axial Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Table 5-8Relative Exposure through Vessel at Maximum Fluence Point at the End of Cycle 21 (28.088 EFPY).

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Non-Proprietary

Table 5-9 Calculated Maximum Vessel Exposure (Shell 2) at the End of Cycle 21

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Table 5-10 Azimuthal Location Ranges of Plates in Vessel Shell 1, Shell2, and Shell 3.

Plate ID	Azimuthal Angle Range (degrees)	Equivalent Azimuthal Angle Range in First Quadrant (degrees)
Shell 1		
PC MK 21-1-1 Plate Heat A1113	0-120	0-90
PC MK 21-1-2 Plate Heat C2557	120-240	0-60
PC MK 21-1-3 Plate Heat C2506	240-360	0-90
Shell 2		
PC MK 22-1-1 Plate Heat C2593	322-52	0-52
PC MK 22-1-2 Plate Heat C2594	52-142	38-90
PC MK 22-1-3 Plate Heat C2594	142-232	0-52
PC MK 22-1-4 Plate HeatA1224	232-322	38-90
Shell 3		
PC MK 23-1-1 Plate Heat C2741	349-109	0-90
PC MK 23-1-2 Plate Heat C2779	109-229	0-71
PC MK 23-1-3 Plate Heat C2741	229-349	11-90

Table 5-11 Calculated Maximum Vessel Shell 1 Exposure at the End of Cycle 21 (28.088 EFPY) and Projected to Future Exposures.

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Table 5-12Calculated Maximum Vessel Shell 3 Exposure at the End of Cycle 21 (28.088 EFPY) and Projected to Future Exposures.

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Table 5-13 Azimuthal Locations of Vertical Welds in Vessel Shell 1, Shell 2 and Shell 3.

Weld ID	Azimuthal Angle (degrees)	Equivalent Azimuthal Angle in First Quadrant (degrees)
Shell 1		
BA	0	0
BB	120	60
BC	240	60
Shell 2		
BD	52	52
BE	142	38
BF	232	52
BG	322	38
Shell 3		
BH	109	71
BJ	229	49
BK	349	11

Table 5-14 Calculated Maximum Vessel Shell 1 Weld Exposures at the End of Cycle 21 (28.088 EFPY) and Projected to Future Exposures.

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Table 5-15 Calculated Maximum Vessel Shell 2 Weld Exposures at the End of Cycle 21 (28.088 EFPY) and Projected to Future Exposures.

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Table 5-16 Calculated Maximum Vessel Shell 3 Weld Exposures at the End of Cycle 21 (28.088 EFPY) and Projected to Future Exposures.

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Table 5-17Calculated Maximum Vessel Circumferential Weld Exposures at the End of Cycle 21 (28.088 EFPY) and Projected to Future Exposures.

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Table 5-18 GGNS Calculated Vessel Fluence and Fluence Determined using dpa Attenuation.

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Table 5-19 GGNS RPV Nozzle Locations.

Nozzle	Height Above Vessel Zero to Center of Nozzle (inches)	Height To Center of Nozzle Relative to BAF (inches)	Nozzle Weld OD (inches)	Maximum Nozzle Fluence Point Relative to BAF (inches)	Fluence Location Height Specified in Reference [5-9] Relative to Vessel '0' (inches)	Fluence Location Height Specified in Reference [5-9] Relative to BAF (inches)	Nozzle Center Angles in First Quadrant
N1	172.31	-44.00	24.750	-31.62	199.7	-16.61	0°
N2	179.31	-37.00	13.094	-30.45	197.0	-19.31	26°15', 51°45', 77°15'
N12	366.00	149.69	1.880	148.75	364.7	148.39	15°
N6	419.00	202.69	13.250	196.07	-[a]	-[a]	39°

Table 5-20GGNS RPV Nozzle Maximum Fluence Values.

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Table 5-21GGNS RPV Nozzle Region Maximum Fluence Values in the First Quadrant at the Axial Locations Specified in Reference [5-9].

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Table 5-22GGNS RPV Nozzle Region Fluence Values at the Axial and Azimuthal Locations Specified in Reference [5-9].

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Figure 5-1Azimuthal Variation of Maximum Vessel Fluence (E > 1.0 MeV) at the End of Cycle 21 (28.088 EFPY).

Non-Proprietary

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Figure 5-2 Axial Variation of Peak Fluence (E > 1.0 MeV) in the Vessel at the End of Cycle 21 (28.088 EFPY).

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Figure 5-3 GGNS Vessel Roll Out Drawing Showing Beltline Weld Seam and Plate Locations. The Increase in the Axial Extent of the Beltline will Eventually Require that Radiation Damage Effects for Shell 3 Materials be Included in the PT Curve Analysis.

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Figure 5-4Axial Variation of Peak Fluence (E > 1.0 MeV) in the Vessel at 54 EFPY.

6 UNCERTAINTY ANALYSIS

A comprehensive report has been prepared which documents the benchmarking of MPM transport methods for BWR plant analyses [6-1]. For each plant analyzed, a detailed uncertainty analysis was performed to estimate each source of uncertainty in the calculated fluence values. This analysis made use of defined uncertainties and tolerances where possible, but some of the uncertainty estimates had to be based on estimates derived from data variation, such as the detailed power distribution and void fraction variations within a single cycle. This section of the report summarizes the uncertainty analysis performed for GGNS.

The geometry uncertainty assignments are from plant drawings referenced in Tables 2-2 through 2-4, and from generic assessments for cases where dimensional uncertainties are not available. Discussion of each uncertainty assumption is given below. Based on these uncertainty values, detailed evaluations were performed for the shroud, surveillance capsule, and reactor vessel. The uncertainty assessments for TORT results at the shroud, capsule, and vessel over the active fuel elevations are summarized in Table 6-1. The uncertainty assessments for 2D synthesis results at the shroud, capsule, and vessel over the active fuel regions are summarized in Table 6-2. The differences in the uncertainties between TORT and 2D synthesis are in the methods uncertainty as discussed further below.

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6.1 Uncertainty Assumptions

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Non-Proprietary

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6.2 Uncertainty Evaluation over Active Fuel Length

The results for the uncertainty evaluation over the active fuel length are summarized in Tables6-1 and 6-2, based on the assumptions just discussed. These tables are applicable to the shroud, vessel, and surveillance capsule over the extent of the active fuel region [[ ]] A total uncertainty was derived by combining the independent individual contributors in quadrature.

3D TORT Uncertainty

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6.3 Uncertainty Evaluation for Shroud and Top Guide Weld Locations

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6.4 Uncertainty Conclusions

The detailed uncertainty analysis demonstrates that the MPM calculational methods provide fluence results within allowable tolerance bounds (+20%) for the reactor vessel, shroud welds, and surveillance capsules which lie within the axial active fuel region. This satisfies the requirements of RG 1.190. However, RG 1.190 does not address benchmarking outside this region. In the uncertainty analysis, evaluations of structures outside the beltline region have been made, and, especially above the core, [[

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6.4 Section 6 References

[6-1] [[

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[6-2] [[

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[6-3] McElroy, W.N., Ed., "LWR-PV-SDIP: PCA Experiments and Blind Test", NUREG/CR-1861, 1981.

[6-4] [[

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[6-5] Maerker, R.E., et.al., "Application of LEPRICON Methodology to LWR Pressure Vessel Dosimetry", Reactor Dosimetry: Methods, Applications, and Standardization, ASTM STP 1001, 1989, pp 405-414.

[6-6] [[

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[6-7] [[

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[6-8] [[

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[6-9] Fero, A.H., "Neutron and Gamma Ray Flux Calculations for the Venus PWR Engineering Mockup," NUREG/CR-4827 (WCAP-11173), January, 1987.

[6-10] Tsukiyama, T., et. al., "Benchmark Validation of TORT Code Using KKM Measurement and its Application to 800 MWe BWR," Proceedings of the 11th International Symposium on Reactor Dosimetry, Brussels, Belgium, August 18-23, 2002, World Scientific, to be published.

[6-11] [[

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Non-Proprietary

Table 6-1 Grand Gulf Shroud, Capsule, and Vessel Active Fuel Region TORT Calculational FluenceUncertainty.

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Non-Proprietary

Table 6-2 Grand Gulf Shroud, Capsule, and Vessel Active Fuel Region 2D Synthesis Calculational Fluence Uncertainty.

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Table 6-3 Estimated Maximum Uncertainty for Shroud and Top Guide Horizontal Welds.

Horizontal Weld	Fluence Uncertainty at Weld IR (Percent)
H1	[[ ]]
H2	[[ ]]
H3	[[ ]]
H4	[[ ]]
H5	[[ ]]
H6A	[[ ]]
H6B	[[ ]]
H7	[[ ]]

Table 6-4 Estimated Maximum Uncertainty for Shroud and Top Guide Vertical Welds.

Vertical Weld	Uncertainty in Maximum Fluence (Percent)
V1 to V4	[[ ]]
V5, V6	[[ ]]
V7,V8	[[ ]]
V9 to V12	[[ ]]
V13,V14	[[ ]]
V15,V16	[[ ]]
V17,V18	[[ ]]
V19 to V22	[[ ]]
V23, V24	[[ ]]

Non-Proprietary

Table 6-5 Estimated Maximum Weld IR Fluence Uncertainty for Shroud and Top Guide Vertical Welds.

Vertical Weld	Maximum Uncertainty to Weld IR Fluence (Percent)
V1 to V4	[[ ]]
V5, V6	[[ ]]
V7,V8	[[ ]]
V9 to V12	[[ ]]
V13,V14	[[ ]]
V15,V16	[[ ]]
V17,V18	[[ ]]
V19 to V22	[[ ]]
V23, V24	[[ ]]

7 SUMMARY AND CONCLUSIONS

A detailed three-dimensional transport calculation has been completed for GGNS encompassing operation over the first 20 fuel cycles [[ ]]. This work was undertaken to calculate the best estimate neutron fluence, and its uncertainty, to the GGNS reactor pressure vessel, surveillance capsule, core shroud/top guide horizontal and vertical welds, as well as to several below core structures and nozzles. The calculations in the active fuel region were carried out using a three dimensional neutron transport calculation [[

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Based on the calculations and analyses performed, the following conclusions have been made:

- The transport calculations meet all of the criteria of RG 1.190 for evaluation of fluence to the shroud, surveillance capsule, and reactor vessel within the reactor beltline region. The calculational methodology has been benchmarked to reactor mock-ups, calculational benchmarks, and measurements in other BWR plants. In addition, the calculations have been benchmarked against measurements in GGNS. Comparisons with dosimetry measurements at the GGNS surveillance capsule location at the end of cycle 1 were made and excellent agreement was found. [[

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- Fluence results have been obtained for all GGNS shroud and top guide welds. [[

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- Fluence results have been obtained for the reactor vessel and projected for future operation up to 54 EFPY.[[

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- Fluence calculations for the surveillance capsules indicated that the capsule lead factor is [[ ]]. Although the lead factor is less than unity, this is not a concern for GGNS since the BWRVIP surveillance program does not require any future capsule testing for GGNS.

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8 NOMENCLATURE

ASTM	American Society for Testing and Materials
BAF	bottom of active fuel
BWR	boiling water reactor
BWRVIP	BWR Vessel Internals Project
C/M	calculated-to-measured ratio
D	dimension
dpa	displacements per atom
EBZ	enriched bottom zone
EFPS	effective full power seconds
EFPY	effective full power years
ENDF	evaluated nuclear data file
EOC	end-of-cycle
GGNS	Grand Gulf Nuclear Station
ID	inner diameter
IGSCC	irradiation assisted stress corrosion cracking
IR	inner radius
LHGR	linear heat generation rate
LWR	light water reactor
MWd/MTU	megawatt days per metric ton of uranium
MOC	middle-of-cycle
MPM	MP Machinery and Testing, LLC
NMP-1	Nine Mile Point Unit 1
NMP-2	Nine Mile Point Unit 2
NRC	U. S. Nuclear Regulatory Commission
OD	outer diameter
OR	outer radius
ORNL	Oak Ridge National Laboratory
PCA	pool critical assembly
PT	Pressure-Temperature
PWR	pressurized water reactor
RBS	River Bend Station
RG	Regulatory Guide
RSICC	Radiation Safety Information Computational Center
RT_{NDT}	Reference temperature for nil ductility transition
RT_{PTS}	Reference temperature for pressurized thermal shock
SE	sequence exchange
T	vessel wall thickness
TAF	top of active fuel
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A
SHROUD/TOP GUIDE WELD FLUENCE RESULTS AT THE END OF CYCLE 21 (28.088 EFPY EXPOSURE)

This appendix contains calculated fast fluence values (fluence for neutrons with energy above 1 MeV) for welds in the shroud and top guide. Fluence values for each weld are given at the IR, OR, and at positions 1/4, 1/2, and 3/4 of the distance between the IR and OR for an operation time of28.088 EFPY (the calculated end of cycle 21). Values are tabulated versus azimuthal angle for horizontal welds, and versus height above BAF for vertical welds.

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Appendix Table A-1Fast Fluence at Locations in the Top Guide for Weld H1 vs. Azimuth.

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Appendix Table A-2Fast Fluence at Locations in the Top Guide for Weld H2 vs. Azimuth.

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Appendix Table A-3 Fast Fluence at Locations in the Shroud for Weld H3 vs. Azimuth.

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Appendix Table A-4 Fast Fluence at Locations in the Shroud for Weld H4 vs. Azimuth.

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Appendix Table A-5 Fast Fluence at Locations in the Shroud for Weld H5 vs. Azimuth[a].

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Appendix Table A-6 Fast Fluence at Locations in the Shroud for Weld H6A vs. Azimuth[a].

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Appendix Table A-7Fast Fluence at Locations in the Shroud for Weld H6B vs. Azimuth.

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Appendix Table A-8Fast Fluence at Locations in the Shroud for Weld H7 vs. Azimuth.

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Appendix Table A-9 Fast Fluence at Locations in the Top Guide for Welds V1 and V3 vs. Height above BAF.

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Appendix Table A-10 Fast Fluence at Locations in the Top Guide for Welds V2 and V4 vs. Height above BAF.

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Appendix Table A-11Fast Fluence at Locations in the Top Guide for Weld V5 and V6 vs. Height above BAF.

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Appendix Table A-12Fast Fluence at Locations in the Top Guide for Welds V7 and V8 vs. Radial Location.

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Appendix Table A-13 Fast Fluence at Locations in the Top Guide for Welds V9 and V11 vs. Height above BAF.

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Appendix Table A-14Fast Fluence at Locations in the Top Guide for Welds V10 and V12 vs. Height above BAF.

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Appendix Table A-15Fast Fluence at Locations in the Shroud for Weld V13 and V14 vs. Height above BAF.

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Appendix Table A-16 Fast Fluence at Locations in the Shroud for Weld V15 and V16 vs. Height above BAF.

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Appendix Table A-17Fast Fluence at Locations in the Shroud for Welds V17 and V18 vs. Height above BAF.

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Appendix Table A-18 Fast Fluence at Locations in the Shroud for Welds V19 and V21 vs. Height above BAF.

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Appendix Table A-19 Fast Fluence at Locations in the Shroud for Welds V20 and V22 vs. Height above BAF.

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Appendix Table A-20Fast Fluence at Locations in the Shroud for Welds V23 and V24 vs. Height above BAF.

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B
SHROUD/TOP GUIDE WELD FLUENCE RESULTS AFTER 35 EFPY EXPOSURE

This appendix contains calculated fast fluence values (fluence for neutrons with energy above 1 MeV) for welds in the shroud and top guide. Fluence values for each weld are given at the IR, OR, and at positions 1/4, 1/2, and 3/4 of the distance between the IR and OR for an exposure of 35 EFPY (6.912 EFPY beyond the calculated end of cycle 21). Values are tabulated versus azimuthal angle for horizontal welds, and versus height above BAF for vertical welds.

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Appendix Table B-1Fast Fluence at Locations in the Top Guide for Weld H1 vs. Azimuth.

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Appendix Table B-2 Fast Fluence at Locations in the Top Guide for Weld H2 vs. Azimuth.

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Appendix Table B-3Fast Fluence at Locations in the Shroud for Weld H3 vs. Azimuth.

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Appendix Table B-4Fast Fluence at Locations in the Shroud for Weld H4 vs. Azimuth.

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Appendix Table B-15Fast Fluence at Locations in the Shroud for Weld V13 and V14 vs. Height above BAF.

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Appendix Table B-17 Fast Fluence at Locations in the Shroud for Welds V17 and V18 vs. Height above BAF.

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Appendix Table B-20 Fast Fluence at Locations in the Shroud for Welds V23 and V24 vs. Height above BAF.

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C
SHROUD/TOP GUIDE WELD FLUENCE RESULTS AFTER 54 EFPY EXPOSURE

This appendix contains calculated fast fluence values (fluence for neutrons with energy above 1 MeV) for welds in the shroud and top guide. Fluence values for each weld are given at the IR, OR, and at positions 1/4, 1/2, and 3/4 of the distance between the IR and OR for an exposure of 54 EFPY (25.912 EFPY beyond the calculated end of cycle 21). Values are tabulated versus azimuthal angle for horizontal welds, and versus height above BAF for vertical welds.

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Attachment 2

GNRO-2016/00015

<u>MPM Machinery and Testing, LLC (MPM) Neutron Transport Analysis for Grand Gulf Nuclear Station, report MPM-814779, Revision 2</u>

(Non- Proprietary Version)

This is a non-proprietary version of Attachment 3 of letter GNRO-2015/00011 which has the proprietary information removed. Redacted information is identified by blank space enclosed with double brackets [[ ]].

NON-PROPRIETARY VERSION

Report Number MPM-814779
Revision 2

Neutron Transport Analysis For Grand Gulf Nuclear Station

MP Machinery & TESTING, LLC

January, 2015

MPM Technical Report

Neutron Transport Analysis for
Grand Gulf Nuclear Station

MPM Report Number MPM-814779
Revision 2

Final Report, January, 2015

MP Machinery and Testing, LLC

2161 Sandy Drive

State College, PA 16803

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M P Manahan, Sr.

M. P. Manahan, Sr.
President

1/31/2015

Date

J. Nemet
QA Manager

1/31/2015

Date

EXECUTIVE SUMMARY

This work was undertaken to calculate the best estimate neutron fluence, and its uncertainty, to the Grand Gulf Nuclear Station (GGNS) reactor pressure vessel, core shroud and top guide horizontal and vertical welds, as well as to several beltline vessel nozzles. Figures ES-1 and ES-2 show the shroud and top guide weld designations, and Figure ES-3 shows the weld seam and plate locations for the reactor vessel. 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 of MPM methods has been prepared under separate cover.

The neutron transport calculational procedures and dosimetry analysis methods meet standards specified by the NRC and ASTM as appropriate. In particular, the 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. [[

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Summary of Shroud and Top Guide Fluence Results

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Summary of Vessel 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. [[

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The vessel has nozzle penetrations at several locations, and neutron exposure at the nozzles is of concern for neutron damage analysis. Four sets of nozzles were evaluated. For nozzles below the core, the maximum fluence point occurs at the top of the nozzle. The reverse is true for nozzles above the core. [[

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Benchmarking and Uncertainty Analysis

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 [[ ]] These uncertainties are within the value of ± 20% specified by RG 1.190. Moreover, the 3D calculations have been benchmarked [[

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Regulatory Guide 1.190 requires that the overall fluence calculation bias and uncertainty must be determined by an appropriate combination of the analytic uncertainty analysis results and the results of the uncertainty analysis based on the comparisons to the operating reactor and simulator benchmark measurements. The regulatory guide states that this combination may be a weighted average that accounts for the reliability of the individual estimates. The regulatory guide goes on to state that if the analytical uncertainty at the 1 sigma level is greater than 30%, the methodology of the regulatory guide is not applicable and the application will be reviewed on an individual basis. [[

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Table ES-1 Maximum Fluence to GGNS Shroud and Top Guide Horizontal Welds.

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Table ES-2 Maximum Fluence to GGNS Shroud and Top Guide Vertical Welds.

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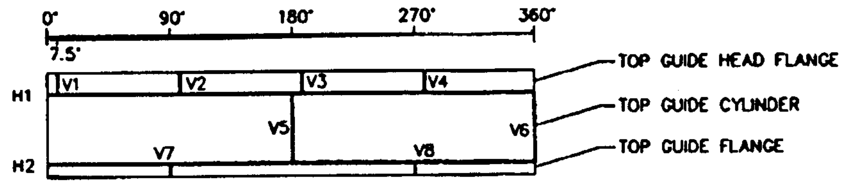

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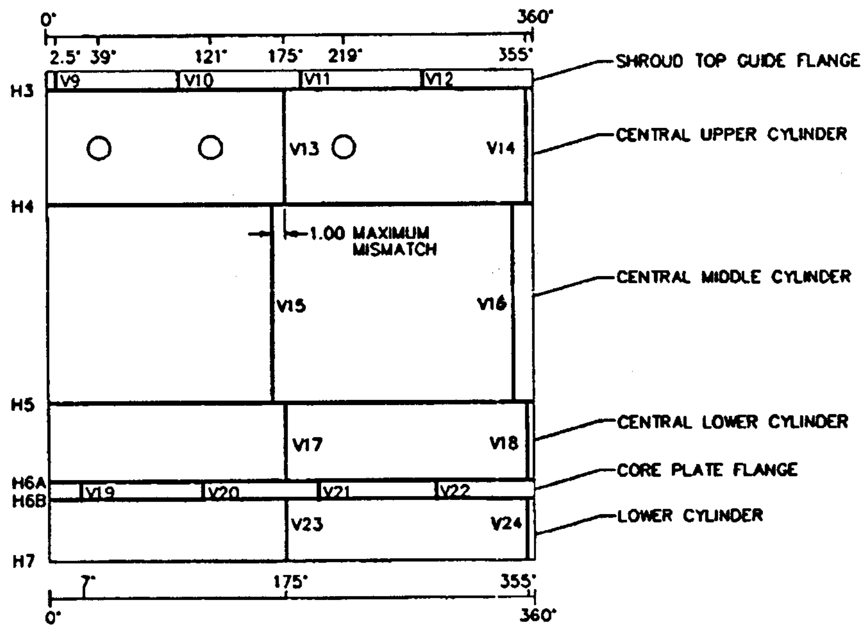

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1 INTRODUCTION

This work was undertaken to calculate the best estimate neutron fluence, and its uncertainty, to the Grand Gulf Nuclear Station (GGNS) top guide, reactor pressure vessel, vessel nozzles, and core shroud horizontal and vertical welds. The neutron exposure at the top guide, shroud, and, in particular, the shroud welds is an important concern for many Boiling Water Reactors (BWRs) since cracks have been observed in several plants. Figures 1-1 and 1-2 show the GGNS top guide and shroud weld designations, and Figure 1-3 shows the GGNS weld seam and plate locations for the reactor vessel. These designations are used throughout the report to identify the vertical and horizontal welds of interest. It is important to note that, with the exception of welds V7 and V8, all of the shroud and top guide welds are located within the cylindrical portion of these components. [[

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The neutron fluence calculations were carried out using a three dimensional (3D)TORT model for each fuel cycle through the end of cycle 21. At the time of the calculations, the plant had not completed cycle 20, and therefore the fuels data has been extrapolated to the estimated end of cycle 20. It will be necessary to update the cycle 20 transport analysis sometime in the future to represent the as-burned cycle. [[

]] The neutron transport models are fully described in Section 2. The calculational procedures meet standards specified by the Nuclear Regulatory Commission (NRC) and American Society for Testing and Materials (ASTM) as appropriate. In particular, the beltline analysis meets the requirements of Regulatory Guide 1.190 (RG 1.190) [1-2]. An important requirement of RG 1.190 is methodology benchmarking. In addition to the plant-specific benchmarking for GGNS, other MP Machinery and Testing, LLC (MPM) benchmark work is summarized in Section 3. A full discussion of the benchmarking of MPM methods is given in Reference [1-3].

The shroud fluence results are presented in Section4 and the Appendices. The vessel fluence calculations are given in Section5. RG 1.190 requires that a detailed uncertainty analysis be performed to identify each source of uncertainty and the impact on the overall accuracy of the calculation. The uncertainty analysis results are reported in Section6. Summary and conclusions are given in Section7.

1.1 Section 1 References

[1-1] [[

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[1-2] Regulatory Guide 1.190, Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence, U. S. Nuclear Regulatory Commission, March 2001.

[1-3] [[

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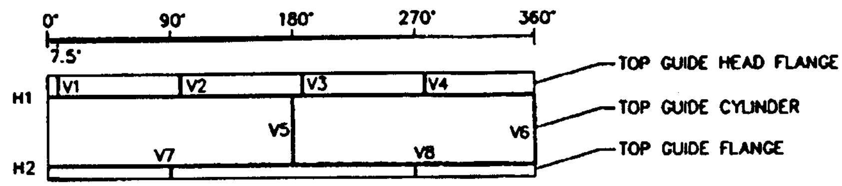

Figure 1-1 Weld Designations for GGNS Top Guide.

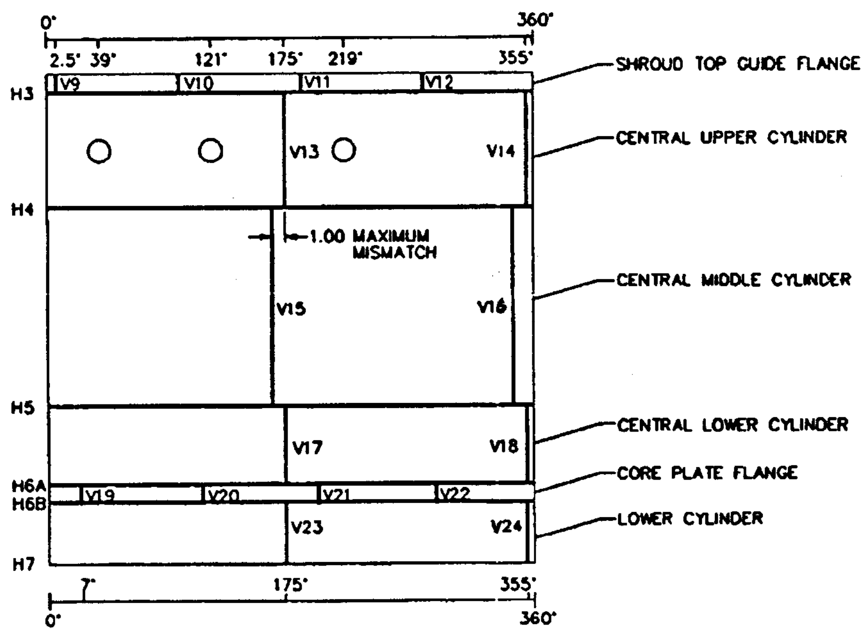

Figure 1-2 Weld Designations for GGNS Shroud.

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Figure 1-3GGNS Vessel Rollout Drawing Showing Beltline Weld Seam and Plate Locations. The Increase in the Axial Extent of the Beltline Requires that Radiation Damage Effects for Shell 1, 2 and 3 Materials be Included in the P-T Curve Analysis.

2
NEUTRON TRANSPORT MODEL DESCRIPTION

The neutron exposure of reactor structures is determined by a neutron transport calculation, or a combination of neutron transport calculations, to represent the distribution of neutron flux in three dimensions. The calculation determines the distribution of neutrons of all energies from their source from fission in the core region to their eventual absorption or leakage from the system. The calculation uses a model of the reactor geometry that includes the significant structures and geometrical details necessary to define the neutron environment at locations of interest. This chapter provides a summary of the neutron source representation used in the GGNS model along with a description of the key model features.

2.1 Source Representation

During reactor operation, the neutron flux level at any point in the shroud or vessel will vary due to changes in fuel composition, power distributions within the core, and water void fraction. These changes occur between fuel cycles due to changes in fuel loading and fuel design, and within a fuel cycle due to fuel and poison burnup and resultant changes in power shape, control rod position, fission contributions by nuclide, and void fraction vs. axial height in each fuel bundle. Power shape throughout a typical cycle's worth of operation has similar characteristics from cycle-to-cycle. Power starts out being preferentially produced in the bottom half of the core, and, as the fuel cycle progresses, the power peak shifts to higher core locations. The fuel burnup decreases the fraction of fissions coming from U235 and increases the fraction of fissions from plutonium isotopes. This results in slight changes in the fission spectrum and in the number of fissions per unit power. The control rod patterns are altered at several discreet time intervals throughout the cycle. These sequence exchanges (SE) result in step changes in the power distribution and in the distribution of reactor water densities, especially in local areas near the control rods. [[

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2.2 Neutron Transport Model

The transport calculations for GGNS were carried out in 2Dand 3D geometries using the DORT two-dimensional and the TORT three-dimensional discrete ordinates codes [2-13], respectively. [[ ]] The DORT code is an update of the DOT code which has been in use for this type of problem for many years. The TORT code has the same calculational methodology as DORT, except that it extends the calculation into the third dimension. [[

]] The energy group boundaries for the 47 groups are given in Table 2-1. [[

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The computer codes were obtained from the Radiation Safety Information Computational Center (RSICC) at Oak Ridge National Laboratory. Each code was then compiled on the computer used by MPM for the calculations and a series of test cases were run to verify the code performance. The test cases all agreed within allowable tolerance with established results. This verification was conducted under the MPM Nuclear Quality Assurance Program. The calculational procedures meet standards specified by the NRC and ASTM as appropriate. In particular, the analysis (including all modeling details and cross-sections) is consistent with RG 1.190 [2-16], and the 3D calculational methodology has been benchmarked to measured plant-specific BWR data as described in Section 3. [[

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2D R-θ Calculations

The R-θ layout at core axial midplane is shown schematically in Figure 2-1. Some of the structures in this figure (the surveillance capsule and jet pumps) are not to scale. A more detailed listing of dimensions for the various structures is given in Tables 2-2 through 2-4. Dimensions were obtained from plant drawings and the GGNS design inputs which are referenced in the tablesthat were transmitted to MPM by Reference [2-19]. As shown in Figure 2-1, all structures outside the core were modeled with a cylindrical symmetry except for the inclusion of a surveillance capsule centered at 3 degrees and jet pump structures located in the downcomer region. [[

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2D R-Z Calculations

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Flux Synthesis

[[ ]] In order to estimate the fluence rate in the three dimensional geometry, the following standard synthesis equation was used to evaluate the flux (φ) [[ ]]

$$\varphi(R,\theta,Z) = \varphi(R,\theta) * \varphi(R,Z) / \varphi(R)$$

[[

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3D TORT Calculations

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2.3 Section 2 References

[2-1] [[

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.[2-2] RSICC Peripheral Shielding Routine Code Collection, PSR-277, LEPRICON, PWR Pressure Vessel Surveillance Dosimetry Analysis System, available from the Radiation Safety Information Computational Center, Oak Ridge National Laboratory, Oak Ridge, TN, June 1995.

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[2-13] RSICC Computer Code Collection, CCC-543, TORT-DORT-PC, Two- and Three-Dimensional Discrete Ordinates Transport Version 2.7.3, available from the Radiation Safety Information Computational Center, Oak Ridge National Laboratory, Oak Ridge, TN, June 1996.

[2-14] [[

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[2-15] ASTM Designation E482-89, Standard Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance, in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1997.

[2-16] Regulatory Guide 1.190, Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence, U. S. Nuclear Regulatory Commission, March 2001.

[2-17] [[

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[2-18] [[

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[2-19] [[

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Table 2-1Neutron Energy Group Structure used in the GGNS Transport Calculations – 47 Groups.

Energy Group	Upper Energy (MeV)	Energy Group	Upper Energy (MeV)
1	1.733E+01	25	2.972E-01
2	1.419E+01	26	1.832E-01
3	1.221E+01	27	1.111E-01
4	1.000E+01	28	6.738E-02
5	8.607E+00	29	4.087E-02
6	7.408E+00	30	3.183E-02
7	6.065E+00	31	2.606E-02
8	4.966E+00	32	2.418E-02
9	3.679E+00	33	2.188E-02
10	3.012E+00	34	1.503E-02
11	2.725E+00	35	7.102E-03
12	2.466E+00	36	3.355E-03
13	2.365E+00	37	1.585E-03
14	2.346E+00	38	4.540E-04
15	2.231E+00	39	2.145E-04
16	1.920E+00	40	1.013E-04
17	1.653E+00	41	3.727E-05
18	1.353E+00	42	1.068E-05
19	1.003E+00	43	5.044E-06
20	8.208E-01	44	1.855E-06
21	7.427E-01	45	8.764E-07
22	6.081E-01	46	4.140E-07
23	4.979E-01	47	1.000E-07
24	3.688E-01		

Table 2-2 GGNS Reactor Component Radial Dimensions.

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Table 2-2 GGNS Reactor Component Radial Dimensions (continued).

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Table 2-2GGNS Reactor Component Radial Dimensions (continued).

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Table 2-3GGNS Reactor Component Azimuthal Locations.

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Table 2-4GGNS Reactor Component Axial Dimensions.

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Figure 2-1GGNS R-θ Geometry used in the TORT Calculations.

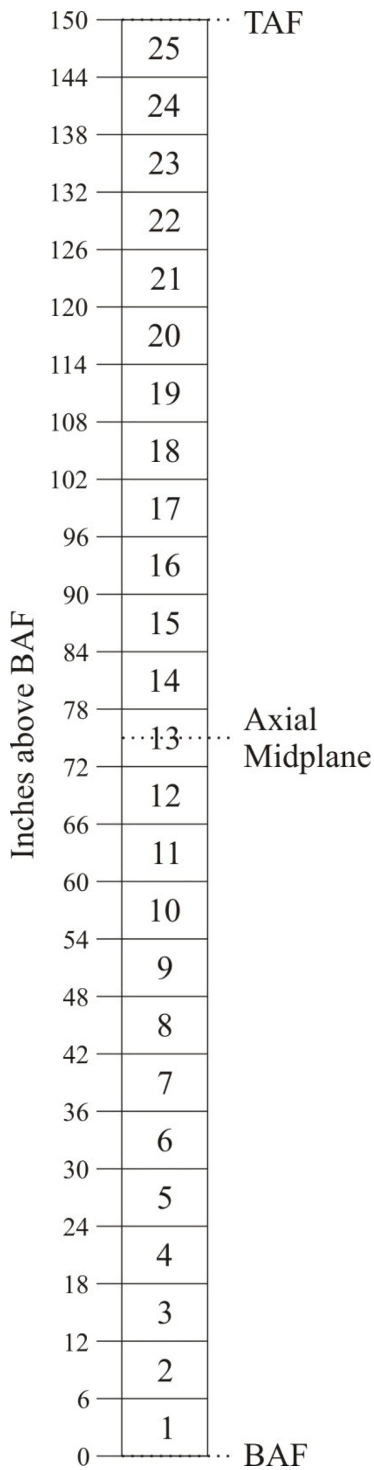

Figure 2-2 Diagram Showing Locations of Fuel Nodes versus Axial Height Relative to BAF.

3
BENCHMARK ANALYSES

NRC Regulatory Guide 1.190 requires that neutron transport methods satisfy compliance requirements and they must also demonstrate specified accuracy through benchmarking. This section of the report summarizes the work done to demonstrate that the MPM methods meet the NRC's requirements.

3.1 Compliance with RG 1.190

The United States Nuclear Regulatory Commission has issued RG 1.190 entitled, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [3-1]. This guide covers recommended practices for neutron transport calculations and applies to other reactor components in addition to the primary emphasis on the pressure vessel. The regulatory positions in the guide that pertain to calculational methodology are summarized in Table 3-1 which is taken directly from RG 1.190. The table references paragraphs in the guide that give more detailed information on each position. The compliance of the GGNS fluence calculations with the guide is summarized below.

<u>Fluence Calculational Methods</u>

Fluence Determination: This calculation was performed using an absolute fluence calculation.
Meets guide requirement.

Modeling Data: The MPM methodology is based on documented and verified plant-specific data. Further, the calculations use as-built data for plant structures and material compositions whenever these data are available. The fuel data is specific for each fuel cycle and includes results for power distributions and water densities taken from the fuel depletion analysis.
Meets guide requirement.

Nuclear Data: The calculations used the [[ ]]cross section set that is based on the latest version (VI) of the Evaluated Nuclear Data File (ENDF/B). The [[ ]] set has undergone extensive testing and benchmarking to ensure its validity for light water reactor (LWR) calculations.
Meets guide requirement.

Cross-Section Angular Representation: The calculations use a P3 angular decomposition in accordance with the guide.
Meets guide requirement.

Cross-Section Group Collapsing: All calculations are performed with the [[ ]] library which is collapsed to 47 neutron groups. Benchmarking has shown that the 47-group structure is adequate for LWR neutron transport calculations.
Meets guide requirement.

Neutron Source: The neutron source is calculated taking into account changes in neutrons per fission, energy per fission, and the average fission spectrum which develops as the U235 is

burned and other isotopes, such as Pu239, increase in fission fraction.
Meets guide requirement.

End-of-Life Predictions: The fluence has been calculated for every cycle up to the present. [[

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Meets guide requirement.

Spatial Representation: The present methodology meets or exceeds these requirements. [[

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Meets guide requirement.

Multiple Transport Calculations: It was not necessary to use bootstrapping for these calculations so this requirement does not apply.

Point Estimates: This requirement only applies to Monte Carlo calculations which are not used here.

Statistical Tests: This requirement only applies to Monte Carlo calculations which are not used here.

Variance Reduction: This requirement only applies to Monte Carlo calculations which are not used here.

Capsule Modeling: The capsule geometry is modeled using as-built drawings. [[

]].
Meets guide requirement.

Spectral Effects on RT_{NDT}: This requirement only applies to extrapolation through the vessel and does not affect the benchmark calculations. However, when fluence within the vessel is required, the displacement per atom (dpa) extrapolation methodology is applied to vessel calculations as specified in RG 1.99, Revision 2. Data are supplied to enable extrapolation using dpa calculated extrapolation or using RG 1.99 Rev. 2 extrapolation to account for the spectral shift.
Meets guide requirement.

Cavity Calculations: Cavity calculations are not important at GGNS, and no cavity dosimetry work has been performed at GGNS.

Methods Qualification: Methods qualification for these calculations is discussed in detail in Sections 3.2 and 3.3, which deals with benchmarking of the methodology. A complete analytical uncertainty analysis is described in Section 6, and this analysis was carried out in accordance with the guide. In summary, 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 ($\pm$ 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%.
Meets guide requirement.

Fluence Calculational Uncertainty: An extensive evaluation of all contributors to the uncertainty in the calculated fluence was made for the BWR plant calculations performed to date. This evaluation indicated that the uncertainty in calculated fluences in the reactor beltline region is below 20% as specified in the guide. In addition, the comparisons with measurements indicate agreement well within the 20% limit. The agreement of calculations with measurements to within ±20% uncertainty indicates that the MPM calculations can be applied for fluence determination with no bias.
Meets guide requirement.

Fluence Measurement Methods

The regulatory positions in the guide that pertain to fluence measurement methods are summarized in Table 3-2.The compliance of the GGNS fluence measurements with the guide is summarized below.

Spectrum Coverage: GGNS does not have dosimetry sets installed to provide spectrum definition. This is in common with GE BWRs which have only limited dosimetry installed in surveillance capsules. The GGNS dosimetry analyzed to date is from iron wires attached to a surveillance capsule and removed after the first cycle of operation. Calculated neutron spectra are validated using the detailed measurements for test reactors and the calculational benchmark included in RG 1.190. Results are documented in Reference [3-2].

Dosimeter Nuclear and Material Properties: The GGNS dosimetry wire material characterization was performed by GE (Reference [3-3]). All nuclear constants and parameters used in the dosimetry counting and analysis follow ASTM standard procedures and use validated nuclear constants.
Meets guide requirement.

Corrections: [[

]]

Meets guide requirement.

Response Uncertainty: Uncertainty analyses for each dosimeter have been reported in all of the surveillance capsule reports performed by MPM. [[

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Meets guide requirement.

Validation: [[

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Fast-Neutron Fluence: M/C ratios are determined for each dosimeter measurement using calculated detector responses (Reference [3-2]).

Meets guide requirement.

Measurement-to-Calculated Ratios: The M/C ratios, standard deviations, and comparisons between calculation and measurement have been done and have been reported as discussed earlier (References [3-2]).

Meets guide requirement.

<u>Reporting Provisions</u>

The regulatory positions in the guide that pertain to neutronfluence and uncertainty reporting are summarized in Table 3-3.The compliance of the GGNS reporting with the guide is summarized below.

Neutron Fluence and Uncertainties: This report on the GGNS calculations, as well as the benchmark report (Reference [3-2]), meets all RG 1.190 reporting requirements.
Meets guide requirement.

Multigroup Fluences: Because of the extensive amount of data, only the fluence distributions for neutrons with energy greater than 1 MeV are reported, except for the vessel where neutrons

with energy greater than 0.1 MeV and dpa are also reported. The multigroup fluence rates are permanently stored for future access if required. These are not seen as having any application except for possible future dosimetry analysis. Multigroup flux values are readily available for all dosimetry locations.
Meets guide requirement.

Bias Reporting: No bias is observed and none is applied.
Meets guide requirement.

Integral Fluences and Uncertainties: These are all reported.
Meets guide requirement.

Comparisons of Calculation and Measurement: These are reported for all dosimetry locations.
Meets guide requirement.

Standard Neutron Field Validation: Not applicable.

Specific Activities and Average Reaction Rates: This is contained in all reports with measured dosimetry data.
Meets guide requirement.

Corrections and Adjustments to Measured Quantities: No corrections made.
Meets guide requirement.

3.2 Summary of MPM Methods

The neutron exposure of reactor structures to determine the spatial distribution of neutrons in the reactor components is determined by either a 3D neutron transport calculation, or by a combination of R-θ, R-Z, and R neutron transport calculations in support of the 2D synthesis method. In particular, the calculation determines the distribution of neutrons of all energies from their source from fission in the core region to their eventual absorption or leakage from the system. The transport calculations use a model of the reactor geometry that includes the significant structures and geometrical details necessary to accurately define the neutron environment at all locations of interest. A brief summary of the MPM methods is included here.

[[

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Whenever possible, the plant geometry is modeled using as-built drawings. In cases where as-built drawings are not available, design drawings are used. During reactor operation, the neutron flux level at any point in the shroud or vessel will vary due to changes in fuel composition, power distributions within the core, and water void fraction. These changes occur between fuel cycles due to changes in fuel loading and fuel design, and within a fuel cycle due to fuel and poison burnup and resultant changes in power shape, control rod position, fission contributions by nuclide, and void fraction vs. axial height in each fuel bundle. To achieve the highest accuracy, changes in the three-dimensional bundle power distributions and water density distributions must be appropriately modeled. [[

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3.3 Benchmarking MPM Methodology

Although RG 1.190 was specifically developed to address calculation of fluence to the vessel, the guide can be considered to apply to other reactor components such as the shroud, surveillance capsules, and internal structures. As mentioned, MPM methods include the standard 2D synthesis, [[ ]] and 3D TORT.In order to meet the methods qualification requirement of RG 1.190, the MPM calculational methodology has been validated by comparison with measurement and calculational benchmarks. Comparisons of calculations with measurements have been made for the Pool Critical Assembly (PCA) pressure vessel simulator benchmark, [[ ]] Comparisons with a BWR calculational benchmark have also been completed.

The qualification of the methods used for the reactor transport calculations can be divided into two parts. The first part is the qualification of the cross-section library and calculational methods that are used to calculate the neutron transport. The second part is the validation of the method by comparison of calculations with dosimetry measurements for both representative plants and also for the current plant being analyzed, which, in the current case, is the GGNS plant. Details of the extensive benchmarking efforts performed by MPM are contained in a separate document [3-2].

Measurement and Calculational Benchmarks

The qualification of the cross-section library and calculational methods is particularly important for vessel fluence calculations because of the large amount of neutron attenuation between the source in the core and the vessel. The cross-sections are first developed as an evaluated file that details all of the reactions as a continuous function of energy. The ENDF/B evaluators take into account various measurements, including integral measurements (such as criticality and dosimetry measurements) that provide a test of the adequacy of the evaluation. For transport calculations using discrete ordinates, the ENDF/B cross-section files must be collapsed into multigroup files. This is done in two steps. First, fine-group cross-sections are calculated (Vitamin B6 library). [[ ]]

contains cross-sections collapsed using a BWR core spectrum, a PWR core spectrum, a PWR downcomer spectrum, a PWR vessel spectrum, and a concrete shield spectrum. These various cross-section libraries are then tested against various benchmarks and compared with measured results and results calculated using the fine-group cross-sections [3-7]. [[

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]] A particularly appropriate benchmark for geometry outside the core which includes the vessel, is the PCA benchmark [3-9]. This benchmark provides validation of the transport through typical reactor structures and a simulated reactor vessel in a simple geometry. The PCA has high-accuracy measurement results extending from inside a simulated thermal shield through to the outside of a simulated vessel. The PCA benchmark was calculated using the MPM methodology and detailed results are reported in [3-2]. [[

]]

Another benchmarking requirement of RG 1.190 is to compare with a suitable calculational benchmark. The calculational benchmarks used to satisfy this requirement are documented in Reference [3-11]. The benchmark problems include 3 different PWR geometries and a single BWR problem. It is intended that the analyst select the benchmark problem or problems appropriate to the plant being analyzed. [[

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Benchmarking MPM Calculations against Measured Data in Operating BWRs

The second part of benchmarking is to compare calculations with measurements in a geometry as close as possible to that which is being analyzed. Comparisons were made with BWR surveillance dosimetry measurements [[

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Calculated activation results from each dosimeter type all fell within ±20% of the measurement. The dosimetry results have been averaged for each set and the results are shown in Table 3-6. If the average results from the nine dosimetry sets are averaged, the average calculated-to-measured (C/M) ratio [[ ]] which indicates that the MPM methodology does not exhibit any consistent bias for BWR calculations. This meets the criterion set by RG 1.190 for acceptability of the calculations.

Benchmarking 3D TORT Methods

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]] very good agreement between the 3D calculation and the measurement,

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3.4 Conclusions

In summary, it is concluded that the RG 1.190 requirement for qualification of the MPM methods for BWR neutron transport analyses by comparisons to measurement and calculational benchmarks has been fully satisfied. Moreover, the agreement of calculations with measurements to within ±20% uncertainty indicates that the MPM calculations can be applied for fluence determination with no bias.

3.5 Section 3 References

[3-1] Regulatory Guide 1.190, Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence, U. S. Nuclear Regulatory Commission, March 2001.

[3-2] [[

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[3-3] [[

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[3-4] W. N. McElroy, “Data Development and Testing for Fast Reactor Dosimetry “, Nucl. Tech. 25, 177 (1975).

[3-5] R. Gold and W. N. McElroy, "The Light Water Reactor Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP): Past Accomplishments, Recent Developments, and Future Directions," Reactor Dosimetry: Methods, Applications, and Standardization, ASTM STP 1001, Harry Farrar and E. P. Lippincott, eds., American Society for Testing and Materials, Philadelphia, 1989, pp 44-61.

[3-6] [[

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[3-9] Remec, I. and Kam, F.B.K., Pool Critical Assembly Pressure Vessel Facility Benchmark, NUREG/CR-6454, (ORNL/TM-13205), USNRC, July 1997.

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[3-11] Carew, J. F., Hu, K., Aronson, A., Prince, A., and Zamonsky, G., PWR and BWR Pressure Vessel Fluence Calculational Benchmark Problems and Solutions, NUREG/CR-6115 (BNL-NUREG-52395), September, 2001.

Table 3-1 Summary of Regulatory Guide 1.190 Positions on Fluence Calculation Methods.

Regulatory Position – Fluence Calculation Methods	Regulatory Section
Fluence Determination. Absolute fluence calculations, rather than extrapolated fluence measurements, must be used for the fluence determination.	1.3
Modeling Data. The calculation modeling (geometry, materials, etc.) should be based on documented and verified plant-specific data.	1.1.1
Nuclear Data. The latest version of the Evaluated Nuclear Data File (ENDF/B) should be used for determining nuclear cross- sections. Cross-section sets based on earlier or equivalent nuclear-data sets that have been thoroughly benchmarked are also acceptable. When the recommended cross-section data change, the effect of these changes on the licensee-specific methodology must be evaluated and the fluence estimates updated when the effects are significant.	1.1.2
Cross-Section Angular Representation. In discrete ordinates transport calculations, a P_3 angular decomposition of the scattering cross-sections (at a minimum) must be employed.	1.1.2
Cross-Section Group Collapsing. The adequacy of the collapsed job library must be demonstrated by comparing calculations for a representative configuration performed with both the master library and the job library.	1.1.2
Neutron Source. The core neutron source should account for local fuel isotopics and, where appropriate, moderator density. The neutron source normalization and energy dependence must account for the fuel exposure dependence of the fission spectra, the number of neutrons produced per fission, and the energy released per fission.	1.2
End-of-Life Predictions. Predictions of the vessel end-of-life fluence should be made with a best-estimate or conservative generic power distribution. If a best estimate is used, the power distribution must be updated if changes in core loadings, surveillance measurements, or other information indicate a significant change in projected fluence values.	1.2
Spatial Representation. Discrete ordinates neutron transport calculations should incorporate a detailed radial- and azimuthal-spatial mesh of ~2 intervals per inch radially. The discrete ordinates calculations must employ (at a minimum) an S_8 quadrature and (at least) 40-80 intervals per octant.	1.3.1
Multiple Transport Calculations. If the calculation is performed using two or more "bootstrap" calculations, the adequacy of the overlap regions must be demonstrated.	1.3.1

Table 3-1 Summary of Regulatory Guide 1.190 Positions on Fluence Calculation Methods (continued).

Regulatory Position – Fluence Calculation Methods	Regulatory Section
Point Estimates. If the dimensions of the tally region or the definition of the average-flux region introduce a bias in the talley edit, the Monte Carlo prediction should be adjusted to eliminate the calculational bias. The average-flux region surrounding the point location should not include material boundaries or be located near reflecting, periodic or white boundaries.	1.3.2
Statistical Tests. The Monte Carlo estimated mean and relative error should be tested and satisfy all statistical criteria.	1.3.2
Variance Reduction. All variance reduction methods should be qualified by comparison with calculations performed without variance reduction.	1.3.2
Capsule Modeling. The capsule fluence is extremely sensitive to the geometrical representation of the capsule geometry and internal water region, and the adequacy of the capsule representation and mesh must be demonstrated.	1.3.3
Spectral Effects on RT_{NDT}. In order to account for the neutron spectrum dependence of RT_{NDT}, when it is extrapolated from the inside surface of the pressure vessel to the T/4 and 3T/4 vessel locations using the > 1-MeV fluence, a spectral lead factor must be applied to the fluence for the calculation of ΔRT_{NDT}.	1.3.3
Cavity Calculations. In discrete ordinates transport-calculations, the adequacy of the S_8 angular quadrature used in cavity transport calculations must be demonstrated.	1.3.5
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.	1.4.1, 1.4.2, 1.4.3
Fluence Calculational Uncertainty. The vessel fluence (1 sigma) calculational uncertainty must be demonstrated to be 20% for RT_{PTS} and RT_{NDT} 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.	1, 1.4.3

Table 3-2 Summary of Regulatory Guide 1.190 Positions on Fluence Measurement Methods.

Regulatory Position - Fluence Measurement Methods	Regulatory Section
Spectrum Coverage. The set of dosimeters should provide adequate spectrum coverage.	2.1.1
Dosimeter Nuclear and Material Properties. Use of dosimeter materials should address melting, oxidation, material purity, total and isotopic mass assay, perturbations by encapsulations and thermal shields, and accurate dosimeter positioning. Dosimeter half-life and photon yield and interference should also be evaluated.	2.1.1
Corrections. Dosimeter-response measurements should account for fluence rate variations, isotopic burnup effects, detector perturbations, self-shielding, reaction interferences, and photofission.	2.1.2
Response Uncertainty. An uncertainty analysis must be performed for the response of each dosimeter.	2.1.3
Validation. Detector-response calibrations must be carried out periodically in a standard neutron field.	2.2
Fast-Neutron Fluence. The E > 1 MeV fast-neutron fluence for each measurement location must be determined using calculated spectrum-averaged cross-sections and individual detector measurements. As an alternative, the detector responses may be used to determine reaction probabilities or average reaction rates.	2.3
Measurement-to-Calculation Ratios. The M/C ratios, the standard deviation and bias between calculation and measurement, must be determined.	2.3

Table 3-3 Summary of Regulatory Guide 1.190 Positions on Fluence and Uncertainty Reporting.

Regulatory Position - Reporting Provisions	Regulatory Section
Neutron Fluence and Uncertainties. Details of the absolute fluence calculations, associated methods qualification and fluence adjustments (if any) should be reported. Justification and a description of any deviations from the provisions of this guide should be provided.	3.1
Multigroup Fluences. Calculated multigroup neutron fluences and fluence rates should be reported.	3.2
Bias Reporting. The value and basis of any bias or model adjustment made to improve the measurement-to-calculation agreement must be reported.	3.2
Integral Fluences and Uncertainties. Calculated integral fluences and fluence rates for E > 1 MeV and their uncertainties should be reported.	3.3
Comparisons of Calculation and Measurement. Measured and calculated integral E > 1 MeV fluences or reaction rates and uncertainties for each measurement location should be reported. The M/C ratios and spectrum averaged cross-section should also be reported.	3.4
Standard Neutron Field Validation. The results of the standard field validation of the measurement method should be reported.	3.5
Specific Activities and Average Reaction Rates. The specific activities at the end of irradiation and measured average reaction rates with uncertainties should be reported.	3.5
Corrections and Adjustments to Measured Quantities. All corrections and adjustments to the measured quantities and their justification should be reported.	3.5

Table 3-4 Comparison of Calculated and Measured Results for PCA.

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Non-Proprietary

Table 3-5 Comparison of Calculated and Benchmark Results for the Reactor Vessel Calculated at Reactor Axial Peak.

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Table 3-6 Tabulation of Dosimetry Results for Operating BWR Plants.

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4
SHROUD AND TOP GUIDE FLUENCE RESULTS

An important concern for many BWR reactors is the neutron exposure of the shroud and top guide welds in these components. Because the shroud is located close to the fuel region, the exposure is relatively high, and this must be taken into account to evaluate the possibility of stress corrosion cracks and crack growth. Of special importance is determining weld locations that exceed the irradiation assisted stress corrosion cracking (IASCC) fast (E>1.0 MeV) neutron threshold fluence. A fluence of 5E+20 n/cm^2 has been proposed as a screening threshold, but more recent data suggest a lower threshold for some materials. In any event, radiation damage effects on the SCC crack growth model must be considered in setting future inspection intervals.

4.1 Shroud Weld Fluence Results

Locations of all the shroud vertical and horizontal welds are shown schematically in Figures 1-1 and 1-2, and the weld locations are listed in Tables4-1 and 4-2. Evaluations of the fluence for all of these welds have been performed using the 3D TORT method described in Section 2. Tables 4-3 and 4-4 summarize the calculated maximum fluences to the shroud and top guide welds. These fluences were calculated at the inner diameter (ID) surface of the welds at the point along the welds where the fluence is a maximum. [[

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4.2 Comparison with Past Results

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4.3 Section4 References

[4-1] [[

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[4-2] [[

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Non-Proprietary

Table 4-1 Locations for Fluence Evaluation in the Shroud and Top Guide Horizontal Welds.

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Table 4-2 Locations for Fluence Evaluation in the Shroud and Top Guide Vertical Welds.

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Non-Proprietary

Table 4-3 Estimated Maximum Fluence to GGNS Shroud and Top Guide Horizontal Welds.

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Non-Proprietary

Table 4-4 Estimated Maximum Fluence to GGNS Shroud and Top Guide Vertical Welds.

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Non-Proprietary

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Figure 4-1Fast Flux (E>1.0 MeV) at the Shroud IR to Weld H4 for Cycles 1 through 21.

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Figure 4-2 Fast Fluence (E>1.0 MeV) to Shroud Weld H4 at the End of Cycle 21 (28.088 EFPY).

Non-Proprietary

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Figure 4-3 Fast Fluence (E>1.0 MeV) to Shroud Weld H4 at 35 EFPY.

Non-Proprietary

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Figure 4-4 Fast Fluence (E>1.0 MeV) to Shroud Weld H4 at 54 EFPY.

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Figure 4-5 Fast Fluence (E>1.0 MeV) to Shroud Vertical Welds V13-V16 at the End of Cycle 21 (28.088 EFPY).

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Figure 4-6 Fast Fluence (E>1.0 MeV) to Shroud Vertical Welds V13-V16 at 35 EFPY.

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Figure 4-7 Fast Fluence (E>1.0 MeV) to Shroud Vertical Welds V13-V16 at 54 EFPY.

5
VESSEL AND CAPSULE FLUENCE RESULTS

5.1 Cycle 1 Dosimetry Analysis

The GGNS cycle 1 dosimetry was originally analyzed using 2D synthesis [5-1]. The past work has been reanalyzed using 3D TORT calculations. [[

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The detailed power history for GGNS cycle 1 operation is presented in Table 5-3 [5-2]. [[

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The removable dosimetry packet consisted of 3 iron wires. The dosimetry packet was located on the side of the capsule. The location of the dosimetry was provided in the plant documentation as indicated in Table 2-3. [[

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5.2 Pressure Vessel Analysis

The analysis of the vessel was carried out using the 3D methodology described in Section 2. The maximum flux location at the vessel shell course plates varies slightly from cycle-to-cycle as a result of the different fuel loadings. Therefore, the maximum fluence for the shell course plates was conservatively calculated [[

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Plots of the azimuthal and axial traverses of fluence at the vessel wetted surface at the end of cycle 21, along with the results at the vessel 1/4T and 3/4T positions, are shown in Figures 5-1 and 5-2, respectively. [[

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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." [[

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As shown in Figure 5-3, the reactor vessel is made up of plates that are welded together. For the purpose of PT curve calculations, fluences have been evaluated for shell 1, shell 2, and shell 3 materials. [[

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Radiation embrittlement effects are often correlated with fast fluence (E > 1 MeV). However, it is generally thought that dpa might be a better correlation parameter since it accounts for spectral effects and, if this is correct, the use of the fast fluence (E > 1 MeV) values within the vessel might under predict the radiation damage at locations within the vessel. Therefore, the NRC requires the use of a dpa attenuation in the vessel for preparation of pressure-temperature operating curves. The fluence attenuation factors within the vessel can be evaluated using calculated dpa attenuation from Table 5-9 or using the dpa formulation specified in the RG 1.99 (Rev 2) [5-6]. The fluence values using both these attenuation methods are given in Table 5-18[[

]]This evaluation of the dpa cross section is based on the ENDF-IV cross section file.[[

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The vessel has nozzle penetrations at several locations, and neutron exposure at the nozzles is of concern for neutron damage analysis. Four sets of nozzles were evaluated, and the locations of these nozzles along with the fluence evaluation points are summarized in Table 5-19. As shown in the table, the nozzle fluences are determined at the maximum point in the nozzle weld as well as at the conservative locations specified in Reference [5-9]. For nozzles below the core, the maximum fluence point occurs at the top of the nozzle. The reverse is true for nozzles above the core. The maximum fluence values at the nozzles are summarized in Table 5-20. Fluences at conservative locations near the nozzles specified by GGNS in Reference [5-9] are given in Tables 5-21 and 5-22. The data in Table 5-21 is at the elevation specified, but at the maximum azimuthally, whereas the data in Table 5-22 is at the locations specified in [5-9]. [[

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Section 6 presents the N6 and N12 maximum nozzle fluence uncertainties. The N12 nozzle elevation is close to that of the active fuel, [[

]] Regulatory Guide 1.190 states that if the analytical uncertainty at the 1 sigma level is greater than 30%, the methodology of the regulatory guide is not applicable and the plant application will be reviewed on an individual basis. [[

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Non-Proprietary

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5.3 Comparison with Past Results

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5.4 Section5 References

[5-1] [[

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[5-2] [[

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[5-3] ASTM Designation E263-00, Standard Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Iron, in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 2003.

[5-4] ASTM Designation E1005-03, Standard Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance, E706(IIIA), in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 2003.

[5-5] [[

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[5-6] Regulatory Guide 1.99, Revision 2, Radiation Embrittlement of Reactor Vessel Materials, U. S. Nuclear Regulatory Commission, May 1988.

[5-7] ASTM Designation E693-94, Standard Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA), E706(ID), in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 2000.

[5-8] ASTM Designation E693-01, Standard Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA), E706(ID), in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 2003.

[5-9] [[

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Table 5-1Calculated Fe(n,p) Reaction Rates at the Cycle 1 Dosimetry Location.

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Table 5-2Flux Spectrum at Dosimetry Location.

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Table 5-3 Grand Gulf Cycle 1 Power History.

Date	Days In Period	Cycle Cumulative (MWd/MTU)	Period Effective Full Power Days	Period Average Fraction Of Full Power
30-Sep-84	Begin	0		
3-Nov-84	35	178	6.789	0.1940
14-Nov-84	11	235	2.174	0.1976
21-Nov-84	7	295	2.288	0.3269
4-Feb-85	75	618	12.320	0.1643
12-Apr-85	67	906	10.985	0.1640
25-Apr-85	13	1071	6.293	0.4841
17-May-85	22	1384	11.938	0.5427
4-Jun-85	18	1666	10.756	0.5976
3-Jul-85	29	2140	18.079	0.6234
15-Jul-85	12	2351	8.048	0.6707
26-Jul-85	11	2596	9.345	0.8495
7-Aug-85	12	2863	10.184	0.8487
21-Aug-85	14	3158	11.252	0.8037
23-Aug-85	2	3187	1.106	0.5531
28-Aug-85	5	3269	3.128	0.6255
5-Sep-85	8	3466	7.514	0.9392
10-Sep-85	5	3591	4.768	0.9535
19-Sep-85	9	3808	8.277	0.9196
22-Sep-85	3	3865	2.174	0.7247
26-Sep-85	4	3914	1.869	0.4672
30-Sep-85	4	4026	4.272	1.0680
12-Oct-85	12	4334	11.748	0.9790
15-Dec-85	64	4360	0.992	0.0155
18-Dec-85	3	4459	3.776	1.2587
28-Dec-85	10	4586	4.844	0.4844
9-Jan-86	12	4861	10.489	0.8741
11-Jan-86	2	4891	1.144	0.5721
31-Jan-86	20	5177	10.909	0.5454
12-Feb-86	12	5376	7.590	0.6325
25-Feb-86	13	5498	4.653	0.3579
28-Feb-86	3	5551	2.022	0.6738
15-Mar-86	15	5812	9.955	0.6637
18-Apr-86	34	6250	16.706	0.4914
28-Apr-86	10	6419	6.446	0.6446

Non-Proprietary

Table 5-3 Grand Gulf Cycle 1 Power History (continued).

Date	Days in Period	Cycle Cumulative (MWd/MTU)	Period Effective Full Power Days	Period Average Fraction of Full Power
9-May-86	11	6612	7.361	0.6692
25-May-86	16	6907	11.252	0.7032
28-May-86	3	6964	2.174	0.7247
13-Jun-86	16	7286	12.282	0.7676
27-Jun-86	14	7615	12.549	0.8963
3-Jul-86	6	7738	4.691	0.7819
18-Jul-86	15	8038	11.442	0.7628
30-Jul-86	12	8229	7.285	0.6071
7-Aug-86	8	8389	6.103	0.7628
18-Aug-86	11	8586	7.514	0.6831
23-Aug-86	5	8701	4.386	0.8773
3-Sep-86	11	8782	3.089	0.2809
5-Sep-86	2	8823	1.564	0.7819

Table 5-4Nuclear Parameters Used in the Evaluation of Neutron Sensors.

Monitor Material	Reaction of Interest	Isotopic Fraction	Approximate Response Threshold	Product Half-Life
Iron	$Fe^{54}(n,p)Mn^{54}$	0.05845	2 MeV	312.3 days

Table 5-5Tabulation of Dosimetry Results.

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Non-Proprietary

Table 5-6 Azimuthal Variation of Maximum Fluence at the Vessel Wetted Surface.

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Table 5-6Azimuthal Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Non-Proprietary

Table 5-6 Azimuthal Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Non-Proprietary

Table 5-6 Azimuthal Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Non-Proprietary

Table 5-6 Azimuthal Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Table 5-6 Azimuthal Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Table 5-6 Azimuthal Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Table 5-7 Axial Variation of Maximum Fluence at the Vessel Wetted Surface.

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Non-Proprietary

Table 5-7 Axial Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Non-Proprietary

Table 5-7 Axial Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Table 5-7 Axial Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Table 5-7 Axial Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Non-Proprietary

Table 5-7 Axial Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Table 5-7 Axial Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Table 5-7 Axial Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Table 5-7Axial Variation of Maximum Fluence at the Vessel Wetted Surface (continued).

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Non-Proprietary

Table 5-8 Relative Exposure through Vessel at Maximum Fluence Point at the End of Cycle 21 (28.088 EFPY).

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Non-Proprietary

Table 5-9 Calculated Maximum Vessel Exposure (Shell 2) at the End of Cycle 21 (28.088 EFPY) and Projected to Future Exposures.

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Non-Proprietary

Table 5-10 Azimuthal Location Ranges of Plates in Vessel Shell 1, Shell2, and Shell 3.

Plate ID	Azimuthal Angle Range (degrees)	Equivalent Azimuthal Angle Range in First Quadrant (degrees)
Shell 1		
PC MK 21-1-1 Plate Heat A1113	0-120	0-90
PC MK 21-1-2 Plate Heat C2557	120-240	0-60
PC MK 21-1-3 Plate Heat C2506	240-360	0-90
Shell 2		
PC MK 22-1-1 Plate Heat C2593	322-52	0-52
PC MK 22-1-2 Plate Heat C2594	52-142	38-90
PC MK 22-1-3 Plate Heat C2594	142-232	0-52
PC MK 22-1-4 Plate HeatA1224	232-322	38-90
Shell 3		
PC MK 23-1-1 Plate Heat C2741	349-109	0-90
PC MK 23-1-2 Plate Heat C2779	109-229	0-71
PC MK 23-1-3 Plate Heat C2741	229-349	11-90

Table 5-11 Calculated Maximum Vessel Shell 1 Exposure at the End of Cycle 21 (28.088 EFPY) and Projected to Future Exposures.

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Table 5-12 Calculated Maximum Vessel Shell 3 Exposure at the End of Cycle 21 (28.088 EFPY) and Projected to Future Exposures.

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Table 5-13 Azimuthal Locations of Vertical Welds in Vessel Shell 1, Shell 2 and Shell 3.

Weld ID	Azimuthal Angle (degrees)	Equivalent Azimuthal Angle in First Quadrant (degrees)
Shell 1		
BA	0	0
BB	120	60
BC	240	60
Shell 2		
BD	52	52
BE	142	38
BF	232	52
BG	322	38
Shell 3		
BH	109	71
BJ	229	49
BK	349	11

Table 5-14 Calculated Maximum Vessel Shell 1 Weld Exposures at the End of Cycle 21 (28.088 EFPY) and Projected to Future Exposures.

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Non-Proprietary

Table 5-15 Calculated Maximum Vessel Shell 2 Weld Exposures at the End of Cycle 21 (28.088 EFPY) and Projected to Future Exposures.

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Table 5-16 Calculated Maximum Vessel Shell 3 Weld Exposures at the End of Cycle 21 (28.088 EFPY) and Projected to Future Exposures.

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Non-Proprietary

Table 5-17Calculated Maximum Vessel Circumferential Weld Exposures at the End of Cycle 21 (28.088 EFPY) and Projected to Future Exposures.

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Table 5-18 GGNS Calculated Vessel Fluence and Fluence Determined using dpa Attenuation.

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Non-Proprietary

Table 5-19 GGNS RPV Nozzle Locations.

Nozzle	Height Above Vessel Zero to Center of Nozzle (inches)	Height To Center of Nozzle Relative to BAF (inches)	Nozzle Weld OD (inches)	Maximum Nozzle Fluence Point Relative to BAF (inches)	Fluence Location Height Specified in Reference [5-9] Relative to Vessel '0' (inches)	Fluence Location Height Specified in Reference [5-9] Relative to BAF (inches)	Nozzle Center Angles in First Quadrant
N1	172.31	-44.00	24.750	-31.62	199.7	-16.61	0°
N2	179.31	-37.00	13.094	-30.45	197.0	-19.31	26°15', 51°45', 77°15'
N12	366.00	149.69	1.880	148.75	364.7	148.39	15°
N6	419.00	202.69	13.250	196.07	-[a]	-[a]	39°

[a] The desired fluence locations for nozzle N6 were not specified in Reference [5-9].

Table 5-20 GGNS RPV Nozzle Maximum Fluence Values.

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Table 5-21GGNS RPV Nozzle Region Maximum Fluence Values in the First Quadrant at the Axial Locations Specified in Reference [5-9].

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Table 5-22GGNS RPV Nozzle Region Fluence Values at the Axial and Azimuthal Locations Specified in Reference [5-9].

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Non-Proprietary

Table 5-23 GGNS RPV Nozzle N6 Maximum Fluence Values With Margin Term Applied to Account for Above Core Fluence Uncertainty.

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Figure 5-1Azimuthal Variation of Maximum Vessel Fluence (E > 1.0 MeV) at the End of Cycle 21 (28.088 EFPY).

Non-Proprietary

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Figure 5-2 Axial Variation of Peak Fluence (E > 1.0 MeV) in the Vessel at the End of Cycle 21 (28.088 EFPY).

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Figure 5-3 GGNS Vessel Roll Out Drawing Showing Beltline Weld Seam and Plate Locations. The Increase in the Axial Extent of the Beltline will Eventually Require that Radiation Damage Effects for Shell 3 Materials be Included in the PT Curve Analysis.

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Figure 5-4Axial Variation of Peak Fluence (E > 1.0 MeV) in the Vessel at 54 EFPY.

6
UNCERTAINTY ANALYSIS

A comprehensive report has been prepared which documents the benchmarking of MPM transport methods for BWR plant analyses [6-1]. For each plant analyzed, a detailed uncertainty analysis was performed to estimate each source of uncertainty in the calculated fluence values. This analysis made use of defined uncertainties and tolerances where possible, but some of the uncertainty estimates had to be based on estimates derived from data variation, such as the detailed power distribution and void fraction variations within a single cycle. This section of the report summarizes the uncertainty analysis performed for GGNS.

The geometry uncertainty assignments are from plant drawings referenced in Tables 2-2 through 2-4, and from generic assessments for cases where dimensional uncertainties are not available. Discussion of each uncertainty assumption is given below. Based on these uncertainty values, detailed evaluations were performed for the shroud, surveillance capsule, reactor pressure vessel, and N6 and N12 vessel nozzles. The uncertainty assessments for TORT results at the shroud, capsule, and vessel over the active fuel elevations are summarized in Table 6-1. The uncertainty assessments for 2D synthesis results at the shroud, capsule, and vessel over the active fuel regions are summarized in Table 6-2. The differences in the uncertainties between TORT and 2D synthesis are in the methods uncertainty as discussed further below. Uncertainty estimates for the shroud and top guide welds are summarized in Tables 6-3 and 6-4. Finally, the vessel nozzle uncertainties are given in Table 6-5.

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6.1 Uncertainty Assumptions

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6.2 2D/3D Uncertainty Evaluation over Active Fuel Length

The results for the uncertainty evaluation over the active fuel length are summarized in Tables6-1 and 6-2, based on the assumptions just discussed. These tables are applicable to the shroud, vessel, and surveillance capsule over the extent of the active fuel [[

]] As previously mentioned, a total uncertainty was derived by combining the independent individual contributors in quadrature.

3D TORT Uncertainty

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6.3 3D Model Uncertainty Evaluation for Shroud, Top Guide Weld, and Nozzle Locations

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6.4 Uncertainty Conclusions

The detailed uncertainty analysis demonstrates that the MPM calculational methods provide fluence results within allowable tolerance bounds (+20%) for the reactor vessel, shroud welds, and surveillance capsules which lie within the axial active fuel region. This satisfies the requirements of RG 1.190. However, RG 1.190 does not address benchmarking outside this region. In the uncertainty analysis, evaluations of structures outside the beltline region have been made, and, especially above the core, [[

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Regulatory Guide 1.190 requires that the overall fluence calculation bias and uncertainty must be determined by an appropriate combination of the analytic uncertainty analysis results and the results of the uncertainty analysis based on the comparisons to theoperating reactor and simulator benchmark measurements. The regulatory guide states that this combination may be a weighted average that accounts for the reliability of the individualestimates. The regulatory guide goes on to state that if the analytical uncertainty at the 1 sigma level is greater than 30%, the methodology of the regulatory guide is not applicable and theapplication will be reviewed on an individual basis. [[

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6.4 Section6 References

[6-1] [[

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[6-2] [[

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[6-3] McElroy, W.N., Ed., "LWR-PV-SDIP: PCA Experiments and Blind Test", NUREG/CR-1861, 1981.

[6-4] [[

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[6-5] Maerker, R.E., et.al., "Application of LEPRICON Methodology to LWR Pressure Vessel Dosimetry", Reactor Dosimetry: Methods, Applications, and Standardization, ASTM STP 1001, 1989, pp 405-414.

[6-6] [[

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[6-7] [[

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[6-8] [[

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[6-9] Fero, A.H., "Neutron and Gamma Ray Flux Calculations for the Venus PWR Engineering Mockup," NUREG/CR-4827 (WCAP-11173), January, 1987.

[6-10] Tsukiyama, T., et. al., "Benchmark Validation of TORT Code Using KKM Measurement and its Application to 800 MWe BWR," Proceedings of the 11th International Symposium on Reactor Dosimetry, Brussels, Belgium, August 18-23, 2002, World Scientific, January 1, 2003.

[6-11] [[

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Table 6-1 Grand Gulf Shroud, Capsule, and Vessel Active Fuel Region TORT Calculational Fluence Uncertainty.

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Table 6-2 Grand Gulf Shroud, Capsule, and Vessel Active Fuel Region 2D Synthesis Calculational Fluence Uncertainty.

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Table 6-3 Estimated Maximum Uncertainty for Shroud and Top Guide Horizontal Welds.

Horizontal Weld	Maximum Fluence Uncertainty (Percent)
H1	[[ ]]
H2	[[ ]]
H3	[[ ]]
H4	[[ ]]
H5	[[ ]]
H6A	[[ ]]
H6B	[[ ]]
H7	[[ ]]

Table 6-4 Estimated Maximum Uncertainty for Shroud and Top Guide Vertical Welds.

Vertical Weld	Maximum Fluence Uncertainty (Percent)
V1 to V4	[[ ]]
V5, V6	[[ ]]
V7,V8	[[ ]]
V9 to V12	[[ ]]
V13,V14	[[ ]]
V15,V16	[[ ]]
V17,V18	[[ ]]
V19 to V22	[[ ]]
V23, V24	[[ ]]

Table 6-5Estimated Maximum Weld IR Fluence Uncertainty for Shroud and Top Guide Vertical Welds.

Vertical Weld	Maximum Uncertainty to Weld IR Fluence (Percent)
V1 to V4	[[ ]]
V5, V6	[[ ]]
V7,V8	[[ ]]
V9 to V12	[[ ]]
V13,V14	[[ ]]
V15,V16	[[ ]]
V17,V18	[[ ]]
V19 to V22	[[ ]]
V23, V24	[[ ]]

Table 6-6 Estimated Maximum Uncertainty forVessel Nozzles N6 and N12.

Nozzle Identification	Maximum Fluence Uncertainty (Percent)
N6	[[ ]]
N12	[[ ]]

7 SUMMARY AND CONCLUSIONS

A detailed three-dimensional transport calculation has been completed for GGNS encompassing operation over the first 20 fuel cycles [[ ]] This work was undertaken to calculate the best estimate neutron fluence, and its uncertainty, to the GGNS reactor pressure vessel, surveillance capsule, core shroud/top guide horizontal and vertical welds, as well as to several below core structures and nozzles. The calculations in the active fuel region were carried out using a three dimensional neutron transport calculation [[

]] Based on the calculations and analyses performed, the following conclusions have been made:

- The transport calculations meet all of the criteria of RG 1.190 for evaluation of fluence to the shroud, surveillance capsule, and reactor vessel within the reactor beltline region. The calculational methodology has been benchmarked to reactor mock-ups, calculational benchmarks, and measurements in other BWR plants. In addition, the calculations have been benchmarked against measurements in GGNS. Comparisons with dosimetry measurements at the GGNS surveillance capsule location at the end of cycle 1 were made and excellent agreement was found. [[

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- Fluence results have been obtained for all GGNS shroud and top guide welds. [[

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- Fluence results have been obtained for the reactor vessel and projected for future operation up to 54 EFPY. [[

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- Fluence calculations for the surveillance capsules indicated that the capsule lead factor is [[ ]] Although the lead factor is less than unity, this is not a concern for GGNS since the BWRVIP surveillance program does not require any future capsule testing for GGNS.
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8 NOMENCLATURE

ASTM	American Society for Testing and Materials
BAF	bottom of active fuel
BWR	boiling water reactor
BWRVIP	BWR Vessel Internals Project
C/M	calculated-to-measured ratio
D	dimension
dpa	displacements per atom
EBZ	enriched bottom zone
EFPS	effective full power seconds
EFPY	effective full power years
ENDF	evaluated nuclear data file
EOC	end-of-cycle
GGNS	Grand Gulf Nuclear Station
ID	inner diameter
IGSCC	irradiation assisted stress corrosion cracking
IR	inner radius
LHGR	linear heat generation rate
LWR	light water reactor
MWd/MTU	megawatt days per metric ton of uranium
MOC	middle-of-cycle
MPM	MP Machinery and Testing, LLC
NMP-1	Nine Mile Point Unit 1
NMP-2	Nine Mile Point Unit 2
NRC	U. S. Nuclear Regulatory Commission
OD	outer diameter
OR	outer radius
ORNL	Oak Ridge National Laboratory
PCA	pool critical assembly
PT	Pressure-Temperature
PWR	pressurized water reactor
RBS	River Bend Station
RG	Regulatory Guide
RSICC	Radiation Safety Information Computational Center
RT_{NDT}	Reference temperature for nil ductility transition
RT_{PTS}	Reference temperature for pressurized thermal shock
SE	sequence exchange
T	vessel wall thickness
TAF	top of active fuel
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A
SHROUD/TOP GUIDE WELD FLUENCE RESULTS AT THE END OF CYCLE 21 (28.088 EFPY EXPOSURE)

This appendix contains calculated fast fluence values (fluence for neutrons with energy above 1 MeV) for welds in the shroud and top guide. Fluence values for each weld are given at the IR, OR, and at positions 1/4, 1/2, and 3/4 of the distance between the IR and OR for an operation time of 28.088 EFPY (the calculated end of cycle 21). Values are tabulated versus azimuthal angle for horizontal welds, and versus height above BAF for vertical welds.

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Appendix Table A-1Fast Fluence at Locations in the Top Guide for Weld H1 vs. Azimuth.

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Appendix Table A-2Fast Fluence at Locations in the Top Guide for Weld H2 vs. Azimuth.

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Appendix Table A-3 Fast Fluence at Locations in the Shroud for Weld H3 vs. Azimuth.

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Appendix Table A-4 Fast Fluence at Locations in the Shroud for Weld H4 vs. Azimuth.

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Appendix Table A-5 Fast Fluence at Locations in the Shroud for Weld H5 vs. Azimuth[a].

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Appendix Table A-6 Fast Fluence at Locations in the Shroud for Weld H6A vs. Azimuth[a].

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Appendix Table A-7Fast Fluence at Locations in the Shroud for Weld H6B vs. Azimuth.

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Appendix Table A-8Fast Fluence at Locations in the Shroud for Weld H7 vs. Azimuth.

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Appendix Table A-9 Fast Fluence at Locations in the Top Guide for Welds V1 and V3 vs. Height above BAF.

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Appendix Table A-10 Fast Fluence at Locations in the Top Guide for Welds V2 and V4 vs. Height above BAF.

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Appendix Table A-11Fast Fluence at Locations in the Top Guide for Weld V5 and V6 vs. Height above BAF.

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Appendix Table A-12Fast Fluence at Locations in the Top Guide for Welds V7 and V8 vs. Radial Location.

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Appendix Table A-13 Fast Fluence at Locations in the Top Guide for Welds V9 and V11 vs. Height above BAF.

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Appendix Table A-14Fast Fluence at Locations in the Top Guide for Welds V10 and V12 vs. Height above BAF.

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Appendix Table A-15Fast Fluence at Locations in the Shroud for Weld V13 and V14 vs. Height above BAF.

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Appendix Table A-16 Fast Fluence at Locations in the Shroud for Weld V15 and V16 vs. Height above BAF.

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Appendix Table A-17Fast Fluence at Locations in the Shroud for Welds V17 and V18 vs. Height above BAF.

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Appendix Table A-18 Fast Fluence at Locations in the Shroud for Welds V19 and V21 vs. Height above BAF.

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Appendix Table A-19 Fast Fluence at Locations in the Shroud for Welds V20 and V22 vs. Height above BAF.

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Appendix Table A-20Fast Fluence at Locations in the Shroud for Welds V23 and V24 vs. Height above BAF.

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B
SHROUD/TOP GUIDE WELD FLUENCE RESULTS AFTER 35 EFPY EXPOSURE

This appendix contains calculated fast fluence values (fluence for neutrons with energy above 1 MeV) for welds in the shroud and top guide. Fluence values for each weld are given at the IR, OR, and at positions 1/4, 1/2, and 3/4 of the distance between the IR and OR for an exposure of 35 EFPY (6.912 EFPY beyond the calculated end of cycle 21). Values are tabulated versus azimuthal angle for horizontal welds, and versus height above BAF for vertical welds.

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Appendix Table B-1Fast Fluence at Locations in the Top Guide for Weld H1 vs. Azimuth.

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Appendix Table B-2 Fast Fluence at Locations in the Top Guide for Weld H2 vs. Azimuth.

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Appendix Table B-3Fast Fluence at Locations in the Shroud for Weld H3 vs. Azimuth.

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Appendix Table B-4Fast Fluence at Locations in the Shroud for Weld H4 vs. Azimuth.

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Appendix Table B-5Fast Fluence at Locations in the Shroud for Weld H5 vs. Azimuth[a].

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Appendix Table B-6Fast Fluence at Locations in the Shroud for Weld H6A vs. Azimuth[a].

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Appendix Table B-7 Fast Fluence at Locations in the Shroud for Weld H6B vs. Azimuth.

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Appendix Table B-8Fast Fluence at Locations in the Shroud for Weld H7 vs. Azimuth.

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Appendix Table B-9Fast Fluence at Locations in the Top Guide for Welds V1 and V3 vs. Height above BAF.

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Appendix Table B-10 Fast Fluence at Locations in the Top Guide for Welds V2 and V4 vs. Height above BAF.

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Appendix Table B-11 Fast Fluence at Locations in the Top Guide for Weld V5 and V6 vs. Height above BAF.

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Appendix Table B-12 Fast Fluence at Locations in the Top Guide for Welds V7 and V8 vs. Radial Location.

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Appendix Table B-13Fast Fluence at Locations in the Top Guide for Welds V9 and V11 vs. Height above BAF.

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Appendix Table B-14Fast Fluence at Locations in the Top Guide for Welds V10 and V12 vs. Height above BAF.

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Appendix Table B-15Fast Fluence at Locations in the Shroud for Weld V13 and V14 vs. Height above BAF.

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Appendix Table B-16 Fast Fluence at Locations in the Shroud for Weld V15 and V16 vs. Height above BAF.

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Appendix Table B-17 Fast Fluence at Locations in the Shroud for Welds V17 and V18 vs. Height above BAF.

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Appendix Table B-18Fast Fluence at Locations in the Shroud for Welds V19 and V21 vs. Height above BAF.

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Appendix Table B-19Fast Fluence at Locations in the Shroud for Welds V20 and V22 vs. Height above BAF.

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Appendix Table B-20 Fast Fluence at Locations in the Shroud for Welds V23 and V24 vs. Height above BAF.

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C
SHROUD/TOP GUIDE WELD FLUENCE RESULTS AFTER 54 EFPY EXPOSURE

This appendix contains calculated fast fluence values (fluence for neutrons with energy above 1 MeV) for welds in the shroud and top guide. Fluence values for each weld are given at the IR, OR, and at positions 1/4, 1/2, and 3/4 of the distance between the IR and OR for an exposure of 54 EFPY (25.912 EFPY beyond the calculated end of cycle 21). Values are tabulated versus azimuthal angle for horizontal welds, and versus height above BAF for vertical welds.

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Appendix TableC-1 Fast Fluence at Locations in the Top Guide for Weld H1 vs. Azimuth.

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Appendix Table C-2Fast Fluence at Locations in the Top Guide for Weld H2 vs. Azimuth.

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Appendix Table C-3Fast Fluence at Locations in the Shroud for Weld H3 vs. Azimuth.

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Appendix Table C-4Fast Fluence at Locations in the Shroud for Weld H4 vs. Azimuth.

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Appendix Table C-5 Fast Fluence at Locations in the Shroud for Weld H5 vs. Azimuth[a].

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Appendix Table C-6 Fast Fluence at Locations in the Shroud for Weld H6A vs. Azimuth[a].

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Appendix Table C-7Fast Fluence at Locations in the Shroud for Weld H6B vs. Azimuth.

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Appendix Table C-8 Fast Fluence at Locations in the Shroud for Weld H7 vs. Azimuth.

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Appendix Table C-9 Fast Fluence at Locations in the Top Guide for Welds V1 and V3 vs. Height above BAF.

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Appendix Table C-10 Fast Fluence at Locations in the Top Guide for Welds V2 and V4 vs. Height above BAF.

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Appendix Table C-11Fast Fluence at Locations in the Top Guide for Weld V5 and V6 vs. Height above BAF.

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Appendix Table C-12 Fast Fluence at Locations in the Top Guide for Welds V7 and V8 vs. Radial Location.

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Appendix Table C-13 Fast Fluence at Locations in the Top Guide for Welds V9 and V11 vs. Height above BAF.

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Appendix Table C-14 Fast Fluence at Locations in the Top Guide for Welds V10 and V12 vs. Height above BAF.

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Appendix Table C-15Fast Fluence at Locations in the Shroud for Weld V13 and V14 vs. Height above BAF.

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Appendix Table C-16Fast Fluence at Locations in the Shroud for Weld V15 and V16 vs. Height above BAF.

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Appendix Table C-17 Fast Fluence at Locations in the Shroud for Welds V17 and V18 vs. Height above BAF.

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Appendix Table C-18 Fast Fluence at Locations in the Shroud for Welds V19 and V21 vs. Height above BAF.

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Appendix Table C-19Fast Fluence at Locations in the Shroud for Welds V20 and V22 vs. Height above BAF.

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Appendix Table C-20 Fast Fluence at Locations in the Shroud for Welds V23 and V24 vs. Height above BAF.

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