ENCLOSURE 2

MFN 06-041

NEDO-32983-A, Revision 2

Non-Proprietary Version

GE Energy, Nuclear
1989 Little Orchard Street
San Jose, CA 95125

NEDO-32983-A
Revision 2
DRF 0000-0012-4185
Class I
January 2006

Licensing Topical Report

General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

PLEASE READ CAREFULLY

The information contained in this document is furnished for the purpose of obtaining NRC approval of a calculation process for determining the reactor pressure vessel neutron fluence. The only undertakings of General Electric Company respecting information in this document are contained in the contracts between General Electric Company and the participating utilities in effect at the time this report is issued, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone other than that for which it is intended is not authorized; and with respect to **any unauthorized use**, General Electric Company makes no representation or warranty, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

CHANGES FROM REV. 0

There is no change to the body of the report. NRC's final safety evaluation regarding removal of methodology limitations was added for this revision. Appendix A was added for Revision 2, documenting GE's responses to the Requests for Additional Information (RAIs) from the NRC staff regarding NEDC-32983P Revision 0. Appendix B was added for Revision 2, documenting GE's responses to the Requests for Additional Information from the NRC staff regarding removal of methodology limitations. Revision 1 was not issued so that the revision number is the same for NEDO-32983-A and NEDC-32983P-A.

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

MFN-05-143

November 17, 2005

Mr. George B. Stramback
Regulatory Services Project Manager
GE Nuclear Energy
175 Curtner Avenue
San Jose, CA 95125

RECEIVED
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GENERAL ELECTRIC COMPANY

SUBJECT: FINAL SAFETY EVALUATION REGARDING REMOVAL OF METHODOLOGY LIMITATIONS FOR NEDC-32983P-A, "GENERAL ELECTRIC METHODOLOGY FOR REACTOR PRESSURE VESSEL FAST NEUTRON FLUX EVALUATION" (TAC NO. MC3788)

Dear Mr. Stramback:

By letters dated January 29, 2003, July 14, September 10, and December 2, 2004, and May 20, 2005, General Electric Nuclear Energy (GENE) submitted information to justify removing methodology limitations associated with Topical Report (TR) NEDC-32983P-A, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation."

By letter dated September 1, 2005, an NRC draft safety evaluation (SE) regarding our approval of TR NEDC-32983P-A was provided for your review and comments. By letter dated September 15, 2005, GENE commented on the draft SE. The staff's disposition of the GENE comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter. In addition, the September 15 letter identified several locations in the draft SE where there was information considered proprietary by GENE. The proprietary information has been removed from the draft SE, which is located at Agencywide Documents Access and Management System Accession No. ML053010126.

The NRC staff has found that TR NEDC-32983P-A is acceptable for referencing in licensing applications for GE-designed boiling water reactors to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that GENE publish accepted proprietary and non-proprietary versions of this TR within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The accepted versions shall include a "-A" (designating accepted) following the TR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, GENE and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

Herbert N. Berkow

Herbert N. Berkow, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 710

Enclosure: Final SE

cc w/encl: See next page

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

FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING REMOVAL OF METHODOLOGY LIMITATIONS FOR NEDC-32983P-A,

"GENERAL ELECTRIC METHODOLOGY FOR REACTOR PRESSURE VESSEL FAST

NEUTRON FLUX EVALUATION"

GENERAL ELECTRIC NUCLEAR ENERGY

PROJECT NO. 710

1.0 INTRODUCTION

On September 14, 2001 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML012400381), the Nuclear Regulatory Commission (NRC) approved the General Electric Nuclear Energy (GENE) boiling-water reactor (BWR) methodology for pressure vessel and core shroud fast neutron flux (E > 1.0 MeV) evaluation (Reference 1). However, the approval was subject to the following limitations:

(1) Within three years from the day of the approval of this methodology, GENE will perform predictive calculations of at least four additional BWR surveillance capsule dosimetry measurements which will be submitted to the NRC staff before initiation of the measurements.

(2) Comparisons of the measurements and calculations will also be submitted to the NRC.

(3) Shroud fluence estimates will be limited to the beltline region, without bias adjustment.

(4) GENE will perform dosimetry analysis to confirm and remove the conservatism in the shroud fluence calculations.

(5) Revisions to the fluence methodology and supporting uncertainty analysis will be provided, if the calculated/measured (C/M) comparisons (for the additional analysis of the vessel and the shroud) are not consistent with the NEDC-32983P fluence methodology.

In the process of removing the limitations, GENE submitted additional information in letters dated January 29, 2003, July 14, September 10, and December 2, 2004, and May 20, 2005 (References 2 to 6, respectively). Information was also exchanged in telephone conferences between the NRC staff and GENE personnel in order to clarify the information submitted in these letters.

2.0 REGULATORY BASIS

Specific fracture toughness requirements for normal operation and for anticipated operational occurrences for power reactors are set forth in Appendix G, "Fracture Toughness

Requirements," of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." The requirements of Appendix G are imposed by 10 CFR 50.60. Additionally, in response to concerns over potential pressurized thermal shock events in pressurized-water reactors, the NRC issued 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events."

To satisfy the requirements of both Appendix G and 10 CFR 50.61, methods for determining the fast neutron fluence (E > 1.0 MeV) are necessary to estimate the fracture toughness of the pressure vessel materials.

Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," describes methods and assumptions acceptable to the NRC staff for determining pressure vessel fluence. This RG is intended to ensure the accuracy and reliability of the fluence determination required by General Design Criteria 14, 30, and 31 of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50. The NRC staff's review of the NEDC-32983P methodology used the guidance contained in RG 1.190 to determine the acceptability of the proposed changes.

3.0 <u>TECHNICAL EVALUATION</u>

The following is a discussion and justification for the removal of those limitations.

<u>Limitation (1)</u>:

> Within three years from the day of the approval of this methodology, GENE will perform predictive calculations of at least four additional BWR surveillance capsule dosimetry measurements which will be submitted to the NRC staff before initiation of the measurements.

GENE stated that there were no surveillance capsules in the pipeline for them to fulfill the condition of four surveillance capsules. Instead, General Electric (GE) proposed a one capsule blind test (from the River Bend plant) and three existing (but not calculated) surveillance capsules. The alternative, i.e., the four GE surveillance capsules for a blind test, would cause an unpredictable delay in removing the limitations. The NRC staff agreed to this arrangement and the River Bend surveillance capsule at 183° azimuth calculated value was submitted on January 29, 2003 (Reference 2). The measured value of the same capsule was published by the Electrical Power Research Institute in June 2003 in BWRVIP-113 (Reference 7). The difference between the pre-calculated and measured values is well within the 20 percent (1σ) guidance in RG 1.190 (Reference 8) and, therefore, it is acceptable. In addition, GE submitted the calculated values for the three existing surveillance capsules (one by GE; two by other vendors) for which GE performed the calculations. The C/M ratios are also within the provisions of RG 1.190 and, therefore, are acceptable.

GE incorporated the additional four data points into its data base. The bias and the associated uncertainty was reduced, however, GE stated that the practice of applying a higher conservative bias will continue. This is acceptable and the requirement to perform additional confirmatory calculations has been fulfilled and therefore, this limitation is being removed.

Limitation (2):

Comparisons of the measurements and calculations will also be submitted to the NRC.

As indicated in the discussion of Limitation (1) above, this requirement has been satisfied and therefore, this limitation is being removed.

Limitations (3) and (4):

Shroud fluence estimates will be limited to the beltline region, without bias adjustment.

GENE will perform dosimetry analysis to confirm and remove the conservatism in the shroud fluence calculations.

Reference 4 documented GE's efforts regarding shroud fluence recalculation and benchmarking. GE identified two shroud samples taken from BWR-4 plants, one from the middle-plane at a 100° azimuth and the other 36 inches below the top guide ring weld at the 316° azimuth. A total of seven samples were created, measured, and calculated. The mean value of C/M ratios for E > 1.0 MeV flux and the associated uncertainty is acceptably low. These values are conservative and GE suggested that this was sufficient to satisfy the requirement for additional work.

The NRC staff expected that GE would present measurements to quantify axial shroud bias. This is important because fluence is used in estimating shroud crack growth rates due to irradiation assisted stress corrosion cracking. Such cracks populate mostly at the beltline region. In the December 2, 2004, submittal (Reference 5), GE stated that it does not possess any additional data to establish the shroud axial dependance of the flux. However, GE presented arguments based on the In-Reactor Irradiation Monitoring (IRIM) experimental data from 36 near-shroud measurements in response to question 8 during the original review to support their position that axial shroud bias was not a significant effect. In addition, GE presented arguments that material properties, for example yield strength versus fluence and intergranular chromium precipitation versus fluence, demonstrate very wide variations for a given fluence value, thus accurate knowledge of the fluence does not add to the accuracy of the knowledge of the material properties. The NRC staff considered in total: the IRIM data not being actual plant data, the existence of two actual plant data points showing good C/M agreement, theoretical arguments advanced by GE that there does not exist a particular cause for such axial bias dependence, the behavior of irradiated material versus fluence and the lower fluence accuracy requirements (compared to vessel) regarding crack propagation rate and decided that the GE fluence methodology is acceptable for shroud fluence calculations. Therefore, Limitations (3) and (4) are being removed.

There is another emerging issue regarding fluence calculations for the shroud and for reactor internals, i.e., that of helium production that affects their weldability. Helium calculations involve both fast and thermal fluence. GE stated (Reference 6) that because its methodology does not calculate thermal flux, it will not be applied to helium calculation problems.

Limitation (5):

> Revisions to the fluence methodology and supporting uncertainty analysis will be provided if the C/M comparisons (for the additional analysis of the vessel and the shroud) are not consistent with the NEDC-32983P fluence methodology.

This limitation is a generic condition that remains unchanged.

4.0 CONCLUSIONS

GENE provided information to justify removing methodology Limitations (1) through (4), listed above, associated with NEDC-32983P-A, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation." The NRC staff has reviewed the information submitted by GENE using the regulatory basis described in Section 2.0 above and concludes that sufficient justification has been provided to remove Limitations (1) through (4). This safety evaluation does not alter any of the other conclusions and applicability statements made in the NRC staff's September 14, 2001, letter approving the use of NEDC-32983P-A. In particular, Limitation (5) remains as a condition of applicability of the methodology.

5.0 REFERENCES

1. NEDC-32983P-A, Licensing Topical Report, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," by S. Sitaraman, et. al., General Electric Nuclear Energy, December 2001 (proprietary submittal - not publicly available in ADAMS).

2. Letter from G. Stramback, GE Nuclear Energy to U.S. Nuclear Regulatory Commission, "GE Flux Calculation Methodology Confirmation Results Part I - Surveillance Capsule Flux at River Bend Station," January 29, 2003 (ADAMS Accession No. ML030310134).

3. Letter from G. Stramback, GE Nuclear Energy to U.S. Nuclear Regulatory Commission, "Confirmatory Information on GE Methodology for RPV Flux Calculation" (Re: NEDC-32983P-A), July 14, 2004 (ADAMS Accession No. ML042020102).

4. Letter from G. Stramback, GE Nuclear Energy to U.S. Nuclear Regulatory Commission, "Confirmatory Information on GE Methodology for Shroud Flux Calculation" (Re: NEDC-32983P-A), September 10, 2004 (ADAMS Accession No. ML042610137).

5. Letter from G. Stramback, GE Nuclear Energy to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information - GE Nuclear Energy Licensing Topical Report NEDC-32983P-A" (TAC No. MC37388), December 2, 2004 (ADAMS Accession No. ML043480399).

6. Letter from G. Stramback, GE Nuclear Energy to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information - GE Nuclear Energy Licensing Topical Report NEDC-32983P-A" (TAC No. MC37388), May 20, 2005 (ADAMS Accession No. ML051600469).

7. BWRVIP-113, "BWR Vessel and Internals Project River Bend 183 Degree Surveillance Capsule Report," by R. Carter, June 2003 (proprietary submittal - not publicly available in ADAMS).

8. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U.S. Nuclear Regulatory Commission, March 2001.

Principal Contributor: L. Lois

Date: November 17, 2005

NRC Staff Responses to GENE Comments on Draft Safety Evaluation

Regarding Removal of Methodology Limitations for NEDC-32983P-A,

"General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation"

GENE Comment 1:

On Page 2, Limitation (1) of draft NRC SE: GE recommends that the statement "GE incorporated the additional four data points into its data base." be revised as follows: "GE incorporated the additional six data points (one data point each from three of the four additional capsule calculations and three data points from the fourth capsule calculation) into its data base."

NRC Staff Response:

In the existing data base, GENE averaged each surveillance capsule and counted them as one data point. If GENE would like to count this as six data points, GENE should go back and unbundle the existing data base.

GENE Comment 2:

On Page 4, Limitation (5) of draft NRC SE: Is this limitation for future calculated/measured (C/M) comparisons? The new C/M comparisons GE performed so far are consistent with the NEDC-32983P fluence methodology and, therefore, are not applicable to this limitation.

NRC Staff Response:

GE stated that it followed the guidance in RG 1.190. In Section 1.4.2.1, the guide states, among others, "As capsule and cavity measurements become available, they should be incorporated into the operating reactor measurements data base, and the calculational biases and uncertainties should be updated as necessary." Therefore, the statement refers to future data and updating of the data base.

ATTACHMENT

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

September 14, 2001

Mr. James F. Klapproth, Manager
Engineering & Technology
GE Nuclear Energy
175 Curtner Ave
San Jose, CA 95125

MFN 01-050

SUBJECT: SAFETY EVALUATION FOR NEDC-32983P, "GENERAL ELECTRIC METHODOLOGY FOR REACTOR PRESSURE VESSEL FAST NEUTRON FLUX EVALUATION" (TAC NO. MA9891)

Dear Mr. Klapproth:

By letter dated September 1, 2000, GE Nuclear Energy (GENE) submitted the subject licensing topical report (LTR) and requested staff review and approval for boiling water reactor (BWR) licensing actions. Additional information was submitted on December 20, 2000, January 5 and 17, 2001, March 2 and 14, 2001, and June 1 and 15, 2001. The NRC staff and Brookhaven National Laboratory (BNL staff consultant) exchanged information with GENE personnel on several occasions in the course of this review.

The proposed methodology employs an analytic approach based on the discrete ordinates neutron transport method to determine the fast ($E > 1.0$ MeV) flux (and fluence) in BWR vessels. The proposed methodology adheres to the guidance in Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The method is using available BWR surveillance capsule dosimetry measurements for the validation of the analytic transport calculations and the estimation of the uncertainty and bias. In addition, the method is compared to the NUREG-6115 benchmark problem and the results of a foreign reactor benchmark provided by GENE.

The staff finds the proposed methodology acceptable for referencing in licensing actions, subject to the limitation that the applicant will demonstrate the method's predictive capability in at least four surveillance capsules within three years from the day of approval of this methodology. The LTR includes a limited amount of information on the method's capability to predict the fluence on and through the core shroud. The staff concluded that the method would yield a conservative fluence estimate on the shroud. In view of the shroud fluence requirements, the staff finds the method acceptable subject to the limitations listed in the summary and limitations section of the enclosed safety evaluation (SE).

A conference call was held on August 14, 2001 between GENE, BNL and the NRC staff to discuss GENE's findings from their review of the draft SE (ADAMS accession no. ML012410011) regarding proprietary information. The conference call determined that there was no proprietary information contained in the SE. GENE requested clarification on the three year requirement for the confirmatory and predictive dosimetry for the vessel and the shroud. The staff stated that: (1) the measurement to calculation comparisons need only include activation dosimetry, (2) RG 1.190 contains the required guidance, and (3) GENE must prepare and submit to the staff a plan, identifying proposed surveillance capsules and a time schedule to satisfy and erase the limitations from the methodology.

The NRC requests that the GENE publish an accepted version of the revised NEDC-32983P within 3 months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed SE between the title page and the abstract, and add an "-A" (designating accepted) following the report identification number (i.e., NEDC-32983-A).

If the NRC's criteria or regulations change so that its conclusion in this letter that the LTR is acceptable is invalidated, GENE and/or the applicant referencing the LTR will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the LTR without revision of the respective documentation.

If you have any questions, please contact Robert Pulsifer, GENE Project Manager, at (301) 415-3016.

Sincerely,

FOR

Stuart A. Richards, Director
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 710

Enclosure: Safety Evaluation

cc w/encl: See next page

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

GE NUCLEAR ENERGY TOPICAL REPORT NEDC-32983P

"GENERAL ELECTRIC METHODOLOGY FOR REACTOR PRESSURE VESSEL FAST NEUTRON FLUX EVALUATIONS"

PROJECT NO. 710

1.0 INTRODUCTION

By letter dated September 1, 2000, GE Nuclear Energy (GENE) submitted their methodology for reactor pressure vessel fast neutron flux evaluations and requested NRC review and approval (Reference 1). The proposed methodology is intended for the determination of the fast neutron fluence accumulated by the pressure vessel and internal components of US boiling water reactor (BWR) plants. The methodology has evolved from earlier GENE fluence methods. The proposed licensing topical report (LTR) (NEDC-32983P) fluence evaluation employs an analytic approach using the most recent fluence calculational methods and nuclear data sets. In the proposed methodology, the vessel fluence is determined by a discrete ordinates transport calculation in which the core neutron source is explicitly represented and the neutron flux is propagated from the core through the downcomer and the jet pumps and jet pump risers whenever present, to the vessel (rather than by an extrapolation of the measurements). The method proposed for predicting the dosimeter response and the vessel inner-wall fluence is generally consistent with Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Reference 2).

The LTR provides a description of the application of the proposed methodology to the calculation of the Brookhaven National Laboratory (BNL) pressure vessel fluence benchmark problem described in NUREG/CR-6115 (Reference 3). The LTR also describes the application of the methodology to the analysis of a GENE dosimetry benchmark experiment (References 4 and 5). This includes a description of both the discrete ordinates DORT (Reference 6) and the MCNP (Reference 7) Monte Carlo transport calculations of the measurements and the techniques used to interpret the in-vessel dosimeter response. Representative BWR surveillance measurements and comparisons to GENE calculations are provided as additional qualification of the calculational methods. The GENE dosimetry measurements are used to validate the DORT vessel fluence methodology and determine the calculational biases and uncertainties.

The LTR fluence calculation and uncertainty methodology is summarized in Section 2. The evaluation of the important technical issues raised during this review is presented in Section 3 and the summary and limitations given in Section 4. The staff was assisted in this review by BNL personnel as consultants.

2.0 SUMMARY OF THE NEDC-32983P VESSEL FLUENCE METHODOLOGY

2.1 Pressure Vessel Fluence Calculation Methodology

The proposed methodology provides a best-estimate prediction of the fluence rather than the conservative prediction as was the case with earlier methods. The fluence calculations are performed with the DORT discrete ordinates transport code. The LTR provides a description of the DORT calculation used to determine the vessel fluence, as well as the calculations used to predict the GENE measured dosimetry and validate the transport model. The calculational model includes a representation of the peripheral fuel assemblies and the core-internals, downcomer and vessel geometry. Calculations are performed to determine the pin-by-pin and bundle-average power distribution in the peripheral fuel bundles for input to the DORT core neutron source. Calculations employ a relatively fine (r, θ, z) spatial mesh and are carried out using both an S_8 and an S_{12} angular quadrature set.

The eighty-group MATXS (Reference 8) cross section library is the basic nuclear data set. This library is used in performing the energy and spatial self-shielding and removal calculations. The scattering cross sections are represented using a P_3 Legendre expansion. The calculations are performed in (r, θ) and (r, z) geometries. A synthesis technique is used to determine the three-dimensional fluence distribution and to some extent account for the effect of axial leakage between the core and the cavity.

Predictions of the dosimeter response measurements are required to determine the calculation-to-measurement (C/M) data base used to validate the fluence calculation methods. The predictions are made for the in-vessel dosimetry using essentially the same methods used to determine the vessel fluence. The proposed methodology includes dosimeter response adjustments for the half-lives of the reaction products and the core power history. In order to ensure an accurate prediction of the dosimeter response, a detailed spatial representation of the capsule geometry is included in the DORT model. The measured dosimeter reaction rates are calculated using the dosimeter-specific reaction cross sections. The calculated dosimeter response is determined for the irradiation period up to the time the capsule was withdrawn.

2.2 Calculation of the BNL Pressure Vessel Fluence Benchmark Problem

As part of the qualification of the fluence calculational methodology, GENE has calculated the BNL NUREG/CR-6115 BWR pressure vessel fluence benchmark problem. The NUREG/CR-6115 report provides the detailed specification and corresponding numerical solutions for the BWR fluence benchmark problem. The calculation of the benchmark problem allows a detailed assessment and verification of the numerical procedures, code implementation, and the various modeling approximations relative to a representative BWR operating configuration. The geometry, materials and space and energy dependent source are fixed by the problem specification and the reference solutions allow comparisons of the predicted fluence at the vessel locations of interest.

The LTR describes the calculations performed using both the proposed DORT discrete ordinates method and the MCNP Monte Carlo method. The DORT calculations were performed using the proposed "current" method. The calculational model included the complete radial geometry from the core out through the concrete biological shield and axially from the core inlet up to the steam separator. As part of the analysis of the benchmark problem, GENE performed a series of sensitivity calculations in which various modeling assumptions were evaluated. Calculations were performed with three downcomer models: (1) a conservative model in which the jet pumps and risers are neglected, (2) an approximate model in which the materials of the jet pumps and risers are homogenized over the volume of the downcomer, and (3) a model in which the components in the downcomer are treated explicitly as heterogeneous material zones. Calculations were performed using both ENDF/B-V and ENDF/B-VI nuclear data sets. The effect of using a more accurate angular quadrature set was evaluated by comparing calculations performed with S_8 and S_{12} quadratures. The effect of the peripheral radial flux gradient on the core neutron source and vessel fluence was evaluated by calculating the fluence with: (1) a model in which a uniform bundle-average power is assigned to each peripheral fuel bundle, and (2) a model in which spatially dependent power distributions (provided with the problem specification) are assigned to the outer three rows of fuel bundles.

Comparisons of the GENE and BNL NUREG/CR-6115 vessel fluence predictions are provided. As an additional verification of the GENE fluence methodology, GENE has performed MCNP Monte Carlo calculations of the BNL vessel fluence benchmark problem. The MCNP model included an essentially exact octant representation of the core, shroud, jet pump/riser and vessel geometry specified in the NUREG/CR-6115 report. The calculations were performed using a continuous energy representation of the nuclear data. The cross section data used in these calculations is based on the ENDF/B-V nuclear data except for iron, hydrogen and oxygen. Since the cross sections for these elements have changed significantly in the more recent ENDF/B-VI data set, ENDF/B-VI cross sections were used for iron, hydrogen and oxygen. Calculations were performed using two models for describing the power/source distribution in the peripheral fuel bundles: (1) a uniform bundle-average power model, and (2) a pin-wise power distribution model. The source normalization used in the MCNP calculation was taken to be the same as that used in the DORT calculation of the benchmark problem. Variance reduction was accomplished by defining a set of importance regions which allowed particle splitting. In addition to the base calculation, a series of MCNP sensitivity calculations was performed to determine the effect of: (1) including the jet pumps and riser materials, (2) variations in the fuel actinide inventory, and (3) the ENDF/B-V to ENDF/B-VI cross section updates.

The LTR includes comparisons of the GENE and BNL NUREG/CR-6115 MCNP fluence predictions. Comparisons are provided for the $E > 1.0$ MeV fluence at the axial midplane for locations in the downcomer and on the vessel inner-wall.

2.3 Calculation of the BWR Neutron Dosimetry Benchmark Measurements

In order to provide a measurement benchmark for qualifying the DORT and MCNP calculational methodology, GENE has performed an in-reactor dosimetry benchmark experiment (References 4 and 5). The experiment included the irradiation of a set of passive dosimeters for one cycle in an operating (non-US) BWR. The measurements included Fe-54, Nb-93, and Ni-58 threshold dosimeters as well as U-238, Th-232 and Np-237 fission dosimeters. The dosimeters were located in the downcomer at three axial elevations, three azimuths and three radial locations. The dosimeter activation counting and related measurements were performed at the GE Vallecitos Nuclear Laboratory.

The neutron dosimetry provides a direct measurement of the activity (dps/gm) associated with the individual dosimeter-specific reactions. The measured dosimeter activities were converted to specific full power reaction rates (rps/nucleus) averaged over the period of irradiation. This conversion accounted for the physical characteristics of the sensor (e.g., weight of the target isotope in the sample), the operating history of the reactor, the energy response of the sensor (e.g., reaction cross section), decay of the target isotope, and in the case of fission dosimeters, the number of product atoms produced per reaction. In order to allow comparison of the measured and calculated dosimeter reaction rates, the measured reaction rates have also been corrected for target depletion.

The in-vessel dosimetry measurements were used to benchmark and validate the proposed calculational methodology. The validation included both DORT and MCNP calculations of the measured dosimeter reaction rates. The calculational models used in the prediction of the measurements are based on the proposed methodology described in Section 2.1. The models include a detailed representation of the peripheral fuel assemblies and the core internals, downcomer and vessel geometry. The DORT calculations employ a relatively fine (r, θ, z) spatial mesh and were carried out using an S_{12} quadrature and a P_3 expansion of the scattering cross sections.

The calculations of the dosimeter response measurements are used to determine the calculation-to-measurement data base used to validate the fluence calculation methods. The analysis of the C/M data indicates that: (1) the DORT calculations using an adjusted downcomer model result in a mean C/M value ranging from 0.9 to 1.1 for Fe-54 and Nb-93; and (2) the MCNP calculations result in C/M values ranging from 0.9 to 1.1.

3.0 EVALUATION

The review of the NEDC-32983P methodology focused on the details of the fluence calculation methods, their compliance with the guidance in RG 1.190 and the qualification of the methodology provided by the GENE C/M data-base. As a result of the review, several technical issues were identified which required additional information and clarification from GENE. Requests for additional information (RAI) were transmitted in References 9, 13 and 16. The GENE responses were provided in References 10-12, 14-15, and 17-18. This evaluation is based on the material included in the LTR and in the referenced GENE responses to the RAIs. The evaluation of the major issues raised during the review is summarized in the following.

3.1 Pressure Vessel Fluence Calculation Methodology

The DORT transport calculational model is constructed using plant-specific as-built dimensions and actual plant parameters whenever possible (response to RAI-5, Reference 10). The calculations use a fine spatial and angular mesh in both the (r, θ) and (r, z) calculations together with a detailed representation of the core internals, downcomer, and vessel geometry. The calculations employ an S_8 angular quadrature set and a P_3 scattering cross section expansion.

The proposed fluence methodology generally employs a best-estimate approach, however, certain conservative features have been retained from the traditional method. For example, in response to RAI-2 (Reference 10), GENE indicates that the core neutron source used in the

DORT transport calculation is based on the bundle-average power in the peripheral fuel bundles. This results in a conservatism in the fluence estimate because: (1) the fuel pins close to the core edge have reduced power because of neutron leakage from the core, and (2) the fuel pins close to the core edge provide the dominant contribution to the vessel fluence. The magnitude of the effect of using the bundle-average power rather than the pin-wise power distribution is calculated for the BWR pressure vessel fluence benchmark problem (Reference 3). In addition, in the response to RAI-2 (Reference 15), GENE indicated that this conservatism in the fluence calculation is applicable to all core designs. However, GENE has indicated in response to RAI-18 (Reference 10) that credit for this conservatism will be taken in determining the adjustment that must be applied to the calculated fluence to determine the best-estimate fluence value. Therefore, while the proposed current methodology includes this conservatism and over-predicts the fluence due to the use of bundle-average power, this conservatism is removed in the application of the methodology when the best-estimate fluence is determined, by a downward adjustment of the calculated fluence.

The nuclear cross section library used in the fluence transport calculations employs a P_3 Legendre expansion of the anisotropic cross sections. However, because of the relatively strong axial dependence of the void distribution in the core and the presence of the jet-pump and jet-pump riser arrangement in the downcomer, there was concern that the third order Legendre expansion may not be sufficiently detailed to accurately model the streaming and shadowing effects at the vessel inner-wall. In order to evaluate this effect, GENE has performed a series of sensitivity calculations using a P_5 expansion of the anisotropic scattering cross section. The results of these calculations are presented in the response to RAI-7 (Reference 10) and indicate that the effect of this approximation on the vessel fluence and dosimetry reaction rates is negligible.

3.2 Calculation of the BNL Pressure Vessel Fluence Benchmark Problem

The BNL pressure vessel fluence benchmark problem was calculated as part of the validation and testing of the NEDC-32983P fluence methodology. The calculations were carried out using the proposed GENE methodology (response to RAI-5, Reference 2) and were compared with the tabulated benchmark reference predictions. The analysis of the benchmark problem included a set of sensitivity calculations which evaluated and confirmed the validity of several modeling assumptions included in the methodology. The GENE and reference calculations of the vessel peak inner-wall fluence were found to be in good agreement.

In the proposed methodology, the DORT transport calculations are performed using a nuclear cross section set that has been collapsed by averaging the nuclear data over a multi-group energy structure. Following the guidance in RG 1.190 (Section-1.1.2.2), GENE tested and evaluated the averaging procedure used in collapsing the cross sections. The evaluation included a series of DORT transport calculations which were carried out for the BNL vessel fluence benchmark problem using several sets of collapsed cross sections. The results of these calculations are included in the GENE response to RAI-3 (Reference 10). Calculations were performed for a 26-group cross section set, a 44-group cross section set and a 47-group cross section set (calculated by BNL). Comparisons of the $E > 1.0$ MeV flux and the flux spectrum were made at the shroud, downcomer, surveillance capsule, vessel inner-wall, vessel quarter-thickness and vessel outer-wall locations. Based on these comparisons and additional calculations performed by GENE, it is concluded that the use of the collapsed cross section library introduces a bias into the fluence prediction (response to RAI-3, Reference 10). GENE has indicated in response to RAI-18 (Reference 10) that, in order to account for this approximation, the fluence calculated with the NEDC-32983P methodology will be adjusted to determine the best-estimate fluence value. Therefore, while the proposed current methodology includes this calculational bias due to the cross section averaging procedure, this bias will be removed when the best-estimate fluence is determined.

3.3 Calculation of the BWR Benchmark Dosimetry Measurements

The BWR neutron dosimetry experiment includes an extensive set of in-vessel fast and thermal neutron dosimeter measurements. The irradiation of the dosimeters was performed during a single cycle of operation at an operating BWR. The dosimeter activation and associated measurements were performed at the GE Vallecitos Nuclear Laboratory. The inferred reaction rates are proportional to the measured specific activities and include adjustments for the actual plant operating history and the decay of the reaction product isotope. The reaction rates were used to construct the C/M benchmark data base and determine the calculational bias and uncertainty.

The initial analysis of the BWR neutron dosimetry experiment did not include C/M comparisons for the dosimetry measurements at the 71° azimuth. However, in response to RAI-9 (Reference 15), GENE has updated the C/M data base to include this data. This additional C/M data is generally consistent with data taken at 4° and 20°. In order to allow valid benchmarking C/M comparisons of the calculations and the dosimetry experiment measurements, reliable estimates of the uncertainty in the dosimetry measurements are required. In response to RAI-11 (Reference 11), GENE has provided the uncertainty analysis for the dosimetry experiment measurements. The statistical uncertainty in the specific dosimeter activity measurement is provided for both the fast and thermal dosimeters. The measurement uncertainty resulting from the uncertainty in the capsule location is based on: (1) the mechanical tolerance for capsule displacement, and (2) the sensitivity of the dosimeter response to capsule displacement. Since the spatial variation of the fast and thermal flux (and associated displacement sensitivity) is different, the measurement uncertainty due to capsule displacement is determined for both the fast and thermal dosimeters.

In addition to the BWR neutron dosimetry experiment, the GENE dosimetry benchmark data base includes a set of surveillance capsule flux measurements. This surveillance capsule data base includes a range of plant measurements that have been made over the past decade. The activity measurements were carried out using a set of standard fast neutron threshold dosimeters. The GE Vallecitos Nuclear Laboratory analyzed the activity measurements and determined the analysis uncertainty. The activation measurement is converted to flux using a dosimeter specific cross section determined by a series of specially controlled experiments. In response to RAI-3 (Reference 14), GENE has indicated that the methods used to analyze these surveillance dosimetry measurements are compliant with the ASTM standards for measuring fast-neutron reaction rates by radioactivation of iron, copper and nickel; ASTM E-263-93 (Reference 19), ASTM E-523-92 (Reference 20) and ASTM E-264-92 (Reference 21), respectively.

3.4 C/M Comparisons and Uncertainty Analysis

The qualification of the NEDC-32983P pressure vessel neutron fluence methodology includes comparisons of fluence calculations and measurements for: (1) the operating reactor benchmark dosimetry experiment and (2) the BWR surveillance capsule dosimetry measurements. The methods benchmarking is based on both the BWR dosimetry experiment and the surveillance capsule measurements. The benchmark experiment measurements include a set of fast neutron threshold dosimeters located in the downcomer at three axial elevations, three azimuths and three radial locations. The BWR surveillance measurements are for capsules located at various locations in the downcomer including within the shadow and the penumbra of the jet pumps and jet pump risers. The dosimetry experiment provides a continuous fluence measurement during the single cycle of irradiation, while the surveillance capsule measurements provide a continuous fluence measurement from initial startup to the time of capsule removal which represent a variety of irradiation time intervals. These operating reactor measurements provide an indication of the effect of the as-built geometry and material compositions on the fluence calculations. The benchmarking is based on the calculation-to-measurement (C/M) comparisons of the measured reaction rates. The measurements provide a number of C/M comparisons and a statistical estimate of the calculational bias and uncertainty.

The benchmark experiment comparisons are made for each location as a function of dosimeter type (e.g., Fe-54 and Nb-93). In the response to RAIs 10 and 14 (Reference 12), GENE has provided the C/M ratios and analysis for the dosimetry benchmark experiment. In addition to the Fe-54 and Nb-93 bare capsule dosimeters included in the LTR, Ni-58 and Nb-93 shielded capsules were also evaluated. The C/M analysis for the dosimetry benchmark experiment indicates that the calculations are within 20 percent (one-σ) for the vessel measurements.

In the responses to RAI-17 (Reference 12) and RAI-7 (Reference 15), GENE has provided a statistical analysis of the C/M comparisons for the BWR capsule surveillance measurements. The analysis included in the responses to RAIs 17 and 18 (Reference 12) and RAI-7 (Reference 15) indicates that the proposed methodology is biased relative to the measurements. The C/M bias and its uncertainty have been determined using statistical techniques. In the proposed methodology, the best-estimate fluence is determined by applying the C/M bias to the calculated fluence. In addition, GENE has indicated in response to RAI-8 (Reference 15) that as new measurements become available these comparisons will be updated. If necessary, the bias and its uncertainty will be updated and the adjustment to the calculated fluence will be revised.

In order to provide an independent estimate of the bias and uncertainty in the NEDC-32983P fluence calculational methodology, GENE has performed an analytic uncertainty estimate. The significant sources of bias/uncertainty were identified by a set of DORT fluence sensitivity calculations. These calculations concerned the treatment of the nuclear cross section data, core neutron source, angular quadrature, and geometrical representation of the downcomer. In addition, in response to RAI-6 (Reference 15), GENE has included the effect of the BWR fuel bundle nodal and pin-wise power distribution uncertainty on the calculated fluence. Estimates of the important uncertainty contributors were made and the effect of these uncertainties was propagated through the fluence calculation using the calculated sensitivities. In the response to

RAI-18 (Reference 12) and RAI-6 (Reference 15), the analytically determined fluence calculational uncertainty is shown to be less than 20 percent.

The significant sources of calculational bias were determined to be: (1) the effect of using the bundle-average power rather than the pin-wise power distribution in the peripheral fuel bundles, and (2) the effect of using a specific flux-averaged multi-group cross section set. In the response to RAI-18 (Reference 12), the overall fluence calculational bias is determined analytically as a combination of these individual components. The bias determined using the analytic method was found to be slightly less but well within the uncertainty range of the bias determined based on the surveillance dosimetry measurements. In the conclusion of the response to RAI-7 (Reference 15), GENE stated that the calculational bias based on the dosimetry measurements will be applied to the fluence calculated using the NEDC-32983P fluence methodology.

While the uncertainty analysis based on the surveillance dosimetry C/M comparisons is generally consistent with the analytic uncertainty, it is noted that several substantial adjustments are required to account for approximations made in the calculations of the surveillance data. In addition the uncertainty in the fluence adjustment is not substantially smaller than the adjustment itself. Therefore, in order to provide additional confidence in the benchmarking of the proposed fluence methodology, within three years GENE is required to perform predictive calculations of at least four additional BWR capsule dosimetry activity measurements. These calculations should be submitted to the NRC staff prior to the completion of the measurements. After the measurements are completed, comparisons of the measurements and calculations should also be submitted to the NRC. If the C/M comparisons are not consistent with the proposed NEDC-32983P fluence methodology and supporting benchmark uncertainty analysis, the necessary revisions to the uncertainty analysis and methodology should be provided in the submittal. This requirement was discussed and agreed upon with GENE in a NRC/GENE/BNL conference call on June 25, 2001.

3.5 Core Shroud

In addition to the calculation of pressure vessel fluence, GENE has indicated that the proposed fluence methodology may be required for material evaluations of the core shroud. GENE has described the shroud fluence calculational procedure and provided an analytic estimate of the calculational uncertainty in response to RAI-8 (Reference 17).

As benchmarking for the shroud fluence calculation, in Figure 5-4 of the LTR and in the response to RAI-8 (Reference 17), GENE has provided comparisons of reaction rates calculated with the proposed methodology and reaction rates determined from measurements for capsules located close to the shroud. No direct shroud data were provided. The benchmark experiment C/M comparisons for the shroud indicate a conservative bias and a systematic over-prediction of the measurement data. However, review of this data indicates that the C/M comparisons for these dosimeters include large differences that are outside the expected calculation and measurement uncertainties. Consequently, because the bias is based on a single experiment and there is no surveillance data to confirm this result, this conservatism is not considered sufficiently reliable to reduce the calculated shroud fluence.

However, shroud fluence values are used mainly for the estimation of shroud crack growth propagation rates. The phenomenon is associated with a threshold fluence value. Therefore, the staff finds the proposed method acceptable for shroud fluence calculations provided that: (1) the estimates are limited within the beltline region, and (2) the bias is not deducted from the calculated value. To provide additional confidence to the predicted shroud fluence, GENE is required within three years from the approval of this methodology to perform and provide to the staff additional dosimetry analysis, directly related to the shroud, demonstrating the capability of this method.

4.0 SUMMARY AND LIMITATIONS

The staff reviewed NEDC-32983P entitled, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," and supporting documentation provided in References 10-12, 14-15 and 17-18. Based on this review, it is concluded that the NEDC-32983P methodology provides an acceptable best-estimate prediction of the pressure vessel neutron fluence for US BWR plants. As discussed in Section 3.4 of this SE, the best-estimate vessel fluence prediction is determined by the application of the calculated-bias adjustment to the fluence estimate using the NEDC-32983P fluence methodology.

However, this acceptance is subject to the following limitations and requirements (Sections 3.4 and 3.5):

(1) Within three years from the day of the approval of this methodology, GENE will perform predictive calculations of at least four additional BWR surveillance capsule dosimetry measurements which will be submitted to the staff before initiation of the measurements.

(2) Comparisons of the measurements and calculations will also be submitted to the NRC.

(3) Shroud fluence estimates will be limited to the beltline region, without bias adjustment.

(4) GENE will perform dosimetry analysis to confirm and remove the conservatism in the shroud fluence calculations.

(5) Revisions to the fluence methodology and supporting uncertainty analysis will be provided, if the C/M comparisons (for the additional analysis for the vessel and the shroud) are not consistent with the NEDC-32983P fluence methodology.

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Date: September 14, 2001

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ABSTRACT

This document presents the calculation methodology developed by the General Electric Company (GE) for the determination of reactor pressure vessel fast neutron flux. The adequacy of the GE methodology is demonstrated through a detailed description of the calculation procedures and examples showing agreement between GE practices and the standards and requirements set forth in the Regulatory Guide 1.190.

Validation of the methodology is demonstrated through GE solutions to the BWR benchmark problem. Benchmark calculations of the in-reactor irradiation sample reaction rates provide additional validation. Sensitivity studies of calculation variables, as well as uncertainty and bias assessments, are also included.

A calculational bias currently exists in the GE-calculated fluences compared to data collected through surveillance samples. The improved methodology described in this LTR eliminates some of the excess conservatism and provides a more realistic flux distribution within the reactor vessel, while meeting the requirements of Regulatory Guide 1.190.

1.0 INTRODUCTION

This document presents an improved General Electric Company (GE) flux calculation methodology for determination of reactor pressure vessel (RPV) and internals neutron fluence. Similar methods and processes have been in use by GE for the past decade for the evaluation of fast neutron fluence in the reactor pressure vessel and internal components.

In order to demonstrate that the GE methodology is in agreement with the intent of Regulatory Guide 1.190 (and its draft version, DG-1053), Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence [1], the following topics are covered in this report:

- GE flux synthesis methodology: Geometric and material representations of the calculation model, cross section library, neutron source distribution, etc.
- GE solution to the NUREG/CR-6115 BWR benchmark problem[2]: GE methodology comparable to Brookhaven National Laboratory (BNL) solution is used for the benchmark calculations. GE results and BNL results are compared. Sensitivity studies are performed for calculation variables including: the effects of steel components in the downcomer, ENDF[1]/B-VI vs. ENDF/B-V iron cross sections, S_8 vs. S_{12} angular quadratures, pin-by-pin vs. bundle-average power density, etc.
- Monte Carlo solution to BWR benchmark problem: Comparison of GE 3-D Monte Carlo technique vs. BNL 3-D solution.
- Benchmark through in-reactor measurements: Calculated reaction rates vs. dosimetry data collected via an in-reactor irradiation monitoring project.
- Correlation of a set of in-vessel surveillance data vs. GE-calculated results.
- Uncertainty and bias assessments.

For the past two decades, GE has provided services in the area of RPV fluence evaluations, using both calculations and dosimetry. The fluence calculation methodology employed by GE has been standardized in the past decade. The calculated ratio of the surveillance sample flux to the peak flux at RPV defines a lead factor. This lead factor is applied to the sample dosimetry data for determination of the RPV peak fluence, which is required for the vessel fracture toughness evaluations.

In order to comply with the provisions of RG 1.190 for a best-estimate fluence, GE has revised several aspects of the fluence evaluation processes. In the current method, the cross-section data for iron, oxygen, and hydrogen are updated with the ENDF/B-VI values. In addition, the material composition and geometric outline of the steel components in

[1] Evaluated Nuclear Data File (ENDF). See "ENDF/B Summary Documentation," BNL-NCS-17541 (ENDF-201), R, Kinsey, ed. (July 1979).

downcomer are explicitly modeled. Consequently, the current method provides more realistic neutron flux distributions in the RPV and its internal components.

2.0 GE METHODOLOGY

For the evaluation of RPV fast neutron flux, GE has traditionally employed the flux synthesis technique where a combination of two-dimensional calculations are performed and the results combined to synthesize a three-dimensional flux distribution.

The two-dimensional code used by GE is DORTG01V, which is a discrete ordinates code package based on CCC-543 TORT-DORT Version 2.8.14 issued by Oak Ridge National Laboratory (ORNL) in 1984[3]. DORTG01V is a controlled code in the GE Engineering Computer Program (ECP) library[1].

2.1 DISCRETE ORDINATES METHOD

2.1.1 (r,θ) Model

[1] ECP library contains controlled computer codes and cross section libraries approved for design applications.

In the angular coordinate, θ, the mesh size is $^1/_2$ degree or less per mesh step. Mesh size in the radial direction varies with each region. Generally, a fine mesh is provided near material interfaces, where significant flux gradients are expected. Fine meshes are also applied near the capsule, the RPV clad, and the innermost portion of the RPV. Sufficient fine mesh steps are provided to simulate the outer profile of the core. The mesh step is fine enough such that the (r,θ) representation would reproduce the true physical bundle area to within ~0.5%.

2.1.2 (r,z) Model

2.1.3 Coolant Density

2.1.4 Neutron Source Distribution

The spatial distribution of neutron source density is assumed to be proportional to the relative cycle-averaged energy production at each fuel node and each bundle location. A typical core-averaged relative power density variation is shown in Table 2-2 for (r,θ) calculation. A typical core-zone averaged variation in the axial direction is shown in Table 2-3 for (r,z) calculation.

2.1.5 Material Compositions

The composition in each material zone is treated as a homogenized mixture. The volume fractions of solid material and coolant in the core regions are calculated based on the bundle design data.

2.1.6 Cross-Section Library

The cross-section data used in the DORT calculation are processed with the nuclear cross-section processing package in the GE ECP library. The basic cross-section library used is the MATXS library[5], which was generated by Los Alamos National Laboratory for reactor physics application[5]. The MATXS library contains the 80-group infinite dilute neutron cross sections for various temperatures and self-shielding parameters (σ_o). This library is used in performing the resonance self-shielding, spatial self-shielding, elastic removal correction, reactor and cell flux solutions, and cross-section condensation to fewer groups.

The nuclide atom densities described in Sections 2.1.3 and 2.1.5 are incorporated, in conjunction with the microscopic cross section set, to create the macroscopic mixture cross sections which approximate the anisotropic scattering cross sections with 3rd-order Legendre polynomial expansions (P3). These data sets are further transformed to a group-organized format compatible with the DORT inputs.

2.1.7 Results of Discrete Ordinates Method

Figure 2-3 shows an example of the calculated fast neutron flux (E>1MeV) vs. azimuth along the reactor shroud inner radius. Figure 2-4 shows the axial flux profile and the elevation of peak flux on the same radius.

2.2 CURRENT VS. TRADITIONAL METHODOLOGIES

2.3 MONTE-CARLO TECHNIQUE

Table 2-1
Sample Nodal Coolant Density

	Coolant Density (g/cc)				
Node	**R1**	**R2**	**R3**	**R4**	**Core Average**
25	0.4429	0.3902	0.3635	0.3657	0.3896
24	0.4454	0.3920	0.3652	0.3680	0.3917
23	0.4497	0.3948	0.3679	0.3714	0.3949
22	0.4552	0.3984	0.3714	0.3759	0.3992
21	0.4618	0.4028	0.3757	0.3813	0.4043
20	0.4692	0.4080	0.3807	0.3872	0.4101
19	0.4774	0.4137	0.3864	0.3928	0.4164
18	0.4864	0.4201	0.3926	0.3992	0.4233
17	0.4963	0.4272	0.3994	0.4065	0.4311
16	0.5155	0.4472	0.4202	0.4272	0.4513
15	0.5267	0.4554	0.4279	0.4355	0.4600
14	0.5390	0.4643	0.4362	0.4446	0.4696
13	0.5543	0.4755	0.4467	0.4562	0.4817
12	0.5716	0.4886	0.4590	0.4696	0.4956
11	0.5907	0.5039	0.4736	0.4854	0.5117
10	0.6115	0.5216	0.4909	0.5038	0.5302
9	0.6336	0.5421	0.5114	0.5253	0.5513
8	0.6567	0.5657	0.5355	0.5506	0.5753
7	0.6802	0.5931	0.5641	0.5802	0.6026
6	0.7031	0.6250	0.5979	0.6148	0.6335
5	0.7242	0.6619	0.6381	0.6547	0.6682
4	0.7408	0.7016	0.6839	0.6971	0.7048
3	0.7500	0.7369	0.7282	0.7346	0.7370
2	0.7535	0.7526	0.7514	0.7522	0.7524
1	0.7552	0.7552	0.7552	0.7552	0.7552

Table 2-2
Sample Bundle Relative Power Density

I\J	1	2	3	4	5	6	7	8	9	10	11
1	1.2147	1.4181	1.3014	1.3900	1.2992	1.5350	1.3712	1.4265	1.1662	1.0239	0.5845
2	1.3896	1.3202	1.4103	1.3801	1.4863	1.4536	1.4978	1.2650	1.2915	0.9654	0.5434
3	1.2649	1.4028	1.2828	1.4728	1.3410	1.5163	1.3296	1.3524	1.2156	0.8058	0.4865
4	1.3796	1.3772	1.4720	1.3713	1.4449	1.4094	1.4058	1.3133	0.9378	0.6598	
5	1.2975	1.4848	1.3407	1.4462	1.2752	1.4106	1.1927	1.1810	0.7944		
6	1.5363	1.4559	1.5186	1.4149	1.4152	1.1538	1.1898	1.0263	0.6002		
7	1.3728	1.5000	1.3332	1.4120	1.1982	1.1914	0.9299	0.7586	0.4949		
8	1.4283	1.2667	1.3549	1.3171	1.1842	1.0267	0.7580	0.5497			
9	1.1655	1.2943	1.2184	0.9396	0.8044	0.5975	0.4899				
10	1.0298	0.9695	0.8084	0.6643							
11	0.6196	0.5525	0.4911								

Table 2-3
Sample Normalized Power Density

Node	R1	R2	R3	R4	Core Avg
25	0.1125	0.1663	0.2348	0.2911	0.2007
24	0.3051	0.4136	0.5431	0.6729	0.4820
23	0.4059	0.5644	0.7327	0.8921	0.6471
22	0.4752	0.6971	0.8963	1.0698	0.7835
21	0.5185	0.7945	1.0122	1.1827	0.8768
20	0.5461	0.8654	1.0918	1.2381	0.9365
19	0.5637	0.9163	1.1451	1.2359	0.9686
18	0.5754	0.9546	1.1819	1.2319	0.9909
17	0.5831	0.9864	1.2101	1.2329	1.0092
16	0.5698	0.9857	1.2050	1.2010	0.9974
15	0.5783	1.0177	1.2333	1.2138	1.0185
14	0.6627	1.1353	1.3764	1.3440	1.1384
13	0.6759	1.1749	1.4119	1.3708	1.1677
12	0.6861	1.2161	1.4514	1.3997	1.1983
11	0.6914	1.2520	1.4840	1.4206	1.2227
10	0.6933	1.2841	1.5094	1.4332	1.2414
9	0.6929	1.3124	1.5205	1.4304	1.2510
8	0.6884	1.3432	1.5414	1.4390	1.2656
7	0.6802	1.3750	1.5608	1.4510	1.2799
6	0.6670	1.4050	1.5765	1.4625	1.2914
5	0.6463	1.4246	1.5865	1.4688	1.2956
4	0.6144	1.4107	1.5718	1.4483	1.2757
3	0.5645	1.3082	1.4754	1.3464	1.1878
2	0.4700	1.0270	1.1865	1.0735	0.9507
1	0.1683	0.3287	0.4093	0.3679	0.3225
Avg	0.5534	1.0144	1.2059	1.1967	1.0000

Table 2-4
Group Structure for 80-Group Neutron Cross-Section Data

Group Number	Upper Energy (eV)	Lethargy Width	Group Number	Upper Energy (eV)	Lethargy Width
1	2.0000E+7	0.168	41	1.5034E+4	0.125
2	1.6905E+7	0.125	42	1.3268E+4	0.125
3	1.4918E+7	0.100	43	1.1709E+4	0.125
4	1.3499E+7	0.125	44	1.0333E+4	0.125
5	1.1912E+7	0.175	45	9.1188E+3	0.125
6	1.0000E+7	0.250	46	8.0473E+3	0.125
7	7.7880E+6	0.250	47	7.1017E+3	0.125
8	6.0653E+6	0.250	48	6.2673E+3	0.125
9	4.7237E+6	0.250	49	5.5308E+3	0.125
10	3.6788E+6	0.250	50	4.8810E+3	0.125
11	2.8650E+6	0.250	51	4.3074E+3	0.125
12	2.2313E+6	0.250	52	3.8013E+3	0.125
13	1.7377E+6	0.250	53	3.3546E+3	0.125
14	1.3534E+6	0.125	54	2.9604E+3	0.125
15	1.1943E+6	0.125	55	2.6126E+3	0.125
16	1.0540E+6	0.125	56	2.3056E+3	0.125
17	9.3014E+5	0.125	57	2.0347E+3	0.125
18	8.2085E+5	0.125	58	1.7956E+3	0.125
19	7.2440E+5	0.125	59	1.5846E+3	0.125
20	6.3928E+5	0.125	60	1.3984E+3	0.125
21	5.6416E+5	0.125	61	1.2341E+3	0.125
22	4.9787E+5	0.125	62	1.0891E+3	0.125
23	4.3937E+5	0.125	63	9.6112E+2	0.250
24	3.8774E+5	0.250	64	7.4852E+2	0.250
25	3.0197E+5	0.250	65	5.8295E+2	0.250
26	2.3518E+5	0.250	66	4.5400E+2	0.250
27	1.8316E+5	0.250	67	3.5358E+2	0.250
28	1.4264E+5	0.250	68	2.7536E+2	0.500
29	1.1109E+5	0.250	69	1.6702E+2	0.500
30	8.6517E+4	0.250	70	1.0130E+2	0.500
31	6.7380E+4	0.250	71	6.1442E+1	0.500
32	5.2475E+4	0.250	72	3.7266E+1	0.500
33	4.0868E+4	0.250	73	2.2603E+1	0.500
34	3.1828E+4	0.125	74	1.3710E+1	0.500
35	2.8088E+4	0.075	75	8.3153E+0	0.500
36	2.6058E+4	0.050	76	5.0435E+0	0.500
37	2.4788E+4	0.125	77	3.0590E+0	1.000
38	2.1875E+4	0.125	78	1.1253E+0	1.000
39	1.9304E+4	0.125	79	4.1399E-1	1.000
40	1.7036E+4	0.125	80	1.5230E-1	7.000
			Minimum	1.3888E-4	

Table 2-5
Group Structure for 26-Group Neutron Cross-Section Data

Figure 2-1. Schematic View of (r, θ) Model

Figure 2-2. Schematic View of (r, z) Model

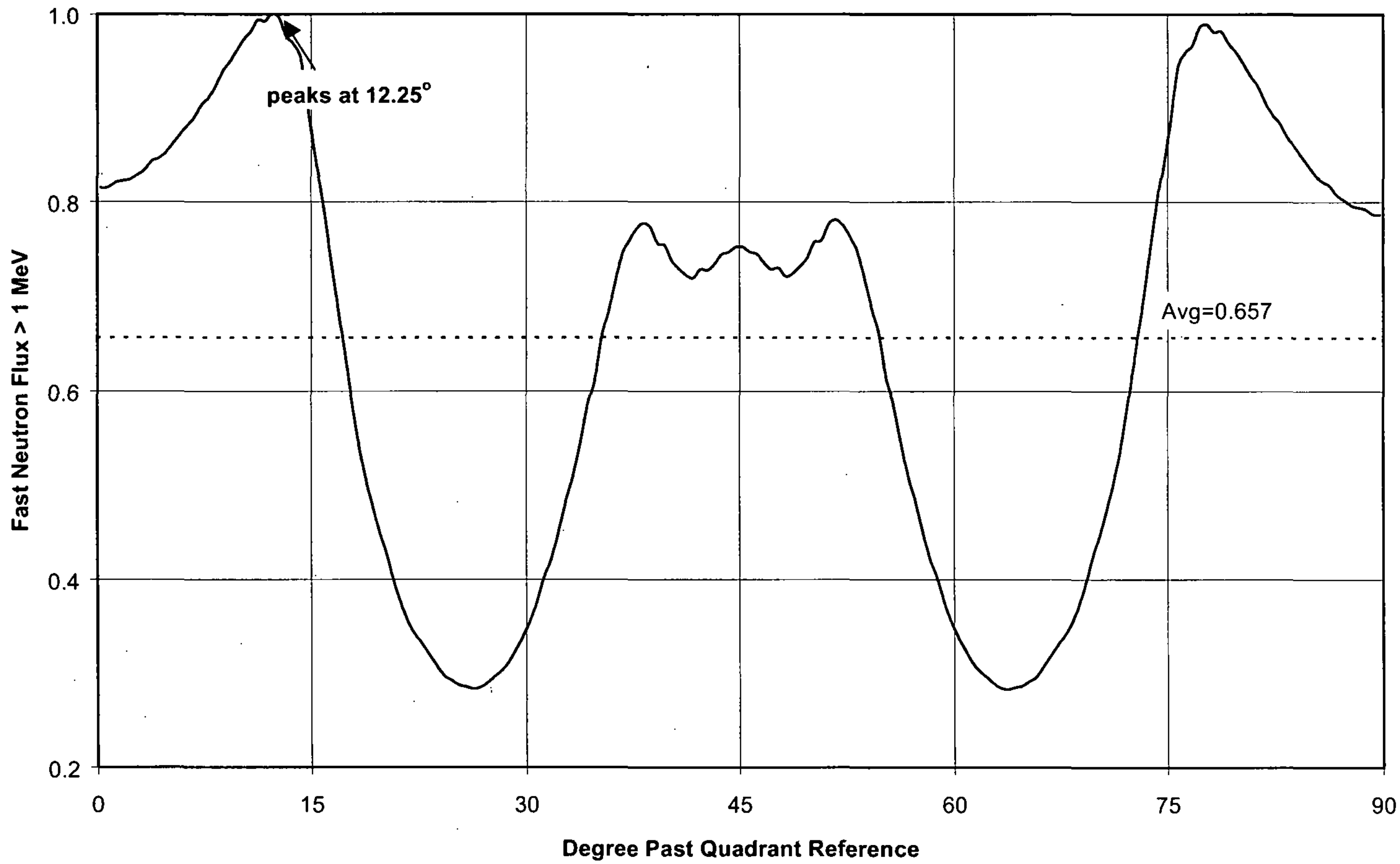

Figure 2-3. Sample Relative Neutron Flux (E>1 MeV) vs. Azimuth at Shroud ID

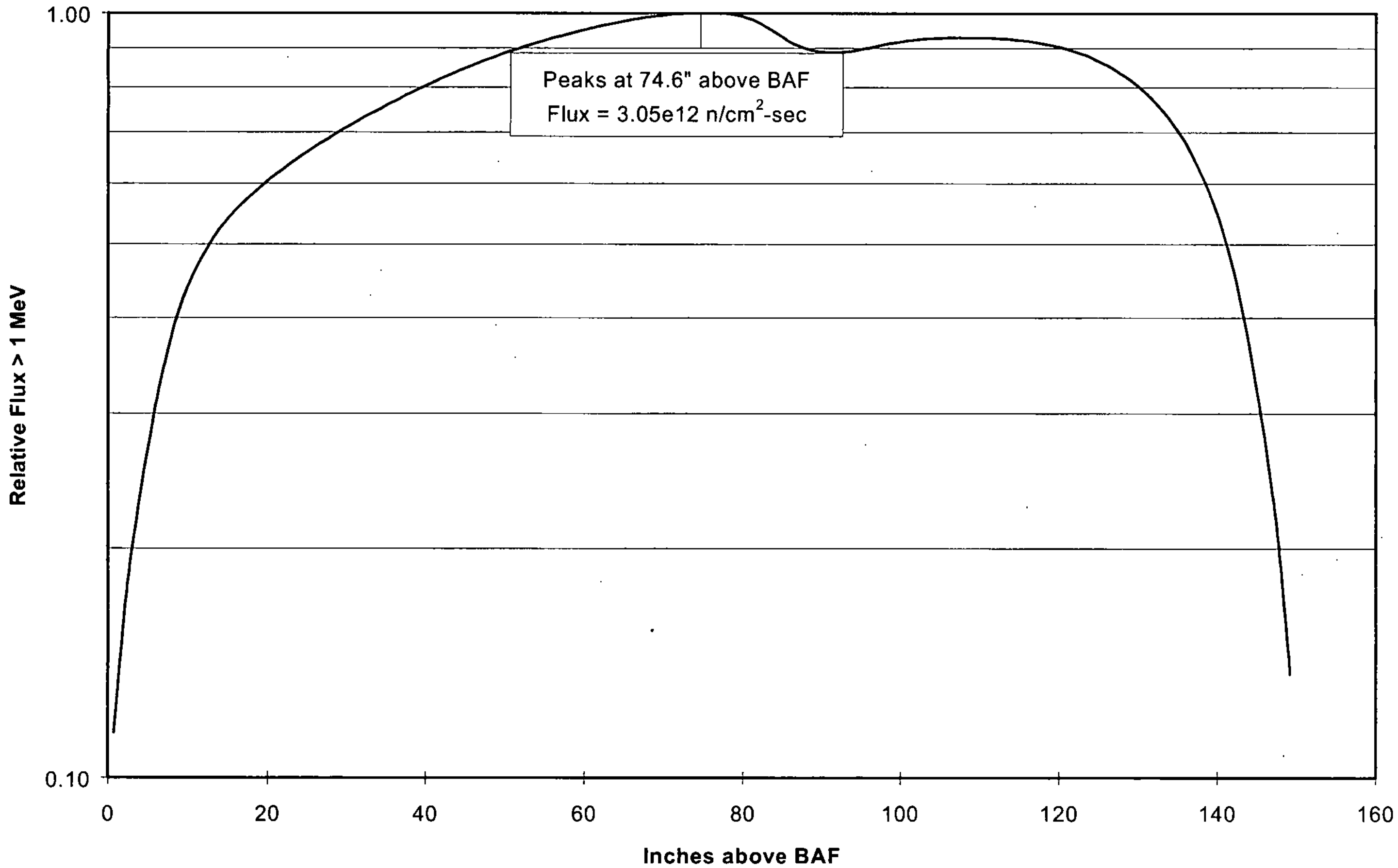

Figure 2-4. Sample Relative Neutron Flux (E>1 MeV) vs. Axial Elevation at Shroud ID

3.0 GE SOLUTIONS TO BWR BENCHMARK PROBLEM

This section documents the GE discrete ordinates solutions to the BWR benchmark problem as defined in NUREG/CR-6115, which was published by Brookhaven National Laboratory (BNL) for the NRC as one of the benchmarks for validating fluence calculation methodologies.

The solution provided in NUREG/CR-6115 (herein called the BNL solution) determines the RPV flux by combining the results of DORT(r,θ), DORT(r,z), and DORT(r) calculations. The GE solution to the same benchmark problem, using similar discrete ordinates methodology is described below.

3.1 GE SOLUTION USING DISCRETE ORDINATES METHOD

3.1.1 Core Configuration

The benchmark problem is modeled for a BWR core with 800 fuel bundles. The active core height is 381 cm. The core configuration, including radii of various material regions, are described in Figures 3-1 and 3-2 for the (r,θ) and (r,z) calculations, respectively. The RPV has an inner radius of 321.786 cm and is 16.129 cm thick, with a 0.476 cm stainless steel liner on the inner surface.

The (r,θ) calculation models the core and outward surroundings including the shroud, jet pumps and risers, the RPV, and an outer concrete biological shield. To be consistent with the BNL solution edits, where the RPV peak flux occurs at elevation 306.605 cm, the (r,θ) calculation model is selected at the same elevation.

In addition to the components modeled in the (r,θ) calculation, the (r,z) calculation models the core inlet and core plate which are below the active fuel region as well as upper reflector, top guide, and steam separator, which are above the core. The (r,z) calculation model is chosen to simulate the core dimension at 40.24° azimuth where the maximum core radius occurs.

Table 3-1 shows the radial, axial, and azimuthal meshes used in the GE DORT(r,θ) and (r,z) calculations. These data are almost identical to those in Table 4.2.2.1 of NUREG/CR-6115. The only exceptions are the inner and outer radii of the shroud, which were listed incorrectly in NUREG/CR-6115. Hence, GE made the necessary corrections to these data. Similarly, radial meshes near the capsule were modified slightly in order to account for the inner surface as well as the centerline of the capsule.

3.1.2 Power Distribution

In the benchmark problem, the assumed bundle power and exposure simulated those of an equilibrium core. The relative power of each fuel bundle was given in Table 2.2.2.4 of NUREG/CR-6115. These relative powers had been rearranged and re-normalized so that the average power of a single bundle is one, as shown in Table 3-2. Since Table 3-2 shows a quadrant of the core, the values of diagonal bundles have been doubled from those in NUREG/CR-6115 to account for the actual relative power.

NUREG/CR-6115 also provided the nodal relative power in the axial direction. The core power distribution consists of three radial zones. For example, at elevation 306.605 cm (axial zone 18) the nodal power in the outermost layer is 0.0516 (or 1.29 times the core average, which is 0.04). The second layer is 0.0484 (1.21 times core average), and the inner core region is 0.0441 (1.1025 times core average). Since the DORT(r,θ) calculation is performed at 306.605 cm, these axial factors are multiplied by the relative power density in Table 3-2. The resulting data are listed in Table 3-3.

To simulate the BNL solution, one set of GE (r,θ) solutions also assumes a pin-by-pin or 8x8 power grid for each bundle at the outermost row, a 4x4 power grid for each second tier bundle, and 2x2 power grid for each third tier bundle. The remaining core regions are assumed to have uniform power density within the bundle. A second set of GE solution takes the traditional GE approach of assuming bundle-average power for inner core as well as for peripheral bundles. The end results of these two sets of solution will be compared to justify the calculation bias.

The axial power distribution for the (r,z) calculation combines the radial zone-average of Table 3-2 with the normalized axial power given in Table 2.2.2.6 of NUREG/CR-6115. The resulting power density map used for the actual calculation is shown in Table 3-4.

3.1.3 Material Composition and Coolant Density

The BWR benchmark problem assumes that each of the 800 fuel bundles consists of 62 fuel rods plus 2 water rods. The fuel channel thickness is 0.3048 cm, channel OD is 13.8557 cm. The assembly pitch is 15.24 cm (6") and fuel pin pitch is 1.61544 cm.

These volume fractions were used to generate the weighted atom density for each of the inner and outer core zones. The results are listed in Table 3-5. The downcomer is modeled in three different ways and details of which are addressed in Section 3.1.6.

3.1.4 Cross-Section Processing

Nuclear cross-section data processing in the GE discrete ordinates method has been described in Section 2.1.6. The atom densities of Table 3-5 are used as the basis for generating macroscopic cross sections for the DORT calculations.

The isotopic fractions of these isotopes, 5.9%, 91.72%, 2.1%, and 0.28%, respectively, are taken from the "Nuclides and Isotopes"[6].

3.1.5 Neutron Source Calculation

A fixed source distribution proportional to the power density is assumed for the neutron source. Since the actual neutron source strength was not provided in NUREG/CR-6115, GE performed a lattice depletion calculation with an assumed 8x8 array fuel of 3.78% average enrichment.

where

3833 MWt is the thermal power level.

0.125 represents 1/8th of the core.

381 cm is the total height of the active core.

For the (r,z) calculation, the goal is to generate a relative, rather than absolute, flux magnitude. Therefore a precise neutron source description is not essential. A typical number of 1.0E20 n/sec was used as the fixed source input for the calculation.

3.1.6 Treatment of Downcomer Region

Precise modeling of the BWR downcomer components and materials is not practical in a two-dimensional calculation. In order to assess the effect of neutron interactions in the downcomer, three variations of downcomer model were used in the GE solutions of (r,θ) calculations.

The first model treats the downcomer region as composed solely of subcooled water, without any metal components. This is a conservative approach traditionally employed by GE.

The second model assumes that the downcomer is a homogenized mixture of coolant and metal, the effective neutron scattering by steel is accounted for to a certain extent.

The third model considers the downcomer as composed of heterogeneous material zones, with each jet pump and riser as individual component.

The results of these three downcomer models are presented in the next section and in Section 3.2 as part of the sensitivity studies.

For the (r,z) calculation, the second model is used to simulate the material compositions in the downcomer region.

3.1.7 GE Discrete Ordinates Solution Results

GE calculated flux results (E>1 MeV) using the discrete ordinates method described above are presented in the following figures:

Figure 3-3	Downcomer Flux with Various Downcomer Models
Figure 3-4	RPV ID Flux with Various Downcomer Models
Figure 3-5	RPV 1/4T Flux with Various Downcomer Models
Figure 3-6	RPV T Flux with Various Downcomer Models
Figure 3-7	Axial Flux Profile at Various Radial Locations
Figure 3-8	GE vs. BNL Azimuthal Flux Profile at Downcomer
Figure 3-9	GE vs. BNL Azimuthal Flux Profile at RPV ID
Figure 3-10	GE vs. BNL Flux Spectra at RPV ID
Figure 3-11	GE vs. BNL Flux Spectra at RPV 1/2T
Figure 3-12	GE vs. BNL Flux Spectra at Capsule

The extracted edits are listed in Table 3-6, together with the results of the sensitivity studies described in Section 3.2. As mentioned in Section 3.1.2, the pin-by-pin power densities are used for the three downcomer models. A fourth calculation was performed to validate the current GE methodology. This calculation uses the bundle-average power in conjunction with the third downcomer model and is designated Case 4 in Table 3-6. Results of Case 4 are used as the basis of comparison for all other cases.

Case 6 simulates the traditional model employed by GE, except it uses ENDF/B-VI library instead of the ENDF/B-V library. The over-prediction of vessel ID peak flux by this model should be less than 17% (difference between Case 6 and BNL minus Case 9). As stated above, for BWRs with different configurations, the over-prediction should be less if the locations of jet pumps are not coincident with the peak flux azimuth.

The calculation model of Case 3 is almost identical to that of BNL solution. Consequently, the peak vessel ID flux is within 0.5% of the BNL result.

Comparisons between the results of current GE methodology and BNL solution are presented in Figures 3-8 through 3-12. Along the circumference of the RPV inner surface, the two solutions differ by less than 10%. However, both solutions predicted the same peak flux locations at 42.5°.

3.1.8 Flux >0.1 MeV

Figures 3-13 and 3-14 present the comparison between the GE and BNL solutions for the >0.1 MeV fluxes at the downcomer and RPV ID, respectively. The differences between the two solutions are consistent with those displayed in Figures 3-8 and 3-9 for neutron flux above 1 MeV.

3.2 SENSITIVITY STUDIES

Sensitivity studies of variables in the discrete ordinates calculations were performed and the effect of each variable is assessed. The study results are summarized in Table 3-6. The most significant among these variables is the treatment of downcomer components, which was discussed in the previous section. Overestimation of vessel flux due to omission of steel components in downcomer is especially prominent if the flux peaks at locations shielded by a jet pump or riser component, as is the case here.

The following figures demonstrate the effects of calculation variables such as cross section library, pin-by-pin power distribution, and angular quadratures:

Figure 3-15	ENDF/B-VI vs. ENDF/B-V Flux at RPV ID
Figure 3-16	ENDF/B-VI vs. ENDF/B-V Flux at RPV 1/4T
Figure 3-17	ENDF/B-VI vs. ENDF/B-V Flux at RPV T
Figure 3-18	Pin-By-Pin vs. Bundle-Average Flux at Shroud ID
Figure 3-19	Pin-By-Pin vs. Bundle-Average Flux at RPV ID
Figure 3-20	S_{12} vs. S_8 Flux at Shroud ID
Figure 3-21	S_{12} vs. S_8 Flux at RPV ID

The results of a sensitivity study of the effects of cross section library on the fast neutron flux are presented in Table 3-7. The base case uses ENDF/B-V library (Case 7). Case 8 uses ENDF/B-V library overridden with ENDF/B-VI iron cross sections. Case 9 uses ENDF/B-V library overridden with ENDF/B-VI iron, oxygen, and hydrogen cross sections. The calculation model for this study is consistent within the study group, however it is slightly different from those of Table 3-6. Therefore, the comparisons are only made within the group.

The effect of pin-by-pin power vs bundle-average power is demonstrated through Figures 3-18 and 3-19, and Table 3-6.

When this effect is taken into consideration, the traditional GE method produces a flux value almost identical to the BNL result in the downcomer near the shroud outer surface, as indicated in Table 3-6 Case 6.

Angular quadrature higher than S_8 produces less than 1% change in the calculated flux, as shown in Figures 3-20 and 3-21 as well as Table 3-6 Case 3a. This conclusion is consistent with that provided by Table 3.4.2 of NUREG/CR-6115.

Using a reflective boundary condition to approximate the innermost core region provides almost identical flux results as a full core model, as demonstrated in Table 3-6 Case 5. This is further proof that the traditional practice adopted by GE to economize computational effort did not sacrifice the accuracy of calculated flux.

3.3 CONCLUSIONS

GE solutions to the BWR benchmark problem were performed using the GE controlled version of the DORT code, with various calculation models simulating the peripheral bundle powers and downcomer material compositions.

In the downcomer near the outer surface of the shroud, GE and BNL solutions produce almost identical flux results.

The methodology traditionally employed by GE produces peak vessel flux approximately 16% higher than the BNL result, due to the placement of jet pump riser which coincides with the peak flux azimuth. With the same method, when the conservatism created by bundle-average power is excluded, the traditional GE approach and BNL solution produce almost identical flux results in the downcomer near the outer surface of the shroud.

Table 3-1
Meshes for DORT Calculations

Node	R (cm)	Z (cm)	θ (revolution)
1	0.00000	0.00000	0.00000E+00
2	7.50000	3.09750	6.24722E-03
3	15.0000	6.19500	7.63889E-03
4	22.5000	9.29250	9.02778E-03
5	30.0000	12.3900	1.04194E-02
6	37.5000	15.4767	1.37889E-02
7	45.0000	18.5633	1.53167E-02
8	52.5000	21.6500	1.79556E-02
9	60.0000	24.7367	1.95444E-02
10	67.5000	27.8233	2.00389E-02
11	75.0000	30.9100	2.06556E-02
12	82.5000	33.9967	2.15389E-02
13	90.0000	37.0833	2.36278E-02
14	97.5000	40.1700	2.37944E-02
15	105.000	43.2567	2.47944E-02
16	112.500	46.3433	2.63556E-02
17	120.000	49.4300	3.06500E-02
18	127.500	52.4780	3.39278E-02
19	135.000	55.5260	3.89833E-02
20	142.500	58.5740	4.15722E-02
21	150.000	61.6220	4.45389E-02
22	157.500	64.6700	4.71278E-02
23	165.000	67.7180	4.94056E-02
24	172.500	70.7660	5.26833E-02
25	180.000	73.8140	5.56500E-02
26	182.602	76.8620	5.67889E-02
27	185.560	79.9100	6.12056E-02
28	186.709	82.9580	6.37944E-02
29	187.785	86.0060	6.67611E-02
30	188.896	89.0540	6.80222E-02
31	189.782	92.1020	6.88444E-02
32	190.616	95.1500	7.01389E-02
33	191.625	98.1980	7.16222E-02
34	192.706	101.246	7.21278E-02
35	193.369	104.294	7.54611E-02
36	194.008	107.342	7.76833E-02
37	194.782	110.390	7.95444E-02
38	195.444	113.438	8.04611E-02
39	196.207	116.486	8.20389E-02
40	198.098	119.534	8.32111E-02
41	198.825	122.582	8.48778E-02
42	199.580	125.630	8.60167E-02
43	200.562	128.678	8.75944E-02

Table 3-1
Meshes for DORT Calculations (Continued)

Node	R (cm)	Z (cm)	θ (revolution)
44	201.416	131.726	8.80056E-02
45	202.458	134.774	9.03722E-02
46	203.446	137.822	9.29611E-02
47	204.500	140.870	9.42056E-02
48	205.506	143.918	9.71278E-02
49	206.520	146.966	9.86445E-02
50	207.440	150.014	1.01506E-01
51	208.610	153.062	1.05650E-01
52	209.491	156.110	1.09689E-01
53	210.474	159.158	1.11206E-01
54	211.606	162.206	1.12328E-01
55	212.585	165.254	1.15372E-01
56	213.795	168.302	1.17356E-01
57	214.611	171.350	1.18811E-01
58	215.643	174.398	1.22222E-01
59	216.308	177.446	1.23611E-01
60	217.732	180.494	1.24861E-01
61	218.318	183.542	1.25000E-01
62	219.483	186.590	
63	220.282	189.638	
64	221.543	192.686	
65	222.394	195.734	
66	223.735	198.782	
67	225.048	201.830	
68	226.423	204.878	
69	227.237	207.926	
70	228.431	210.974	
71	229.450	214.022	
72	230.719	217.070	
73	231.344	202.118	
74	232.630	223.116	
75	233.895	226.214	
76	234.874	229.262	
77	235.625	232.310	
78	236.176	235.430	
79	237.228	238.430	
80	238.470	241.430	
81	239.300	244.430	
82	240.227	247.550	
83	240.783	250.598	
84	241.970	253.646	
85	242.570	256.694	
86	243.672	259.742	

Table 3-1
Meshes for DORT Calculations (Continued)

Node	R (cm)	Z (cm)	θ (revolution)
87	244.461	262.790	
88	244.974	265.838	
89	245.616	268.886	
90	247.540	271.934	
91	248.091	274.982	
92	249.019	278.030	
93	250.597	281.078	
94	251.148	284.126	
95	252.568	287.174	
96	252.267	290.222	
97	254.077	293.270	
98	255.598	297.080	
99	257.727	300.890	
100	258.519	304.700	
101	259.530	308.510	
102	260.326	310.430	
103	261.122	313.430	
104	261.918	316.430	
105	262.715	319.430	
106	263.511	323.750	
107	264.307	326.798	
108	265.103	329.846	
109	265.899	332.894	
110	266.695	335.942	
111	267.491	338.990	
112	268.288	344.070	
113	269.557	349.150	
114	270.288	354.230	
115	272.098	359.310	
116	273.367	364.390	
117	274.073	369.470	
118	274.778	374.550	
119	275.368	379.630	
120	276.189	384.710	
121	276.895	389.790	
122	277.600	394.870	
123	278.600	399.950	
124	278.878	405.030	
125	279.627	410.110	
126	280.376	415.190	
127	281.125	420.270	
128	281.875	425.350	

Table 3-1
Meshes for DORT Calculations (Continued)

Node	R (cm)	Z (cm)	θ (revolution)
129	282.624	430.430	
130	283.454	436.603	
131	284.284	442.777	
132	285.114	448.950	
133	285.944	455.123	
134	286.774	461.297	
135	287.604	467.470	
136	288.435	473.643	
137	289.265	479.817	
138	290.095	485.990	
139	290.925	495.637	
140	291.755	505.284	
141	292.585	514.931	
142	293.415	524.578	
143	294.245	534.225	
144	295.075	543.872	
145	295.905	553.519	
146	296.735	563.166	
147	297.565	572.814	
148	298.396	582.461	
149	299.226	592.108	
150	300.056	601.755	
151	300.886	611.402	
152	301.716	621.049	
153	302.546	630.696	
154	303.376	640.343	
155	304.125	649.990	
156	304.875	659.637	
157	305.624	669.284	
158	306.373	678.931	
159	307.122	688.578	
160	307.919	698.225	
161	308.716	707.872	
162	309.513	717.519	
163	310.310	727.166	
164	311.156	736.813	
165	312.002	746.461	
166	312.848	756.108	
167	313.695	765.755	
168	314.541	775.402	
169	315.387	785.049	
170	316.233	794.696	

Table 3-1
Meshes for DORT Calculations (Continued)

Node	R (cm)	Z (cm)	θ (revolution)
171	317.079	804.343	
172	317.925	813.990	
173	318.772		
174	319.618		
175	320.464		
176	321.310		
177	321.786		
178	322.786		
179	324.500		
180	325.318		
181	326.318		
182	327.834		
183	329.351		
184	330.351		
185	331.867		
186	333.383		
187	334.383		
188	336.149		
189	337.915		
190	340.790		
191	346.290		
192	351.790		
193	351.949		
194	354.806		
195	357.663		
196	360.520		
197	365.611		
198	370.701		
199	375.792		
200	380.883		
201	385.973		
202	391.064		
203	396.155		
204	401.245		
205	406.336		
206	411.427		
207	416.517		
208	421.608		
209	426.699		
210	431.789		

Table 3-1
Meshes for DORT Calculations (Continued)

Node	R (cm)	Z (cm)	θ (revolution)
211	436.880		
212	437.500		
213	441.500		
214	445.500		
215	449.500		
216	453.500		
217	457.500		
218	461.500		
219	465.500		
220	469.500		
221	473.500		
222	477.500		

Table 3-2
BWR Benchmark Problem Bundle Power Density

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
1	1.007	1.074	1.058	1.114	1.101	1.107	1.356	1.082	1.08	1.076	1.317	1.063	1.192	0.82	0.602	0.326
2	1.074	1.292	1.122	1.333	1.108	1.358	1.116	1.323	1.078	1.319	1.096	1.288	0.991	0.999	0.587	0.32
3	1.058	1.122	1.125	1.114	1.36	1.128	1.344	1.087	1.109	1.105	1.33	1.058	1.173	0.791	0.542	
4	1.114	1.333	1.114	1.346	1.101	1.331	1.089	1.117	1.098	1.34	1.098	1.27	0.968	0.94	0.471	
5	1.101	1.108	1.36	1.101	1.094	1.083	1.106	1.096	1.344	1.109	1.318	1.041	1.122	0.735	0.438	
6	1.107	1.358	1.128	1.331	1.083	1.325	1.102	1.341	1.114	1.328	1.065	1.205	0.898	0.836	0.392	
7	1.356	1.116	1.344	1.089	1.106	1.102	1.343	1.101	1.325	1.079	1.24	0.947	0.971	0.586	0.327	
8	1.082	1.323	1.087	1.117	1.096	1.341	1.101	1.083	1.068	1.251	0.979	1.037	0.668	0.428		
9	1.08	1.078	1.109	1.098	1.344	1.114	1.325	1.068	1.25	0.997	1.079	0.739	0.484			
10	1.076	1.319	1.105	1.34	1.109	1.328	1.079	1.251	0.997	0.544	0.781	0.605	0.361			
11	1.317	1.096	1.33	1.098	1.318	1.065	1.24	0.979	1.079	0.781	0.629	0.462	0.265			
12	1.063	1.288	1.058	1.27	1.041	1.205	0.947	1.037	0.739	0.605	0.462	0.32				
13	1.192	0.991	1.173	0.968	1.122	0.898	0.971	0.668	0.484	0.361	0.265					
14	0.82	0.999	0.791	0.94	0.735	0.836	0.586	0.428								
15	0.602	0.587	0.542	0.471	0.438	0.392	0.327									
16	0.326	0.32														

Average		
	R1	1.1346
	R2	0.6864
	R3	0.3926

Table 3-3
Power Density at Elevation z=306.6 cm for DORT(r,θ) Calculation

I\J	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
1	1.1103	1.1835	1.1662	1.2279	1.2135	1.2207	1.4953	1.1926	1.1901	1.1866	1.4514	1.1715	1.3137	0.9042	0.7279	0.4211
2	1.1835	1.4248	1.2367	1.4693	1.2220	1.4976	1.2308	1.4590	1.1886	1.4544	1.2081	1.4197	1.0929	1.1016	0.7102	0.4129
3	1.1662	1.2367	1.2405	1.2279	1.4994	1.2432	1.4822	1.1984	1.2229	1.2183	1.4659	1.1661	1.2937	0.9574	0.6989	
4	1.2279	1.4693	1.2279	1.4841	1.2136	1.4669	1.2006	1.2314	1.2101	1.4778	1.2105	1.3998	1.0673	1.1379	0.6080	
5	1.2135	1.2220	1.4994	1.2136	1.2056	1.1942	1.2195	1.2081	1.4815	1.2225	1.4525	1.1479	1.2371	0.8889	0.5647	
6	1.2207	1.4976	1.2432	1.4669	1.1942	1.4611	1.2144	1.4783	1.2281	1.4640	1.1745	1.3288	0.9898	1.0117	0.5053	
7	1.4953	1.2308	1.4822	1.2006	1.2195	1.2144	1.4809	1.2133	1.4603	1.1897	1.3675	1.0444	1.0707	0.7096	0.4224	
8	1.1926	1.4590	1.1984	1.2314	1.2081	1.4783	1.2133	1.1942	1.1774	1.3791	1.0798	1.1435	0.8086	0.5523		
9	1.1901	1.1886	1.2229	1.2101	1.4815	1.2281	1.4603	1.1774	1.3777	1.0988	1.1898	0.8940	0.6245			
10	1.1866	1.4544	1.2183	1.4778	1.2225	1.4640	1.1897	1.3791	1.0988	0.6003	0.8611	0.7315	0.4653			
11	1.4514	1.2081	1.4659	1.2105	1.4525	1.1745	1.3675	1.0798	1.1898	0.8611	0.6940	0.5590	0.3422			
12	1.1715	1.4197	1.1661	1.3998	1.1479	1.3288	1.0444	1.1435	0.8940	0.7315	0.5590	0.4133				
13	1.3137	1.0929	1.2937	1.0673	1.2371	0.9898	1.0707	0.8086	0.6245	0.4653	0.3422					
14	0.9042	1.1016	0.9574	1.1379	0.8889	1.0117	0.7096	0.5523								
15	0.7279	0.7102	0.6989	0.6080	0.5647	0.5053	0.4224									
16	0.4211	0.4129														

Table 3-4
Power Density for (r,z) Calculation

Node	R1	R2	R3
1	0.0000	0.0000	0.0000
2	0.0144	0.0069	0.0039
3	0.0455	0.0211	0.0104
4	0.0564	0.0271	0.0133
5	0.0589	0.0295	0.0146
6	0.0574	0.0301	0.0150
7	0.0548	0.0302	0.0154
8	0.0525	0.0301	0.0157
9	0.0509	0.0302	0.0160
10	0.0499	0.0304	0.0164
11	0.0494	0.0307	0.0169
12	0.0490	0.0311	0.0174
13	0.0496	0.0316	0.0179
14	0.0500	0.0322	0.0185
15	0.0498	0.0327	0.0192
16	0.0505	0.0331	0.0196
17	0.0508	0.0334	0.0200
18	0.0500	0.0332	0.0203
19	0.0491	0.0328	0.0203
20	0.0488	0.0323	0.0199
21	0.0467	0.0309	0.0193
22	0.0424	0.0284	0.0181
23	0.0395	0.0256	0.0163
24	0.0332	0.0212	0.0136
25	0.0243	0.0155	0.0101
26	0.0103	0.0062	0.0044

Table 3-5
Material Compositions

Mixture	Element	# atom/b-cm
INNER CORE 1 (+WATER)	U-235	9.9196E-05
	U-238	5.3633E-03
	O (fuel)	1.0927E-02
	Zr	6.2833E-03
	H	3.0048E-02
	O	1.4269E-02
INNER CORE 2 (+WATER)	U-235	9.9196E-05
	U-238	5.3633E-03
	O (fuel)	1.0927E-02
	Zr	6.2833E-03
	H	2.7789E-02
	O	1.3894E-02
INNER CORE 3 (+WATER)	U-235	9.9196E-05
	U-238	5.3633E-03
	O (fuel)	1.0927E-02
	Zr	6.2833E-03
	H	2.2109E-02
	O	1.1054E-02
INNER CORE 4 (+WATER)	U-235	9.9196E-05
	U-238	5.3633E-03
	O (fuel)	1.0927E-02
	Zr	6.2833E-03
	H	1.8516E-02
	O	9.2576E-03
INNER CORE 5 (+WATER)	U-235	9.9196E-05
	U-238	5.3633E-03
	O (fuel)	1.0927E-02
	Zr	6.2833E-03
	H	1.6170E-02
	O	8.0849E-03
INNER CORE 6 (+WATER)	U-235	9.9196E-05
	U-238	5.3633E-03
	O (fuel)	1.0927E-02
	Zr	6.2833E-03
	H	1.4664E-02
	O	7.3319E-03
INNER CORE 7 (+WATER)	U-235	9.9196E-05
	U-238	5.3633E-03
	O (fuel)	1.0927E-02
	Zr	6.2833E-03
	H	1.3696E-02
	O	6.8478E-03

Table 3-5
Material Compositions (Continued)

Mixture	Element	# atom/b-cm
OUTER CORE 1 (+WATER)	U-235	9.9196E-05
	U-238	5.3633E-03
	O (fuel)	1.0927E-02
	Zr	6.2833E-03
	H	3.0048E-02
	O	1.5024E-02
OUTER CORE 2 (+WATER)	U-235	9.9196E-05
	U-238	5.3633E-03
	O (fuel)	1.0927E-02
	Zr	6.2833E-03
	H	3.0012E-02
	O	1.5006E-02
OUTER CORE 3 (+WATER)	U-235	9.9196E-05
	U-238	5.3633E-03
	O (fuel)	1.0927E-02
	Zr	6.2833E-03
	H	2.8166E-02
	O	1.4083E-02
OUTER CORE 4 (+WATER)	U-235	9.9196E-05
	U-238	5.3633E-03
	O (fuel)	1.0927E-02
	Zr	6.2833E-03
	H	2.4841E-02
	O	1.2421E-02
OUTER CORE 5 (+WATER)	U-235	9.9196E-05
	U-238	5.3633E-03
	O (fuel)	1.0927E-02
	Zr	6.2833E-03
	H	2.1441E-02
	O	1.0721E-02
OUTER CORE 6 (+WATER)	U-235	9.9196E-05
	U-238	5.3633E-03
	O (fuel)	1.0927E-02
	Zr	6.2833E-03
	H	1.9053E-02
	O	9.5266E-03
OUTER CORE 7 (+WATER)	U-235	9.9196E-05
	U-238	5.3633E-03
	O (fuel)	1.0927E-02
	Zr	6.2833E-03
	H	1.8042E-02
	O	9.0210E-03

Table 3-5
Material Compositions (Continued)

Mixture	Element	# atom/b-cm
Reflector Water	H	4.9284E-02
	O	2.4642E-02
Shroud (SS-304)	Fe	5.8300E-02
	Cr	1.7400E-02
	Ni	8.5500E-03
	Mn	1.5200E-03
	Si	8.9300E-04
	C	2.3700E-04
Downcomer Model 1 (Water only)	H	5.0455E-02
	O	2.5228E-02
Downcomer Model 2 (Water +Jet Pump)	H	4.8801E-02
	O	2.4401E-02
	Fe	2.0904E-03
	Ni	2.2599E-04
	Cr	5.0849E-04
RPV Liner (SS-304)	Fe	5.8300E-02
	Cr	1.7400E-02
	Ni	8.5500E-03
	Mn	1.5200E-03
	Si	8.9300E-04
	C	2.3700E-04
RPV Wall (Steel)	Fe	8.1900E-02
	Mn	1.1200E-03
	Ni	4.4400E-04
	Cr	1.2700E-04
	C	9.8100E-04
	Si	3.7100E-04
Cavity	O	9.6200E-06
insulation Liner (SS-304)	Fe	5.8300E-02
	Cr	1.7400E-02
	Ni	8.5500E-03
	Mn	1.5200E-03
	Si	8.9300E-04
	C	2.3700E-04

Table 3-5
Material Compositions (Continued)

Mixture	Element	# atom/b-cm
Insulation	A1	6.0603E-03
Cavity	O	9.6200E-06
Concrete Wall	Fe	6.0976E-04
	H	1.5137E-02
	C	2.2403E-04
	O	8.5327E-02
	Na	2.0455E-03
	Mg	2.8832E-04
	A1	4.6560E-03
	Si	3.0778E-02
	K	1.3500E-03
	Ca	4.4612E-03
Inlet Region	H	3.5415E-02
	O	1.7708E-02
	Zr	7.9747E-03
	Cr	1.9749E-03
	Mn	1.7252E-04
	Fe	6.6171E-03
	Ni	9.7043E-04
	Si	1.0136E-04
	C	2.6900E-05
Core Plate	H	4.6642E-02
	O	2.3321E-02
	Cr	1.3154E-03
	Mn	1.1491E-04
	Fe	4.4075E-03
	Ni	6.4638E-04
	Si	6.7511E-05
	C	1.7917E-05
Top	H	1.2153E-02
	O	6.0767E-03
	Zr	7.6896E-03
Upper Reflector	H	9.8223E-03
	O	4.9112E-03
	Zr	7.5125E-03
	Cr	2.8153E-03
	Mn	2.4594E-04
	Fe	9.4329E-03
	Ni	1.3834E-04
	Si	1.4449E-04
	C	3.8347E-05
Steam Separater	H	1.4785E-02
	O	7.3926E-03

Table 3-6
Sensitivity of DORT Calculated Flux (E>1MeV) to Input Parameters

Table 3-7
Sensitivity of Varying Cross Section Library on Flux at 44.97°

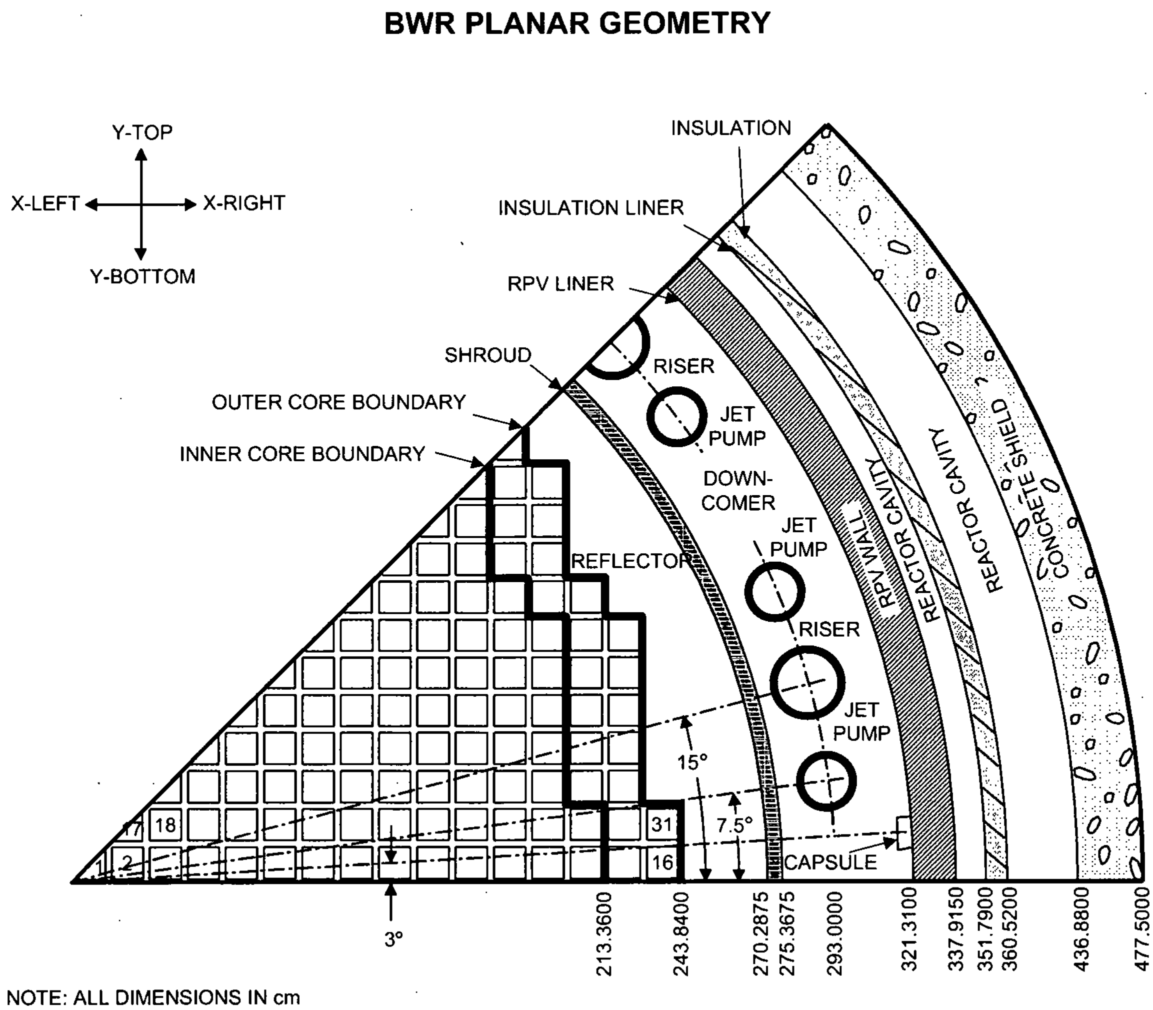

Figure 3-1. BWR Planar Geometry

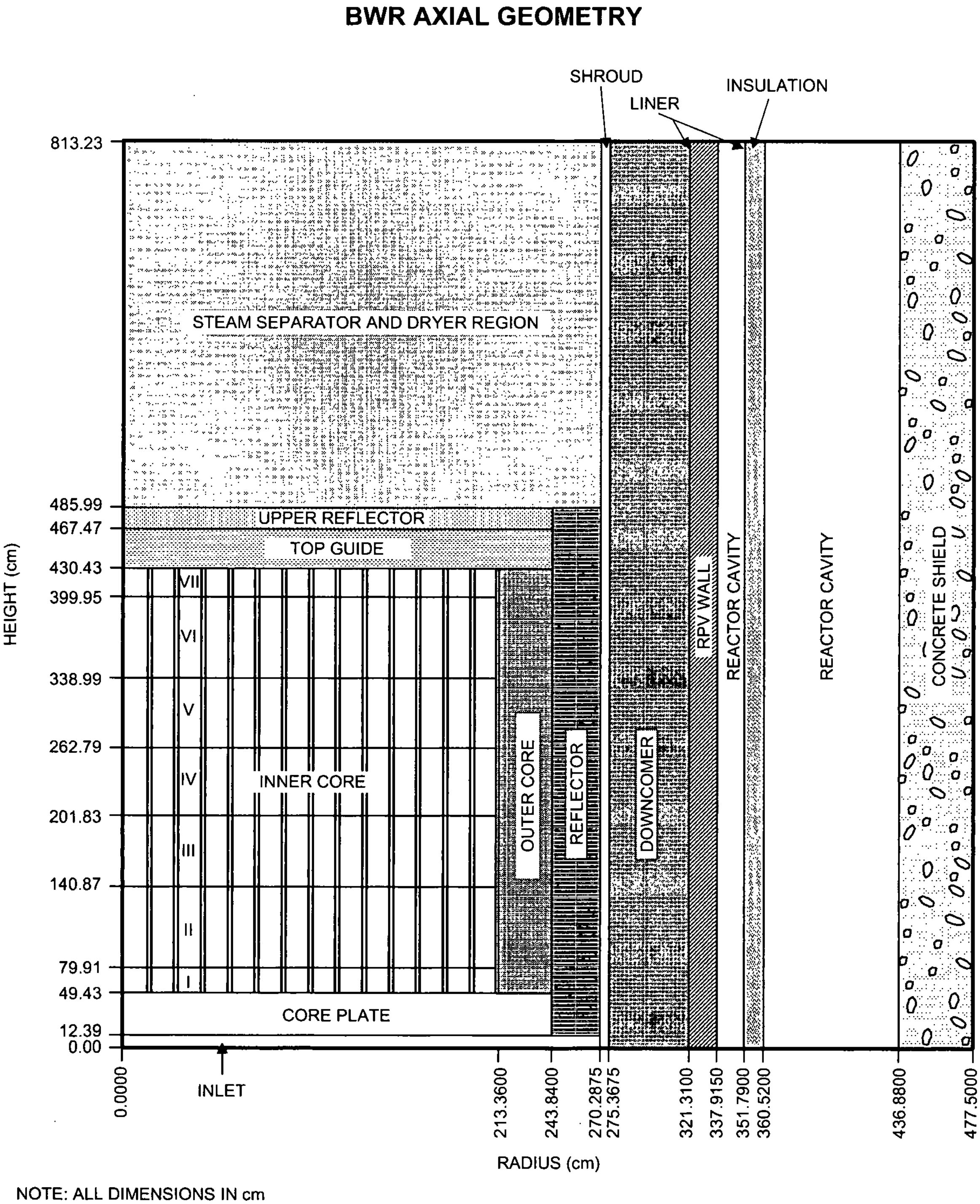

Figure 3-2. BWR Axial Geometry

Figure 3-3. Downcomer Flux (E>1 MeV) with Various Downcomer Models

Figure 3-4. RPV ID Flux (E>1 MeV) with Various Downcomer Models

Figure 3-5. RPV 1/4T Flux (E>1 MeV) with Various Downcomer Models

Figure 3-6. RPV T Flux (E>1 MeV) with Various Downcomer Models

Figure 3-7. Axial Flux Profile (E>1 MeV) at Various Radial Locations

Figure 3-8. GE vs. BNL Azimuthal Flux Profile (E>1 MeV) at Downcomer

Figure 3-9. GE vs. BNL Azimuthal Flux Profile (E>1 MeV) at RPV ID

Figure 3-10. GE vs. BNL Flux Spectra at RPV ID

Figure 3-11. GE vs. BNL Flux Spectra at RPV 1/2T

Figure 3-12. GE vs. BNL Flux Spectra at Capsule

Figure 3-13. GE vs. BNL Azimuthal Flux Profile (E>0.1 MeV) at Downcomer

Figure 3-14. GE vs. BNL Azimuthal Flux Profile (E>0.1 MeV) at RPV ID

Figure 3-15. ENDF/B-VI vs. ENDF/B-V Flux at RPV ID

Figure 3-16. ENDF/B-VI vs. ENDF/B-V Flux at RPV 1/4T

Figure 3-17. ENDF/B-VI vs. ENDF/B-V Flux at RPV OD

Figure 3-18. Pin-By-Pin vs. Bundle-Average Flux at Shroud ID

Figure 3-19. Pin-By-Pin vs. Bundle-Average Flux at RPV ID

Figure 3-20. S_{12} vs. S_8 Flux at Shroud ID

Figure 3-21. S_{12} vs. S_8 Flux at RPV ID

4.0 MONTE CARLO SOLUTION TO BWR BENCHMARK PROBLEM

4.1 INTRODUCTION

This section describes the GE Monte Carlo model of the benchmark problem developed for the NRC by BNL[2], using the code MCNP01A, a GE ECP version of MCNP[7]. The results obtained are compared with the BNL benchmark results. Two MCNP models are described: one where node average powers in the (r,θ) direction will be used to sample the source distribution, and the second where rod by rod power distributions will be used in varying degrees (2x2, 4x4, 8x8) for the three outermost sets of bundles. The calculations were performed using continuous energy ENDF/B-V data for all isotopes except oxygen (in water) and iron, where ENDF/B-VI data were used. Additionally, a third set of calculations was performed using ENDF/B-V data for all isotopes and a rod-by-rod description of the source term in the three outermost sets of bundles. A fourth set of calculations was performed for the rod-by-rod case to determine the sensitivity of the results to the actinide composition of the fuel.

4.2 ASSUMPTIONS

The model of the reactor and internals configurations is based on the description of the benchmark problem [2]. The actinide composition of the fuel is assumed to be uniform for all the fuel types and is taken to be that corresponding to a burnup of 14.12 GWd/t[2]. In the benchmark problem, the three burnup groups are given as 13.831, 13.788, and 12.755 GWd/t[2]. However, since the benchmark specifications did not provide data[2] at any of these specific values, a set closest to the 13.831 GWd/t value was taken to represent the actinide composition. A sensitivity case where the composition was changed to that corresponding to a burnup of 17.88 GWd/t was also made.

4.3 MODEL AND MATERIAL SPECIFICATIONS

The details of the geometric model can be found in Figures 3-1 and 3-2. The MCNP model does not include regions beyond the reactor cavity (just outside the RPV). From a standpoint of fuel composition, the model divided the core into an inner core and outer core in the (r,θ) plane, with the two outermost bundles assigned to the outer core, and all other bundles assigned to the inner core. Seven axial zones were modeled based on the description in Reference 2. An octant of the core and the ex-core regions was modeled with reflective boundary conditions.

The fuel material was modeled as homogenized fuel, clad, and in-channel water, and occupying a square inner box of dimension 12.924 cm on the side. The ex-channel water and channel zircalloy were homogenized into a second material that occupied everything outside of the fuel mixture in the node, which is the standard 15.24 cm on each side. The material compositions are given in Table 3-5, with the exception of the jet pumps and risers. The jet pumps and risers in the three-dimensional calculations were modeled explicitly and therefore used the material compositions presented for jet pump and riser metal in Reference 2.

As described above, two basic models were developed, one with only one fuel region in all bundles and the other with 2x2, 4x4, and 8x8 regions in the second bundle from the periphery, the first bundle from the periphery, and the peripheral bundles, respectively. In all cases, the fuel composition remained the same; the reason for dividing the fuel region into finer regions was to apply the appropriate power levels at each axial node. The first of these models is referred to as the "smeared" case and the second would be referred to as the "rod" case during the rest of this Chapter. The repeated-structures capability in MCNP was used in the core region for both models.

The reflector region, shroud, downcomer, RPV liner, and RPV were modeled with sub-regions in order to facilitate importance sampling as a variance reduction technique. Beyond the RPV outer wall, the cavity region was also modeled. In the axial direction, the region below the core plate, core plate, top guide region, upper reflector region, and the steam separator regions were modeled. Figures 4-1, 4-2, and 4-3 show the MCNP representation of the smeared and rod (r,θ) views and the (r,z) view (generally representative of both models), respectively.

4.4 SOURCE SPECIFICATIONS

The neutron source was modeled using the generalized source card and volume sources. The cell feature was used for both the smeared and the rod cases. The probability distribution for the cells was obtained from the source distribution provided in Reference 2. The general (r,θ) source distribution for the smeared case is presented in Table 3-2. This source distribution is re-normalized such that $1/8^{th}$ of the total core power equals 1.0.

In the rod case, the x-y source distributions of the outer three layers of bundles were obtained from Reference 2. However, the rod-by-rod numbers presented in the benchmark report were multiplied by a peak axial factor (1.2575 for the full sub-nodes and 2.5150 for the half sub-nodes) for use in the 2-D calculations. In order for the rod-by-rod numbers to add up to the proper node average power, the rod-by-rod numbers were divided by the peak axial factors such that the total power is 1.0 for a $1/8^{th}$ core. There were some sub-nodes in bundles 103 and 106 that were outside the geometry of the system. Since this would present a problem for the source sampling, these nodes had their probabilities set to zero and the corresponding mirror reflected sub-node (that was within the geometry) had its probability doubled. Thus, once again, the total power was preserved at 1.0.

The z source distribution (axial shape) was obtained from Table 2.2.2.6 of Reference 2. The distribution labeled 1 was used for the all the bundles except the last two layers. The distribution labeled 2 was used for the bundles next to the peripheral bundles. The distribution labeled 3 was used for the peripheral bundles.

The source energy distribution was based on a burnup of 14.12 GWd/t and was sampled from the thermal fission neutron distribution for U-235, U238, Pu-239 or Pu-241 with the probability of each isotope based on the distribution given in Table 2.2.1.1 of Reference 2. The fission spectrum and energy group structure for each of these isotopes were also obtained from Reference 2. As mentioned earlier, a sensitivity assessment to the burnup was performed and will be discussed later.

4.5 TALLY SPECIFICATIONS

All cases were run using a sample size of 320 million histories. All tally regions in the octant were divided into 20 azimuthal sectors each of 2.25°. The axial locations were chosen based on those chosen for the benchmark problem in Reference 2. Tallies were scored in the following regions, the axial extents of which were 4 cm:

- Downcomer in a region between 278.877 cm and 277.323 cm, with the center at 278.1 cm. Two cells with axial midpoints of 240 cm and 306 cm were tallied.
- RPV liner region with axial midpoints of 240 cm and 306 cm.
- RPV quarter thickness between 323.802 cm and 325.818 cm, with the center at 324.81 cm. Two axial cells with midpoints of 240 and 306 cm were tallied.
- RPV full thickness between 337.915 cm and 335.899 cm, with the center at 336.9 cm. Two cells with axial midpoints at 240 cm and 302 cm were tallied.
- Shroud tallies were between 271.304 cm and 270.288 cm with the center at 270.75 cm. Two cells with axial midpoints at 240 cm and 302 cm were tallied.
- A tally was made for a special run between the inner wall of the RPV at 321.786 cm and 324.5 cm in order to compare the MCNP results with the BNL MCNP results.

The tallies were made with volumes of 1 cm^3 for each region and the correct volume of each region was incorporated later. A problem cut-off of 0.1 MeV was used to speed up the calculation.

The fuel temperature used was chosen as 793K. The moderator and structural material, including the fuel cladding, were at the standard operating temperature of 559K. All runs were made using

Digital Alpha Stations which ran histories at the rate of approximately 75,000 – 87,000 histories per CPU minute.

4.6 ANALYSIS

4.6.1 MCNP Calculations and MCNP BNL Benchmark Calculations

The rod model was used as a base model and the results from these calculations were compared to the BNL benchmark MCNP results. MCNP results were presented in Reference 2 for two locations: the downcomer and quarter T (see Section 4.5 for definitions of tally regions). Figure 4-4 and Table 4-1 present the comparison of the two sets of data. A similar comparison can be made for the inner RPV location, and these are presented in Table 4-2 and Figure 4-5. All points in the figures are shown with 1σ error bars.

4.6.2 Sensitivity to the Cross Sections

The base rod case was run using ENDF/B-V data for all isotopes except oxygen and the iron isotopes for which ENDF/B-VI data was used. Two sensitivity runs were made to study the effects of the cross sections on the fluxes. The first used ENDF/B-V cross sections for all isotopes and the second used ENDF/B-VI cross sections for iron only. These results will be presented for the downcomer, liner, RPV- quarter T and full T locations. The error bars on the ENDF/B-VI for iron only are not shown for purposes of clarity in the figures. The ratios shown are taken with respect to the ENDF/B-V data, since the intent is to show the effect of moving to the ENDF/B-VI data for iron and oxygen from the existing ENDF/B-V data set.

The downcomer location data is presented in Figure 4-6. The results indicate that the effect of the cross sections at this location is very small and generally within one standard deviation of the ratios.

The next set of results in Figure 4-7 represent data at the RPV liner. These results also show an effect, which is within the uncertainty of the data. The quarter T position results are presented in Figure 4-8 and the full T position results are presented in Figure 4-9.

4.6.3 Sensitivity to Jet Pumps

The next series of sensitivities were performed to determine the effect of the jet pumps. The rod case was modified to replace the jet pumps and risers with water and tallies were made at the downcomer, RPV liner, RPV quarter T and the RPV full T and these were compared to the rod case.

The results at the downcomer are presented in Figure 4-10. As expected, the jet pumps have no effect at the downcomer locations. The liner results are presented in Figure 4-11. The effect of the jet pumps and risers are seen in the case of the liner. On the average, the fluxes in the case without the jet pump are about 15 percent higher than in the base case.

4.6.4 Sensitivity to Fission Source Distribution

The base case runs used a distribution of actinides based on a burnup of 14.12 GWd/t. Thus, based on the relative abundance of U-235, U-238, Pu-239, and Pu-241, the appropriate fission energy distribution was selected for each particle in the simulation. The purpose of this study was to determine the effect of changing the relative abundance of these four isotopes. A new run was made using the rod model and the distribution of these actinides based on 17.88 GWd/t[2]. The new source had a relative abundance of 51% U-235 (compared to 56% at the original burnup), 8% of U-238 (compared to 8% at the original burnup), 36% Pu-239 (compared to 32% at the original burnup), and 5% Pu-241 (compared to 4% at the original burnup).

4.6.5 Sensitivity to Bundle Source Distribution Model

The last set of sensitivity studies was performed to determine the effect of using rod-by-rod source distributions in the peripheral bundles compared to smeared sources. Figures 4-16 and 4-17 show the comparisons between the two cases for the downcomer and the RPV quarter T locations.

4.7 CONCLUSIONS

Table 4-1
Comparison of MCNP Calculations with Benchmark Data at 240 cm for Downcomer

Table 4-2
Comparison of MCNP Calculations with Benchmark Data at 240 cm for Inner RPV Location

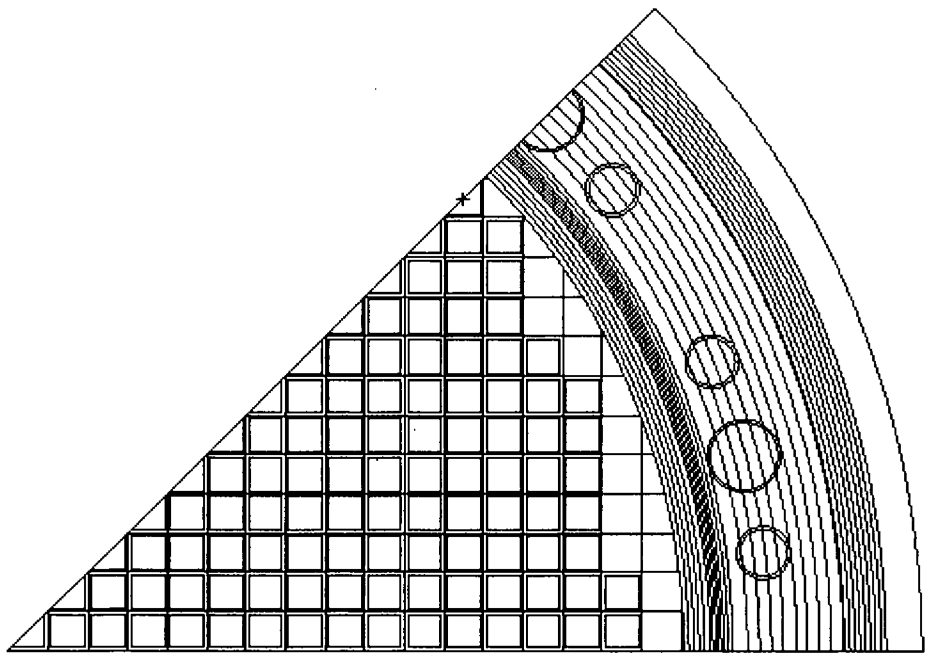

Figure 4-1. (R,θ) View of the Smeared Model

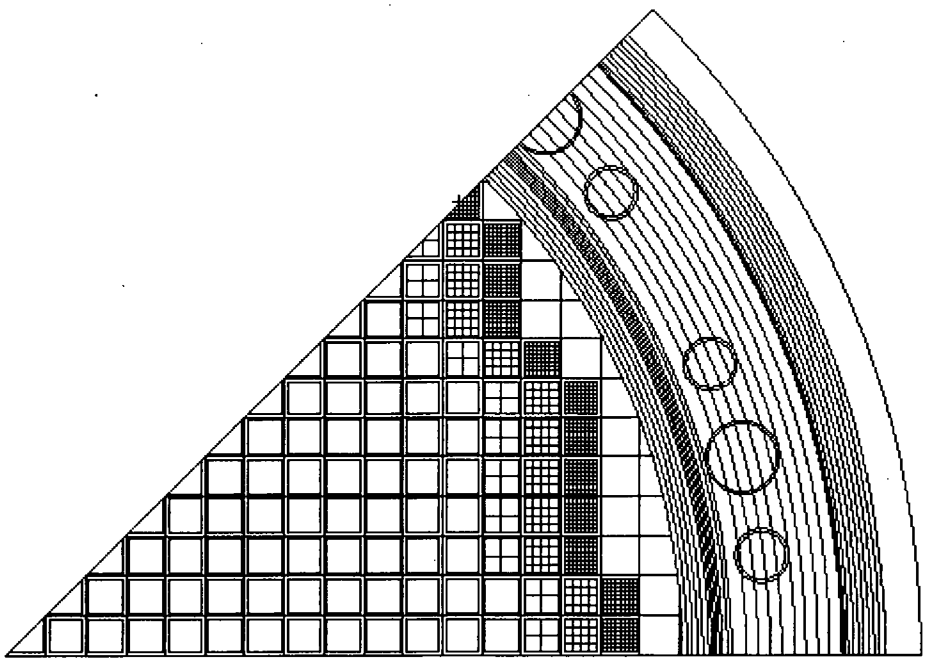

Figure 4-2. (R,θ) View of the Rod Model

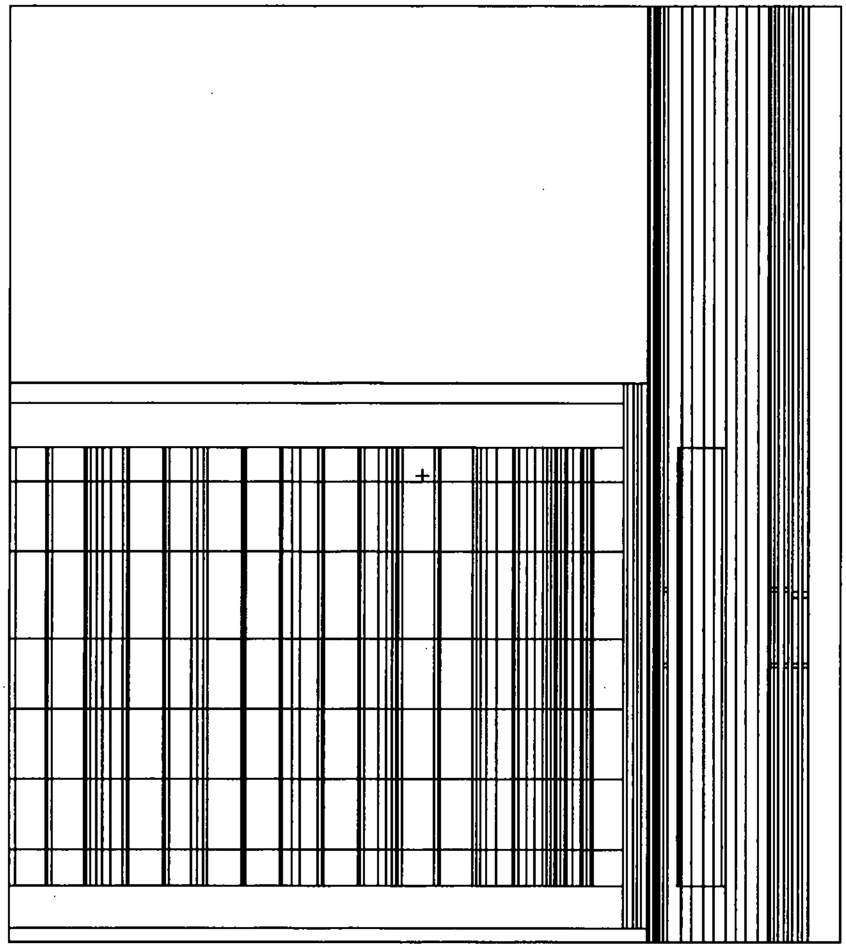

Figure 4-3. (R,Z) View of the Models

Figure 4-4. Comparison of MCNP and Benchmark MCNP Downcomer Fluxes at 240 cm

Figure 4-5. Comparison of MCNP and Benchmark MCNP Inner RPV Fluxes at 240 cm

Figure 4-6. Effect of Cross Section Sets on MCNP Flux at the Downcomer at 306 cm

Figure 4-7. Effect of Cross Section Sets on MCNP Flux at the RPV Liner at 306 cm

Figure 4-8. Effect of Cross Section Sets on MCNP Flux at the RPV Quarter T at 306 cm

Figure 4-9. Effect of Cross Section Sets on MCNP Flux at the RPV Full T at 302 cm

Figure 4-10. Effect of Jet Pumps on MCNP Fluxes at Downcomer at 306 cm

Figure 4-11. Effect of Jet Pumps on MCNP Fluxes at RPV Liner at 306 cm

Figure 4-12. Effect of Jet Pumps on MCNP Fluxes at RPV Quarter T at 306 cm

Figure 4-13. Effect of Jet Pumps on MCNP Fluxes at RPV Full T at 302 cm

Figure 4-14. Effect of Burnup on MCNP Fluxes at Downcomer at 306 cm

Figure 4-15. Effect of Burnup at RPV Quarter T at 306 cm

Figure 4-16. Effect of Bundle Source Distribution Model at Downcomer at 306 cm

Figure 4-17. Effect of Bundle Source Distribution Model at Quarter T at 306 cm

Figure 4-18. Effect of Bundle Source Distribution Model at Shroud at 306 cm

5.0 IN-REACTOR DOSIMETRY BENCHMARK

5.1 BACKGROUND

Regulatory Guide 1.190 calls for validation of the fluence calculation methodology by comparing the calculated results with both measurements and calculational benchmark. The fluence calculation methods must be validated against (1) operating reactor measurements that provide in-vessel surveillance capsules dosimetry or ex-vessel cavity measurements or both, (2) a pressure vessel simulator benchmark that provides measurements at the inner surface and at the T/4 and 3T/4 positions within the vessel, and (3) available fluence calculation benchmarks. The methods used to determine the plant-specific data and to calculate the benchmark solutions must be consistent to the extent possible with those used to calculate the vessel fluence. That is, the same cross sections, transport technique, and transport code parameters that are to be used in the reactor licensing application must be employed in the calculation of the benchmark measurements and reference calculations.

The calculation-to-measurement comparisons are used to identify biases in the calculations and to provide reliable estimates of the fluence uncertainties. When the measurement data are of sufficient quality and quantity that they allow a reliable estimate of the calculational biases (i.e., they represent a statistically significant measurement data base), the comparisons to measurement may be used to (1) determine the effect of the various modeling approximations and any calculational bias and, if appropriate, (2) modify the calculations by applying a correction to account for bias or by model adjustment or both.

In this section, calculations are compared to a set of in-reactor dosimetry measurements obtained from an operating BWR. These measurements serve the purpose of the simulator benchmark stipulated in RG 1.190.

5.2 DESCRIPTION OF MEASUREMENT BENCHMARK

A set of radiation and damage monitors was successfully installed in an overseas BWR4 plant during the summer of 1997, and removed following a cycle of operation during the summer of 1998[8,9]. The activated monitors were shipped to GE Vallecitos Nuclear Center (VNC) for processing. At VNC, the monitors were removed from their holders and sorted for counting and measurements to obtain the reaction rates of these monitors.

5.3 DORT CALCULATIONAL MODEL

The GE discrete ordinates flux solution methodology described in Section 2.1 is used to provide the 3-D flux distribution by constructing the synthesized results of (r,θ) and (r,z) calculations. All flux solution calculations are performed using an S_{12} quadrature and a P_3 Legendre polynomial expansions of the scattering cross sections.

5.3.1 Core Configuration

5.3.2 Cross Section Processing

5.3.3 Neutron Source Calculation

where,

1097 MWt is the rated thermal power

0.125 represents 1/8th of the core

381 cm is the total height of the core.

For the (r,z) calculation, the goal is to generate a relative, rather than absolute, flux magnitude. Therefore, a precise neutron source description is not essential. A typical number of 1.0E20 n/sec is used.

5.3.4 Modeling of Downcomer and Bypass Region

Since the focus of this benchmark is the dosimetry reaction rates in the downcomer at the 4°, 20°, and 71° azimuths, which are free of interference from the jet pumps and risers, these components are not modeled in this calculation.

5.3.5 Reaction Rate Calculation

The dosimeters in the capsules, after irradiation and post-processing of these dosimeters, result in measurement data in the unit of disintegration per second per gram of dosimeter isotope (dps/g). The corresponding calculated values can be derived and expressed in the following equation:

$$\frac{dps}{g} = \frac{N_A}{M} \sum_g \sigma_g \phi_g \sum_i p_i (1 - e^{-\lambda' \Delta t_i}) e^{-\lambda (t_{EOI} - t_i)}$$

where,

N_A = Avogadro's number (6.022×10^{23})

M = atomic mass of the dosimetry material

σ_g = dosimetry cross section for neutron energy group g

ϕ_g = neutron flux for group g

λ = decay constant for the daughter isotope of dosimeter isotope of interest

λ' = isotope removal constant, $\lambda + \sigma^r \phi$

σ^r = isotope removal cross section

t_i = time at the end of irradiation time step i

Δt_i = duration of irradiation time step i

t_{EOI} = time at the end of cycle irradiation

p_i = relative power for time step i.

5.3.6 Results

5.4 MCNP CALCULATIONS

5.4.1 Calculational Methodology

The In-Reactor Dosimetry experiment (see Section 5.2) was simulated using three-dimensional Monte Carlo methodology. The computational model was based on the code MCNP01A, a GE ECP version of MCNP[7], used in conjunction with ENDF/B-V and -VI cross section data. The model was developed in two stages.

In the first stage, a quadrant of the core was modeled with the full geometric and material details in the core. This included rod-by-rod description in each of 25 axial nodes (15.24 cm) for all 60 bundles in the quadrant. The important structural components inside the RPV were also included and detailed water density distributions inside and outside the core were modeled. Four exposure points during the cycle were studied. At each point, a criticality run was performed and fast flux profiles, particularly in peripheral bundles, were compared with plant data. When these results compared within 10% at the periphery[10], the second stage of the calculations was performed using sources saved at the core periphery.

The fixed source calculations, one at each of the four exposure points, were used to obtain activation rates at the various wires. The fixed source model included details of the dosimetry packages present in the downcomer region, including clamp hardware used to keep the dosimeters in place during irradiation. Detailed water density distributions were used in the downcomer in the radial direction. A time integration with appropriate decay terms for each isotope under consideration was performed to properly add the results of the four separate runs to obtain the final specific activity of each wire at the End of Cycle (EOC) at every location. The test data specific activities were also presented at EOC. Figure 5-6 shows an (r,θ) view of the fixed source model (without the core) with dosimetry holders and other structural hardware.

5.4.2 Results

Results for three axial and radial locations for the three azimuths are shown in Tables 5-4a, 5-4b, 5-4c and 5-5a, 5-5b, 5-5c. The detector identification is as follows: "A", "B", and "C" represent the low, middle, and high axial locations. The first numeral "1", "2", and "3" represent the shroud, mid-annulus, and RPV locations. The second numeral represents the azimuth, 1 representing 20° (next to a jet pump), 2 representing 4° (away from jet pumps), and 3 representing 71° (between jet pumps) (see Figure 5-6). Tables 5-4a, 5-4b, and 5-4c show the fast response comparisons as C/M (calculation to measurement) ratios. The uncertainty associated with these fast responses is 13%.

The actinide responses agree well with test data at the shroud and tend to be under-predicted at the RPV. This is principally due to the fact that the neutron-to-gamma ratios fall rapidly when moving radially outward and the gamma induced fission becomes important. The calculations have ignored this contribution because the MCNP code does not have the capability of generating and tracking gamma induced neutrons. Thus the calculations under-predict the activities in a progressive manner as the capsule location changes from the shroud to the RPV. The remaining wires are consistent and are in good agreement with the test data. Tables 5-5a, 5-5b, and 5-5c present the thermal response comparisons, including those obtained from the helium measurements. The agreement between test data and calculational results is good everywhere, especially at the shroud location. The exceptions are at the high axial locations where the center location has a C/M ratio of 1.3 and the RPV location has a C/M ratio of 0.7. This trend was consistent for all the azimuths leading to the conclusion that the attachment hardware was not at the nominal position used in the model, thus perturbing the local thermal field.

5.5 CONCLUSIONS

Calculations of capsule reaction rates in the In-Reactor Irradiation Monitors (IRIM) indicate that good calculation-to-measurement comparisons for the reaction rates in the dosimetry capsules can be obtained using the current GE RPV flux evaluation methodology based on the DORT discrete ordinates transport code and the cross section processing process in the GE ECP library. This is especially true for the capsules at 4° azimuth which is away from the jet pumps perturbation. For the ^{93}Nb dosimeters, the calculated reaction rates are within ~20% of the measurement for most capsule locations with the best estimate downcomer temperature distribution. With the base model, relatively high C/M ratios are found for the ^{54}Fe and ^{58}Ni dosimeters at the capsule locations near the vessel and the mid-annulus at the 20° azimuth. Applying a more realistic downcomer temperature distribution helps bring these C/M ratios closer to unity.

The MCNP results show C/M ratios to be for the most part within the 13% uncertainties associated with the results. The average C/M ratios for the fast wires are: 1.0 (± 0.07) at 4°, 1.02 (± 0.11) at both 20° and 71°. The average C/M ratios for the thermal wires, excluding the mid-annulus and RPV upper axial locations, are: 0.98 (± 0.09) at 4°, 1.02 (± 0.14) at 20° and 1.05 (± 0.13) at 71°.

Table 5-1
Dosimetry Capsule ID and Locations

Table 5-2
C/M Ratios of Reaction Rates for Non-Actinides with Base Model

Table 5-2
C/M Ratios of Reaction Rates for Non-Actinides with Base Model (Continued)

Table 5-3
C/M Ratios of Reaction Rates for Non-Actinides with Alternative Model

Table 5-3
C/M Ratios of Reaction Rates for Non-Actinides with Alternative Model (Continued)

Table 5-4a
Fast C/M Ratios at 4 Degrees

Det.	^{238}U			^{232}Th		^{237}Np				^{93}Nb		^{93}Nb		
ID	Cs	Zr	Ru	Cs	Zr	Cs	Zr	Ru	^{58}Ni	shield	^{54}Fe	bare	Mean*	σ*
A12	1.02	1.02	0.99	1.00	0.98	1.04	1.07	1.02	1.12	1.28	1.10	1.09	1.15	0.09
B12	0.95	0.94	0.98	0.99	0.93	0.96	1.01	1.01	0.96	1.13	0.95	1.03	1.04	0.10
C12	0.88	0.86	0.89	0.82	0.82	0.87	0.89	0.90	1.00	1.05	0.93	0.98	1.00	0.06
A22	0.77	0.74	0.75	0.79	0.71	0.86	0.86	0.84	0.96	1.05	1.07	1.06	1.03	0.05
B22	0.76	0.75	0.79	0.73	0.74	0.78	0.84	0.84	0.99	0.99	1.15	1.11	1.05	0.08
C22	0.74	0.71	0.76	0.70	0.72	0.75	0.78	0.80	0.98	0.95	0.99	0.95	0.96	0.03
A32	0.54	0.54	0.58	0.49	0.50	0.66	0.67	0.68	0.92	0.91	0.89	0.92	0.92	0.02
B32	0.56	0.55	0.60	0.50	0.51	0.65	0.70	0.74	0.97	0.95	0.88	0.89	0.93	0.04
C32	0.59	0.54	0.60	0.49	0.50	0.66	0.69	0.73	0.98	0.98	0.88	0.90	0.94	0.05

*Mean and σ for non-actinides only

Table 5-4b
Fast C/M Ratios at 20 Degrees

Det.	^{238}U			^{232}Th		^{237}Np				^{93}Nb		^{93}Nb		
ID	Cs	Zr	Ru	Cs	Zr	Cs	Zr	Ru	^{58}Ni	shield	^{54}Fe	bare	Mean*	σ*
A11	0.89	0.98	1.02	0.93	1.04	0.97	1.04	1.03	1.09	1.14	0.95	1.03	1.05	0.09
B11	0.91	0.97	1.02	0.97	1.08	0.97	1.03	1.05	1.15	1.18	1.21	1.22	1.19	0.04
C11	0.91	0.88	0.89	0.85	0.96	0.94	0.96	0.95	1.10	1.14	1.05	1.11	1.10	0.04
A21	0.75	0.73	0.74	0.76	0.74	0.85	0.84	0.82	0.90	1.03	0.98	1.00	0.98	0.14
B21	0.82	0.86	0.90	0.80	0.93	0.88	0.92	0.91	1.15	1.07	1.06	1.03	1.08	0.05
C21	0.82	0.78	0.80	0.83	0.84	0.85	0.85	0.84	1.02	1.09	1.07	1.05	1.06	0.10
A31	0.61	0.63	0.64	0.52	0.56	0.66	0.67	0.66	0.92	0.86	0.92	0.84	0.88	0.05
B31	0.56	0.57	0.63	0.48	0.57	0.68	0.71	0.72	0.90	0.89	0.93	0.90	0.91	0.09
C31	0.57	0.55	0.60	0.53	0.60	0.63	0.66	0.68	0.92	0.88	0.90	0.86	0.89	0.03

* Mean and σ for non-actinides only

Table 5-4c
Fast C/M Ratios at 71 Degrees

Det. ID	^{238}U Cs	^{238}U Zr	^{238}U Ru	^{232}Th Cs	^{232}Th Zr	^{237}Np Cs	^{237}Np Zr	^{237}Np Ru	^{58}Ni	^{93}Nb shield	^{54}Fe	^{93}Nb bare	Mean *	σ^*
A13	0.82	0.80	0.79	0.81	1.11	0.89	0.98	0.85	0.79	1.05	0.90	1.03	0.94	0.12
B13	0.94	0.98	1.01	0.95	1.11	0.95	1.04	1.02	1.04	1.18	1.14	1.21	1.14	0.07
C13	0.90	0.93	0.96	0.88	0.89	1.01	1.01	1.03	1.07	1.16	1.18	1.23	1.16	0.06
A23	0.82	0.85	0.86	0.83	1.11	0.88	1.03	0.91	1.10	1.09	1.03	1.01	1.06	0.05
B23	0.83	0.81	0.82	0.85	0.97	0.88	0.91	0.87	0.99	1.07	1.04	1.05	1.04	0.04
C23	0.88	0.88	0.92	0.86	0.84	0.88	0.86	0.91	1.14	1.15	1.12	1.08	1.12	0.03
A33	0.59	0.60	0.61	0.57	0.77	0.61	0.72	0.64	0.84	0.83	0.87	0.81	0.84	0.03
B33	0.63	0.64	0.68	0.57	0.69	0.70	0.72	0.71	0.98	0.96	0.89	0.91	0.93	0.04
C33	0.58	0.56	0.62	0.56	0.56	0.71	0.66	0.72	0.93	0.94	0.93	0.90	0.93	0.02

* Mean and σ for non-actinides only

Table 5-5a
Thermal C/M Ratios at 4 Degrees

Det. ID	^{235}U			^{45}Sc	He	Mean	σ
	Cs	Zr	Ru				
A12	1.11	1.12	1.10	1.11	0.96	1.08	0.07
B12	1.02	1.02	1.18	0.98	1.01	1.04	0.08
C12	1.10	1.10	1.29	0.90		1.10	0.16
A22	0.92	0.94	0.99	0.82	0.85	0.91	0.07
B22	1.00	1.05	1.16	0.81	0.94	0.99	0.13
C22	1.38	1.38	1.59	1.12		1.37	0.20
A32	0.89	0.91	0.97	0.81		0.89	0.07
B32	0.87	0.88	1.00	0.73		0.87	0.11
C32	0.79	0.79	0.94	0.63		0.79	0.12

Table 5-5b
Thermal C/M Ratios at 20 Degrees

Det. ID	^{235}U			^{45}Sc	^{59}Co	He	Mean*	σ*
	Cs	Zr	Ru					
A11	1.02	1.02	0.98	0.97	1.20		1.05	0.09
B11	1.00	1.06	1.20	0.83	1.09		1.04	0.13
C11	0.93	0.96	1.13	0.81	0.99	No He	0.96	0.11
A21	0.81	0.82	0.87	0.73	0.78		0.80	0.05
B21	1.04	1.15	1.33	0.89	1.28		1.14	0.18
C21	1.39	1.38	1.54	1.12	1.38		1.36	0.15
A31	1.21	1.27	1.36	1.04	1.28		1.23	0.12
B31	0.91	0.92	1.06	0.75	1.03		0.94	0.12
C31	0.70	0.70	0.83	0.55	0.68		0.69	0.10

Table 5-5c
Thermal C/M Ratios at 71 Degrees

Det. ID	^{235}U			^{45}Sc	^{59}Co	He	Mean*	σ*
	Cs	Zr	Ru					
A13	0.98	1.00	1.00	1.01	1.30		1.06	0.13
B13	0.99	1.08	1.10	1.01	1.15	1.08	1.07	0.06
C13	1.00	1.03	1.04	0.84	1.14	0.92	1.00	0.10
A23	0.86	0.89	0.88	0.78	0.87		0.86	0.04
B23	1.19	1.29	1.29	0.97	1.15	1.25	1.19	0.12
C23	1.33	1.36	1.36	1.08	1.37	1.38	1.31	0.12
A33	1.23	1.26	1.26	1.06	1.39		1.24	0.12
B33	0.96	0.99	0.99	0.84	1.05		0.96	0.08
C33	0.74	0.72	0.72	0.57	0.84		0.72	0.10

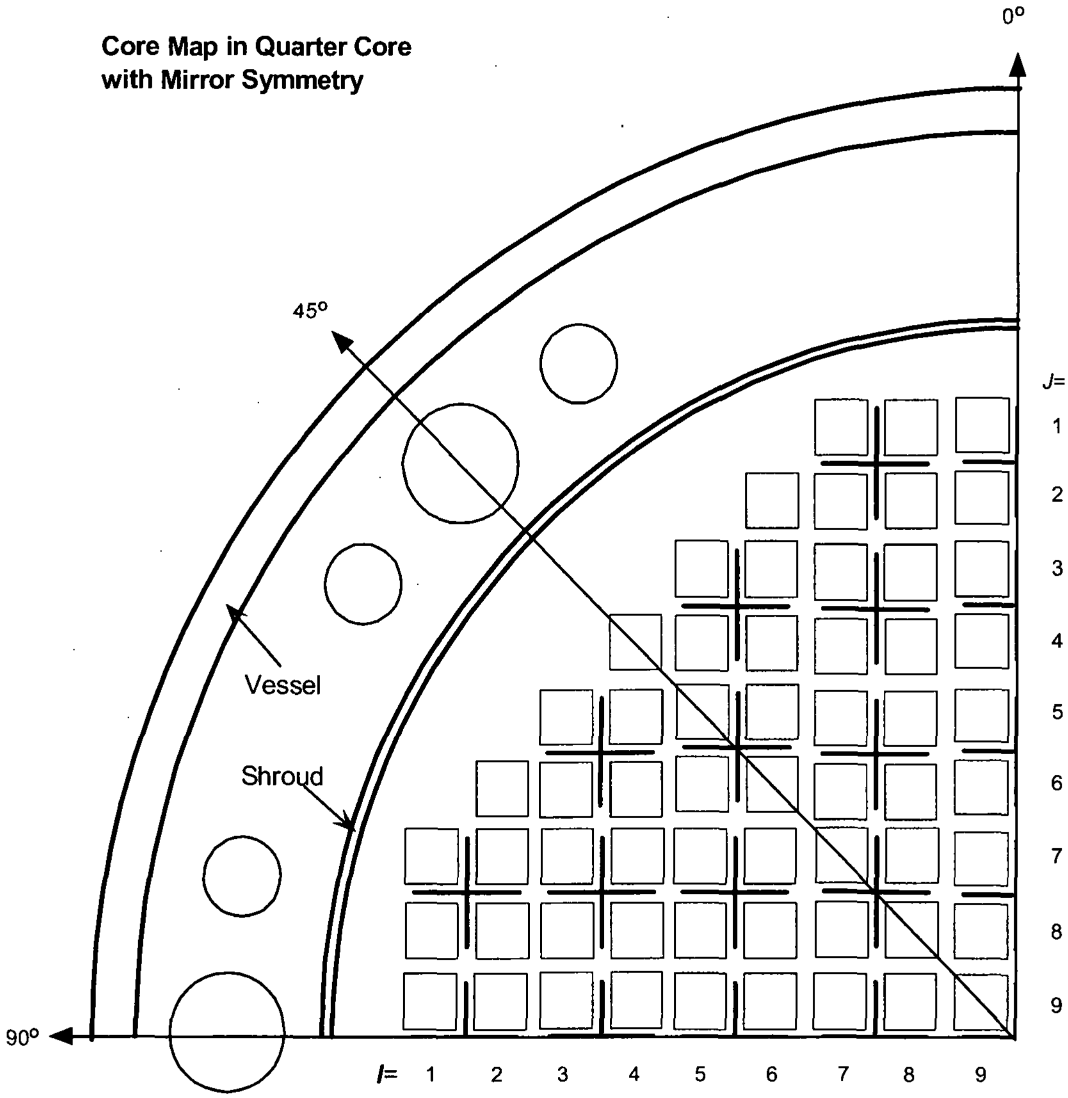

Figure 5-1. Schematic of a Quadrant of the Reactor Core

Figure 5-2. Relative Power Density at Core Midplane

Figure 5-3. Axial Nodal Power of Peripheral Bundles

Figure 5-4. C/M Ratios of Reaction Rates with Base Model

Figure 5-5. C/M Ratios of Reaction Rates with Alternative Model

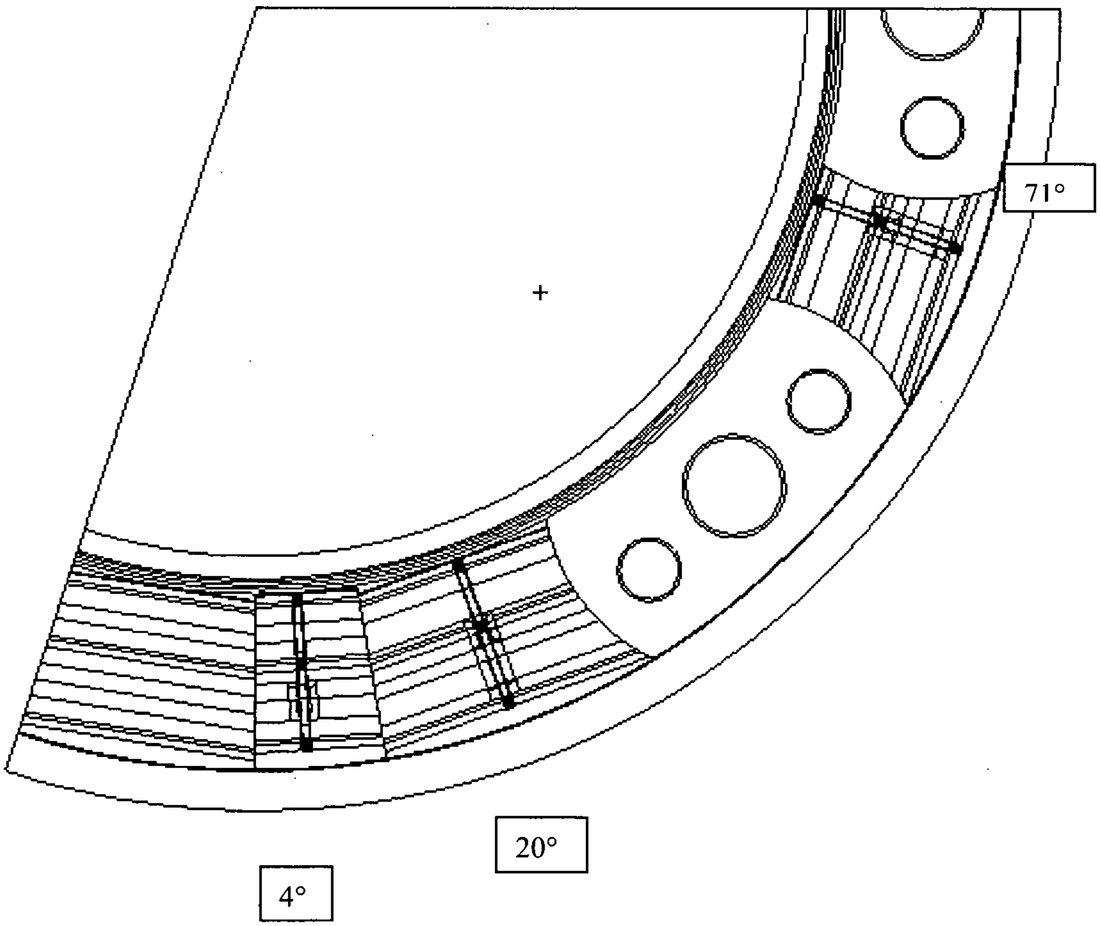

Figure 5-6. (R,θ) View of the MCNP Model with Capsule Holders

6.0 BWR SURVEILLANCE SAMPLE DATA

6.1 SURVEILLANCE DATA

A representative list of flux data generated by GE during the past decade is shown in Table 6-1.

The calculated flux values at surveillance capsule locations were the result of GE traditional discrete ordinates calculations, using the coolant-only downcomer model and bundle-average power density, in conjunction with ENDF/B-V cross section library.

The dosimetry data obtained through surveillance samples were analyzed by the Vallecitos Nuclear Chemistry Facility (VNC). The majority of surveillance capsules contain iron, copper, and nickel wires. Since Co-58 has a relatively short half-life compared to activation products of iron and copper, the nickel wire reading could differ significantly from those of iron and copper. Therefore in some of the dosimetry reports, the result of nickel wires was discounted. In other older reports, data for individual flux wires were not given, instead a single flux value was reported. The methodology and process for the VNC dosimetry analysis are detailed in Section 6.2.

6.2 MEASURED FLUX UNFOLDING FROM SURVEILLANCE DATA

6.2.1 Basic Equations

The power history for use in dosimetry analysis is obtained from plant operating data applicable to period of residence of the dosimetry capsule in the reactor. The power history or the total amount of energy generated can be obtained on a variety of bases ranging from a daily breakdown to larger time periods. Based on this information, the effective full power fraction can be derived. The effective full power fraction, p_i, is defined as P_i/P, where P is the full power of the core and P_i is the power during the time interval $(t_i - t_{i-1})$. Therefore, if ϕ_P is the full power flux, the actual flux during this time interval is

$$\phi_i = p_i \phi_P \qquad (1)$$

The total specific activity of a dosimeter wire in disintegrations per second per gram of target isotope (dps/g) at the end of irradiation (EOI) may be expressed as

$$\frac{dps}{g} \, at\ EOI = \frac{N}{\rho} \sigma \phi_P \sum_i p_i (1 - e^{-\lambda(t_i - t_{i-1})}) \, e^{-\lambda(t_{EOI} - t_i)} \qquad (2)$$

where

N = atomic density of the target isotope (atoms/cm^3)
σ = microscopic activation cross section for target isotope (cm^2)
ρ = mass density of target isotope (g/cm^3)

t_{EOI}= time at end of irradiation (s)
λ = decay constant of daughter isotope (s^{-1})

The first term with the exponential in Equation 2 accounts for the build-up and decay of the daughter isotope at the flux level ϕ_i, while the second term accounts for the decay during the remainder of the time to EOI. It must also be noted that the decay constant used in the first term with the exponential is strictly defined as $\lambda' = \lambda + \sigma\phi$, where $\sigma\phi$ represents the removal of the daughter isotope by neutron absorption. However, typically this term is negligible when compared to the radioactive decay constant. Therefore, the expression in Equation 2 contains only the decay constant, λ. The sum over each time step represents the actual measured specific activity at EOI. Inverting Equation 2,

$$\phi_P = \frac{\frac{dps}{g}}{\frac{N}{\rho}\sigma\sum_i p_i(1-e^{-\lambda(t_i-t_{i-1})})e^{-\lambda(t_{EOI}-t_i)}} \tag{3}$$

Substituting for N,

$$N = \frac{\rho N_A}{M} \tag{4}$$

where,

N_A = Avogadro's number (atoms/mole)
M = mass of one mole of the target isotope (g/mole).

The full power flux

$$\phi_P = \frac{\frac{dps}{g}\cdot M}{N_A\sigma\sum_i p_i(1-e^{-\lambda(t_i-t_{i-1})})e^{-\lambda(t_{EOI}-t_i)}} \tag{5}$$

The accumulated fluence over the period ending at EOI is then given by

$$Fluence = \phi_P\sum_i p_i(t_i - t_{i-1}) \tag{6}$$

The term under the summation sign is the effective time the reactor was operating at full power.

6.2.2 Cross Section Evaluation

In the expressions presented in Section 6.1.1, the effective activation cross section, σ, needs to be defined. Typically, in BWR surveillance programs there are only two wires, Fe and Cu. The cross sections used for this purpose are derived from two correlations developed by GE for the Fe-54 and Cu-63 activation reactions.

A multiple dosimeter, full spectrum unfolding code[11] which determined a neutron spectrum and integral fluxes using activation data from irradiated dosimeters was obtained for application to GE Nuclear experiments.

A qualification program for the generation of neutron spectra from activation detectors was undertaken by the IAEA where two reaction rate sets were sent to international participants for intercomparison. These sets were the only information about the spectra which could be used by the participants, who, in turn, had to use their own program, cross sections, etc., to generate neutron spectra. GE participated in this program with results which ranked fourth (among 59) and first (among 34) for the two cases evaluated[12]. Thus, the spectral unfolding code was independently benchmarked and qualified for use in spectral unfolding experiments at GE.

The code was later modified such that trial input spectra were automatically selected from a library containing several neutron spectra. The differential cross section library input originally was ENDF/B-IV and later updated to ENDF/B-V.

Over the years the GE code was successfully used for several spectral determinations at BWR and General Electric Test Reactor locations, as well as at the Advanced Test Reactor (ATR) at Idaho Falls.

6.2.3 Uncertainty in the Cross Sections

6.2.4 ASTM Standards

All the measurement methodologies applied were compliant with the following ASTM standards:

1. ASTM Designation E181-93, *Standard Test Methods for Detector Calibration and Analysis of Radionuclides*, ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.
2. ASTM Designation E261-90, *Standard Practice for Determining Neutron Fluence Rate, Fluence, and Spectra by Radioactivation Techniques*, ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.
3. ASTM Designation E263-93, *Standard Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Iron*, ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.
4. ASTM Designation E264-92, *Standard Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Nickel*, ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.
5. ASTM Designation E523-92, *Standard Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Copper*, ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.
6. ASTM Designation E844-86 (Reapproved 1991), *Standard Guide for Sensor Set Design and Irradiation for Reactor Surveillance*, ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.
7. ASTM Designation E1005-84, *Standard Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance*, ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.

8. ASTM Designation E1297-89, *Test Method for Measuring Fast Neutron Reaction Rates by Radioactivation of Nb,* ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.

9. ASTM Designation E704-90, *Test Method for Measuring Reaction Rates by Radioactivation of U-238,* ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.

10. ASTM Designation E705-90, *Test Method for Measuring Reaction Rates by Radioactivation of Np-237,* ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.

Table 6-1
Collective RPV Flux Data

Table 6-2
Fast Cross Sections (>1 MeV) for Iron and Copper Activation in BWRs

Table 6-3a
Fast (>1 MeV) Cross Sections in barns for a Dosimetry Capsule Location at a Distance 32.55 inches from the Core Edge

Table 6-3b
Fast (>1 MeV) Cross Sections in barns for a Dosimetry Capsule Location at a Distance 28.33 inches from the Core Edge

Table 6-3c
Fast (>1 MeV) Cross Sections in barns for a Dosimetry Capsule Location at a Distance 29.41 inches from the Core Edge

Table 6-3d
Fast (>1 MeV) Cross Sections in barns for a Dosimetry Capsule Location at a Distance 25.27 inches from the Core Edge

Table 6-3e
Fast (>1 MeV) Cross Sections in barns for a Dosimetry Capsule Location at a Distance 25.58 inches from the Core Edge

7.0 UNCERTAINTY AND BIAS ASSESSMENTS

7.1 CALCULATION UNCERTAINTIES

7.2 CALCULATION BIASES

7.2.1 Analytical Bias Assessment

7.2.2 Bias Derived from Historical Data

7.2.3 Applicability of Calculation Bias

7.3 OVERALL CALCULATION UNCERTAINTY

7.4 BEST-ESTIMATE FLUX AT REACTOR VESSEL

7.5 SHROUD FLUX UNCERTAINTIES AND BIASES

Table 7-1
RPV Flux Data for Bias Determination

8.0 CONCLUSIONS

9.0 REFERENCES

1. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," USNRC, March 2001.
2. NUREG/CR-6115, "PWR and BWR Pressure Vessel Fluence Calculation Benchmark Problems and Solutions," USNRC, May 1997.
3. CCC-543, "TORT-DORT Two- and Three-Dimensional Discrete Ordinates Transport Version 2.8.14," Radiation Shielding Information Center, Oak Ridge National Laboratory.
4. Letter, S.A. Richards (USNRC) to G. A. Watford, "Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, GESTAR II – Implementing Improved GE Steady-State Methods (TAC No. MA6481)," November 10, 1999.
5. R. E. MacFarlane and D. W. Muir, "The NJOY Nuclear Data Processing System, Version 91," LA-12740-M, Los Almos National Laboratory, October 1994.
6. Nuclides & Isotopes, Fifteenth Edition, GE Nuclear Energy 1996.
7. Briesmeister, J., Ed., "MCNP - A Monte Carlo N-Particle Transport Code, Version 4A," LA12625, March 1994.
8. Terhune, J.H., Sitaraman, S., Higgins, J. P., Chiang, R-T, and Asano, K., "Neutron and Gamma Spectra in the BWR- Phase 1 Experimental and Computational Methods," Proceedings of the 5th International Conference on Nuclear Engineering, Nice, France, ICONE5-2020, May 26-30, 1997.
9. Sitaraman, S., Chiang, R-T., Asano, K., and Koyabu, K., "Benchmark for a 3D Monte Carlo Boiling Water Reactor Fluence Computational Package - MF3D", International Conference on Advanced Monte Carlo for Radiation Physics, Particle Transport Simulation and Applications, Lisbon, Portugal, October 23-26,2000.
10. Chiang, R-T. and Sitaraman, S., "Development of 3D MCNP-Based BWR Fluence Computational Software Package: MF3D," *Reactor Dosimetry, ASTM STP 1398,* John G. Williams, David W. Vehar, Frank H. Ruddy and David M. Gilliam, Eds., American Society for Testing and Materials, West Conshohocken, PA, 2000.
11. G. DiCola and A. Rota, "RDMM- A Code for Fast Neutron Spectra Determination by Activation Analysis", EUR-2985e, 1966.
12. "International Intercomparison of Neutron Spectra Evaluating Methods Using Activation Detectors", prepared by A. Fischer, Berichte der Kernforschungsanlage Julich-Nr. 1196, June 1975.

APPENDIX A

GE RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION

1. Letter from J. F. Klapproth, GE Nuclear Energy to R. M. Pulsifer, U.S. Nuclear Regulatory Commission, "Partial Response to Request for Additional Information – GE Nuclear Energy Licensing Topical Report NEDC-32983P," MFN 00-054, December 20, 2000.
2. Letter from J. F. Klapproth, GE Nuclear Energy to R. M. Pulsifer, U.S. Nuclear Regulatory Commission, "Partial Response to Request for Additional Information – GE Nuclear Energy Licensing Topical Report NEDC-32983P Question #11," MFN 01-001, January 5, 2001.
3. Letter from J. F. Klapproth, GE Nuclear Energy to U.S. Nuclear Regulatory Commission, "Completion of Responses to Request for Additional Information – GE Nuclear Energy Licensing Topical Report NEDC-32983P RAI#'s: 6, 9, 10, 14, 17, 18, and 19," MFN 01-003, January 17, 2001.
4. Letter from J. F. Klapproth, GE Nuclear Energy to U.S. Nuclear Regulatory Commission, "Partial Response to Request for Additional Information (Round Two) – GE Nuclear Energy Licensing Topical Report NEDC-32983P," MFN 01-006, March 2, 2001.
5. Letter from J. F. Klapproth, GE Nuclear Energy to U.S. Nuclear Regulatory Commission, "Completion of Response to Request for Additional Information (Round Two) – GE Nuclear Energy Licensing Topical Report NEDC-32983P RAI#'s: 1, 2, 4, 6, 7, 8, and 9," MFN 01-009, March 14, 2001.
6. Letter from J. F. Klapproth, GE Nuclear Energy to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information – GE Nuclear Energy Licensing Topical Report NEDC-32983P RAI#'s: 1 through 8," MFN 01-023, June 1, 2001.
7. Letter from J. F. Klapproth, GE Nuclear Energy to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information – GE Nuclear Energy Licensing Topical Report NEDC-32983P Verbal Request During June 8, 2001 Conference Call," MFN 01-024, June 15, 2001.

GE Nuclear Energy

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December 20, 2000

MFN 00-054

Robert M. Pulsifer
US Nuclear Regulatory Commission
OWFN Building – Mail Stop 8 B1
11555 Rockville Pike
Rockville, MD 20852-2738

Subject: **Partial Response to Request for Additional Information – GE Nuclear Energy Licensing Topical Report NEDC-32983P**

Reference:

1.) US NRC Letter, R. M. Pulsifer to J. F. Klapproth (GE), dated November 15, 2000: Request for Additional Information – Topical Report NEDC-32983P, **General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations**, (TAC No. MA9891).

2.) MFN 00-035, Submittal of GE Proprietary Document NEDC-32983P, **General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations**, September 1, 2000.

As requested in the first reference, this letter provides, as attachments, a partial response to the Request for Additional Information (RAI) associated with the NRC Staff's preliminary review of GE Nuclear Energy LTR NEDC-32983P (Reference 2). The letter provides responses to eleven of the nineteen RAIs identified in Reference 1. The remaining responses will be provided by the middle of January 2001.

The information provided in the attached set of responses to the RAIs is considered proprietary to GE because the evaluation results and detail information within the responses comes from the proprietary licensing topical report submittal (i.e., the referenced report) or the underlying proprietary analysis bases.

Please note that the attachment contains proprietary information of the type that GE maintains in confidence and withholds from public disclosure. The information has been handled and classified as proprietary to GE as indicated in the attached affidavit. GE hereby requests that this information be withheld from public disclosure in accordance with the provisions of 10CFR2.790.

Sincerely,

Bradley J Erbes

For J. F. Klapproth, Manager
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GE Nuclear Energy
(408) 925-5434
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Attachments:

Affidavit by George B. Stramback, dated December 19, 2000 (4 pages)

GE responses to RAI # 1, 2, 3, 4, 5, 7, 8, 12, 13, 15, and 16. (33 pages)

cc:

R. M. Pulsifer	(NRC)	w/ attachments (5 copies)
M. A. Mitchell	(NRC)	w/o attachments
L. Lois	(NRC)	w/o attachments
C. E. Carpenter	(NRC)	w/o attachments
K. E. Wichman	(NRC)	w/o attachments
R. S. Drury	(GE)	w/ attachments

GE Nuclear Energy

Attachment to MFN 00-054

Partial Response to RAI Questions: 1, 2, 3, 4, 5, 7, 8, 12, 13, 15, and 16

Addressing Request for Additional Information Topical Report NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations.

Question 1:

The report does not identify nor describe the method to be utilized in the computation of BWR vessel (or reactor internals) fluence. The method(s) to be used should be the same to that described and benchmarked in the topical report.

Please describe the proposed methods to be used, the proposed range of application (types of BWRs) and the intended application.

Response 1:

The revisions to Section 1, **Introduction**, and Section 2.1.1, **(r,θ) Model,** which are attached, incorporate changes to clarify the proposed method used in the computation of BWR vessel fluence. Thus, these revisions provide a partial response to Question 1 that will be amplified and completed with the response to Question 19.

Also, Section 8, **Conclusions,** will be revised to summarize the proposed methodology. The revision to Section 8 will be provided with the response to Question 19. Thus, only a partial response to Questions 1 is provided at this time.

[To assist in the identification of the text changes, the font attribute of the revised text is changed to "**bold**". There are four additions or revisions: two in Section 1, one within the first paragraph of Section 2.1.1, and the addition of footnote text at the bottom of the page.]

1.0 INTRODUCTION

This document presents an improved General Electric Company (GE) flux calculation methodology for determination of reactor pressure vessel (RPV) and internals neutron fluence. Similar methods and processes have been in use by GE for the past decade for the evaluation of fast neutron fluence in the reactor pressure vessel and internal components.

In order to demonstrate that the GE methodology is in agreement with the intent of Draft Regulatory Guide DG-1053 Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence [1], the following topics are covered in this report:

- GE flux synthesis methodology: Geometric and material representations of the calculation model, cross section library, neutron source distribution, etc.
- GE solution to the NUREG/CR-6115 BWR benchmark problem[2]: GE methodology comparable to Brookhaven National Laboratory (BNL) solution is used for the benchmark calculations. GE results and BNL results are compared. Sensitivity studies are performed for calculation variables including: the effects of steel components in the downcomer, ENDF/B-VI vs. ENDF/B-V iron cross sections, S_8 vs. S_{12} angular quadratures, pin-by-pin vs. bundle-average power density, etc.
- Monte Carlo solution to BWR benchmark problem: Comparison of GE 3-D Monte Carlo technique vs. BNL 3-D solution.
- Benchmark through in-reactor measurements: Calculated reaction rates vs. dosimetry data collected via an in-reactor irradiation monitoring project.
- **Correlation of in-vessel surveillance data vs. GE-calculated results.**
- Uncertainty and bias assessments.

For the past two decades, GE has provided services in the area of RPV fluence evaluations, using both calculations and dosimetry. The fluence calculation methodology employed by GE has been standardized in the past decade. The calculated ratio of the surveillance sample flux to the peak flux at RPV defines a lead factor. This lead factor is applied to the sample dosimetry data for determination of the RPV peak fluence, which is required for the vessel fracture toughness evaluations.

2.1.1 (r,θ) Model

In the angular coordinate, θ, the mesh size is $^1/_2$ degree or less per mesh step. Mesh size in the radial direction varies with each region. Generally, a fine mesh is provided near material interfaces, where significant flux gradients are expected. Fine meshes are also applied near the capsule, the RPV clad, and the innermost portion of the RPV. Sufficient fine mesh steps are provided to simulate the outer profile of the core. The mesh step is fine enough such that the (r,θ) representation would reproduce the true physical bundle area to within ~0.5%.

2.1.2 (r,z) Model

---

[1] **For BWRs without jet pumps, the annulus is treated as composed of subcooled water only. This simplified yet conservative approach has traditionally been used by GE for BWRs with jet pumps as well. The proposed method performs explicit modeling of steel components in the downcomer region.**

[2] With the superior capability of modern computers, economizing of computational effort is no longer a major concern. Applying a reflective boundary condition as a substitute for the innermost core region may be used as an option.

8.0 Conclusions

Question 2:

The report refers frequently to conservatisms in the calculations. Conservatisms are desired and welcomed in some applications, as for example, the calculation of the vessel fluence. However, there are applications, as for example, multiplant and cross plant dosimetry analysis where only an accurate value should be used.

Please specify whether the methodology will be geared toward a conservative or an accurate solution

Response 2:

The proposed GE methodology is intended to produce more accurate solutions. As-built dimension and actual plant parameters are being applied as much as possible when constructing the calculation model, in order to eliminate uncertainties and unwarranted biases. However, the conservative practice of modeling with bundle-power is being retained, since the associated bias is well-defined and is being included in the bias assessment.

Revised texts in Chapters 1, 2, 7 and 8 will provide additional clarification.

Question 3:

The applicant should address the question: are the energy groups adopted in the Report adequate with respect to the energy spectrum and the resulting accuracy associated with the conversion of dosimeter activation measurements to flux with E > 1.0 MeV? Spectrum accuracy is particularly important if Fe-54, Cu-63, etc. dosimeters with threshold energies E > 1.0 MeV are to be analyzed, which were irradiated in the shadow (or penumbra) of the jet pumps or pump risers.

Please demonstrate that the chosen number of energy groups (compared to the 47 groups in the standard BUGLE cross sections) is adequate for activation conversion in the shadow, the vicinity or away from jet pumps or pump risers.

Response 3:

A GE 44-group structure was constructed to emulate the BUGLE-47 library, using the original LANL library, which the GE 26-group was based on.

BWR benchmark problem solutions originally calculated with 26-group are re-calculated with the 44-group cross-sections. The results are presented below:

Table R3-2	Sensitivity of Calculated Flux to Energy Groups
Figure R3-1	Capsule Flux Spectrum – 44 Group vs BUGLE-47
Figure R3-2	Capsule Flux Spectrum – 26 Group vs BUGLE-47
Figure R3-3	Shroud Flux Comparison – 44 vs 26 Group
Figure R3-4	Downcomer Flux – 44 vs 26 vs BUGLE-47
Figure R3-5	RPV ID Flux Comparison – 44 vs 26 vs BUGLE-47
Figure R3-6	RPV 1/4T Flux Comparison–44 vs 26 vs BUGLE-47
Figure R3-7	RPV T Flux Comparison–44 vs 26 vs BUGLE-47

Table R3- 1 Energy Group Structures

Table R3- 1 (Continued)

Table R3- 2 Sensitivity of Flux to Energy Groups

Figure R3- 1 Capsule Flux (GE-44 vs BUGLE-47)

Figure R3- 2 Capsule Flux (GE-26 vs BUGLE-47)

Figure R3- 3 Shroud ID Flux (GE-26 vs GE-44)

Figure R3- 4 Downcomer Flux (GE-26 vs GE-44 vs BUGLE-47)

Figure R3- 5 RPV ID Flux (GE-26 vs GE-44 vs BUGLE-47)

Figure R3- 6 RPV ¼T Flux (GE-26 vs GE-44 vs BUGLE-47)

Figure R3- 7 RPV Full T Flux (GE-26 vs GE-44 vs BUGLE-47)

Question 4:

The differences to the vessel and downcomer azimuthal fluxes between the GE and the BNL MCNP models are very high (e.g., see Tables 4-1 and 4-2, etc.).

Please provide a physically based explanation and understanding of these differences. The report should also establish and justify a threshold of acceptability of the differences.

Response 4:

The BNL 3D MCNP calculation used the 47-group BUGLE cross sections while the GE calculation used pointwise data. In order to establish that this is a potential cause for the differences observed between the calculations, two additional calculations were performed. The first of these calculations was performed using the 30-neutron group standard multigroup set that accompanies the MCNP code. The second calculation was performed using the discrete reaction cross sections that also accompany the MCNP code. Both these sets are based entirely on ENDF/B-V data. However, at the locations where the comparisons are made, the downcomer and the inner RPV, the difference between the ENDF/B-V and the hybrid ENDF/B-V/VI runs (shown in Tables 4-1 and 4-2), is about 3% and is statistically insignificant. (See LTR Section 4.6.2).

The energy group structure makes a difference in the flux results with the numbers increasing from the 30-group treatment to the 263 group and continuous treatments, both of which gave very similar results.

The MCNP and DORT calculations performed by GE show good agreement using similar cross section sets. The MCNP results from GE and BNL, however, show a difference and this seems to indicate a difference in the geometric model used in the BNL and GE MCNP calculations.

Table 4-1a. Comparison of GE MCNP Multigroup Calculations with Benchmark Data at 240 cm for the Downcomer

Table 4-1b. Comparison of GE MCNP Discrete Cross Section Set Calculations with Benchmark Data at 240 cm for the Downcomer

Table 4-2a. Comparison of GE MCNP Multigroup Calculations with Benchmark Data at 240 cm for the Inner RPV Location

Table 4-2b. Comparison of GE MCNP Discrete Cross Section Set Calculations with Benchmark Data at 240 cm for the Inner RPV Location

Table 4-3a. Comparison of GE MCNP Multigroup Calculations with GE DORT Data at 360 cm for the Downcomer

Table 4-3b. Comparison of GE MCNP Multigroup Calculations with GE DORT Data at 360 cm for the RPV Liner

Table 4-3c. Comparison of GE MCNP Multigroup Calculations with GE DORT Data at 360 cm for the RPV Quarter T

Question 5:

In Section 5.1, it is stated that: "The methods used to determine the plant specific data and to calculate the benchmark solution must be consistent to the extent possible with those used to calculate the vessel fluence."

Please state the differences between the benchmarked and the methods used and if necessary, evaluate and justify.

Response 5:

The methods used to calculate the vessel fluence, as described in Section 2 of the LTR, were also used to calculate the BNL benchmark (calculation benchmark) and the In-Reactor Irradiation Monitor (IRIM) benchmark (used in the LTR as a simulator benchmark). The only difference is that jet pumps/risers were not modeled in the IRIM benchmark calculation, as they do not affect the dosimeter reaction rates in the capsules, which are located in the downcomer at 4° and 20° azimuths.

Question 7:

In Section 5.3, S_{12} and P_3 approximations are proposed. However, the BWR depth of penetration is considerable.

Should you consider using a P_5 approximation to better represent the sharply forward component of the flux?

Response 7:

Since the emphasis of the calculation in Section 5 is to determine the dosimeter reaction rates in the downcomer region, not the cavity fluence, a P_3 truncation of the Legendre polynomial expansion is adequate to approximate the anisotropy in the differential scattering cross sections. As a matter of fact, a P_3 approximation is also used in the BNL's RPV fluence benchmark specification (NUREG/CR-6115).

To further validate that a P_3 approximation is adequate for the RPV flux calculation in BWRs, two separate calculations were performed in which the only difference in the model is the order of the Legendre polynomial expansion – one with a P_3 approximation and the other with a P_5 approximation. The resultant reaction rates and fast fluxes are compared in the following table:

Table R7-1 Ratios (P_5 / P_3) of Fast Neutron Flux and Dosimeter Reaction Rate

Question 8:

In Section 5.3.1, a BWR with a relatively small vessel was chosen to perform measurements for the benchmarking.

The report should discuss the applicability of the data to the newer BWRs (BWR4-6) with larger vessel diameters, larger number of fuel assemblies and different core and downcomer configurations. Likewise for earlier BWRs.

Response 8:

The In-Reactor Irradiation Monitoring (IRIM) project was chosen to serve as a pressure vessel simulator benchmark to meet the requirements of Regulatory Position 1.4.2 of DG-1053. That is, in addition to the operating reactor measurements and an available fluence calculation benchmark, a simulator benchmark is required as one of the three validation steps to qualify the calculational methods. The selection of the IRIM project for the simulator benchmark was based on the facts that data analysis for the experiment and a comprehensive MCNP calculation of the experiment have been completed recently.

It is our judgment that effects of vessel diameter, number of fuel assemblies and core/downcomer configurations on the calculated RPV flux should be addressed in the calculation to measurement comparison of plant specific data which cover a wide range of BWR configurations (see response to Question 17). It is impractical to expect that one simulator benchmark would be applicable to every BWR configuration. In fact, some of the simulator benchmarks recommended in DG-1053 (e.g., HB Robinson 2) have smaller vessel diameter than the one we chose for our simulator benchmark.

Question 12:

Section 5.4.2 states that the gamma induced fissions in the actinides have been ignored.

Please discuss why photofission can be ignored using acceptable technical or statistical arguments.

Response 12:

The statement does not imply that the photofission contribution can be ignored. The photofission contributions could not be calculated because the MCNP code does not have the capability to generate and track these contributions. The text will be revised to clarify this. It will now read:

> "*the calculations have ignored this contribution because the MCNP code does not have the capability of generating and tracking gamma induced neutrons. Thus the calculations underpredict the activities in a progressive manner as the capsule location changes from the shroud to the RPV*".

Question 13:

Section 5.4.2 finds large differences in the axial direction as acceptable because the trend is "consistent" for all azimuths leading to the conclusion that the dosimeter locations may be wrong.

The report should justify why consistency is an adequate justification for acceptance. Statistical arguments should be used along with appropriate acceptance criteria for inclusion or rejection of specific data points in the data base.

Response 13:

It must be noted that this statement pertains **to the thermal responses and not the fast responses.** Tables 5-5a,b,c present thermal C/M ratios which are presented for the sake of completeness of the MCNP benchmark calculations. They are, thus, not applicable to the validation of the fast neutron calculational methodology, which is the intent of the LTR. The design of the clamps used to secure the capsule holders was such that there was some room for moving the clamp along the length of the holder. This may have been a practical necessity at the upper axial locations at all three azimuths. Therefore, unlike at the other positions, the clamp was possibly not at the nominal positions that were used in the calculations. The clamp design was such that, depending on the position on the clamp relative to the capsule holders, there would have been an over-shielding effect caused by the clamp material at the mid-annulus location while replacing this with water at the near-vessel location. Thus, there was a consistent underprediction of the thermal response at the near-vessel location and an overprediction at the mid-annulus. This was not observed in the fast responses because these would be far less sensitive to this effect.

Question 15:

Section 5.5, Tables 5.4a, b and c, include specific dosimetry measurements with very large deviations from C/M = 1.0.

Please provide justification for the acceptability of these values and the corresponding acceptability criteria.

Response:

Question 16:

In Section 5.3.1, there seems to be a misdesignation (or a typo) for the I=9, J=3 bundle.

Please clarify.

Response 16:

All occurrence of "I=9, J=3 bundle" in Section 5.3.1 should be replaced by "I=7, J=1 bundle". This correction will be made to the LTR.

GE Nuclear Energy

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January 5, 2001

MFN 01-001

Mr. Robert M. Pulsifer
U. S. Nuclear Regulatory Commission
OWFN Building – Mail Stop 8 B1
11555 Rockville Pike
Rockville, MD 20852-2738

Subject: **Partial Response to Request for Additional Information – GE Nuclear Energy Licensing Topical Report NEDC-32983P Question # 11**

Reference:

1.) MFN 00-054, Letter J. F. Klapproth (GE) to R. M. Pulsifer (NRC), dated December 20, 2000: **Partial Response to Request for Additional Information – GE Nuclear Energy Licensing Topical Report NEDC-32983P.**

2.) US NRC Letter, R. M. Pulsifer to J. F. Klapproth (GE), dated November 15, 2000: Request for Additional Information – Topical Report NEDC-32983P, **General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations.**

3.) MFN 00-035, Submittal of GE Proprietary Document NEDC-32983P, **General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations**, September 1, 2000.

As requested in the second reference, this letter provides, as an attachment, a partial response to the Request for Additional Information (RAI) associated with the NRC Staff's preliminary review of GE Nuclear Energy LTR NEDC-32983P (Reference 3). The letter provides a response to Question #11 identified in Reference 2. The remaining responses will be provided approximately by the middle of January 2001. Providing the responses as they become available minimizes the elapsed time of review and meets the spirit of a timely review as noted in Reference 3.

The information provided in the attached response to the RAI Question #11 is considered proprietary to GE because the evaluation results and detail information within the responses comes from the proprietary licensing topical report submittal (i.e., the referenced report) or the underlying proprietary analysis bases.

Please note that the attachment contains proprietary information of the type that GE maintains in confidence and withholds from public disclosure. The information has been handled and classified as proprietary to GE as indicated in the attached affidavit. GE hereby requests that this information be withheld from public disclosure in accordance with the provisions of 10CFR2.790.

Sincerely,

B.J. Erbes

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Attachments:

Affidavit by David J. Robare, dated January 5, 2000 (4 pages)

GE response to RAI Question # 11. (2 pages)

cc:

R. Pulsifer	(NRC)	w/ attachments
M. A. Mitchell	(NRC)	w/o attachments
L. Lois	(NRC)	w/o attachments
C. E. Carpenter	(NRC)	w/o attachments
K. E. Wichman	(NRC)	w/o attachments
R. Drury	(GE)	w/ attachments

Attachment to MFN 01-001

Partial Response to RAI Questions: No. 11

Addressing Request for Additional Information Topical Report NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations.

Question 11:

Section 5.4.2 states that: "The uncertainties associated with each C/M ratio is of the order of 10 percent including statistical uncertainties."

Please justify this statement with respect to the magnitude of the uncertainty, the uncertainty components and method of analysis.

Response 11:

GE Nuclear Energy

James F. Klapproth
Manager, Engineering and Technology

Nuclear Services
General Electric Company
175 Curtner Avenue, MC 706, San Jose, CA 95125-1088
408 925-5434, Fax: 408 925-3837
james.klapproth@gene.ge.com

January 17, 2001

MFN 01-003

US Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Attention: Chief, Information Management Branch
Program Management
Policy Development and Analysis Staff

Subject: **Completion of Responses to Request for Additional Information – GE Nuclear Energy Licensing Topical Report NEDC-32983P RAI#'s: 6, 9, 10, 14, 17, 18, and 19.**

Reference:

1.) MFN 01-001, **Partial Response to Request for Additional Information-GE Nuclear Energy Licensing Topical Report NEDC-32983P–Question # 11**, January 5, 2001.

2.) MFN 00-054, **Partial Response to Request for Additional Information-GE Nuclear Energy Licensing Topical Report NEDC-32983P**, December 20, 2000.

3.) US NRC Letter, R. M. Pulsifer to J. F. Klapproth (GE), dated November 15, 2000: Request for Additional Information – Topical Report NEDC-32983P, **General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations.**

4.) MFN 00-035, Submittal of GE Proprietary Document NEDC-32983P, **General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations,** September 1, 2000.

As first requested in Reference 3, this letter provides, as an attachment, the remaining final responses to the Request for Additional Information (RAI) associated with the NRC Staff's preliminary review of GE Nuclear Energy LTR NEDC-32983P (Reference 4). The letter provides responses to the remaining seven of the nineteen RAIs identified in Reference 3. Previous letters provided responses to the other RAIs (See Reference 1 and 2). With the attachment to this letter, GE now has provided a response to each of the 19 RAIs as listed in Reference 3. With these responses, GE believes that the NRC has sufficient information to complete its review and provide an SER before the end of February 2001. In support of that schedule, GE offers to review the draft SER for proprietary content, on an expedited basis.

The information provided in the attached set of responses to the RAIs is considered proprietary to GE because the evaluation results and detail information within the responses comes from the proprietary licensing topical report submittal (i.e., the referenced report) or the underlying proprietary analysis bases.

Please note that the attachment contains proprietary information of the type that GE maintains in confidence and withholds from public disclosure. The information has been handled and classified as proprietary to GE as indicated in the attached affidavit. GE hereby requests that this information be withheld from public disclosure in accordance with the provisions of 10CFR2.790.

Sincerely,

TA Caine for

J. F. Klapproth, Manager
Engineering and Technology
GE Nuclear Energy
(408) 925-5434
james.klapproth@gene.ge.com

Attachments:

Affidavit by George B. Stramback, dated January 17, 2001 (4 pages)

GE responses to RAI # 6, 9, 10, 14, 17, 18, and 19.

cc:

R. M. Pulsifer	(NRC)	w/ attachments
M. A. Mitchell	(NRC)	w/o attachments
L. Lois	(NRC)	w/o attachments
C. E. Carpenter	(NRC)	w/o attachments
K. E. Wichman	(NRC)	w/o attachments
R. S. Drury	(GE)	w/ attachments

Attachment to MFN 01-003

Completion of Responses to Remaining RAI Questions: 6, 9, 10, 14, 17, 18, and 19

Addressing Request for Additional Information Topical Report NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations.

Question 6:

In Section 5.1, it is stated: "...when the measurement data are of sufficient quality and quantity that they allow a reliable estimate of the calculated biases.

The adequacy of the quality and the quantity of the database should be supported by appropriate statistical analysis.

Response 6:

The adequacy of the quality and the quantity of the database is supported by appropriate statistical analysis. Such analysis is documented in the response to Questions 9, 17, and 18.

Question 9:

In view of the proposed BWR ISP (integrated surveillance program), it is almost certain that the methodology will be applied to plants without recent credible dosimetry.

The report should assure that the benchmarking is based on all available dosimetry data and data from all types of reactors for which the methodology is intended and are included in the ISP. Any such justification should be based on acceptable statistical arguments.

Response 9:

Intent of the LTR is to use all available dosimetry data and data from reactors that GE has serviced and will service in the future. For this reason, the IRIM data were chosen for benchmarking in lieu of simulator benchmarking.

In response to this RAI, GE Corporate Research & Development (CR&D) performed a statistical analysis based on a complete set of calculated and measured flux data. In the process of preparing the evaluation database, GENE discovered discrepancies in the subset data C, which contains calculated and measured plant data that were both produced by GENE, and was adopted as Table 6-1 of the LTR. Table 6-1 is therefore modified slightly. The statistical analysis by CR&D was based on the revised Table 6-1, which is included with the Response to RAI#17.

The result of GE CR&D statistical analysis, shown in Attachment R9-1, concludes that subset C is representative of the entire fleet.

Attachment R9-1

Figure 1

Question 10:

In Section 5.3.6, the data used are limited to the Fe-54 and Nb-93. The available data are much broader, however, the exclusion of the remaining data has not been justified.

Please justify data exclusion with appropriate statistical arguments.

Response 10:

It is true that the available data are much broader in the IRIM benchmark. The following explains the data exclusions:

1. All dosimetry wires that are intended for thermal neutron response were excluded since the emphasis of this LTR is in the calculation of fast neutron flux, not the thermal neutron flux.
2. All actinide wires were excluded due to unavailability of photon fission data. Based on conclusions inferred from the MCNP calculations, photon induced fission has significant contribution to the reaction rates of actinide wires near the RPV. Exclusion of this contribution would introduce large underprediction and skew the statistical analysis.
3. All dosimetry wires at 71-degree azimuth were excluded in order to simplify the analysis process. That is, the analysis can be done with octant symmetry for the capsules at 4- and 20-degree azimuths, instead of the use of quarter core symmetry.

As a supplement to the Fe-54 and Nb-93 (bare capsules) wires analyzed in the LTR, additional dosimetry wires were analyzed – namely, Ni-58 and Nb-93 in the shielded capsules. The results of this additional analysis are combined with the results of the original analysis, which were corrected for a minor calculational error found after the LTR submittal. The expanded results are summarized below.

Table R10-1. Summary of C/M Ratios of Reaction Rates (Alternative Model)

Figure R10-1. C/M Ratios of Reaction Rates (Alternative Model)

Question 14:

Section 5.5 concludes that: " good C/M can be obtained."

Please justify the above statement based on acceptable statistical arguments and stated acceptance criteria.

Response 14:

The acceptance criterion used in the assessment of C/M ratios is that the calculated reaction rates agree with the measurements to within about 20% (1 sigma) for in-vessel surveillance capsules. That is, the C/M ratios should be in the acceptance range of 0.8 – 1.2. This 20% uncertainty is generally considered a typical value for this type of flux calculation and, in fact, used in the DG-1053 (Section 1.4.2) as one of the criteria.

Question 17:

In Section 6, the available data (from the plants to which the methodology has been applied) was sampled. All of the available data should be used.

Please justify why selected data are acceptable. In Table 6-1 C/M values from plants A, B, G, J and K show large deviations from C/M = 1.0. Why are these values acceptable? The report should also state what methodology was used for the calculated flux, what kinds of dosimetry is represented in the measured values, what adjustments (if any) were made on the measured values and what spectra were used for the conversion of activation to flux (fluence).

Response 17:

The intent of Table 6-1 was to include all of the plant data, of which both calculation and measurement were generated by GE in the past decade. The original LTR submittal did not include BWR 3s & 6s, due to insufficient data in these categories.

In response to RAI#9, GE Corporate Research and Development (CR&D) performed a statistical analysis based on a complete set of flux data. During the process, some discrepancies were discovered and remedied, which changed the contents of Table 6-1 slightly. For instance, calculated Plant B data was the result of 1-D approximation, which is considered non-representative of the current GE methodology. Therefore, it's deleted from the revised Table 6-1.

In addition to Table 6-1, the text of Chapter 6 has also been revised to include more descriptions of calculation and measurement methods associated with the presented data. The revised Chapter 6 is attached as Attachment R17-2.

Attachment R17-1

Attachment R17-2

6.0 BWR SURVEILLANCE SAMPLE DATA

Surveillance capsule flux data generated by GE during the past decade are shown in Table 6-1. Table 6-1 includes the complete list of plants, of which both the calculation and the measurement were performed by GE.

Table 6-1 Flux at Surveillance Capsule

Figure 6-1 BWR Capsule Flux: Measurement vs Calculation

[redacted]

Question 18:

In Section 7.0, the uncertainties listed are by no means exhaustive and important uncertainty components are not recognized. The same holds true for the calculated bias values.

The report should present and justify the method of uncertainty and bias calculation and the qualified data base which these estimates are based on

Response 18:

Chapter 7 Uncertainty and Bias Assessments has been revised to include more sensitivity study results in the uncertainty assessment. Revised calculation bias, as well as discussions of bias based on historical data and benchmark biases, are provided in the revised text.

The revised Chapter 7 is attached as Attachment R18-1.

Attachment R18-1

7.0 UNCERTAINTY AND BIAS ASSESSMENTS

7.1 CALCULATION UNCERTAINTIES

Question 19:

In Section 8 (or other suitable section), the report should define the proposed method, acceptance criteria for the data base, should state the uncertainty and bias to be applied to the calculated values (if any) and spell out the intended applications and the reactor population to which is considered applicable

Response 19:

The conclusion section of the LTR, Chapter 8, has been revised to clearly identify the characteristics of current GE methodology. Associated uncertainty and bias are included in the formula for best-estimate flux evaluation. The applicable reactor population is being addressed as well.

The revised Chapter 8 is attached as Attachment R19-1.

Attachment R19-1

8.0 CONCLUSIONS

GE flux calculation methodology has been updated from the traditional conservative approach to a more realistic modeling, in order to provide the best-estimate fluence at the BWR pressure vessel. The current methodology can be characterized by the following attributes:

- DORT (r,θ) and (r,z) synthesized

GE Nuclear Energy

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March 2, 2001

MFN 01-006

US Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Attention: Chief, Information Management Branch
Program Management
Policy Development and Analysis Staff

Subject: **Partial Response to Request for Additional Information (Round Two)- GE Nuclear Energy Licensing Topical Report NEDC-32983P**

As requested in Reference 1, this letter provides, as an attachment, a partial response to the second round Request for Additional Information (RAI-2) associated with the NRC Staff's review of GE Nuclear Energy LTR NEDC-32983P (Reference 2). The letter provides specific responses to questions 3, 5, and 10. Previous letters have provided responses to the first round RAIs (See References 3, 4, and 5). The remaining responses to the round two RAIs will be provided approximately by March 9, 2001. We are providing the responses as they become available, to help expedite issuance of the staff's Safety Evaluation Report.

Please note that the attachment contains proprietary information of the type that GE maintains in confidence and withholds from public disclosure. The information has been handled and classified as proprietary to GE as indicated in the attached affidavit. GE hereby requests that this information be withheld from public disclosure in accordance with the provisions of 10CFR2.790.

Sincerely,

J. F. Klapproth

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GE Nuclear Energy
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Attachments:

Affidavit by George B. Stramback, dated March 2, 2001 (4 pages)

GE responses to Round Two RAI's: # 3, #5, and #10.

cc:

R. M. Pulsifer	(NRC)	w/ attachments
G. S. Shukla	(NRC)	w/o attachments
M. A. Mitchell	(NRC)	w/o attachments
L. Lois	(NRC)	w/o attachments
C. E. Carpenter	(NRC)	w/o attachments
K. E. Wichman	(NRC)	w/o attachments
R. S. Drury	(GE)	w/ attachments

REFERENCE LIST

1.) US NRC Letter, G. S. Shukla to J. F. Klapproth (GE), dated February 12, 2000: Request for Additional Information – Topical Report NEDC-32983P, **General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations.**

2.) MFN 00-035, Submittal of GE Proprietary Document NEDC-32983P, **General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations**, September 1, 2000.

3.) MFN 01-003, **Completion of Responses to Request for Additional Information – GE Nuclear Energy Licensing Topical Report NEDC-32983P RAI#'s: 6, 9, 10, 14, 17, 18, and 19**, January 17, 2001.

4.) MFN 01-001, **Partial Response to Request for Additional Information-GE Nuclear Energy Licensing Topical Report NEDC-32983P–Question # 11**, January 5, 2001.

5.) MFN 00-054, **Partial Response to Request for Additional Information-GE Nuclear Energy Licensing Topical Report NEDC-32983P**, December 20, 2000.

Attachment to MFN 01-006

Partial Response to Round Two RAI Questions: 3, 5, and 10

Addressing Request for Additional Information Topical Report NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations.

The following nomenclature will be used to reference a specific RAI to help minimize any potential confusion of the part of the reader: RAI-1-X, where X would represent the specific question number. For example, "RAI-1-3" would refer to the third question in the first set of RAIs as shown in the 11/15/2000 NRC letter.

RAI-1 US NRC Letter, R. M. Pulsifer to J. F. Klapproth (GE), dated November 15, 2000: Request for Additional Information – Topical Report NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations. (Questions 1 through 19)

RAI-2 US NRC Letter, G. S. Shukla to J. F. Klapproth (GE), dated February 12, 2001: Request for Additional Information – Topical Report NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations.
(Questions 1 through 10)

Question 3:

Regarding the methods used to analyze the surveillance dosimetry measurements of Table 6-1, how does this analysis account for the effects of power history, isotopic buildup, and decay that contribute to the activity measurement? Is the methodology consistent with the applicable American Society for Testing and Materials (ASTM) dosimetry standards (e.g., ASTM E263-93, "Standard Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Iron," ASTM El297996, "Standard Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Niobium," etc.). If not, provide justification for the method used.

Do each of the measurements shown in Table 6-1, represent several dosimeter measurements? If so, how were the individual measurements combined to yield the reported value?

Response 3:

The power history for use in dosimetry analysis is obtained from plant operating data applicable to the period of residence of the dosimetry capsule in the reactor. The power history or the total amount of energy generated can be obtained on a variety of bases ranging from a daily breakdown to larger time periods. Based on this information, the effective full power fraction can be derived. The effective full power fraction, p_i, is defined as P_i/P, where P is the full power of the core and P_i is the power during the time interval $(t_i - t_{i-1})$. Therefore, if ϕ_P is the full power flux, the actual flux during this time interval is

$$\phi_i = p_i \, \phi_P \quad {}^{n}\!/_{cm^2\text{-sec}} \qquad (1)$$

The total specific activity of a dosimeter wire in disintegrations per second /gram of target isotope (dps/g) at the end of irradiation (EOI) may be expressed as

$$\text{dps/g at EOI} = \sum_i (N/\rho)\sigma \, p_i \, \phi_P \left(1 - e^{-\lambda(t_i - t_{i-1})}\right) e^{-\lambda(t_{eoi} - t_i)} \qquad (2)$$

where

N= atomic density of the target isotope (atoms/ cm^3)

σ= microscopic activation cross section for target isotope (cm^2)

ρ= mass density of target isotope (g/cm^3)

t_{eoi}= time at end of irradiation (s)

λ=decay constant of daughter isotope (s^{-1})

The first term with the exponential in equation (2) accounts for the build-up and decay of the daughter isotope at the flux level ϕ_i, while the second term accounts for the decay during the remainder of the time to EOI. It must also be noted that the decay constant used in the first term with the exponential is strictly defined as $\lambda' = \lambda + \sigma\phi$, where $\sigma\phi$ represents the removal of the daughter isotope by neutron absorption. However, typically this term is negligible when compared to the radioactive decay constant. Therefore, the expression in equation 2 contains only the decay constant, λ. The sum over each time step represents the actual measured specific activity at EOI. Inverting equation (2),

$$\phi_P = (dps/g) \,/\, [\, (N/\rho)\sigma \,(\sum_i p_i \,(1-e^{-\lambda(t_i - t_{i-1})})\, e^{-\lambda(t_{eoi} - t_i)})] \qquad (3)$$

Substituting for N, which can be expressed as

$$N = \rho\, N_A/M \qquad (4)$$

where,

N_A = Avogadro's number (atoms/mole)

M = mass of one mole of the target isotope (g/mole)

The full power flux

$$\phi_P = (dps/g)M \,/\, [N_A\sigma \,(\sum_i p_i \,(1-e^{-\lambda(t_i - t_{i-1})})\, e^{-\lambda(t_{eoi} - t_i)})] \qquad (5)$$

The accumulated fluence over the period ending at EOI is then given by

$$\text{Fluence} = \phi_P \sum_i p_i(t_i - t_{i-1}) \qquad (6)$$

The term under the summation sign is the effective time the reactor was operating at full power.

In these expressions, the effective activation cross section, σ, needs to be defined. Typically, in BWR surveillance programs there are only two wires, Fe and Cu. The cross sections used for this purpose are derived from two correlations developed by GE for the Fe-54 and Cu-63 activation reactions. Further discussion of these cross sections will be presented in the response to RAI-2-4.

All the measurement methodologies applied are compliant with the following ASTM standards:

1. ASTM Designation E181-93, *Standard Test Methods for Detector Calibration and Analysis of Radionuclides*, ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.

2. ASTM Designation E261-90, *Standard Practice for Determining Neutron Fluence Rate, Fluence, and Spectra by Radioactivation Techniques*, ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.

3. ASTM Designation E263-93, *Standard Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Iron*, ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.

4. ASTM Designation E264-92, *Standard Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Nickel*, ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.

5. ASTM Designation E523-92, *Standard Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Copper*, ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.

6. ASTM Designation E844-86 (Reapproved 1991), *Standard Guide for Sensor Set Design and Irradiation for Reactor Surveillance*, ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.

7. ASTM Designation E1005-84, *Standard Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance*, ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.

8. ASTM Designation E1297-89, *Test Method for Measuring Fast Neutron Reaction Rates by Radioactivation of Nb,* ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.

9. ASTM Designation E704-90, *Test Method for Measuring Reaction Rates by Radioactivation of U-238,* ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.

10. ASTM Designation E705-90, *Test Method for Measuring Reaction Rates by Radioactivation of Np-237,* ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.

In GE response to RAI-1-17 (Reference 1)[♦], the revised Table 6-1 column: Measured Capsule Flux contains data which are consistent with those in the dosimeter measurement reports. With the exception of Plant H, these reports were prepared by GE Vallecitos Nuclear Center. Attachment R2.3-1 lists all the available data contained in these dosimetry reports. Next to the last column: Reported Sample Flux was the flux value recognized by each report. These 21 sets of data indicated that reported flux falls into one of the following three categories:

- In the majority of cases with multiple wires, the average of measurement results from iron, copper, and nickel wires was reported (9 cases).
- Since Co-58 has a relatively short half-life compared to activation products of iron and copper, results of nickel wires can easily differ from the others. In these instances, the average of iron and copper was reported (3 cases). For these plants, taking the average of all three wires created only minor deviations (less than 6%). The nickel reading was discounted based on engineering judgment at the time of the report.
- In older reports, data for individual flux wires were not given; instead a single flux value was reported (9 cases).

♦ References for Response #3 (or RAI-2-3) are shown at the end of the response or section.

Response 3: REFERENCE LIST

1.) The response to Question #17 (or RAI-1-17) as transmitted in "MFN 01-003, Completion of Responses to Request for Additional Information – GE Nuclear Energy Licensing Topical Report NEDC-32983P RAI#'s: 6, 9, 10, 14, 17, 18, and 19, January 17, 2001."

Question 5:

In the response to the previous RAIs, several corrections to the original NEDC-32983 analysis have been identified. For example, in response to RAI-10, it is stated that the "original analysis" was corrected for a calculational error and, in response to RAI-17, several of the C/M values in Table 6-1 have been changed (e.g., plants C and D) or deleted (e.g., plant I). Please provide the specific reason for these changes.

Response 5:

Response 5A (with reference to RAI-1-10): The calculation error referred in Reference 2 [♦] was an incorrect nuclide density value used in the cross-section mixing process. Specifically, the oxygen nuclide density which was used to form the mixture for the peripheral bundle region was too high, resulting in under-prediction of dosimetry reaction rates in the capsules. This error was corrected and the new results were documented in the response to RAI-1-10.

Response 5B (with reference to RAI-1-17): Revised Table 6-1 in Reference 3 differs from the original table of Reference 1, specifically regarding the following plants:

♦ References for Response #5 (or RAI-2-5) are shown on the following page.

Response 5: REFERENCE LIST

1.) NEDC-32983P as transmitted in "MFN 00-035, Submittal of GE Proprietary Document NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations, September 1, 2000."

2.) The response to Question #10 (or RAI-1-10) as transmitted in "MFN 01-003, Completion of Responses to Request for Additional Information – GE Nuclear Energy Licensing Topical Report NEDC-32983P RAI#'s: 6, 9, 10, 14, 17, 18, and 19, January 17, 2001."

3.) The response to Question #17 (or RAI-1-17) as transmitted in "MFN 01-003, Completion of Responses to Request for Additional Information – GE Nuclear Energy Licensing Topical Report NEDC-32983P RAI#'s: 6, 9, 10, 14, 17, 18, and 19, January 17, 2001."

Question 10:

Is the "statistical" argument presented in R9-1 valid in view of your finding in response to RAI-17 that separate subsets of the data base yields two different biases?

Response 10:

The original question [RAI-1-9, Reference 1 [♦]] was this:

> In view of the proposed BWR ISP (integrated surveillance program), it is almost certain that the methodology will be applied to plants without recent credible dosimetry. The report should assure that the benchmarking is based on all available dosimetry data and data from all types of reactors for which the methodology is intended and (sic) are included in the ISP. Any such justification should be based on acceptable statistical arguments.

Our approach has been to use those cases for which both the results of dosimetry, and the results of calculation are available, to estimate calculational bias, and generally to validate the computational methodology by comparison with historical, empirical data.

Our database comprises 68 cases of BWRs with dosimetry data, for only 21 of which we presently have also the corresponding, calculated fluences.

Clearly, one cannot estimate bias by comparison with dosimetry for the 68-21=47 cases for which no calculated fluences are yet available. Therefore, the issue of whether the subset of 21 indeed is, in some sense, representative of the set of 68, cannot be addressed on the basis of the difference that there may exist between measurement and calculation - it has to be addressed on other bases.

In our reply to that question 9 (Reference 2), we chose to address the issue on the basis of the measured fluences, and asked how typical the measured fluences in the aforementioned subset of 21 would be if this subset were a random sample of the set of 68, rather than merely an accidental sample.

We concluded that they would be rather typical, in the following sense: about 19% of the random samples that could have been drawn from the set of 68 would have been more dissimilar from this set than the subset of 21 we do have, when dissimilarity is gauged by a distance between empirical distribution functions (of fluence values in the full set, and in the sample). A similar analysis, applied to the values of the water gap, led to a similar conclusion.

♦ References for Response #10 (or RAI-2-10) are shown at the end of the response or section.

Recapitulating: it is impossible to ascertain that the subset of 21 is representative of the set of 68 it was drawn from, on the count of calculational bias, for the simple reason that we do not yet have computed fluences for the remaining 47 cases; but we can answer the question, in the affirmative, for characteristics like measured fluence or size of the water gap, whose values we know for all the cases.

It should be noted that the representativeness of the subset of 21 is quite unrelated to the homogeneity of the set of 68 with regards to bias, or with regards to any other characteristic. Indeed, although the set of 68 is far from homogeneous on the count of measured fluence, for example, its subset of 21 does capture this heterogeneity sufficiently well to be deemed "representative" of the whole.

Response 10: REFERENCE LIST

1.) US NRC Letter, R. M. Pulsifer to J. F. Klapproth (GE), dated November 15, 2000: Request for Additional Information – Topical Report NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations.

2.) The response to Question #9 (or RAI-1-9) as transmitted in "MFN 01-003, Completion of Responses to Request for Additional Information – GE Nuclear Energy Licensing Topical Report NEDC-32983P RAI#'s: 6, 9, 10, 14, 17, 18, and 19, January 17, 2001."

GE Nuclear Energy

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Manager, Engineering and Technology

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MFN 01-009

March 14, 2001

US Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Attention: Chief, Information Management Branch
Program Management
Policy Development and Analysis Staff

Subject: **Completion of Response to Request for Additional Information (Round Two)-GE Nuclear Energy Licensing Topical Report NEDC-32983P RAI#'s: 1, 2, 4, 6, 7, 8, and 9**

As requested in Reference 1, this letter provides, as an attachment, the remaining final response to the second round Request for Additional Information (RAI-2) associated with the NRC Staff's review of GE Nuclear Energy LTR NEDC-32983P (Reference 2). Attachment 1 provides specific responses to the remaining seven of the additional ten RAIs identified in Reference 1. Attachment 2 provides an update to the Section 7.0 originally provided in Reference 2. Previous letters have provided responses to the first round RAIs (See References 3, 4, and 5) and a partial response to the second round RAIs (Reference 6).

GE has now provided a response to each of the nineteen first round RAIs and to each of the ten second round RAIs. With the completion of the second round responses, GE believes that the NRC has sufficient information to complete its review and provide an SER by April 2001. In support of that schedule, GE offers to review the draft SER for proprietary content, on an expedited basis.

Please note that the attachment contains proprietary information of the type that GE maintains in confidence and withholds from public disclosure. The information has been handled and classified as proprietary to GE as indicated in the attached affidavit. GE hereby requests that this information be withheld from public disclosure in accordance with the provisions of 10CFR2.790.

Sincerely,

J.F. Klapproth

J. F. Klapproth, Manager
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Attachments:

Affidavit by David J. Robare, dated March 14, 2001 (4 pages)

GE responses to Round Two RAI's: 1, 2, 4, 6, 7, 8, and 9. (22 pages)

Revised Section 7.0 of NEDC-32983P (4 pages)

cc:

R. M. Pulsifer	(NRC)	w/ attachments
G. S. Shukla	(NRC)	w/o attachments
M. A. Mitchell	(NRC)	w/o attachments
L. Lois	(NRC)	w/o attachments
C. E. Carpenter	(NRC)	w/o attachments
K. E. Wichman	(NRC)	w/o attachments
R. S. Drury	(GE)	w/ attachments

REFERENCE LIST

1.) US NRC Letter, G. S. Shukla to J. F. Klapproth (GE), dated February 12, 2000: Request for Additional Information – Topical Report NEDC-32983P, **General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations**.

2.) MFN 00-035, Submittal of GE Proprietary Document NEDC-32983P, **General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations**, September 1, 2000.

3.) MFN 01-003, **Completion of Responses to Request for Additional Information – GE Nuclear Energy Licensing Topical Report NEDC-32983P RAI#'s: 6, 9, 10, 14, 17, 18, and 19**, January 17, 2001.

4.) MFN 01-001, **Partial Response to Request for Additional Information-GE Nuclear Energy Licensing Topical Report NEDC-32983P–Question # 11**, January 5, 2001.

5.) MFN 00-054, **Partial Response to Request for Additional Information-GE Nuclear Energy Licensing Topical Report NEDC-32983P**, December 20, 2000.

6.) MFN 01-006, **Partial Response to Request for Additional Information (Round Two)-GE Nuclear Energy Licensing Topical Report NEDC-32983P**, March 2, 2001.

Attachment 1 to MFN 01-009

Completion of Response to Round Two RAI Questions: 1, 2, 4, 6, 7, 8, and 9

Addressing Request for Additional Information Topical Report NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations.

The following nomenclature will be used to reference a specific RAI to help minimize any potential confusion of the part of the reader: RAI-1-X, where X would represent the specific question number. For example, "RAI-1-3" would refer to the third question in the first set of RAIs as shown in the 11/15/2000 NRC letter.

RAI-1 US NRC Letter, R. M. Pulsifer to J. F. Klapproth (GE), dated November 15, 2000: Request for Additional Information – Topical Report NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations. (Questions 1 through 19)

RAI-2 US NRC Letter, G. S. Shukla to J. F. Klapproth (GE), dated February 12, 2001: Request for Additional Information – Topical Report NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations. (Questions 1 through 10)

Question 1:

The synthesis equation of Section 2.1 does not account for the axial dependence of the fluence when the fluence is calculated at $z = z'$ (as appears to be the case in the Chapter-3 calculations). Please discuss this approximation and its effect on: (1) the calculations of the Chapter-3 benchmark problem, (2) the Chapter-5 dosimetry benchmark measurements, and (3) the BWR licensing calculations.

Response 1:

The 2-D flux synthesis approach is based on the premise that the three-dimensional flux distribution can be approximated with the combined results of one or more lower dimensional flux calculations. The synthesis equation used in DG-1053 is

$$\phi(r,\theta,z) = \phi(r,\theta) * L(r,z) \qquad \text{(Eq. R2-1)}$$

where $\phi(r,\theta)$ is the flux distribution in (r,θ) geometry for a representative z-plane and $L(r,z)$ is the axial shape factor. DG-1053 also recommended the determination of the axial shape factor by the following equation:

$$L(r,z) = \phi(r,z) / \phi(r) \qquad \text{(Eq. R2-2)}$$

where $\phi(r,z)$ and $\phi(r)$ are two- and one-dimensional flux solutions, respectively, for a cylindrical representation of the geometry that preserves the important axial source and attenuation characteristics.

Figure R2-1 Sample Plant Normalized Peripheral Bundle Power Density

Question 2:

Is the conservatism resulting from the use of bundle power in determining the core neutron source identified in Table 3-6 (Case-3 vs. Case4) applicable to all core designs? For example, are there core designs where the peripheral bundle pin power distribution is relatively flat and the bias is significantly less (e.g., in fuel bundles at higher elevations where the void fraction is maximum) or even non-conservative?

The recent trend toward thermal power uprates has led to flatter radial power distributions which complicates the reactor core calculations. In light of this fact, describe the methods used to calculate the source for the transport solution and justify its use for this application including reference to low power range monitor (LPRM) data as near to the core periphery as possible.

Response 2:

In general, the local pin power distribution near the periphery of a BWR core slopes downward (in a conservative direction) toward the edge of the core. This principle is the result of the design goal of a relatively flat pin power distribution (when the bundle is in the interior of the core) combined with the effect of the high leakage environment for the bundle when it is located at the periphery of the core. However, depending on various circumstances, such as the initial enrichment distribution, lattice void distribution and pin burnup distribution, there may be localized situations where there is a reduction or possibly a reversal of this edge effect. Therefore, it cannot be proven in all circumstances that the flat interior local power distribution is conservative for every rod in every node. The fluence on the reactor vessel wall is determined by the cumulative effect of many rods within several bundles, which are not likely to be non-conservative at the same time. Furthermore, any local peaking, which may exist at the beginning of a cycle, can be expected to be reduced as a result of burnup by the end of the cycle, thus reducing the effect on fluence.

The source distribution for the neutron flux solution is based on the nodal power distribution from the GE BWR simulator. The simulator is composed of a three-dimensional, one and one half group, coarse-mesh nodal diffusion theory model coupled with a multi-channel, two-phase thermal hydraulic flow and void distribution model. The simulator is qualified for calculating the nodal and radial power distributions for all operating BWRs including those that have undergone power uprates. The qualification is based on comparisons with gamma scan data (including peripheral bundles) as well as Traversing In-core Probe (TIP) data from operating plants. [Local Power Range Monitor (LPRM) data by itself is not useful for steady-state qualification, since it must always be calibrated by the TIP system and since there are only four data points per instrument assembly, compared with the twenty-four data points available in a TIP trace.] The TIP data by itself is not particularly useful for qualification of the peripheral bundle powers, since four separate bundles contribute directly to each TIP reading and since only one of the four, at most, is a peripheral bundle.

Question 4:

The determination of the semi-empirical cross sections used to relate the flux (E > 1 .0 MeV) to. the count rate measurements should be described. Also, please describe the spectrum unfolding method used to relate the measured reaction rate to the flux (E > 1 .0 MeV) and its validation. Please provide and justify the definition and describe - the application of: (1) the cross section as a function of the Fe-54/Cu-63 reaction rate ratio, (2) the Fe-54/Cu-63 reaction rate ratio as a function of the size of the water gap, and (3) the semiempirical cross section method in the analysis of nickel dosimetry measurements. Please provide the validation of these cross sections and the determination of the cross section uncertainty.

Response 4:

---

* References are shown at the end of the section.

Question 6:

The fluence calculation uncertainty analysis of Section 7.1 should be updated to include the effect of the uncertainty in the bundle and pin power distributions on the calculated fluence.

Response 6:

The fluence calculation uncertainty analysis of Section 7.1 has been updated to include the effect of bundle and pin power distribution uncertainty (see attachment). The following is an explanation of the methods for estimating the uncertainties.

[redacted]

Question 7:

The C/M comparisons of Table 6-1 indicate a substantial systematic underprediction of the BWR 3/6 fluence by the proposed current, method. This underprediction is substantially larger than the increase being applied for BWR 4/5s. In view of the magnitude of this bias and the much smaller bias being applied for BWR 4/5s, provide: (1) an explanation for this bias, and (2) justification for using the smaller bias being applied to BWR 4/5s fluence calculations as an initial estimate of the bias for BWR 3/6s. (Are the two operating BWR 2s left out of your proposed methodology?)

Response 7:

Question 8:

It is stated in Section-8 that for BWR 3/6s the initial flux estimate will assume the bias based on the BWR 4/5 surveillance data, and the flux "may be benchmarked against actual measurements to justify whether it is necessary to re-adjust the bias." In view of the substantial differences observed between the BWR 4/5 and BWR 3/6 benchmarking comparisons, provide the criteria and basis that will be used to determine (what?) when the benchmarking against actual measurements will be performed.

Response 8:

In the answer to Question 7 we explained our current approach to the estimation of bias based on historical data: that all BWR reactors, irrespective of their model, are treated on the same footing, once their corresponding calculated fluences have been adjusted, when necessary, for the shadowing effect of the jet pumps.

Our intention is still to continue to compare calculated and measured fluences as fresh data become available: however, while a new estimate of bias, based on new and old data, will remain within the current bias plus or minus its current standard error, then no modification will be undertaken.

If a modification is undertaken, then the estimates of both the bias and of its uncertainty will be revised, and, if necessary, the formula that is used to produce the corrected fluence will be revised also, according to the specification in Section 1.4.3 of NRC's Draft Regulatory Guide DG-1053.

Question 9:

In your response to RAI-10 the actinide measurements were excluded "due to unavailability of photo-fission data." The fact that the photo-fission calculations have not been performed, when they can be performed with existing models, is not an acceptable reason for the exclusion of this data. Also the 71-degree azimuth wire data has been excluded "..in order to simplify the analysis..." Simplification of the analysis is not an acceptable reason for excluding this data. These exclusions amount to selective data rejection. Please provide the M/C comparisons for the excluded data or justify the exclusions with valid statistical arguments.

Response 9:

Additional DORT calculations have been performed to analyze (1) the dosimetry wires at the 71-degree azimuth and, (2) fast fission response of the actinide wires in the shielded capsules. The results of these additional calculations are combined with the results of the dosimetry wires at the 4- and 20-degree azimuth, previously reported in our response to Question 10 of RAI-1. The expanded results are summarized below.

71-Degree Dosimeters

Attachment 2 to MFN 01-009

Completion of Response to Round Two RAI Revised Section 7.0 of NEDC-32983P.

The following nomenclature will be used to reference a specific RAI to help minimize any potential confusion of the part of the reader: RAI-1-X, where X would represent the specific question number. For example, "RAI-1-3" would refer to the third question in the first set of RAIs as shown in the 11/15/2000 NRC letter.

RAI-1	US NRC Letter, R. M. Pulsifer to J. F. Klapproth (GE), dated November 15, 2000: Request for Additional Information – Topical Report NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations. (Questions 1 through 19)
RAI-2	US NRC Letter, G. S. Shukla to J. F. Klapproth (GE), dated February 12, 2001: Request for Additional Information – Topical Report NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations. (Questions 1 through 10)

Attachment RAI-2-1

GE Nuclear Energy

James F. Klapproth
[illegible]

[illegible]
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[illegible]
[illegible]
[illegible]

MFN 01-023

June 1, 2001

US Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Attention: Chief, Information Management Branch
Program Management
Policy Development and Analysis Staff

Subject: **Response to Request for Additional Information--**
GE Nuclear Energy Licensing Topical Report NEDC-32983P
RAI#'s: 1 through 8

As requested in Reference 1, this letter provides, as an attachment, the responses to an additional Request for Additional Information (RAI-3) associated with the NRC Staff's review of GE Nuclear Energy LTR NEDC-32983P (Reference 2). Attachment 1 provides specific responses to the eight additional RAIs identified in Reference 1. Previous letters have provided responses to the first round RAIs (See References 3, 4, and 5) and to the second round RAIs (Reference 6 and 7).

GE has now provided a response to each of the nineteen first round RAIs and to each of the ten second round RAIs, as well as the eight additional RAIs (RAI-3). With the completion of this third set of responses, GE believes that the NRC has sufficient information to complete its review and provide an SER by July 2001. In support of that schedule, GE offers to review the draft SER for proprietary content, on an expedited basis.

In response to Reference 1, GE revised its previous submittal of the calculation bias and uncertainty assessment of GE flux evaluation methodology. This revision took full advantage of all historical plant data. Although statistically sound, this approach resulted in a bias with an uncertainty in the bias being larger than the bias itself. The NRC Staff viewed this result as not acceptable. The results were discussed, during a conference call, with the Staff on April 20, 2001.

To address the NRC Staff's concern, GE has sought to increase the source data set by taking full advantage of individual wire measurements and the inclusion of data from the In-Reactor Irradiation Monitoring (IRIM) experiment. This increased data set is now the basis for the calculational bias assessment presented in this set of responses (i.e., RAI-3).

GE believes the attached responses meet the spirit of the technical discussion of 4/20/01. However, to facilitate understanding of the responses and the resolution of the NRC staff's review of the additional RAI responses, GE requests a conference call with the NRC staff, before 6/8/01, to facilitate resolution of the review.

Please note that the attachment contains proprietary information of the type that GE maintains in confidence and withholds from public disclosure. The information has been handled and classified as proprietary to GE as indicated in the attached affidavit. GE hereby requests that this information be withheld from public disclosure in accordance with the provisions of 10CFR2.790.

Sincerely,

J. F. Klapproth

J. F. Klapproth, Manager
Engineering and Technology
GE Nuclear Energy
(408) 925-5434
james.klapproth@gene.ge.com

Attachments:

Affidavit by George B. Stramback, dated June 1, 2001 (4 pages)
GE response to 4/27/01 RAI's (18 pages)

cc:

R. M. Pulsifer	(NRC)	w/ attachments
G. S. Shukla	(NRC)	w/o attachments
M. A. Mitchell	(NRC)	w/o attachments
L. Lois	(NRC)	w/o attachments
C. E. Carpenter	(NRC)	w/o attachments
K. E. Wichman	(NRC)	w/o attachments
R. S. Drury	(GE)	w/ attachments

REFERENCE LIST

1.) US NRC Letter, R. Pulisfer (G. S. Shukla) to J. F. Klapproth (GE), dated April 27, 2000: Request for Additional Information – Topical Report NEDC-32983P, **General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations**.

2.) MFN 00-035, Submittal of GE Proprietary Document NEDC-32983P, **General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations**, September 1, 2000.

3.) MFN 01-003, **Completion of Responses to Request for Additional Information – GE Nuclear Energy Licensing Topical Report NEDC-32983P RAI#'s: 6, 9, 10, 14, 17, 18, and 19**, January 17, 2001.

4.) MFN 01-001, **Partial Response to Request for Additional Information-GE Nuclear Energy Licensing Topical Report NEDC-32983P–Question # 11**, January 5, 2001.

5.) MFN 00-054, **Partial Response to Request for Additional Information-GE Nuclear Energy Licensing Topical Report NEDC-32983P**, December 20, 2000.

6.) MFN 01-009, **Partial Response to Request for Additional Information (Round Two)-GE Nuclear Energy Licensing Topical Report NEDC-32983P**, March 14, 2001.

7.) MFN 01-006, **Partial Response to Request for Additional Information (Round Two)-GE Nuclear Energy Licensing Topical Report NEDC-32983P**, March 2, 2001.

Attachment to MFN 01-023

Response to Round Three RAIs

Addressing Request for Additional Information Topical Report NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations.

The following nomenclature will be used to reference a specific RAI to help minimize any potential confusion of the part of the reader: RAI-1-X, where X would represent the specific question number. For example, "RAI-1-3" would refer to the third question in the first set of RAIs as shown in the 11/15/2000 NRC letter.

RAI-1 US NRC Letter, R. M. Pulsifer to J. F. Klapproth (GE), dated November 15, 2000: Request for Additional Information – Topical Report NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations. (Questions 1 through 19)

RAI-2 US NRC Letter, G. S. Shukla to J. F. Klapproth (GE), dated February 12, 2001: Request for Additional Information – Topical Report NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations.
(Questions 1 through 10)

RAI-3 US NRC Letter, R. M. Pulsifer to J. F. Klapproth (GE), dated April 27, 2001: NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations.
[Request for Additional Information – (Questions 1 through 8)]

Note that each RAI response may have response specific references, which are listed at the end of each section and denoted by super-scripted brackets: [x].

Question 1:

In RAI-2-7, the bias in the fluence calculation is given as -6.4% ± 11.6% based on historical data. In attachment RAI-2-1, this bias is used to adjust the calculated pressure vessel fluence. The 11.6% (one-σ) uncertainty in this bias is substantially larger than the bias itself and does not provide the required level of confidence in the bias estimate and is therefore not acceptable.

Responses to Questions 1, 2, & 7:

References

1. US NRC Letter, G. S. Shukla to J. F. Klapproth (GE), dated February 12, 2001: **Request for Additional Information – Topical Report NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations.** (Questions 1 through 10).
2. Attachment 1 to MFN 01-009, **Completion of Response to Round Two RAI Questions: 1, 2, 4, 6, 7, 8, and 9**, March 14, 2001.
3. US NRC Letter, R. M. Pulsifer to J. F. Klapproth (GE), dated April 27, 2001: **Request for Additional Information – NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations.** (Questions 1 through 8.)
4. Attachment 2 to MFN 01-009, **Completion of Response to Round Two RAI Revised Section 7.0 of NEDC-32983P**, March 14, 2001
5. Teleconference GE to NRC, April 20, 2001.

Question 2:

The entries in the corrected Table 6-1 (Attachment R2.3-1) show wide variations, resulting in a large uncertainty. The number of plants represented in the table entries is small relative to the total number of plants to which the methodology will apply. A larger number of entries of the same quality should reduce the uncertainty.

Response 2:

(The response to Question 2 is included in the response to Question 1.)

Question 3:

Can you please clarify the last paragraph on page R2-10?

Response 3:

Question 4:

In R2-4, Table 1, the cross section uncertainties are estimated to be 10% (one-σ). What analysis has been performed to support this value?

Response 4:

Question 5:

In R2-4, Tables 2, 3, 4, 5 and 6, how is the size of the water gap calculated? Does this correlation assume an equivalent cylindrical surface?

Response 5:

Question 6:

In the R2-7, adjustment of 15.5% (estimated from Figure 3-5) to account for jet pump shadowing (R18-1), how do you account for the spectrum variation in the back of the pump or pump riser? If Figure 3-4 was used, the value of the adjustment would be higher. What is the justification for using Figure 3-5?

Response 6:

Question 7:

In R2-7, the use of engineering judgment as a substitute for BWR-2 data is not an acceptable practice. In view of the lack of C/M surveillance data for BWR-2s, provide justification for the application of the proposed methodology to BWR-2 plants.

Response 7:

(The response to Question 7 is included in the response to Question 1.)

Question 8:

Provide an explanation for the radially-dependent bias in the benchmark experiment C/M data of Figures 5-4, 5-5 and R9-1. Does this indicate that calculations based on the proposed methodology will over-predict the fluence at locations inside the vessel inner-wall? Provide justification for not including this bias in the calculation methodology. In view of the data in Figure 9-1:

- *Is the proposed bias adjustment indeed applicable to the shroud?*
- *Should another bias adjustment be generated?*
- *Should the bias adjustment include an axial (z) dependence?*

Response 8:

The alternative model for IRIM benchmark calculations utilized radial-dependent water density in the downcomer, based on a temperature distribution in the downcomer region (Section 5.3.4 of NEDC-32983P). The objective was to obtain best-estimate calculational results for the experiment, compared to the measured reaction rates in each capsule. The radial dependent bias in the benchmark C/M data is a direct result of utilizing this model. The results from MCNP calculation (Tables 5-4a, 5-4b, and 5-4c of NEDC-32983P) also show this radial dependency; but with a smaller radial bias than that of the DORT results.

In light of this radial bias, we have used separate bias estimates for the shroud and the vessel in our calculation methodology. The bias and uncertainty estimate for the vessel is documented in the response to Questions 1 and 2 of this round of RAI. The bias and uncertainty estimate for the shroud is addressed here.

Analytical Bias and Uncertainty for Shroud Flux

The calculational uncertainties estimated for the shroud are similar to those for the vessel. The breakdown of uncertainty items for the vessel is documented in NEDC-32983P Section 7.0 [1]. For the shroud, the principal differences are the following:

1. Downcomer water density will not affect the shroud flux and, therefore, is set to zero.
2. Shroud dimension variation and impact to the flux is different from the vessel dimension.

Benchmark Bias and Uncertainty for Shroud Flux

Unlike the vessel flux, comparison of calculation and measurement data from operating plants for the shroud is not available and, therefore, no equivalent bias can be obtained.

Combined Bias and Uncertainty

1. Unlike for the vessel flux, there is no historical operating plant measurement data to supplement the simulator benchmark results. Using the bias estimate from one calculation alone for adjusting all future calculations, in our view, is not a practical and technically sound approach.
2.

Axial Bias

References

1. Attachment 2 to MFN 01-009, **Completion of Response to Round Two RAI, Revised Section 7.0 of NEDC-32983P**, March 14, 2001.
2. Attachment 1 to MFN 01-009, **Completion of Response to Round Two RAI, Questions: 1, 2, 4, 6, 7, 8, and 9**, March 14, 2001.

GE Nuclear Energy

James F. Klapproth
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MFN 01-024

June 15, 2001

US Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Attention: Chief, Information Management Branch
Program Management
Policy Development and Analysis Staff

Subject: **Response to Request for Additional Information--**
GE Nuclear Energy Licensing Topical Report NEDC-32983P
Verbal Request During June 8, 2001 Conference Call

As requested during the June 8, 2001 conference call (Reference 1), this letter provides, as an attachment, the response to an additional request for clarification associated with the NRC Staff's review of GE Nuclear Energy LTR NEDC-32983P (Reference 2). The Attachment provides the specific response to the NRC Staff's request for supplemental information regarding the source of the calculated and measured flux data for the In-Reactor Irradiation Monitoring (IRIM) experiment, as shown in Table R3-1 of Reference 3.

With this follow-on clarification to RAI-3 (Reference 3), GE believes that the NRC has sufficient information to complete its review and provide an SER by July 2001. In support of that schedule, GE offers to review the draft SER for proprietary content, on an expedited basis.

Please note that the attachment contains proprietary information of the type that GE maintains in confidence and withholds from public disclosure. The information has been handled and classified as proprietary to GE as indicated in the attached affidavit. GE hereby requests that this information be withheld from public disclosure in accordance with the provisions of 10CFR2.790.

Sincerely,

M. E. Harding for

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Attachments:

Affidavit by George B. Stramback, dated June 15, 2001 (4 pages)
GE response to 6/8/01 verbal request (6 pages)

cc:

R. M. Pulsifer	(NRC)	w/ attachments
G. S. Shukla	(NRC)	w/o attachments
M. A. Mitchell	(NRC)	w/o attachments
L. Lois	(NRC)	w/o attachments
C. E. Carpenter	(NRC)	w/o attachments
K. E. Wichman	(NRC)	w/o attachments
R. S. Drury	(GE)	w/ attachments

REFERENCE LIST

1.) Conference Call: GE Nuclear Energy and NRC Staff, June 8, 2001, **Initial Discussion of RAI-3 Responses**.

2.) MFN 00-035, Submittal of GE Proprietary Document NEDC-32983P, **General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations**, September 1, 2000.

3.) MFN 01-023, **Response to Request for Additional Information–GE Nuclear Energy Licensing Topical Report NEDC-32983P: RAI#'s 1 through 8**, June 1, 2001.

Attachment to MFN 01-024

Supplemental Information on IRIM Flux Data

Supplemental Information on IRIM Flux Data

During a teleconference between GE and NRC on June 8, 2001, the NRC staff requested supplemental information regarding the source of the calculated and measured flux data for the In-Reactor Irradiation Monitoring (IRIM) experiment, as shown in Table R3-1 of GE response to Question 1 of the 3rd round of RAI's [1]. This note provides the requested information.

Calculated Flux

The IRIM experiment was used as a simulator benchmark to meet the guidelines set forth in Regulatory Guide 1.190 and its predecessor, Draft Regulatory Guide DG-1053. The GE discrete ordinates flux solution methodology, as described in Section 2.1 of the Licensing Topical Report (LTR), NEDC-32983P [2], was used to provide the 3-D flux distribution by synthesizing 2-D flux solutions in (r,z) and (r,θ) models. Detailed descriptions of the flux calculation model using the DORT computer code were provided in Section 5.3 of the LTR.

The 18 calculated fluxes shown in Table R3-1 (Reference 1) represent the fast neutron flux ($E > 1$ MeV) at each of the 18 capsule locations near the RPV, calculated with the GE synthesized flux solution methodology. The first nine fluxes are for the bare capsules for the three axial elevations (low, middle, and high) at each of the three azimuths (20, 4, and 70 degrees). Similarly, the next nine fluxes are for the shielded capsules near the RPV.

Reaction rates, as reported in Tables 5-2 and 5-3 of the LTR and expanded in Table R9-1 of GE response to the 2nd round of RAI's [3] in the form of C/M ratios for each dosimeter in the capsule, are determined with the combination of the capsule fluxes and dosimetry reaction cross sections. The procedure for folding the fluxes and the cross sections to obtain the dosimeter reaction rates (reactions per second per nucleus) and the specific activities at the end of irradiation, dps/g (disintegrations per second per gram of dosimeter isotope), is documented in Section 5.3.5 of the LTR. The same calculation is used to determine the calculated fluxes in Table R3-1 and the reaction rate ratios in Table R9-1. Hence, these results are self-consistent.

Measured Flux

The 18 measured fluxes shown in Table R3-1 (Reference 1) represent the fast neutron flux ($E > 1$ MeV) at each of the 18 capsule locations near the RPV. Each capsule flux was derived from the measured reaction rates of all non-actinide dosimeters within the capsule, using a multi-group flux unfolding procedure. Corresponding to the calculated fluxes, the first nine fluxes are for the bare capsules and the next nine fluxes are for the shielded capsules near the RPV.

The multi-group flux unfolding procedure, developed for the IRIM experiment, is an iterative process consisting of several calculation steps with the goal of obtaining an optimal neutron spectrum for each capsule location. These steps include the following:

<u>Consistency Check of Capsule Fluxes and Dosimeter Reaction Rates</u>

Conclusions

Based on the data analysis performed for the near-RPV capsules, the C/M ratios for the fast flux and dosimeter reaction rate in these capsules are self-consistent. It is concluded that the bias estimation based on the calculated and measured capsule fluxes is not significantly different from that based on the calculated and measured dosimeter reaction rates.

References

1. Attachment to MFN 01-023, "Response to Request for Additional Information – GE Nuclear Energy Licensing Topical Report NEDC-32983P RAI#'s: 1 through 8," June 1, 2001.
2. Licensing Topical Report, NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations," August 2000.
3. Attachment 1 to MFN 01-009, "Completion of Response to Round Two RAI Questions: 1, 2, 4, 6, 7, 8, and 9," March 14, 2001.

Table R3A-1 IRIM Calculated and Measured Fast Fluxes at Near-RPV Capsules

Table R3A-2 IRIM Reaction Rate C/M's at Near-RPV Capsules

APPENDIX B

GE RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION REGARDING REMOVAL OF METHODOLOGY LIMITATIONS

1. Letter from G. B. Stramback, GE Nuclear Energy to U.S. Nuclear Regulatory Commission, "Proprietary Content – GE Flux Calculation Methodology Confirmation Result Part I – Surveillance Capsule Flux at River Bend Station," MFN 03-017, March 13, 2003.
2. Letter from G. B. Stramback, GE Nuclear Energy to U.S. Nuclear Regulatory Commission, "Confirmatory Information on GE Methodology for RPV Flux Calculation (Re: NEDC-32983P-A)," MFN 04-068, July 14, 2004.
3. Letter from G. B. Stramback, GE Nuclear Energy to U.S. Nuclear Regulatory Commission, "Confirmatory Information on GE Methodology for Shroud Flux Calculation (Re: NEDC-32983P-A)," MFN 04-097, September 10, 2004.
4. Letter from G. B. Stramback, GE Nuclear Energy to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information – GE Nuclear Energy Licensing Topical Report NEDC-32983P-A (TAC No. MC3788)," MFN 04-128, December 2, 2004.
5. Letter from G. B. Stramback, GE Nuclear Energy to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information – GE Nuclear Energy Licensing Topical Report NEDC-32983P-A (TAC No. MC3788)," MFN 05-039, May 20, 2005.

GE Nuclear Energy

General Electric Company
175 Curtner Avenue
San Jose, CA 95125

MFN-03-017
March 13, 2003

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

Attention: Chief, Information Management Branch
Program Management
Policy Development and Analysis Staff

Subject: **Proprietary Content - GE Flux Calculation Methodology Confirmation Result Part I - Surveillance Capsule Flux at River Bend Station**

Reference: MFN 03-005, "GE Flux Calculation Methodology Confirmation Result Part I - Surveillance Capsule Flux at River Bend Station," January 29, 2003.

The referenced letter transmitted the result of a predictive calculation for the flux surveillance capsule at River Bend Station. Information contained in the attachment of the referenced letter was designated as proprietary. GE has subsequently reviewed the information and has determined that the information designation as proprietary can be more limited.

Attachment 1 hereto provides a re-designation of the proprietary information. The information designated as proprietary is information that GE customarily maintains in confidence and withholds from public disclosure. Attachment 2 provides a non-proprietary version suitable for public disclosure.

Attachment 3 provides an affidavit, which identifies that the designated information has been handled and classified as proprietary to GE. GE hereby requests that the designated proprietary information be withheld from public disclosure in accordance with the provisions of 10 CFR 2.790 and 9.17.

The attachment provides only a change to the information designated as proprietary. No other changes have been made to the information originally provided by the referenced letter, and the discussions and requests of the transmittal letter are unaffected by this letter.

If you have any questions about the information provided here please contact me.

Sincerely,

Bradley J Erbes
for George Stramback

Attachments:
1. River Bend Station Capsule Flux Calculation (Proprietary)
2. River Bend Station Capsule Flux Calculation (Non-Proprietary)
3. Proprietary Affidavit

cc: L.Lois - USNRC
A. B Wang- USNRC
J.F. Klapproth
I. Nir
A. Chung
S. Sitaraman
T. Wu
S. Wang

Attachment 2

MFN-03-017

River Bend Station Capsule Flux Calculation

Non-Proprietary

ATTACHMENT 2

River Bend Station Surveillance Capsule Flux Calculation

This surveillance capsule has resided near the reactor pressure vessel wall of River Bend Station at azimuth 183° since the beginning of operation, and was withdrawn during the refueling outage in 2000. The calculated flux is as follows.

Full Power Fast (> 1 MeV) Neutron Flux [ ]

This flux prediction is calculated based on the methodology described in NEDC-32983P-A "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation,". Because this result will be added to the original database of calculation-to-measurement ratios, the value presented here is the un-corrected result of calculation alone. The bias adjustment factor stated in NEDC-32983P-A has not been applied to this flux value.

GE Nuclear Energy

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MFN-04-068
July 14, 2004

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

Attention: Chief, Information Management Branch
Program Management
Policy Development and Analysis Staff

Subject: **Confirmatory Information on GE Methodology for RPV Flux Calculation (Re: NEDC-32983P-A)**

GE is submitting in Enclosure 2 the results of four additional reactor pressure vessel (RPV) flux calculations together with data from the dosimetry measurements, to resolve and remove the limitations set forth in the NRC safety evaluation on the GE Methodology for RPV flux evaluations (Reference 1). These additional calculation/measurement data were combined with the original data in NEDC-32983P-A (Reference 2), and a new analysis was performed to estimate the bias and the uncertainty on the bias. The result reaffirms that the RPV flux calculational bias assessed in NEDC-32983P-A was appropriate.

This submittal completes the vessel fluence confirmatory items identified during the NRC's review of GE's plan (Reference 3) for addressing the NRC safety evaluation limitations. The other confirmatory item regarding shroud fluence will be provided in a future submittal.

The calculational results in Enclosure 2 contain GE proprietary information, as defined by 10CFR2.390. GE customarily maintains this information in confidence and withholds it from public disclosure. A non-proprietary version of the calculational results is provided in Enclosure 1.

The affidavit contained in Enclosure 3 identifies that the information contained in Enclosure 2 has been handled and classified as proprietary to GE. GE hereby requests that the information of Enclosure 2 be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390 and 9.17.

If you have any questions, please contact, Sylvia Wang at (408) 925-1594 or myself.

Sincerely,

George Stramback
Manager, Regulatory Services
GE Nuclear Energy
(408) 925-1913
george.stramback@gene.ge.com

Project No. 710

References:

1. MFN 01-050, Stuart A. Richards (NRC) to James F. Klapproth (GE), *Safety Evaluation for NEDC-32983P "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation" (TAC No. MA9891)*, September 14, 2001.
2. NEDC-32983P-A Revision 1, *General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation*, December 2001.
3. MFN 02-048, Alan Wang (NRC) to George Stramback (GE), *Plan for Addressing NRC Safety Evaluation Limitations on NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation" (TAC NO. MB2774)*, August 7, 2002
4. MFN 03-017, George Stramback to NRC, *Proprietary Content - GE Flux Calculation Methodology Confirmation Result Part I - Surveillance Capsule Flux at River Bend Station,* March 13, 2003

Enclosures:
1. Re-Assessment of Calculational Fluence Bias (Non-Proprietary Information)
2. Re-Assessment of Calculational Fluence Bias (Proprietary Information)
3. Affidavit, George B. Stramback, dated July 14, 2004.

cc: MB Fields (NRC)
L Lois (NRC)
AB Wang (NRC)
AK Chung (GE/San Jose)
JF Klapproth (GE/San Jose)
I Nir (GE/San Jose)
S Sitaraman (GE/San Jose)
SS Wang (GE/San Jose)
T Wu (GE/San Jose)
eDRF 0000-0012-4185

ENCLOSURE 1

MFN 04-068

Re-Assessment of Calculational Fluence Bias

Redacted and Non-Proprietary Information

ANALYSIS AND RESULTS

Table A lists the result of the four additional RPV flux calculations performed by GE, using the approved GE Flux Evaluation Methodology. Calculation for Plant R was performed as a double-blinded predictive evaluation, the result of which was submitted to the NRC (Reference 4) prior to the measurement data becoming available. Calculations for Plants S, T and U required elaborated extraction of old operation data from each plant in order to adequately reflect the surveillance capsule environment. Dosimetry measurement for the Plant U flux wire was performed by GE Vallecitos Nuclear Center, whereas those for Plants R, S and T were performed by other (non-GE) vendors. The calculated flux in Table A represents the un-corrected (bias not factored in) result of calculation. The measured result for Plant T is the average of Fe, Cu and Ni wires. The results from the other plants represent individual flux wires.

A revised data base combines the data in Table 7-1 of NEDC-32983P-A with the additional six pairs of data represented in Table A. [[ ]] A rigorous statistical analysis similar to the original analysis for NEDC-32983P-A was performed to determine the calculational bias of the combined data set, taking into consideration the analytical uncertainty of the calculations and the measurement uncertainties. The next section (Re-Assessment of Calculational Fluence Bias) describes the details of such analysis. The final bias assessed is [[ ]], with associated uncertainty [[.]]

[[

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Table A

[[

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Re-Assessment of Calculational Fluence Bias

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The data, after application of these corrections, are depicted in Figure A.

[[

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[[

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Figure A: ***Bias Estimation***.[[

]]

GE Nuclear Energy

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MFN-04-097
September 10, 2004

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

Attention: Chief, Information Management Branch
Program Management
Policy Development and Analysis Staff

Subject: **Confirmatory Information on GE Methodology for Shroud Flux Calculation (Re: NEDC-32983P-A)**

GE is submitting in Enclosure 2 the result of two additional shroud flux calculations together with data from the shroud sample dosimetry measurements, to resolve and remove all remaining limitations set forth in the NRC safety evaluation (Reference 1) on the GE Methodology for RPV flux evaluations (Reference 2). The result of these additional calculations and comparison with the measurement data reaffirms that GE's flux calculational method yields conservative fluence estimates on the core shroud.

This submittal completes the shroud fluence confirmatory items identified during the NRC's review of GE's plan (Reference 3) for addressing the NRC safety evaluation limitations. The other confirmatory item regarding RPV fluence was provided in a previous submittal (Reference 4).

The calculational results in Enclosure 2 contain GE proprietary information, as defined by 10 CFR 2.390. GE customarily maintains this information in confidence and withholds it from public disclosure. A non-proprietary version of the calculational results is provided in Enclosure 1.

The affidavit contained in Enclosure 3 identifies that the information contained in Enclosure 2 has been handled and classified as proprietary to GE. GE hereby requests that the information of Enclosure 2 be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390 and 9.17.

If you have any questions, please contact Tang Wu at (408) 925-2209 or myself.

Sincerely,

George Stramback

George Stramback
Manager, Regulatory Services
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(408) 925-1913
george.stramback@gene.ge.com

Project No. 710

References:

1. MFN 01-050, Stuart A. Richards (NRC) to James F. Klapproth (GE), *Safety Evaluation for NEDC-32983P "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation" (TAC No. MA9891)*, September 14, 2001.
2. NEDC-32983P-A Revision 1, *General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations*, December 2001.
3. MFN 02-048, Alan Wang (NRC) to George Stramback (GE), *Plan for Addressing NRC Safety Evaluation Limitations on NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation" (TAC NO. MB2774)*, August 7, 2002.
4. MFN 04-068, George Stramback (GE) to NRC, *Confirmatory Information on GE Methodology for RPV Flux Calculation (Re: NEDC-32983P-A)*, July 14, 2004.

Enclosures:

1. Confirmatory Shroud Flux Calculations (Non-Proprietary Information)
2. Confirmatory Shroud Flux Calculations (Proprietary Information)
3. Affidavit, George B. Stramback, dated September 10, 2004.

cc: MB Fields (NRC)
L Lois (NRC)
AB Wang (NRC)
AK Chung (GE/San Jose)
JF Klapproth (GE/San Jose)
I Nir (GE/San Jose)
S Sitaraman (GE/San Jose)
SS Wang (GE/San Jose)
T Wu (GE/San Jose)
eDRF 0000-0012-4185

ENCLOSURE 1

MFN 04-097

Confirmatory Shroud Flux Calculations

Redacted and Non-Proprietary Information

ANALYSIS AND RESULTS

Tables A and B list the result of the two additional shroud flux calculations performed by GE and comparison of calculated results with data from the shroud sample dosimetry measurements.

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Shroud flux calculations for both plants were performed with the core operating data for the cycle in which the shroud segment was removed at the end of the cycle, so that shroud sample environment is properly modeled. All shroud flux calculations were performed using the approved GE Flux Evaluation Methodology. No bias adjustment has been applied to the calculated result in Tables A and B.

The mean value for the [[ ]] pairs of calculated/measured (C/M) ratios of reaction rate is [[ ]], with a 1σ standard deviation of [[ ]]. The mean value for the C/M ratios of fast neutron flux (E > 1 MeV) is [[ ]], with a 1σ standard deviation of [[ ]]. The result of these additional calculations and comparison with the measurement data reaffirms that GE's flux calculational method yields conservative fluence estimates on the core shroud. Consequently, it is not necessary to further apply a bias to the calculated results. Application of the shroud flux formula in Section 7.5 of NEDC-32983P-A will be continued.

Table A

Calculated and Measured Reaction Rate (dps/nucleus) for Shroud Samples

Class	Plant	Shroud Sample	Calculated Reaction Rate	Measured Reaction Rate	C/M
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Table B

Calculated and Measured Fast Neutron Flux (E > 1 MeV) for Shroud Samples

Class	Plant	Shroud Sample	Calculated Flux (n/cm^2-s)	Measured Flux (n/cm^2-s)	C/M
[[					

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GE Energy

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MFN 04-128
December 2, 2004

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20852-2738

Attention: Chief, Information Management Branch
Program Management
Policy Development and Analysis Staff

Subject: **Response to Request for Additional Information – GE Nuclear Energy Licensing Topical Report NEDC-32983P-A (TAC No. MC3788)**

GE is submitting in Enclosure 2 the response to the Requests for Additional Information (RAIs) (Reference 1) associated with the NRC Staff's review of GE Nuclear Energy submittals of confirmatory information on GE Methodology for RPV flux calculation and shroud flux calculation (References 2 and 3).

With this response to the RAIs, GE believes that the NRC has sufficient information to complete its review of the confirmatory information that addresses the NRC safety evaluation limitations (Reference 4) on the GE Methodology for RPV and shroud flux evaluations (Reference 5).

The RAI responses in Enclosure 2 contain GE proprietary information, as defined by 10 CFR 2.390. GE customarily maintains this information in confidence and withholds it from public disclosure. A non-proprietary version of the RAIs is provided in Enclosure 1.

The affidavit contained in Enclosure 3 identifies that the information contained in Enclosure 2 has been handled and classified as proprietary to GE. GE hereby requests that the information of Enclosure 2 be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390 and 9.17.

If you have any questions, please contact Tang Wu at (408) 925-2209 or myself.

Sincerely,

George Stramback
Manager, Regulatory Services

Project No. 710

References:

1. MFN 04-118, Alan Wang (NRC) to James F. Klapproth (GE), *Request for Additional Information – Licensing Topical Report NEDC-32983P-A, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation" (TAC NO. MC3788)*, November 1, 2004.
2. MFN 04-068, George Stramback (GE) to NRC, *Confirmatory Information on GE Methodology for RPV Flux Calculation (Re: NEDC-32983P-A)*, July 14, 2004.
3. MFN 04-097, George Stramback (GE) to NRC, *Confirmatory Information on GE Methodology for Shroud Flux Calculation (Re: NEDC-32983P-A)*, September 10, 2004.
4. MFN 01-050, Stuart A. Richards (NRC) to James F. Klapproth (GE), *Safety Evaluation for NEDC-32983P "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation" (TAC No. MA9891)*, September 14, 2001.
5. NEDC-32983P-A Revision 1, *General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations*, December 2001.

Enclosures:
1. Response to RAI Questions (Non-Proprietary Information)
2. Response to RAI Questions (Proprietary Information)
3. Affidavit, George B. Stramback, dated December 2, 2004.

cc: MB Fields (NRC)
L Lois (NRC)
AB Wang (NRC)
JF Klapproth (GE/Wilmington)
I Nir (GE/San Jose)
DK Rao (GE/San Jose)
S Sitaraman (GE/San Jose)
SS Wang (GE/San Jose)
T Wu (GE/San Jose)
eDRF 0000-0012-4185

ENCLOSURE 1

MFN 04-128

Response to RAI Questions

Redacted and Non-Proprietary Information

Question 1:

By letter dated July 14, 2004, GENE provided information and requested removal of the limitations regarding vessel fluence calculations in licensing topical report (TR) NEDC-32983P-A. The U.S. Nuclear Regulatory Commission (NRC) staff requests that GE provide the measured and calculated values in Figure A of the July 14, 2004, letter in a tabular form.

Response 1:

Table 1 presents the tabular form of the measured and calculated values. These data were combined from data in Table 7-1 of Reference 1 and Table A of Reference 2.

Response 1 References:

1. NEDC-32983P-A Revision 1, *General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations*, December 2001.
2. MFN 04-068, George Stramback (GE) to NRC, *Confirmatory Information on GE Methodology for RPV Flux Calculation (Re: NEDC-32983P-A)*, July 14, 2004.

Table 1

RPV Flux Data for Bias Determination

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Table 1 (Cont.)

RPV Flux Data for Bias Determination

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Question 2:

By letter dated September 10, 2004, GENE, supplemented the July 14, 2004, letter and provided additional information to support its request for removal of limitations regarding boiling water reactor (BWR) shroud fluence calculations in the licensing TR. Due to the limited information provided in the submittal, the NRC staff requests the following additional information:

a. Please clarify the purpose of this submittal and the basis for any request.

b. The main issue with the existing shroud calculations in the NEDC-32983P-A topical is the axial bias of the C/M values. The submittal indicates a radial dependence of the samples but does not indicate that there is elevation dependence of the samples. Given that the axial shroud weld cracking, attributed to irradiation assisted stress corrosion, is an important BWR issue, how do you justify lifting the limitation?

c. The samples were taken from two BWR4 plants H and V. Why is it appropriate to generalize these results to all BWRs?

d. The currently submitted data were from samples irradiated in the early 1990s, why was the data not part of the initial submittal? Are the old data to be ignored?

Response 2:

a. The purpose of this transmittal is to resolve and remove limitations set forth in the NRC safety evaluation on the GE Methodology for shroud flux evaluation with the submittal of the result of two additional shroud flux calculations together with data from the shroud sample dosimetry measurements. The result of these additional calculations and comparison with the measurement data reaffirms that GE's flux calculational method yields acceptable fluence estimates on the core shroud. The basis of this request is consistent with the staff's response to GE's plan for addressing NRC limitations [References 1 (item (e)) and 2].

b. The samples from the two additional shroud fluence measurements included in the submittal were taken from a fixed elevation and, therefore, axial dependence cannot be evaluated. GE is not in possession of any elevation-dependent shroud measurement.

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As indicated in the safety evaluation of NEDC-32983P, the staff found that GE's methodology acceptable for shroud fluence calculations provided that (1) the estimates are limited within the beltline region, and (2) the bias is not deducted from the calculated value. The results from the two additional dosimetry analyses, directly related to the shroud, reaffirm the acceptability of the GE methodology as stated in the staff's safety evaluation of NEDC-32983P.

c. The governing parameters for the shroud flux calculation are the fuel bundle design (enrichment, volume fractions, etc.), bundle power distribution, and void distribution. These parameters depend principally on the core design and, in general, are not a function of BWR types. Therefore, it is appropriate to generalize these results as being applicable to all BWR types.

d. The initial submittal focused on the RPV flux evaluation and used readily available shroud data for benchmarking the methodology for the shroud flux. GE subsequently found two additional sets of data after the initial submittal that could be used for benchmarking purposes. All measured shroud data submitted to the staff were used for benchmarking purposes by GE.

Response 2 References:

1. MFN 02-048, Alan Wang (NRC) to George Stramback (GE), *Plan for Addressing NRC Safety Evaluation Limitations on NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation" (TAC NO. MB2774)*, August 7, 2002.
2. MFN 02-015, George Stramback (GE) to NRC, *Plan for Addressing NRC SER Limitations on NEDC-32983P*, March 19, 2002.
3. Attachment to MFN 01-023, James F. Klapproth (GE) to NRC, *Response to Request for Additional Information –GE Nuclear Energy Licensing Topical Report NEDC-32983P RAI#'s: 1 through 8*, Question 8, June 1, 2001.

GE Energy
Nuclear

George B Stramback
GE Nuclear Energy
Manager, Regulatory Services

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MFN-05-039
May 20, 2005

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

Attention: Mel Fields, Senior Project Manager
Section 2, Program Directorate IV
Division of Licensing Project Management

Subject: **Response to Request for Additional Information – GE Nuclear Energy Licensing Topical Report NEDC-32983P-A (TAC No. MC3788)**

During a conference call between GE and the NRC on January 10, 2005, the NRC staff raised a question regarding the statistical justification of combining the plant operating data and the In-Reactor Irradiation Monitoring (IRIM) experimental data for determining the calculational bias of the GE fluence methodology. The response to the NRC question is enclosed.

With this response to the RAI, GE believes that the NRC has sufficient information to complete its review of the confirmatory information and remove the limitations set forth in the NRC safety evaluation (Reference 1) on the GE Methodology for RPV and shroud flux evaluations (Reference 2). Based on discussions with the responsible NRC Project Manager, it was estimated that the limitation could be removed in 2 months. GE finds that schedule to be acceptable.

The RAI response in Enclosure 2 contains GE proprietary information, as defined by 10 CFR 2.390. GE customarily maintains this information in confidence and withholds it from public disclosure. A non-proprietary version of the calculational results is provided in Enclosure 1.

The affidavit contained in Enclosure 3 identifies that the information contained in Enclosure 2 has been handled and classified as proprietary to GE. GE hereby requests that the information of Enclosure 2 be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390 and 9.17.

If you have any questions, please contact Mike Lalor at (408) 925-2443 or myself.

Sincerely,

George Stramback
Manager, Regulatory Services

Project No. 710

References:

1. MFN 01-050, Stuart A. Richards (NRC) to James F. Klapproth (GE), *Safety Evaluation for NEDC-32983P "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation" (TAC No. MA9891)*, September 14, 2001.

2. NEDC-32983P-A Revision 1, *General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations*, December 2001.

Enclosures:
1. Response to NRC Questions (Non-Proprietary Information)
2. Response to NRC Questions (Proprietary Information)
3. Affidavit, George B. Stramback, dated May 20, 2005

cc: L Lois (NRC)
AB Wang (NRC)
JF Klapproth (GE/Wilmington)
I Nir (GE/ Wilmington)
DK Rao (GE/San Jose)
S Sitaraman (GE/San Jose)
SS Wang (GE/ Wilmington)
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eDRF 0000-0012-4185

ENCLOSURE 1

MFN 05-039

Response to NRC Questions

Redacted and Non-Proprietary Information

This is a non-proprietary version of Enclosure 2 which has the proprietary information removed. The portions of this enclosure that have been removed are indicated by an open and closed double bracket as shown here [[ ]].

GE Response to NRC Question 1
Statistical Analysis of RPV Flux Data for Bias Determination

During a conference call between GE and the NRC on January 10, 2005, the NRC staff raised a question regarding the statistical justification of combining the plant operating data and the In-Reactor Irradiation Monitoring (IRIM) experimental data for determining the calculational bias of the GE fluence methodology. These calculated and measured capsule flux values were documented in Table 1 of Reference 1. [[

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References for Response 1:

1. MFN 04-128, George Stramback (GE) to NRC, *Response to Request for Additional Information – GE Nuclear Energy Licensing Topical Report NEDC-32983P-A (TAC No. MC3788)*, December 2, 2004.

2. MFN 04-068, George Stramback (GE) to NRC, Confirmatory Information on GE Methodology for RPV Flux Calculation (Re: NEDC-32983P-A), July 14, 2004

GE Response to NRC Question 2
Shroud Fluence Related Issues

During a conference call between GE and the NRC on April 25, 2005, the NRC staff raised a question regarding the effect of fast neutron fluence on the shroud material property and performance. This note is GE's response to the NRC question.

Shroud Material Properties vs. Fluence:

The fluence at the shroud position is used to assess material properties that are subsequently used to evaluate the environmental and mechanical properties of stainless steels used in BWRs. It is well known that the basic material properties such as hardness and tensile properties as well as the environmental properties such as stress corrosion crack growth rates are affected by radiation damage. The rate of change in these properties is a continuous function of the amount of damage as measured by the fluence (dpa or n/m^2). However, it is also well known that small variations in fluence (e.g., ±20%) will only lead to small differences in these properties, especially compared to material (e.g., 304 vs. 316) or heat-to-heat variations. Looking only at microchemical (Cr depletion, Figure 1) or mechanical (yield strength, Figure 2) properties, there is typically greater than a 20% variation in the 304 stainless steel data. Therefore, there is little sensitivity to the projected properties within the uncertainty level from the fast fluence predictions made using the GE fluence methodology.

This is substantiated by test experience at GE and other test labs. For example, it is normal in the best run stress corrosion crack growth tests performed using un-irradiated materials to see a 50% variation between test sequences and between test materials that are conducted under identical test conditions. Tests with irradiated materials will exhibit at least that much variability. Therefore, the fluence estimate variance would have little impact on the ways these irradiated materials would be evaluated given this general variability in properties.

Another important point that should be discussed is the historical basis underlying the estimates of material degradation vs. fluence. The fluence level historically assigned to BWR materials has been based on the older GE fluence methodology. Given that its conservative bias is similar to the current fluence methodology, one must accept that the current GE calculations are more likely to represent the actual level of degradation in the material properties, such as the accelerated stress corrosion crack growth rate, even though that may be a conservative bias.

On the issue of helium production in stainless steel, this phenomenon is dominated by the thermal fluence and the current GE licensed methodology does not produce estimates of the fluence at thermal energies.

In summary, the uncertainty in the material property of the core shroud is significantly higher than the conservative bias of the GE fast fluence prediction for the shroud. The GE materials community feels strongly that the conservative bias of the fast fluence that is predicted using the GE fluence methodology at the shroud location will not have a significant effect on the estimated material properties and any subsequent materials related evaluation.

Availability of Additional Shroud Measurement Data:

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References for Response 2:

1. G.S. Was and P.L. Andresen, "SCC Behavior of Alloys in Aggressive Nuclear Reactor Core Environments," *Topical Research Symposium on Corrosion in Aggressive Environments*, Corrosion/05, NACE, Houston, 2005.
2. MFN 04-097, George Stramback (GE) to NRC, Confirmatory Information on GE Methodology for Shroud Flux Calculation (Re: NEDC-32983P-A), September 10, 2004.
3. MFN 01-050, Stuart A. Richards (NRC) to James F. Klapproth (GE), Safety Evaluation for NEDC-32983P "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation" (TAC No. MA9891), September 14, 2001.
4. NEDC-32983P-A Revision 1, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations, December 2001.
5. MFN 04-128, George Stramback (GE) to NRC, Response to Request for Additional Information – GE Nuclear Energy Licensing Topical Report NEDC-32983P-A (TAC No. MC3788), December 2, 2004

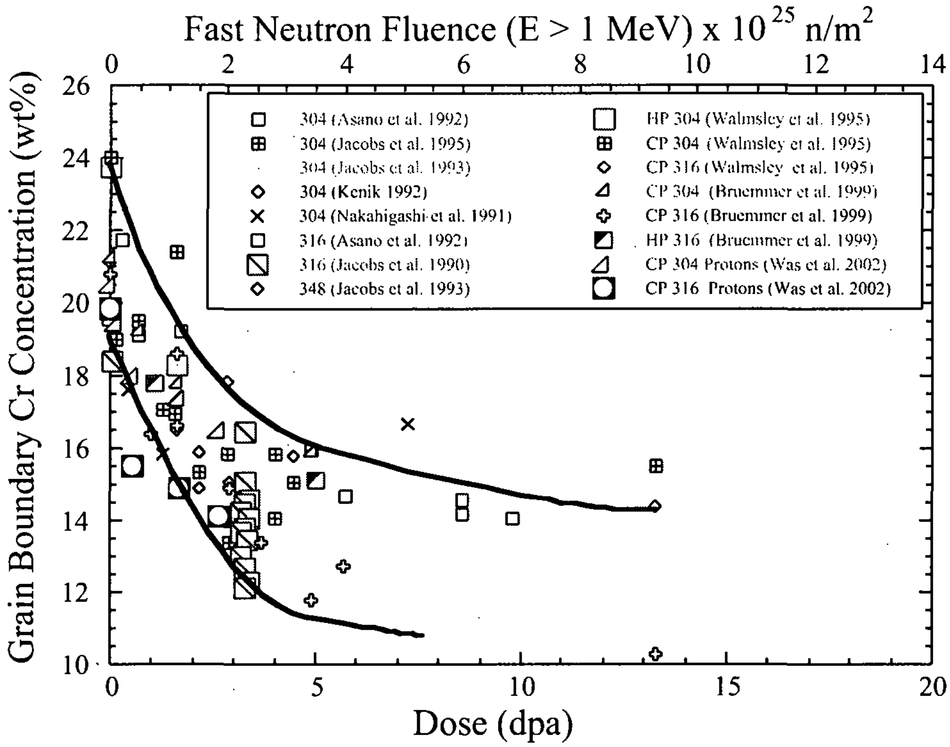

Figure 1. Dose dependence of grain boundary chromium concentration for several 300-series austenitic stainless steels irradiated at a temperature of about 300°C [Reference 1].

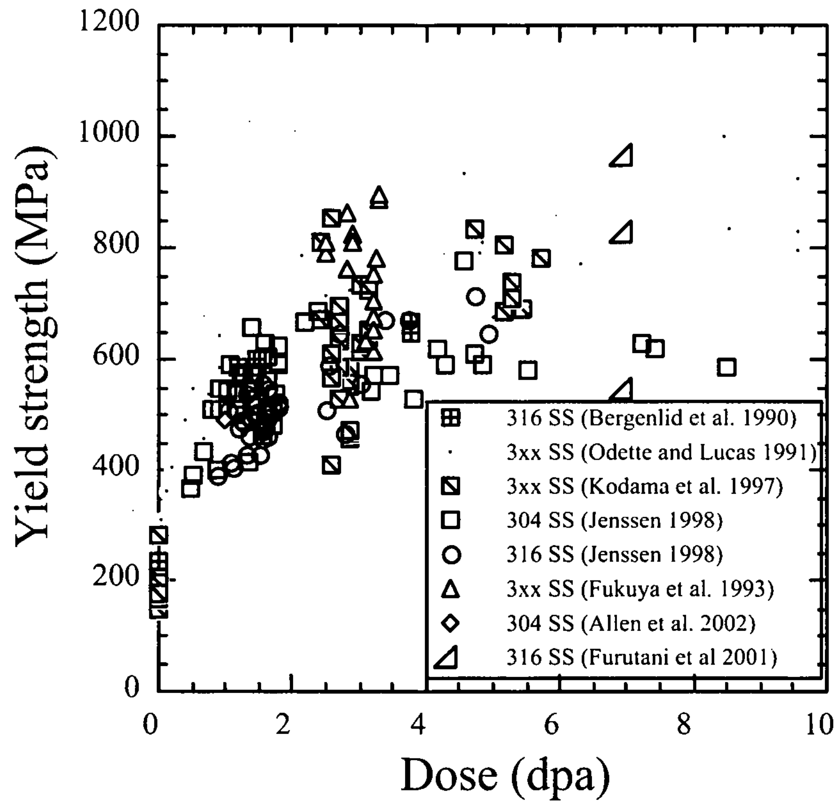

Figure 2. Irradiation dose effects on measured tensile yield strength for several 300-series stainless steels, irradiated and tested at a temperature of about 300°C. [Reference 1]

ENCLOSURE 3

MFN 06-041

Affidavit

General Electric Company

AFFIDAVIT

I, George B. Stramback, state as follows:

(1) I am Manager, Regulatory Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in GE proprietary report NEDC-32983P-A, *General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations,* Revision 2, Class III (GE Proprietary Information), dated January 2006. The proprietary information in the body of the report and in Appendix A is identified by side bar markings where the proprietary information is located and in the proprietary information in Appendix B is delineated by a double underline inside double square brackets. Figures and large equation objects are identified with double square brackets before and after the object. In each case, the superscript notation{3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, <u>Critical Mass Energy Project v. Nuclear Regulatory Commission,</u> 975F2d871 (DC Cir. 1992), and <u>Public Citizen Health Research Group v. FDA</u>, 704F2d1280 (DC Cir. 1983).

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

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

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

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

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

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

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

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

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

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains the methodology and detailed results of analytical models, methods, and processes, including computer code extension, which GE has developed, and applied to perform fast neutron flux calculations associated with BWR reactor pressure vessel evaluations.

The development of these methods to perform fast neutron flux calculations was achieved at a significant cost, on the order of ¼ million dollars, to GE.

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

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

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

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

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

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

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

Executed on this 23rd day of January 2006.

George B. Stramback

George B. Stramback
General Electric Company

NRC FORM 8C
(7-94)
NRCMD 3.57

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