FRAMATOME ANP

Proj 693

March 26, 2002
NRC:02:017

Document Control Desk
ATTN: Chief, Planning, Program and Management Support Branch
U.S. Nuclear Regulatory Commission
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Rockville, MD 20852-2738

Submittal of the Approved Version of Topical Report BAW-2241 (P) (A) Revision 1, "Fluence and Uncertainty Methodologies"

Ref.: 1. Letter, NRC to B&WOG, "Acceptance for Referencing of Licensing Topical report BAW-2241 (P), Revision 1, 'Fluence and Uncertainty Methodologies' (TAC No. M98962)," April 5, 2000

Ref.: 2. Letter, B&WOG to NRC, "B&WOG Topical report BAW-2241 (P), 'Fluence and Uncertainty Methodologies,'" May 22, 1997

Enclosed are 15 copies of the proprietary and 12 copies of the non-proprietary versions of topical report BAW-2241 (P) (A), "Fluence and Uncertainty Methodologies." The NRC found the use of this report to be acceptable for licensed applications in Referenced 1. Framatome ANP plans to apply the methodologies described in the topical report to perform fluence evaluations on Westinghouse and Combustion Engineering-designed reactors. These applications will take into account the three conditions specified in the SER.

The NRC's acceptance letter and SER are included at the front of the topical report. Responses to the NRC's requests for additional information are included as Appendices D and F to the report.

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Framatome ANP considers some of the information contained in the enclosure to this letter to be proprietary. The affidavit provided with the original submittal of BAW-2241 (P) (Reference 2) satisfies the requirements of 10 CFR 2.790(b) to support the withholding of this information from public disclosure.

Very truly yours,

James F. Mallay, Director
Regulatory Affairs

lmk

Enclosure

cc: D. G. Holland
Project 693

BAW-2241NP-A
Revision 1
Volume 2
December, 1999

Fluence and Uncertainty Methodologies

FRAMATOME ANP

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

April 5, 2000

Mr. J. J. Kelly, Manager
B&W Owners Group Services
3315 Old Forest Road
P.O. Box 10935
Lynchburg, VA 24506-3663

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT BAW-2241P, REVISION 1, "FLUENCE AND UNCERTAINTY METHODOLOGIES" (TAC NO. M98962)

Dear Mr. Kelly:

The U.S. Nuclear Regulatory Commission (NRC) staff has completed its review of the subject topical report, which was submitted by the Babcock and Wilcox Owners Group (B&WOG) by letter dated April 30, 1999. The report was prepared by Framatome Technologies, Inc. (FTI), acting on behalf of the B&WOG. The staff has found that this report is acceptable for referencing in licensing applications to the extent specified and under the limitations delineated in the report and the associated NRC safety evaluation, which is enclosed. The evaluation defines the bases for acceptance of the report. The staff will not repeat its review of the matters described in the BAW-2241P, Revision 1, when the report appears as a reference in license applications, except to ensure that the material presented applies to the specific plant involved.

In accordance with procedures established in NUREG-0390, the NRC requests that the B&WOG publish accepted versions of the submittal, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed safety evaluation between the title page and the abstract, and an -A (designating accepted) following the report identification symbol. The staff's requests for additional information (RAIs) and the B&WOG responses to RAIs during the review cycle shall be included as an appendix in the approved version of the topical report.

Pursuant to 10 CFR 2.790, the staff has determined that the enclosed safety evaluation does not contain proprietary information. However, the staff will delay placing the safety evaluation in the public document room for 10 calendar days from the date of this letter to allow you the opportunity to comment on the proprietary aspects only. If, after that time, you do not request that all or portions of the safety evaluation be withheld from public disclosure in accordance with 10 CFR 2.790, the safety evaluation will be placed in the NRC Public Document Room.

If the NRC's criteria or regulations change so that its conclusion that the submittal is acceptable is invalidated, the B&WOG and/or the applicant referencing the topical report will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the topical report without revision of the respective documentation.

Should you have any questions or wish further clarification, please call Stewart Bailey at (301) 415-1321 or Lambros Lois at (301) 415-3233.

Sincerely

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

Project No. 693

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

TOPICAL REPORT BAW-2241P, REVISION 1

"FLUENCE AND UNCERTAINTY METHODOLOGIES"

BABCOCK AND WILCOX OWNERS GROUP

1.0 INTRODUCTION

By letter dated May 14, 1997, the Babcock and Wilcox Owners Group (B&WOG) submitted Topical Report BAW-2241P, regarding a methodology for determining the pressure vessel fluence and associated uncertainties for NRC review (Reference 1). The submittal was prepared by Framatome Technologies, Inc. (FTI) on behalf of the B&WOG. The proposed methodology was intended for application to PWR plants and included numerous updates and improvements to the methods described in References 2 and 3. The approach used in BAW-2241-P is semi-analytic using the most recent fluence calculational methods and nuclear data sets. In the proposed methodology, the vessel fluence is determined by a transport calculation in which the core neutron source is explicitly represented and the neutron flux is propagated from the core through the downcomer to the vessel. The dosimeter measurements are only used to determine the calculational bias and uncertainty. The staff evaluation was completed on February 28, 1998, and found the proposed methodology acceptable for application to Babcock and Wilcox (B&W) plants. The B&WOG subsequently submitted additional information to demonstrate the applicability of the methodology to Westinghouse (W) and Combustion Engineering (CE) plants.

On April 30, 1999, the B&WOG submitted BAW-2241P, Revision 1, which consists of BAW-2241P, with added Appendix E (Reference 4). Review of BAW-2241P, Revision 1, has been completed and is the subject of this safety evaluation. The review and the evaluation were conducted in accordance with the provisions of Draft Regulatory Guide DG-1053 on neutron dosimetry, and BAW-2241P is found to be generally consistent with DG-1053.

The topical report provides a detailed description of the application of the proposed methodology to the calculation of the recent Davis-Besse cavity dosimetry experiment (References 7-9). This includes a description of both the discrete ordinates transport calculation and the techniques used to interpret the in-vessel and cavity dosimeter response. The Davis-Besse measurements have been included in the FTI benchmark data-base and are used to determine the measurement biases and uncertainties. The fluence calculation and uncertainty methodology presented in BAW-2241P, Revision 1, 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 are in Section 4.

2.0 SUMMARY OF THE TOPICAL REPORT

2.1 Semi-Analytic Calculational Methodology

The FTI semi-analytic fluence calculational methodology is the result of a series of updates and improvements to the BAW-1485 methodology developed for the 177-fuel assembly plants, described in References 2 and 3. These updates were made to improve the accuracy of the fluence prediction and to further quantify the calculational uncertainty. The improvements include the implementation of the BUGLE-93 ENDF/B-VI multi-group nuclear data set (Reference 9). The fluence calculations are performed with the DOT discrete ordinates transport code (Reference 10). The prediction of the best-estimate fluence is based on a direct calculation and includes an energy-dependent adjustment based on measurement. The BAW-2241P, Revision 1, approach incorporates most of the provisions of DG-1053 for predicting both the vessel fluence and the dosimeter response.

Predictions of the dosimeter response measurements are required to determine the calculation-to-measurement (C/M) data base. The FTI methodology includes dosimeter response adjustments for the half-lives of the reaction products, photo-fission contributions to the fission dosimeters, and dosimeter impurities. The predictions are made for both in-vessel and cavity dosimetry using the same methods used to determine the vessel fluence. In order to ensure an accurate prediction of the dosimeter response, a detailed spatial representation of the dosimeter holder tube/surveillance capsule geometry is included in the DOT model. Perturbation factors which account for the effect of the support beams and the instrumentation were calculated and applied to the predicted dosimeter responses. Energy-dependent axial synthesis factors are included to account for the axial dependence of the fluence.

2.2 Davis-Besse Cavity Dosimetry Benchmark Experiment

BAW-2241P, Revision 1, provides an extensive description of the Davis-Besse, Unit-1, Cycle-6, cavity dosimetry benchmark program. The program included both in-vessel and cavity experiments and provides a demonstration of the FTI dosimetry measurement methodology. The Davis-Besse dosimetry included an extensive set of activation foils, fission foils and cavity stainless steel chain segments. The in-vessel dosimetry consisted of standard dosimeter sets with energy thresholds down to 0.5 MeV. The in-vessel capsules were located at the azimuthal peak fluence location while the cavity holders were distributed azimuthally. The cavity chains extended from the concrete floor up to the seal plate (spanning the active core height) and were used to determine the axial fluence distribution. The measurement program included eighty dosimetry sets which were installed prior to Cycle 6 and removed in February 1990, after a full cycle (380 effective full power days) of irradiation.

The Davis-Besse dosimetry set included Cu-63 (n,α), Ti-46 (n,p), Ni-58 (n,p), Fe-54 (n,p), U238 (n,f) and Np-237 (n,f) threshold dosimeters. In addition, solid state track recorders (SSTRs) and helium accumulation fluence monitors (HAFMs) were included in the dosimetry set. The fissionable dosimeters were counted using two techniques: (1) the foils and wires were counted directly, and (2) the oxide powders were dissolved and diluted prior to counting. The detector was calibrated using a NIST-traceable mixed gamma standard source. The dosimeter measurements were corrected for dosimeter/detector geometry, self-absorption and photo-fission induced activity. When the foil or dosimeter thickness was large and/or the distance to

the detector was small, the geometry correction was determined with the NIOBIUM special purpose Monte Carlo program.

The measurement technique used for the non-fissionable dosimeters and chain dosimeters was essentially the same as that used for the fissionable dosimeters, although no dissolution was required. A NIST-traceable mixed gamma standard source was used for calibrating the detector and corrections for self-absorption and geometry were included. The Fe-54 (n,p) and Co-59 (n,γ) activities were used to determine the axial fluence shapes from the chain measurements.

2.3 Calculation-to-Measurement (C/M) Data Base and Uncertainty Analysis

FTI uses the comparisons of the calculated and measured dosimeter responses to benchmark and qualify the fluence methodology. Specifically, the data-base of calculation-to-measurement (C/M) values is used to determine the calculation bias and uncertainty (i.e., standard deviation). The data-base is large including a full set of dosimeter types and both in-vessel and cavity measurements. The data-base includes 35 capsule analyses (including two from the PCA benchmark experiment), three standard cavity measurements and the Davis-Besse cavity benchmark experiment.

The measured data is evaluated by material and dosimeter type and is adjusted to account for the dependence on power history and decay since shutdown. The statistical analysis of the C/M data indicates that the calculational model can predict: (1) the measured dosimeter response to within a standard deviation of seven percent or less, and (2) the end-of-life vessel fluence to within a standard deviation of less than twenty percent.

3.0 SUMMARY OF THE TECHNICAL EVALUATION

Topical Report BAW-2241P, Revision 1, provides the FTI methodology for performing pressure vessel fluence calculations and the determination of the associated calculational uncertainty. The review of the FTI methodology focused on: (1) the details of the fluence calculation methods, and (2) the conservatism in the estimated calculational uncertainty. As a result of the review of the methodology, several important technical issues were identified which required additional information and clarification from FTI. The request for additional information (RAI) was transmitted in References 11 to 13 and was discussed with FTI in a meeting at NRC Headquarters on August 5 and 6, 1998. The information requested was provided by FTI in the responses included in References 14 to 16. This evaluation is based on the material presented in the topical report and in References 14 to 16. The evaluation of the major issues raised during the review are summarized in the following subsections.

3.1 Semi-Analytic Calculational Methodology

The FTI semi-analytic calculational methodology is used to determine the pressure vessel fluence, predict the surveillance capsule fluence, determine dosimeter response for the benchmark experiments and perform fluence sensitivity analyses. The neutron transport calculation, selection and processing of the nuclear data and analysis of the Davis-Besse benchmark experiment generally follows the approach described in DG-1053.

DG-1053 notes that as fuel burnup increases the number of plutonium fissions increases, resulting in an increase in the number of neutrons per fission and a hardening of the neutron spectrum. Neglect of either of these effects results in a nonconservative prediction of the vessel fluence. In Responses 1-3 and 1-10 of Reference 14, FTI describes the method used to incorporate these effects in the methodology. It is indicated that the uranium and plutonium isotopic inventory is tracked for each fuel assembly and the uranium and plutonium neutron emission rates are determined for the individual isotopes. The fuel inventory is determined for each depletion time-step and is tracked in three dimensions using a program that is benchmarked to in-core detector data. In Response 1-10 (Reference 14), FTI evaluates the approximation used to determine the burnup-dependent core neutron spectrum. This evaluation indicates that the effect of the spectrum approximation used in the methodology is negligible.

Typically, PWR internals include steel former plates for additional support between the core shroud and barrel. These plates provide additional core-to-vessel fluence attenuation and can have a significant effect on the surveillance capsule dosimeters and the neutron fluence at the vessel. In Response 1-4 (Reference 14), FTI stated that several designs include core shroud former plates and that these plates have been included in the data-base fluence transport analyses. In addition, FTI has provided DOT calculated fluence profiles which quantify the fluence reduction introduced by the former plates.

3.2 Measurement Methodology

The FTI vessel fluence methodology includes an extensive set of plant surveillance capsule fluence measurements as well as the Davis-Besse benchmark measurements. These measurements are important since they are used to determine the calculational uncertainty and bias. In response to RAI 1-16, FTI has stated in Reference 13 that the dosimeter measurements conform to the applicable ASTM standards. In addition, in conformance with DG-1053, FTI performed a reference field measurement validation, which has been provided to the NRC in Reference 15.

The dosimeter reaction rate is determined by measuring the activity due to a specific reaction product. Before the reaction rate can be determined the effect of interfering reactions must be removed. Typically, this will involve the interference from: (1) the fission products resulting from plutonium buildup in the U-238 dosimeters, (2) the fission products resulting from U-235 impurities, (3) the fission products resulting from photo-fission reactions in the U-238 dosimeters, and (4) impurities having decay energies close to the reaction product being measured. FTI has stated in Response 1-16 (Reference 14) that these effects have been evaluated and, when they were significant, have been accounted for in determining the dosimeter response.

The determination of the photo-fission correction for the U-238 (n,f) dosimeters requires a coupled gamma/neutron transport calculation (which is not required for the analysis of the (n,p) dosimeters). This calculation is sensitive to both the neutron and photon cross sections. To ensure the accuracy of these calculations, FTI has stated in Response 1-14 (Reference 14) that photo-fission corrections determined using an alternate neutron/photon cross section library agree (to within a percent) with the corrections used in the BAW-2241P, Revision 1, analysis.

The FTI data-base includes two distinct types of U-238 fission dosimeters. The statistical analysis of the C/M data-base is made without any recognition of the difference between these two sets of dosimetry data. In Response 1-12 (Reference 14), FTI has evaluated the two sets of U-238 data in order to identify any significant difference in either the uncertainty or bias inferred from this data. The evaluation showed no significant difference between the two U-238 data sets.

3.3 Calculation-to-Measurement (C/M) Data Base and Uncertainty Analysis

DG-1053 requires that the vessel fluence calculational methodology be benchmarked against reactor surveillance dosimetry data. The FTI topical report includes an extensive set of calculation-to-measurement benchmark comparisons. FTI has evaluated the C/M data statistically in order to estimate the uncertainty in the fluence predictions and determine the calculational bias.

The plant-to-plant variation in the as-built core/internals/vessel geometry, core power and exposure distributions, and the plant power history are major contributors to the uncertainty in the vessel fluence calculation. The contribution of these uncertainty components can be minimized by selecting the C/M data from only a few plants. In fact, as part of the integrated vessel material surveillance program (BAW-1543A), several of the FTI data sets were taken at a single host plant. FTI has identified the specific data sets and host plant in Response 2-13 (Reference 16). In order to ensure that these data sets have not resulted in an erroneous reduction in the data-base calculation uncertainty, the uncertainty for these plants has been evaluated separately. This evaluation indicated a larger uncertainty for the C/M data taken at the surrogate plants and that use of the surrogate data was not resulting in a non-conservative calculational uncertainty.

The C/M data-base includes a relatively complete set of Np-237(n,f) dosimeters. However, while the calculation-to-measurement agreement is generally good for most dosimeter types, the agreement for the Np-237 dosimeters is poor. In Response 2-18 (Reference 16), FTI has indicated that it is presently evaluating the calculation-to-measurement discrepancies for Np-237. It is important to note, however, that the BAW-2241-P fluence methodology does not include the Np-237(n,f) dosimeter data in the determination of the calculation uncertainty and bias.

The BAW-2241-P analysis includes a detailed evaluation of the measurement uncertainty. This evaluation is based on estimates of the various uncertainties that affect the measurement process and analytic calculations of the sensitivity of the measurement process to these uncertainty components (Reference 16). The calculational uncertainty is determined using the overall data-base C/M variance and the estimated measurement uncertainty. In order to ensure a conservative estimate of the calculational uncertainty, FTI has increased the estimated calculational uncertainty by about 50 percent.

The FTI calculational procedure includes the application of a group-wise multiplicative bias to the calculated > 1-MeV fluence. This bias is based on comparisons of calculation and measurement for both in-vessel capsules and cavity dosimetry and is to be applied to determine the best-estimate fluence. The application of the bias is conservative and results in a relatively small, but positive, increase in the calculated > 1-MeV fluence.

3.4 Application to Westinghouse and Combustion Engineering Plants

The BAW-2241P, Revision 1, methodology is intended for application to W and CE plants, as well as B&W plants. As justification for the application to W and CE plants, FTI has included both W and CE plant dosimetry data in the C/M data-base. In response to request for additional information (RAI) number 1 (RAI-1 in Reference 17) concerning the consistency of the C/M data, FTI has stated that the dosimetry measurements and calculations for the W and CE plants were performed with the same methods used to determine the C/M data for the B&W plants (i.e., the methods described in BAW-2241P, Revision 1). In addition, in response to RAI-2 (Reference 17), it is stated that no W or CE C/M data has been eliminated from the comparisons.

The review of the C/M data-base indicated that the standard deviation between the calculations and measurements is smaller for the CE plants than for the W and B&W plants. It is therefore conservative to apply the larger overall data-base uncertainty to the CE plants. However, the inclusion of the C/M data for the CE plants in the FTI data-base may result in an erroneous reduction in the uncertainty applied to the W and B&W plants. In Response 7 of Reference 17, FTI has evaluated the increase in calculational uncertainty when the C/M data for the CE plants is excluded from the FTI data-base. The resulting increase in calculational uncertainty is found to be very small compared to: (1) the conservatism included in the estimated calculational uncertainty, and (2) the uncertainty requirements of DG-1053.

4.0 SUMMARY AND LIMITATIONS

Topical Report BAW 2241P, Revision 1, "Fluence and Uncertainty Methodologies," and its supporting documentation provided in References 14 and 16 have been reviewed in detail. Based on this review, it is concluded that the proposed methodology is acceptable for referencing in licensing applications for determining the pressure vessel fluence of W, CE and B&W designed reactors.

The following limitations apply:

1. The FTI dosimetry C/M data-base includes an extensive set of PWR core/internals/vessel configurations. However, the dosimetry set is not complete and there are certain designs that are not included in the data-base (e.g., cores including partial-length fuel assembly designs). FTI has indicated (Response-9 of Reference-17) that in the case where the BAW-2241P, Revision 1, methodology is applied to a plant including a feature not included in the FTI data-base, an additional evaluation will be performed. This will include an evaluation of the effect on the dosimetry measurements, calculation-to-measurement ratios and the analytical uncertainties. FTI has stated that the fluence calculational uncertainty will be increased if this evaluation indicates that the uncertainties given in BAW-2241P, Revision 1, are not adequate.

2. Should there be changes in the input cross section of this methodology, the licensee will evaluate the changes for their impact and , if necessary, will modify the methodology accordingly.

3. The licensee will provide the staff with a record of future modifications of the methodology.

The NRC staff will require licensees referencing this topical report in licensing applications to document how these conditions are met.

5.0 REFERENCES

1. "B&WOG Topical Report BAW 2241-P, 'Fluence and Uncertainty Methodologies'," Letter, from J. H. Taylor (B&WOG) to US NRC, dated May 14, 1997.

2. BAW-1485P, Revision 1, "Pressure Vessel Fluence Analysis for 177-FA Reactors," S. Q. King, et al., Framatome Technologies, Inc., April 1998.

3. BAW-1485, "Pressure Vessel Fluence Analysis for 177-FA Reactors," C.L. Whitmarsh, Babcock and Wilcox Corporation, June 1978.

4. BAW-2241P, Revision 1, "Fluence and Uncertainty Methodologies," R. J. Worsham III, Framatome Technologies, Inc., April 1999.

5. Draft Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, June 1996.

6. BAW-1875-A, "The B&W Owners Group Cavity Dosimetry Program," S. Q. King, Babcock and Wilcox Corporation, July 1986.

7. Coor, Jimmy L., "Analysis of B&W Owner's Group Davis-Besse Cavity Dosimetry Benchmark Experiment"" Volumes I, II and III, B&W Nuclear Environmental Services, Inc. (NESI), NESI # 93:136112:02, May 1993, FTI Doc. # 38-1210656-00, Released May 30, 1995.

8. BAW-2205-00, "B&WOG Cavity Dosimetry Benchmark Program Summary Report," J. R. Worsham III, et al., Framatome Technologies, Inc., December 1994.

9. "BUGLE-93: Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," Radiation Shielding Information Center (RSIC), Oak Ridge National Laboratory (ORNL), DLC-175, April 1994.

10. "DOT4.3: Two Dimensional Discrete Ordinates Transport Code," Hassler, L. A., et al., (B&W Version of RSIC/ORNL Code DOT4.3), FTI Doc. # NPD-TM-24, July 1986.

11. "Request for Additional Information for Topical Report BAW-2241-P," Letter, Joseph L. Birmingham (NRC) to J. J. Kelley (BWOG), dated January 30, 1998.

12. "Request for Additional Information for Topical Report BAW-2241-P," Letter, Joseph L. Birmingham (NRC) to J. J. Kelley (BWOG), dated April 8, 1998.

13. "Request for Additional Information for Topical Report BAW-2241-P," Letter, S. Bailey (NRC) to J. J. Kelley (BWOG), dated October 26, 1999.

14. "Response to NRC Request for Additional Information for Topical Report BAW-2241-P, 'Fluence and Uncertainty Methodologies'," Letter, OG-1708, R. W. Clark (BWOG) to J. L. Birmingham (NRC), dated May 29, 1998.

15. Letter from R.W. Clark, B&WOG to US NRC, and attached report, "Standard and Reference Field Validation," by T. Worsham and Q. King dated May 19, 1999.

16. "Response to NRC's April 8, 1998 Request for Additional Information for Topical Report BAW-2241-P, 'Fluence and Uncertainty Methodologies'," Letter, OG-1726, R. W. Clark (BWOG) to J. L. Birmingham (NRC), dated October 30, 1998.

17. "Response to NRC's October 26, 1999, Request for Additional Information - Framatome Topical Report BAW-2241-P, Revision 1, 'Fluence and Uncertainty Methodologies'," Letter, FTI-99-3850, J. R. Worsham III (FTI) to S. Bailey (NRC), dated November 30, 1999.

Principal Contributor: L. Lois

Date: April 5, 2000

Framatome Technologies, Inc.
Lynchburg, Va, 24506

Topical Report BAW-2241P-A

Volume 2
Revision 1
December, 1999

Fluence and Uncertainty Methodologies

J. R. Worsham III

Abstract

The results presented in this topical demonstrate that Framatome Technologies, Inc. (FTI) has a high degree of accuracy in their unbiased, best - estimate fluence calculations, and a high degree of confidence in the very small fluence uncertainties. The methodologies in this topical are applicable to any PWR with the results showing the same accuracy and uncertainties.

Numerous improvements and updates have been made in the FTI fluence and uncertainty methodologies that are used to calculate the fast neutron fluence throughout the reactor system, including the vessel materials and welds. These improvements and updates enhance the accurate determination of vessel fluence and establish a statistically sound methodology for estimating the bias and uncertainty in the calculated fluence. The methodology presented herein is calculational-based. Dosimetry measurements are used only in the estimation of biases and uncertainties. The results of B&WOG Cavity Dosimetry Benchmark Experiment were the key (a) in this update of the measurement biases and uncertainties for the entire FTI dosimetry database, and (b) in the development of calculational biases and uncertainties.

RECORD OF REVISIONS

Rev. No.	Change Section/paragraph	Description of Change
0		Initial Release
1	*Appendix E*	Added Appendix
1	*Appendix F*	Added Appendix

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Volume 1 Table of Contents - CONTINUED -

Volume 2 Table of Contents

Appendix D FTI Responses to the -

Request for Additional Information for Topical BAW-2241P *Fluence and Uncertainty Methodologies* *

Set 1 - Question 1

The topical report states that the B&W owners will revalidate the analytical monitoring of the pressure vessel by performing vessel fluence analyses and benchmark comparisons to cavity measurements. How will the results of these analyses be used and will they be submitted in separate topical reports ?

Response

In the introductory section (1.0) of the Topical, on page 1 – 3, the following remarks were made as part of the discussion concerning why the B & W Owners were submitting a topical at this time.

> In the interim period however, before the draft guide (DG-1053) is finalized, most of the owners will be updating their reactor coolant system pressure – temperature limits for heat-ups and cool-downs. In addition, most owners will be revalidating the analytical monitoring of their vessels by performing vessel fluence analyses that include absolute calculations of the fluence and benchmark comparisons of the calculations to cavity dosimetry measurements.

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*This *Appendix* contains its own Reference section. References D1 and D2 refer to the two sets of NRC requests for additional information.

The question concerns how the fluence results will be used for the updated pressure - temperature limits, and will they be submitted in a topical report. The results of the analyses from each B & W owner (a) revalidating the monitoring of their vessel (using best - estimate calculational results), and (b) performing a benchmark comparison of the calculations to cavity dosimetry measurements, will be used in reactor vessel embrittlement evaluations. The embrittlement evaluations are submitted to the NRC in updates to the plant Technical Specifications for revised pressure - temperature curves by each respective owner. The fluence values and uncertainties are referenced in the Technical Specification change submittal. They are not included in separate topical reports.

Embrittlement evaluations are based on the correlation of increasing fluence levels to increasing reference temperatures in the nil-ductility transition properties of specimens of the vessel materials. The embrittlement evaluations include a "Margin" term which is based on the uncertainties in the correlation.[D3] The NRC has suggested that the fluence uncertainties, forming part of the bases for the correlation uncertainties, can be represented by a standard deviation of 20 percent.[D4] The benchmark comparison of dosimetry calculations to measurements, for each B & W owner in their updated evaluation of reactor coolant system pressure - temperature limits for heat-ups and cool-downs, is used to determine if the calculations for each specific plant evaluation are consistent with the *Fluence and Uncertainty Methodologies* topical. Consistency between (a) the plant-specific uncertainties, and (b) the biases, standard deviations, and confidence levels in the topical, ensure that the plant-specific evaluations are consistent with the embrittlement correlations.

Set 1 - Question 2

Provide a detailed description of the dosimeter, capsule and structural support geometry and how the modeling of this detail was validated.

Response

The permanent dosimetry holder (capsule) consists of

as shown in Figure D1 below.

Figure D1. Cross section of , dosimetry holder, and dosimetry can.

Figure D2. Geometrical model of dosimetry holder and surrounding structures.

Set 1 - Question 3

Describe how the effect of increased Pu in the high burnup fuel is included in the source calculation. Does this treatment allow for the cycle-specific variations ?

Response

This question, concerning the effects of increased plutonium (Pu) concentrations in the high burnup fuel, and Question Set 1 - 9, concerning the dependence of the number of neutrons produced per fission on burnup, and Question Set 1 - 10, concerning the

neutron source spectra as a function isotopic production weighting, are all related. Therefore, in addition to the following explanation, the explanations for Question Sets 1 - 9 and 1 - 10 should also be reviewed.

Set 1 - Question 4

Do the internals of the B&W plants include core shroud former plates and, if so, how is the effect of these plates included in the calculations ?

Response

The B & W design includes core formers. The effects of the former plates are explicitly accounted for in the DORT analyses, as described below.

Set 1 – Question 5

Are there differences between the calculation and measurement methods used for Davis Besse and the methods used for the other plants included in the Appendix-A data base ? For example, were the methods used to determine the dosimeter corrections for the Appendix-A measurements the same as used for Davis Besse ?

Response

The first sentence of this question involves two questions, one concerning calculational methods, and the other concerning measurement methods. The differences between the measurement methods used for Davis Besse and the methods used for the other plants included in *Appendix A* will be addressed first.

The measurement methodology is described in Section 5 of the Topical. The measured results involve (1) a specific activity for the radiometric dosimeters described in Section 5.1, (2) the fissions per target atomic density for the solid state track recorders described in Section 5.2, and (3) helium concentrations in atomic parts per target concentration for the helium accumulation fluence monitors described in Section 5.3. The measurement methods used to obtain these results for the Davis Besse benchmark are the same as the methods used for the other plants referenced in *Appendix A*. This

includes the methods used to determine the dosimeter corrections for the *Appendix A* measurements.

Set 1 – Question 6

Will the BAW-2241-P methodology be applied to cores with partial length fuel assemblies and, if so, how will the (*r, z*) source of Section-3.1.2.2 be determined ?

Response

The methodology can be applied to partial length fuel assembly poison inserts.

In general, the multi-planar $r\theta$ sources and multi-channel rz sources are produced from the results of pin-by-pin, three-dimensional, time-averaged source distributions. The three-dimensional source distributions come from explicit three-dimensional fuel-cycle calculations, such as those from the NEMO or PDQ codes. The calculations of the sources are produced during core-follow benchmarks of the code results to measured power densities.

Set 1 - Question 7

The Model-C (*r, z*) calculation results in negative fluxes and an unacceptable solution. Can this error in the Model-C calculation affect the results of the Model-B calculation ? For example, what is the sensitivity of the Model-B calculation to the albedo boundary conditions ?

Response

The negative fluxes encountered in the *r, z* Model C DORT calculation occurred high up in the air cavity between the vessel and the concrete, and were determined to be the result of computer-memory-related inabilities to specify a large enough quadrature and/or small enough interval dimensions. Once that was ascertained, the Model C DORT run was abandoned, effectively reducing the size of the problem in the axial direction.

The cavity fluxes over the Model B elevation were determined by synthesis

Set 1 – Question 8

Please Provide Reference 21.

Response

It is enclosed in this submittal.

Set 1 - Question 9

In view of the large variation in fuel burnup between assemblies and the dependence of the number of neutrons produced per fission (v) on fuel burnup, what uncertainty is introduced by neglecting this dependence in Equation (4.1) ?

Response

This question, concerning the dependence of the number of neutrons produced per fission on burnup, and Question Set 1 - 3, concerning the effects of increased plutonium (Pu) concentrations in the high burnup fuel, and Question Set 1 - 10, concerning the neutron source spectra as a function isotopic production weighting, are all related. Therefore, in addition to the following explanation, the explanations for Question Sets 1 - 3 and 1 - 10 should also be reviewed.

As explained when Question Set 1 - 3 was addressed above, the variation in fuel burnup between assemblies is modeled explicitly. This modeling includes, core - follow calculations which are compared to the measured core operational data, quasi-static time steps to appropriately treat time dependent behavior, explicit representation of the isotopics within the fuel assembly, and three-dimensional representation of the fuel pins and geometrical detail within the assembly. Thus, the dependence on the changing isotopics as a function of burnup, and the corresponding changes in the number of neutrons produced per fission in the fuel volume is not neglected. The burnup dependence of the neutrons produced per fission within a fuel assembly is included in the neutron source calculation.

The uncertainty in neutron production due to the uncertainty in the burnup of the fuel assemblies can be modeled with the uncertainty in the power distribution. The uncertainty in the power distribution is not normal when it is defined on a relative basis. However, an absolute deviation in the relative power distribution does represent a normal distribution. Using an upper bounding deviation with a 95 percent confidence level in the analytic sensitivity, indicated that the local uncertainty would be about 18 percent with a relative peripheral power of 0.50, and about 30 percent with a relative peripheral power of 0.30.

Set 1 - Question 10

The core neutron source spectrum is determined by a neutron production weighting of the individual assembly neutron spectra. What uncertainty is introduced by the Equation (4.2) power weighting of the assembly spectra ?

Response

This question, concerning the neutron source spectra as a function isotopic production weighting, and Question Set 1 - 3, concerning the effects of increased plutonium (Pu) concentrations in the high burnup fuel, and Question Set 1 - 9, concerning the dependence of the number of neutrons produced per fission on burnup, are all related. Therefore, in addition to the following explanation, the explanations for Question Sets 1 - 3 and 1 - 9 should also be reviewed.

As explained when Question Set 1 - 3 was addressed above, the neutron source spectrum is evaluated for each fuel assembly

In Equation 4.2 (now Equation 3.2), the assembly average fission emission spectrum is the result of the weighting from the isotopics, et cetera. The assembly fission emission spectrum is used in Equation 4.1 (now Equation 3.1) to define the neutron source spectrum for the core - fuel region. However, as noted by the spatial and spectral indices of the source term in Equation 4.1 (Equation 3.1), the source in the DOT models is not a constant spectrum as a function of space.

Thus, each finite mesh block in the DOT models of the fuel region within the core contains neutron source spectra that are unique to the fuel assembly region represented by the mesh block. Consequently, the core neutron source spectrum is not a single spectrum that has been weighted by the neutron productions throughout the core - fuel

region. The core source is represented by unique fuel assembly spectra appropriately applied to the respective mesh blocks within the fuel regions.

Set 1 – Question 11

Describe in detail how the dependence of the dosimeter response on the axial separation between the vessel support beams and the dosimeters is included. Is the method used for including the effect of the support beams at Davis Besse also used for ANO-1 ?

Response

Set 1 - Question 12

Does the dissolution process used in the measurement of the powder fissionable dosimeters introduce more uncertainty than the process used to measure the wire dosimeters ? Is the C/M bias and standard deviation for the powder dosimeters different than for the dosimeter wires ?

Response

The dissolution process used in the measurement of the four powder U-238 dosimeters (page B - 2) and the three powder Np-237 dosimeters (page B - 4) does not introduce

more uncertainty than the process used to measure the wire dosimeters. Page 7 – 18 shows the mean relative standard deviation for all sixteen U-238 dosimeters

Therefore, it appears that the powder and wire dosimeters have the same bias and standard deviation.

Set 1 - Question 13

How does the NIOBIUM prediction compare with the analytic result of Equation (5.1) for the limiting geometry ?

Response

Set 1 – Question 14

The photo-fission corrections for the U-238(n,f) and the Np-237(n,f) dosimeters appear low compared to the results of other investigators. Have the predictions used to determine these corrections been compared to calculations made with the BUGLE-93 library ? Also, what photo-fission cross sections were used for U-238 and Np-237 and what is the basis for these values ?

Response

With respect to the results of other investigators, one reason for a disparity can be explained by the fact that the neutron to gamma flux ratio differs

This point is illustrated by the following comparison of photo-fission factors (as defined above); the same photo-fission cross sections were used in each analysis.

- Davis Besse, Cycle 6: In-vessel U-238 PF correction factor = 1.050
- W reactor: In-vessel U-238 PF correction factor = 1.186

These photo-fission factors vary by 13.0 percent.

Regarding the question: "have comparisons been made to photo-fission corrections using BUGLE-93 data ?" Yes, for example, the ONS2 Cycles 9 - 14 fluence analysis used the Caldwell photo-fission cross sections[D5,D6,D7] with the BUGLE-93 material cross sections (for cavity dosimeters only).

Set 1 - Question 15

What is the effect on the dosimeter response of Pu build-up, U-235 content and impurities ? Why aren't dosimeter response corrections required for these effects ?

Response

U-235 Content in U-238 Dosimeters

Corrections were made to account for the effect of U-235 content in the U-238 dosimeters. The capsule dosimeters and a few of the cavity dosimeters had large U-235 concentrations (about 350 ppm), however the majority of the U-238 dosimeters had small U-235 concentrations (12 ppm). (See page 5 - 6 of the Topical Report).

Pu Build-up in the U-238 Dosimeters

The effect of plutonium build-in was analyzed and found to be negligible. For Davis Besse, Cycle 6, the operational time was 380.25 EFPD. The fraction of the total

Cs-137 produced from Pu fissions in the U-238 dosimeters during 380.25 EFPDs of operation is estimated to be less than 1.0 percent (see Figure D4).

Other Corrections for Impurities

Set 1 - Question 16

Do the dosimeter response measurements conform to the applicable ASTM standards ? If no, justify any differences.

Response

The dosimeter measurements conform to the applicable ASTM standards. The discussion of the "Measurement Methodology" in Section 5.0, and the discussions of the "Measurement Techniques" for (1) fissionable and activation radiometric dosimeters in Sections 5.1.1 and 5.1.2 respectively, (2) solid state track recorders in Section 5.2.1, and (3) helium accumulation fluence monitors in Section 5.3.1, indicate that the techniques and procedures agree with the ASTM standards. The ASTM standards refer to additional ASTM standards for "Spectrum Adjustment Methods", "Application for Reactor Vessel Surveillance", et cetera. These additional standards refer to techniques that differ from those explained in the "Semi-Analytical (Calculational) Methodology", in Section 3.0, and the "Uncertainty Methodology", in Section 7.0. These additional standards refer to the application of the measurements, to infer measured fluences, and are neither applicable to the measurements themselves, nor to vessel fluence predictions. The ASTM standards also refer to precision, bias and uncertainty in terms that are conflicting and inconsistent with mathematical statistics and the National Institute of Standards and Technology (NIST). Section 7.0 of the topical, "Uncertainty Methodology", explains the treatment of the measurement

uncertainties. Section 7.0 also notes the validation of the measurement uncertainties in a NIST reference field.

Set 1 - Question 17

Why isn't a NIOBIUM calculation required for determining geometry and self-absorption corrections for the non-fissionable dosimeters ?

Response

The measured activity of each dosimeter was determined by the B & W radiochemistry laboratory, using QA - approved and certified procedures, data, and equipment.

The results produced by the present methods for determining geometry and self-absorption corrections have been shown to be reliable and accurate by the QA validation of the B & W laboratory during the Benchmark Experiment uncertainty analyses.

Set 1 – Question 18

Provide Table B-2.2-1 including the SSTR measurement results.

Response

The use of SSTRs was evaluated for the B&WOG cavity dosimetry program. However, as discussed on page 7 - 9 of the Topical, the standards for fissionable mass deposits and fission product track counts are still being developed. Therefore, SSTRs have not been validated for implementation to support the B&WOG vessel and material monitoring program with the methodologies presented in the Topical. The reference to SSTR measurements was inadvertent and will be changed (please see below).

5.2.2 **Measured Results**

Numerous SSTR fission-rate measurements were evaluated for the Davis Besse Benchmark Experiment. The initial set of SSTR C/M ratios evaluated for the experiment were in poor agreement with other dosimetry C/M ratios and M/M ratios. Several iterations were required before SSTR measurements were obtained that were consistent with the other dosimetry C/M and M/M ratios. While the final set of C/M ratios for the SSTRs were excellent, the only parameter that changed during the iterations was the SSTR measured results. It has been concluded that, while SSTRs do have some potential advantages over other dosimeter types, the state of development of SSTR technology is insufficiently advanced to justify their use as standard dosimeters in the B&WOG fluence analyses methodologies.

D.2 Question Set 2

Question Set 2 will be addressed in a different format from Question Set 1. The format for Question Set 1 was straightforward in that the NRC sent FTI, and the B & W Owners a set of questions.[D1] FTI and the B & W Owners responded as shown in the previous 26 pages, (D - 1 through D - 26). To reduce costs, and have a better understanding of the questions and explanations on the second set, FTI and the B & W Owners met with the NRC and their contractor in a working meeting on August the fifth and sixth, 1998. This working meeting accomplished the goals of reducing the costs and improving communications. All of the NRC's 19 requests for additional information (RAIs) were satisfactorily addressed. In addition, very detailed discussions on the application of statistical methods were reviewed. Following the meeting, the NRC and their contractor requested that the statistical methods outlined during the review be briefly documented in the response to the second set of RAIs. They also requested that five additional points that they raised during the discussion be documented, and the complete explanations included.

The "Statistical Methods" section, D.2.1, provides a brief outline of the statistical methods used in the topical and explained during the August fifth and sixth NRC meeting. Section D.2.2, "RAI Set 2 Responses", refers to the Statistical Methods, and includes a few brief statements summarizing the discussion during the meeting on each of the 19 requests for additional information. The section following the RAI responses, D.2.3, discusses the "Statistical Processing of Table A-1 Data". The last section,

D.2.4, "Additional Explanations", lists the five additional questions that the NRC raised during the meeting, and provides the requested explanations.

D.2.1 Statistical Methods

A predominant theme throughout the second set of RAIs, concerned the fundamental expressions of mathematical statistics. Therefore, the meeting on the fifth of August began with a review of the expressions which are the bases for the equations in the "Uncertainty Methodology" section (7.0) of the topical. Since nearly all references on statistical evaluations of uncertainty are based on the concept that the mean value of a predicted parameter is unbiased, the review began with the concept that uncertainty includes the possibility of multiple biases (systematic deviations), in addition to the usual random deviations.

It was noted that the definition of the best-estimate fluence implied that the calculational methodology was unbiased

Also noted, was the fact that it is not possible to use the methods of mathematical statistics to estimate the unbiased uncertainty in the vessel fluence, if the biases in the calculational methodology have not been uniquely identified and removed.

The area that incurred the most discussion and explanation, concerned the combination of uncertainties in the independent random variables (that may be functionally related, or correlated) to estimate the variance in the dependent variable. The discussions centered on topical Equation 7.6.

Equation D.5

can be derived from a Taylor series, which represents dependent random variable y in terms of independent random variables x_i. During the meeting discussion, concerning the development of Equation 7.6 from Equation D.5, there were several issues

regarding (a) the truncation of the Taylor series, and (b) subsequent cross product dependencies between the random variables. To ensure that responses to the RAIs and "Additional Explanations" are clear, this discussion of statistical methods begins with the Taylor series relating two random variables, x and y, as shown by Equation D.1.

$$y = g(x) = \sum_{n=0}^{\infty} \frac{\frac{\partial^n g(\bar{x})}{\partial x^n}\left(x - \bar{x}\right)^n}{n!} \tag{D.1}$$

Independent of which specific parameters the variables x and y represent in Equation D.1, dependent variable y is a function of independent variable x. Thus, y cannot be determined without a value for x, and the uncertainty in the value of y cannot be determined without the uncertainty in the value of x.

$$\left(K_y \ \sigma_y\right)^2 = \tag{D.2}$$

The uncertainty in the value of y is represented by the product of a confidence factor (K) and the standard deviation (σ) as shown by the left side of Equation D.2. The confidence factor for the dependent variable y is directly related to the confidence level for the independent variables.

When Equation D.1 is expanded into multiple x variables (x_i), and substituted into Equation D.2, the resulting uncertainty expression for the dependent random variable y is represented by Equation D.3.

$$\left(K_y\ \sigma_y\right)^2 = \left(\sum_{n=1}^{\infty} \frac{\left[\sum_i K_{x_i}\ \sigma_{x_i}\ \frac{\partial}{\partial x_i}\right]^n y\left(x_1, x_2, \ldots, x_i\right)}{n!} \right)^2 \tag{D.3}$$

During the meeting, Equation D.3 was the focus of considerable discussions, questions, and explanations. To provide clarity in the following discussion, Equation D.3 has been modified as expressed by Equation D.4.

$$\begin{aligned}\left(K_y\ \sigma_y\right)^2 = \Bigg(& \left(K_{x_1}\ \sigma_{x_1}\right) \frac{\partial y}{\partial x_1} + \left(K_{x_2}\ \sigma_{x_2}\right) \frac{\partial y}{\partial x_2} + \\ & \frac{1}{2}\left(K_{x_1}\ \sigma_{x_1}\right)^2 \frac{\partial^2 y}{\partial x_1^2} + \frac{1}{2}\left(K_{x_2}\ \sigma_{x_2}\right)^2 \frac{\partial^2 y}{\partial x_2^2} + \\ & \frac{2}{2}\left(K_{x_1}\ \sigma_{x_1}\ K_{x_2}\ \sigma_{x_2}\right) \frac{\partial^2 y}{\partial x_1 \partial x_2} + \\ & \sum_{n=3}^{\infty} \frac{\left[K_{x_1}\ \sigma_{x_1}\ \frac{\partial}{\partial x_1} + K_{x_2}\ \sigma_{x_2}\ \frac{\partial}{\partial x_2}\right]^n y\left(x_1, x_2\right)}{n!} \Bigg)^2\end{aligned} \tag{D.4}$$

In Equation D.4, the first and second order terms in the Taylor series have been explicitly included, and the independent variables have been reduced to x_1, and x_2.

The cross product dependencies between the independent variables, are included in the second and higher order derivatives.

While there are statistical applications where the second, and higher order derivatives are used, the discussions during the meeting focused on the fundamentals of

mathematical statistics. The mathematical statistics expression for the variance in y due to the propagation of uncertainties in the variables x_i, is based on a covariance matrix of x_i uncertainties. The expression for the covariance matrix, can be derived by truncating the Taylor series after the first derivative in Equations D.3 and D.4.

$$\sigma_y^2 \quad = \qquad \text{(D.5)}$$

Equation D.5 provides the form of the fundamental expression for uncertainty propagation in mathematical statistics. During the meeting, there was some confusion regarding the derivation of the covariances and response functions using the first order Taylor series terms. The appropriateness of using first and second order terms, as expressed by the first three lines in Equation D.4, was questioned.

References discussing the propagation of uncertainties generally divide Equation D.5 into two arrays, or matrices as shown by Equations D.6 and D.7. The first array (Equation D.6) is usually termed the response function matrix, or the sensitivity array, or the response surface.

Response Function Matrix

$$\sum_i \sum_j \frac{\partial y}{\partial x_i} \frac{\partial y}{\partial x_j} \tag{D.6}$$

This array is formed by squaring the derivatives on the right side of Equation D.5. The i index is the same as in Equation D.5, and the j index simply repeats the i values. If the indices in Equation D.6 were to represent the Equation D.4 variables, the values would be 1 and 2. It is understood that the product of the response function matrix (Equation D.6) and the covariance matrix (Equation D.7) produces Equation D.5.

Covariance Matrix

(D.7)

The covariance matrix includes the products and cross products of the uncertainties (standard deviations with consistent levels of confidence),

(D.8)

(D.9)

Equation D.10 is the same as Equation D.5 with the independent variables reduced to x_1, and x_2. As explained above when discussing Equation D.7, the variance in the dependent variable can be defined in terms of a unique covariance matrix of the independent variables.

$$\sigma_y^2 \quad = \qquad \text{(D.10)}$$

The covariance matrix for Equation D.10 is expressed below using matrix notation.

Covariance Matrix

$$\begin{vmatrix} \sigma_{11} & \sigma_{12} \\ \sigma_{21} & \sigma_{22} \end{vmatrix}$$

Equation 11 provides the expansion of the covariance matrix terms above into the product of standard deviations and correlation coefficients.

Covariance Matrix Expansion
Using Correlation Coefficients

$$\begin{aligned} \sigma_{11} &= \sigma_{x_1}\,\sigma_{x_1}\,\rho_{x_1 x_1} \\ \sigma_{12} &= \sigma_{x_1}\,\sigma_{x_2}\,\rho_{x_1 x_2} \\ \sigma_{21} &= \sigma_{x_2}\,\sigma_{x_1}\,\rho_{x_2 x_1} \\ \sigma_{22} &= \sigma_{x_2}\,\sigma_{x_2}\,\rho_{x_2 x_2} \end{aligned} \qquad \text{(D.11)}$$

The covariance is derived from the integral of the bivariate distribution in Equation 8 and related to the correlation coefficient in the bivariant form of Gauss' distribution function. Again, the expression in Equation 8[D8] is the same as Equation D.11 in this appendix.

Equation 1 represents the covariance matrix as expressed by Equation D.7 (in this appendix). Since the coefficient for each independent random variable in Equation 1[D8] is unity, the response function matrix (Equation D.6) in this appendix would be unity. Therefore, the covariance matrix represents Equation D.5. Equation 2 is the same as Equation 1,[D8] except that the coefficient for each independent variable (x_i) is a constant term (a_i). Thus, the products and cross products of the "a" terms represent the response function matrix (Equation D.6) in this appendix. Consequently, Equations 1 and 2 from Reference D8 are equivalent to Equations D.5, D.6 and D.7 in this appendix.

Equations 1 and 2 provide a linear relation between a dependent variable and a number of independent random variables that are functionally or correlatively related with correlation coefficients of unity. The coefficient of each expected independent variable is expressed using the symbol (k). The response function matrix would thereby be noted by an array of $k_i k_j$ symbols. The combination of the response function and covariance matrices is represented by Equation 3

In Equation 6,[D8] the expression for the dependent variable standard deviation includes the square of the first derivative of the dependent variable with respect to the independent variables. There are no second or higher order derivatives. With the square of the first derivative, Equation 6[D8] is the same as Equation D.5 in this appendix, when there is no dependency between the independent variables.

Equation D.5 appears to be the fundamental expression for uncertainty propagation in applications of mathematical statistics. Equation D.5 can be derived by truncating the Equation D.3 Taylor series after the first order terms

If the set of "i" - dimensional variables in Equation D.5 is reduced to two (x_1, and x_2), then Equation D.5 is reduced to Equation D.10. The two-dimensional Taylor series is expressed by Equation D.4, which includes explicit representation of first and second order derivatives. If the two-dimensional variables in Equation D.10 are reduced to one-dimension (x_1), Equation D.10 is reduced to Equation D.12.

$$\sigma_y^2 = \left(\frac{\partial y}{\partial x_1} \sigma_{x_1} \right)^2 \qquad \text{(D.12)}$$

If Equation D.4 is reduced to one-dimension (x_1), it would continue to include second order, and third order terms, et cetera. If the derivation of Equations D.5, D.10 and D.12 included the approximation of truncating the Taylor series after second, or higher order terms, then Equation D.12 would include the square of the standard deviation, squared, $\left(\sigma^2\right)^2$, as well as the square of the second derivative as shown by Equation D.4.

During the meeting, the explanations for several of the RAIs included applying Equation D.5 to Equations 7.3, 7.4 and 7.6. The following discussion outlines the application of Equation D.5 to these equations as presented during the meeting. The application involves defining the functional relation between the dependent and independent random variables. This functional relation is then applied to Equation D.5.

Uncertainty in Foil Dosimeter
Self-Absorption Correction

$$I(s) = I(d)\, e^{\mu x} \tag{D.13}$$

The self-absorption of gamma-rays, or other radiation, in a dosimeter, reduces the radiation intensity (I) from the dosimeter source (s). When the source intensity is measured with a detector (d), the measurements must be increased by the self-absorption loss to obtain an accurate intensity $\{I(s)\}$. The loss is a direct function of

the attenuation coefficient (μ) and the dosimeter thickness (x). The integral of Equation D.13 provides a sufficient expression for determining the self-absorption.

(D.14)

Equation D.13 is also sufficient to substitute into Equation D.5 for estimating the effects of dosimeter thickness uncertainties on the self-absorption uncertainty as shown by Equation D.14.

Weight Uncertainty

$$\mathrm{SpA} = {}^{\mathrm{A}}\!/_{\mathrm{gm}} \qquad \text{(D.15)}$$

The results of the dosimeter measurements are defined in terms of specific activity (SpA). The units are micro-Curies (A) from the product isotope per gram (gm) of the dosimeter parent isotope.

(D.16)

When the specific activity functional relation from Equation D.15 is substituted into Equation D.5, the effects of uncertainties in the dosimeter mass on the specific activity can be estimated as shown by Equation D.16.

When Equations D.14 and D.16 were derived and discussed during the meeting, it was clear that the uncertainties in the independent random variables were functionally unrelated and therefore independent of one another.

The equations are thereby reduced to the square root of the sum of the squares. The familiar form of Equations D.14 and D.16 cleared up the questions concerning the uncertainties in the dosimetry measurements related to Equations 7.3 and 7.4.

There were four generally obscure areas related to Equation 7.6. The discussions in these areas included : (1) (2) the response function sensitivity terms, (3) the measured value associated with the measurement uncertainty, and the need to have the degrees of freedom represented by a set, and (4) the relation of the Equation 7.6 measurement uncertainty to the total dosimetry database of measurement uncertainties. The following discussion relates Equations D.5, D.7, D.9 and D.11 to clarify the obscurity in the four areas associated with Equation 7.6.

(D.17)

$$\frac{\partial \mathrm{M}}{\partial m_d} = w_d \tag{D.18}$$

While the dosimeter uncertainties are not dependent on one another, they are dependent on the same set of constants. Consequently, the appropriate treatment of the correlation coefficients should reflect a direct relationship. This treatment means that the values in the set of correlation coefficients, is unity.

As suggested by Equation D.5, the propagation of uncertainties with response functions determined from the Equation D.18 functional relation, should represent the appropriate material and dosimeter weights for evaluation of the measurement uncertainty, as shown by Equation 7.6.

As noted during the meeting, the values in the set of material dependent correlation coefficients were assessed to be unity. This assessment was based on the same type of evaluation used to determine the appropriate values for the set of dosimeter correlation coefficients. With different weights for the materials and the dosimeters associated with each material, and the values in the material and dosimeter sets of correlation coefficients being unity, Equation 7.6 represents a covariance matrix of material dependent dosimeter uncertainties nested within a covariance matrix of material uncertainties.

FTI and the B & W Owners have interpreted ASTM E 185 requirements, that are referenced in Appendix H, of 10 CFR 50, to be appropriately satisfied with four dosimeter material types.

the maximum value for the denominator counter is determined by sets of at least four dosimeters, each of a different material type. In the B & W Owners Group Cavity Dosimetry Experiment referenced in the topical, it is

noted that there are twenty-four dosimeter - material sets. In most B & W Owners' capsules, there are four sets. In many other FTI analyses, there is just one set.

Appendix A of the topical, Tables A-1 and A-2 list 39 capsules and cavities in the dosimetry database (*DD*). Thus, the results from Equation 7.6 are evaluated for each capsule and cavity in the database. The measurement uncertainty (σ_M) for the entire dosimetry database (*DD*) is evaluated using Equation D.19 to appropriately combine the capsule and cavity uncertainties ($\sigma^2_{M(i)}$).

$$\sigma^2_{M(DD)} = \sum_{i}^{DD} w_i \, \sigma^2_{M(i)} \qquad \text{(D.19)}$$

The value of the database variance in Equation D.19 is estimated to be less than, or equal to 49 percent. This gives a measurement uncertainty of 7 percent or less.

D.2.2 RAI Set 2 Responses

This section provides the responses to the set of requests for additional information (RAIs) that were transmitted to the B&WOG in reference D2. The responses to each of the 19 RAIs are based on the discussions during the FTI, B & W Owners Group - NRC meeting. They also refer to the Statistical Methods section, which summarizes explanations discussed during the meeting. The responses include a few brief statements referencing the meeting.

Set 2 - Question 1

Equations (7.4) and (7.5) appear to incorrectly combine (%) relative errors and absolute errors (e.g., measured in cm or mg). Please explain this apparent inconsistency.

Response

Table 7-1, on page 7 – 1 2 of the topical, shows some measurement errors as absolute values, and some as relative values. As shown by Equations D.14 and D.16 in Section D.2.1, on page D - 41, all errors were converted to relative values for the propagation of uncertainties in Equations 7.1 through 7.5.

Set 2 - Question 2

Why is the helium concentration of samples DB-BEC, 9/26, 9/27, 4/10 and 4/12, and DB-Li-5A and 5B (Tables B-4.2-1 and B-4.2-2) a factor of ~ 10 less than the other samples ? Are these samples shielded ?

Response

The helium concentrations in the HAFM dosimeter samples noted in the question are approximately an order of magnitude less than other samples because these dosimeters

are located in the nozzle and seal plate elevations as shown in Figure 4.2, on page 4 - 16. They are not shielded dosimeters.

Set 2 - Question 3

Provide the values and basis for the measurement errors assumed in determining the dosimeter uncertainties of Tables 7-2 and 7-3.

Response

Two of the four volumes from the "Uncertainty Assessment and Results of Niobium Analysis for Davis Besse Cavity Dosimetry Benchmark Experiment" were provided during the meeting. The information included (a) the values, and (b) the basis for the measurement errors assumed in determining the dosimeter uncertainties. "Meeting Question 1", under the heading of "Additional Explanations", Section D.2.4, addresses the other two volumes.

Set 2 - Question 4

Why are the dosimeter measurement uncertainties of Tables 7-2 and 7-4 different ? Which values are used in the FTI analysis ?

Response

Set 2 - Question 5

Using a conservatively large or bounding value for the measurement uncertainty with Equation (7.9) results in a nonconservative estimate for the calculation uncertainty. A conservative calculation uncertainty should be determined using a minimum value for the measurement uncertainty.

Response

based on the values which experimentalist assign to cross section measurements using the same activation techniques, the value of 7.0 percent is estimated as an appropriate measurement uncertainty as explained during the meeting.

Set 2 - Question 6

The form of Equation (7.6) appears to be incorrect. Also, provide the values and basis for w_{mat}, ρ_{mat}, w_d, ρ_d, $\sigma_{mat,d}$ and $N_{\{mat,d|\geq 4\}}$ in Equation (7.6).

Response

The "Statistical Methods" presented in Section D.2.1, summarizes the derivation of Equation 7.6, and explains the basis for its form on pages D - 42 through D - 46. During the meeting discussion, the correlation coefficients for ρ_{mat} and ρ_d were explained to have values of unity.

The standard deviations come from Table 7-4, on page 7 - 18, and include the covariance matrix with the combined set of correlation coefficients.

Set 2 - Question 7

In the application of Equation (3.17), what irradiation period was used in determining the effect of the power history on the dosimeter response ? If the power history used in Equation (3.17) was averaged over an irradiation interval larger than one month, provide an estimate of the effect of this approximation on the dosimeter response.

Response

The total irradiation period was from December 5, 1988, to January 26, 1990, which constituted the operation of Davis Besse Cycle 6. While the duration of the time steps used to calculate the fraction of saturation varied, none of them were greater than 1 day.

Set 2 - Question 8

Equations (7.1)-(7.5) assume that the contribution to the measurement error from a given error source is equal to the error in the source. For example, the error in the measurement due to dimensional errors is taken to be the same as the error in the dimensions. Since this is not generally valid, standard uncertainty analyses relate the error source and resulting measurement error using sensitivity factors which express the sensitivity of the measurement to errors in the source variable. These sensitivity factors can be significantly different than unity when the

measurement has a weak nonlinear dependence on the source variable (e.g., in the case of the exponential dependence of the absorption correction on the dosimeter thickness). These sensitivity factors should be included in the uncertainty equations.

Response

Equations 7.1 through 7.5 do appear to assume that the contribution to the measurement error from a given error source is equal to the error in the source. However, as explained during the meeting, and shown in Equation D.14 on page D - 41, Section D.2.1, the non-linear sensitivity factors, or response functions, are appropriately included in the respective uncertainty terms.

Set 2 - Question 9

Provide the value and basis for the weighting ρ_{mat} used in Equation (7.13). Is the value the same as used in Equation (7.6) ?

Response

The value and basis for the weighting ρ_{mat} used in Equation 7.13, results from the fact that the material dependent C/M benchmarks for any capsule or cavity come from a single calculational process. Thus, all material dependent C/M results are related. Consequently, the correlation coefficients are unity. The material correlation coefficients used in Equation 7.6 are also unity.

Set 2 - Question 10

Please define the denominator in Equation (7. 10).

Response

The total number of independent capsule and cavity data sets in the dosimetry database (*DD*) is thirty-nine as described on page 7 - 28 of the topical.

Set 2 - Question 11

Because of the strong fluence attenuation between the core and vessel, the dosimeter response is very sensitive to the methods and data used in these calculations. As a result, typical pressure vessel fluence calculations are expected to provide an accuracy of ~ 15% when predicting (> 1-MeV) dosimeter response. The major contributors to this uncertainty are the (1) relative core/vessel/dosimeter geometry (2) nuclear cross sections and fission spectra (3) determination of the core neutron source (4) methods and modeling approximations and (5) the Equation (3.17) adjustment for irradiation and decay times. In view of the fact that the observed M/C uncertainty is substantially less than 15 %, provide an explanation for this reduced M/C uncertainty. Have any adjustments (other than those explicitly identified in the report) been made to improve the M/C agreement ?

Response

The C/M benchmark uncertainty in the topical is percent. While the question suggests that the industry uncertainty is around 15.0 percent, draft regulatory guide DG-1053, dated June, 1996, and Table 2-1 (page 2 - 3) in the topical, indicate that the industry uncertainty is generally considered to be more than 20.0 percent, and nearly 30.0 percent when FTI predictions are not included to lower the average. FTI's high degree of precision is also noted in the PCA blind test, where the FTI predictions are

within the measurement uncertainty, while those of others have deviations of twice the uncertainty (NUREG/CR-1861 discusses the PCA results, Reference 37 in the topical).

The reasons for the outstanding accuracy and precision in the FTI predictions are generally costs and expertise. The FTI analyses are performed by senior analysts, who have developed very detailed models for the calculations, including pin by pin fission rates, et cetera. The analyses are therefore more costly than others in the industry.

The measurements are performed by an independent laboratory, and the results independently reported. For the Cavity Dosimetry Experiment, the experimental methods, results, and uncertainties were also checked by independent consultants. The calculations come from standard computer codes, and the FTI procedures are described in topical Section 3.0. The results of the calculations and measurements are shown in Table A-1. The NRC has confirmed the reduced value of the C/M uncertainty by statistically processing the Table A-1 data.

Set 2 - Question 12

In the third column of Table A-2, the value of 1 - C/M is provided instead of $\sigma_{C/M}$. Provide the plant dependent value of $\sigma_{C/M}$.

Response

The third column in Table A-2 will be removed. Since the value of $\overline{\sigma_{C/M}}$ is defined by Equation 7.15, there is no value for each individual capsule and cavity.

Set 2 - Question 14 (Due to formatting difficulties, Question 13 follows Question 15)

The BAW-2241-P fluence methodology does not include the analytic determination (based on numerical sensitivities) of the fluence calculation uncertainty as described in DG-1053. Please identify any other calculation or measurement differences between the proposed methodology and the guidance of DG-1053.

Response

The first statement, that the BAW-2241-P fluence methodology does not include the analytic determination (based on numerical sensitivities) of the fluence calculation uncertainty as described in DG-1053, is not accurate. It is not possible to infer the vessel fluence uncertainty from either benchmark uncertainties of calculations to measurements, or from measurement uncertainties. Therefore, as stated on page 1 - 2 of the topical, analytical vessel fluence uncertainties were integrated with capsule and cavity benchmark uncertainties.

In addition, the draft guide requirement that the measurement uncertainty include a reference field validation, is in progress. The Owners have agreed to send the NRC a copy of the report (for information only) after it is completed in 1999.

Whether the topical meets all of the requirements of DG-1053, or whether there may be differences, the NRC agrees that the topical includes the most advanced fluence technology and most comprehensive uncertainty methodology developed to date, to meet the requirements of the draft guide.

Set 2 - Question 15

The calculational perturbation factors of Appendix-C were determined using the BUGLE-80 fluence methodology rather than the most recent BUGLE 93 Semi Analytic approach. What is the effect on the M/C data-base and associated biases and uncertainties of using this earlier methodology ?

Response

As discussed in the meeting, the effect of using the BUGLE-80 results on the C/M database, and thus on the bias and uncertainty, is negligible.

Set 2 - Question 13

Were the benchmark data-base capsule and cavity measurements of Table A-1 which are identified by plant actually made at the assigned plant, or were the dosimeters/capsules from the assigned plant irradiated in a different (or surrogate) plant ? Please identify any measurements that were not actually installed and measured at the indicated plant.

Response

As discussed in the meeting, and described in the "Integrated Reactor Vessel Material Surveillance Program" topical (BAW-1543A[10]), most B & W plant capsules were irradiated at a host (surrogate) plant. All cavity measurements were actually made at the indicated plant.

Set 2 - Question 16

The standard deviation of the M/Cs (from the overall average M/C) in the Appendix-A data-base appears to be almost a factor of two larger than the value given in the text (on p. 7-33). Please provide an explanation for this difference.

Response

The difference is a result of the energy dependent bias removal function for neutron energies above 0.1 MeV. Section D.2.4, "Additional Explanations" discusses the bias removal function in "Meeting Question 3". The evaluation of the NRC standard deviation, and the FTI value is discussed in the "Statistical Processing of Table A-1 Data" section (D.2.3) that follows.

Set 2 - Question 17

The calculation uncertainty is determined by combining the measurement uncertainty, σ_M, and the standard deviation, $\sigma_{C/M}$, of Equation (7.16). However, it is not evident that these two quantities refer to the determination of the same response (as required). For example, it appears that σ_M refers to the uncertainty in the measurement of a specific nuclide (e.g., Ni-58(n,p)) while $\sigma_{C/M}$ refers to the C/M deviation for the <u>average</u> of all nuclides of a given <u>capsule</u>. Please explain this apparent discrepancy and justify any differences in the response being used in the definitions of σ_M and $\sigma_{C/M}$.

Response

The statement that the calculational uncertainty is determined by combining the measurement uncertainty, σ_M, and the benchmark standard deviation, $\overline{\sigma_{C/M}}$ of

Equation 7.16, is not accurate.

Set 2 - Question 18

BAW 2241-P states that the BUGLE-93 calculations of one of the dosimeter responses is erroneous (p. 7-29) and that the BUGLE-93 calculated C/Ms for this type of dosimeter have been removed from the analysis (p. 7-31). In addition, it is stated (p. 6-4) that this dosimeter has "special problems." What is the C/M bias for this type of dosimeter and is this improved by the use of a BUGLE-93 (rather than CASK) photo-fission correction ?

Response

Set 2 - Question 19

Recent calculations described in NUREG/CR-6453 suggest that the BUGLE-93 cross section library results in an underprediction (relative to BUGLE-96 and SAILOR-95) of the Fe-54, Ni-58, U-238 and Np-237 cavity dosimeter reaction rates of 1%, 2%, 4%, and 10%, respectively. (The prediction of the in-vessel dosimeter reaction rates for the three libraries agree to within 1% .) In view of the difference between these libraries, please review and update the FTI M/C data-base and methodology, as necessary. Will this update allow the inclusion of the threshold dosimeter measurements that were excluded ?

Response

In 1980, the BUGLE-80 library was considered to be the best in the industry for fluence analyses. By 1988, the NRC had convinced FTI and the B & W Owners that the fluence technology using the CASK library was too outdated. Therefore, in concert with the B & W Owners Group Cavity Dosimetry Experiment, FTI performed the analysis using both the CASK and BUGLE-80 libraries.

While the BUGLE-80 results showed a bias in the cavity dosimetry benchmark, the CASK results did not. In the capsule dosimetry benchmark results, neither library showed any statistically significant bias. Furthermore, the uncertainty in the capsule results from both libraries was statistically the same with greater than a 95 percent level of confidence.

Due to the BUGLE-80 bias, the NRC stopped recommending that library and began recommending the BUGLE-93 one. The B & W Owners again paid for a

comprehensive analysis and uncertainty evaluation of the Cavity Dosimetry Experiment. The reanalysis of the results showed no biases in either the capsule or the cavity. However, the Np-237 dosimeter results indicated a bias in that dosimeters cross sections. This was particularly evident in the capsule compared to both the BUGLE-80 and CASK results.

Now, the NRC is recommending another update, from the BUGLE-93 library to the BUGLE-96 one. It appears that the BUGLE-93 results could possibly be biased and under-predict the fluences relative to BUGLE-96.

Technically, FTI agrees that the Np-237 is probably biased, and the calculations under-predict the reactions (see Table 6-1, on page 6 - 3, and Table 6-2, on page 6 - 5 of the topical). Furthermore, updating the library to the best available one, is technically better than any other option. However, from economical considerations, updating the library is the least cost effective option. The topical already notes that Np-237 appears to have biased cross sections that cause the calculations to under-predict the measurements. The B & W Owners have funded a program, that will be completed by 1999, to evaluate the cause of the Np-237 bias. As noted in the "Meeting Question 3" discussion for "Additional Explanations" (Section D.2.4), FTI utilizes an energy dependent bias removal function to treat the effects of calculational biases in the Fe-54, Ni-58 and U-238 dosimeters.

Since the FTI calculational methodology, using the BUGLE-93 library, is not biased, and the deviations in the Fe-54, Ni-58 and U-238 reaction rates between the BUGLE-93 and BUGLE-96 calculations are well within the uncertainties of the FTI, methodology,

FTI and the B & W Owners believe that a BUGLE-96 update is neither technically warranted, nor economically cost effective. Thus, in the future, only the Np-237 bias will be evaluated and corrected.

D.2.3 Statistical Processing of Table A-1 Data

During the August the fifth and sixth meeting between FTI, the B & W Owners, and the NRC, the NRC explained that they had statistically processed the Table A-1 data in the topical. The processing included the creation of a dosimeter by dosimeter benchmark of M/C ratios for the specific activities. The M/C ratios for the 728 dosimeters in the dosimetry database were averaged to determine a mean value of 0.9940. The dosimeters were assumed to have independent uncertainties. Therefore, a benchmark uncertainty for the database was estimated by appropriately evaluating the standard deviation. The statistical procedures for the evaluation included assuming a bias,

The differences in each of the 728 benchmark ratios were squared, summed, and divided by 727 degrees of freedom. This gave a relative standard deviation

The NRC's conclusion from this evaluation was that the FTI methodology had no statistically significant bias

However, the uncertainty of percent was considerably larger than FTI value of percent, noted on page 7 - 33, and calculated with Equations 7-12, 7-13, and 7-15 from the Table A-2 data. Consequently, the NRC wanted an explanation

concerning the validity of the FTI benchmark uncertainty, particularly explaining why a value closer to percent would not more appropriately represent the methodology.

In the past, when the M/C ratio was used to convert calculations to measurements, the measurement bias and the conversion process, or unfolding uncertainty, were related to this ratio. However, when the NRC suggested in DG-1053, that vessel fluence predictions would not have an appropriate uncertainty, unless they were based on calculations, the M/C ratio lost its physical significance. In the topical and meeting discussion, the C/M ratio is referenced as the appropriate term for determining the bias and standard deviation

When the Table A-1 data was processed to determine the mean C/M ratio for the 728 dosimeters, the resulting value was 1.0310. Assuming that the mean value represents a bias, the standard deviation was computed to be percent. This computation of the standard deviation assumes that all dosimeter benchmarks are independent of one another. Thus, the cross product dependency parameters in the covariance matrix are represented by a null set.

The value of percent is based on the assumption that all dosimeter benchmarks are independent of one another. However, as discussed in the "Statistical Methods" section (D.2.1, page D - 43, below Equation D.18, through page D - 45), and in the "Addition Explanations" section (D.2.4), on "Meeting Question 4", the dosimeter uncertainties for each capsule and cavity analyses in Table A-1 are not independent. The uncertainties are directly related to the five constant parameters in Table 7-1, on page 7 - 12 of the topical. Therefore, the appropriate treatment of the correlation coefficients, representing the cross product dependency between dosimeters, should reflect a direct relationship. This treatment means that the values in the set of correlation coefficients are unity.

The explanation and evaluation of the benchmark standard deviation, is focused on the C/M ratio, correlation coefficients of unity, the bias, and the standard deviation difference between a value of percent, estimated in the topical (page 7 - 34, just

below Equation 7.19) and a value of percent or greater, estimated from the above discussions. The difference between C/M and M/C ratios demonstrates why relative standard deviations are frequently closer to a natural logarithm normal distribution than a standard normal distribution. However, the topical discusses the fact that the database of benchmark deviations fits within Student's central "t" distribution with a probability greater than 95 percent. Thus, the C/M ratio is not a parameter that causes the standard deviation difference.

A set of null values for the correlation coefficient, will generally produce a lower standard deviation than a set with values of unity.

The correlation coefficients used in the topical are determined from the physics of the functional relations. Consequently, the correlation coefficient values in Equations 7.12, 7.13 and 7.15 are uniquely determined and do not represent a statistical approximation.

The bias evaluation is the key to understanding the difference between the standard deviation values greater than percent, and the topical estimate or the Equation 7.15 benchmark result of percent. As discussed in the topical, and demonstrated by the mean M/C and C/M ratios, the fluence calculation, integrated over the energy range greater than 0.1 MeV, shows no indication of a bias.

The discussions addressing "Meeting Question 5" in the "Additional Explanations" section (D.2.4), explain that each capsule and cavity analysis does not represent independent calculations of the dosimeters. There is generally a spatial and spectral fluence function at the dosimetry location. The fluence is multiplied by constant cross section - response functions to obtain the saturated asymptotic specific activity. This activity is multiplied by the analytical expression representing the fraction of saturation to obtain the specific activity for benchmark comparisons to the measurements (pages 3 - 30 through 3 - 32 in the topical). Thus, even though there may be four or more dosimeter materials, the benchmark evaluation uses correlation coefficients with values of one in Equations 7.12 and 7.13.

To evaluate the effects of removing the bias with Equation 7.13, the form of Equations 7.12 and 7.13 was modified to process systematic and random deviations. The processing of the systematic deviations used Equation 7.10 (page 7 - 28) to define biases

The 39 capsule and cavity standard deviations are combined as the root mean square, of the sum of the standard deviations, squared. The modified form of Equation 7.15 continues to have 38 degrees of freedom. The resulting benchmark standard deviation is percent. The fact that this value is statistically within 3.01 percent of the benchmark standard deviation estimated with Equations 7.12, 7.13 and 7.15 in the topical, and is less than the topical value of percent, provides confidence that the topical "Uncertainty Methodology" is appropriate.

The fact that the benchmark standard deviation in the database may be estimated to be greater that percent, appears to be a function of (a) the bias, and (b) the sets of unity correlation coefficients

The value of the mean bias affecting the database is estimated by combining the biases in Table D-1.

The mean effective bias, estimated from the above evaluation is percent. With the correlation coefficient values in the covariance matrix of dosimeter uncertainties represented by sets of unity, combining the bias, as if it represented a standard deviation, with the unbiased benchmark standard deviation, is simply additive. Consequently, the covariance matrix combination of the mean effective material bias, and the benchmark standard deviation, gives a biased standard deviation of percent. This is comparable to the percent biased standard deviation initially estimated by processing the C/M dosimetry benchmark ratios in Table A-1.

The summary of the evaluation is that differences between estimates of a benchmark uncertainty greater than percent, versus the topical value of percent, is due

to the fact that the values greater than percent contain an energy dependent bias. The effects of the bias are furthermore accentuated by the fact that the dosimetry uncertainties for each capsule or cavity analysis are not independent. The dependency between the standard deviations in the covariance matrix, result in the energy dependent bias, directly increasing the unbiased benchmark standard deviation as an additive term.

D.2.4 Additional Explanations

As FTI was addressing the second set of RAI's during the August the fifth and sixth meeting between FTI, the B & W Owners, and the NRC, the NRC questioned five areas that needed in-depth additional explanations. These questions could not be addressed during the meeting, because of time constraints. This section of the appendix lists the five meeting questions, and provides more of an in-depth response than provided at the meeting.

Meeting Question 1

Send the measured data and "Uncertainty Assessment... " documents from the B & W Owners Group Cavity Dosimetry Experiment to the NRC.

Discussion

FTI and the B & W Owners have received the NRC's letter stating that there is no problem with the data being proprietary (from Joseph L. Birmingham, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, dated, October 13, 1998). The documents are in the process of being copied, and will be forwarded to Dr. Lambros Lois when they are ready.

Meeting Question 2

The benchmark uncertainty includes results from CASK, BUGLE-80, and BUGLE-93. The NRC questions: how these three different cross section sets provide a consistent benchmark uncertainty ? It would appear to be necessary to update all capsule and cavity calculations with one consistent cross section set, preferably based on BUGLE-96. Why is this not necessary ?

Discussion

The draft Regulatory Guide, DG-1053, and RAI 19 (Set 2), suggest that updating the technology for fluence analyses, including the latest cross section library, is advisable to ensure sufficiently accurate predictions of vessel fluence values. Technically, this suggestion is appropriate. However, as noted in the response to RAI 19, it is not cost-effective to routinely update the technology, if the current technology is accurate (representing a best-estimate, with no observable biases or errors), and has a well-defined uncertainty. Moreover, when it is warranted from both technical and economical considerations, to update the fluence technology, the most cost-effective option would not be to completely reanalyze all capsules, cavities, and dosimetry in the database. The incremental safety, licensing, operational, et cetera, benefits of such a reanalysis would have to be enormous to adequately offset the commensurate costs.

While reanalyzing all capsules, cavities, and dosimetry in the database would not generally be warranted economically, reanalyzing only one capsule or cavity would not be technically justifiable. As discussed in Section 7.0 of the topical and in this appendix, the B & W Owners Cavity Dosimetry Experiment, which also includes a comprehensive capsule analysis, represents just two degrees of freedom in the statistical evaluation of the benchmark data. Therefore, updating the technology with two

benchmarks would not provide a sufficient level of confidence in the results to ensure consistency with the safety evaluations.

FTI and the B & W Owners were faced with the situation of developing an economical, but technically valid program for updating the fluence technology for the Cavity Dosimetry Experiment. (See pages 1 - 1 and 2 - 11 in the topical, which discuss (1) updating the cross section libraries from CASK to BUGLE-80, and then to BUGLE-93, (2) updating the predictive methodology from measurement based to calculation based, and (3) updating the
uncertainty methodology.) Updating the technology, with changes in both the predictive methods and the cross section libraries, would have been excessively costly if the entire database were reanalyzed. Nonetheless, the incremental gains in safety and licensing margins were considered to be technically important. Therefore, to be cost-effective and technically justifiable, the proposed improvement in the technology included benchmarks of the Cavity Dosimetry Experiment with the CASK, BUGLE-80, and BUGLE-93 libraries.

The measurement database was updated to exclude any effects of the analytical analyses. Thus, there is no dependence on any of the dosimeter measurements from the three libraries. As discussed in the topical, the update of the measurement uncertainties demonstrated that the estimated values were valid for all the previous dosimetry measurements.

The benchmarks to the Cavity Dosimetry Experiment, with calculations based on the CASK, BUGLE-80, and BUGLE-93 libraries, provided a means of assessing the uncertainty in the calculations with respect to each library.

The bias removal function was found to be independent of the libraries, although there were some differences in the energy dependent factors. The calculations using the BUGLE-80 library were clearly biased by the vessel. Therefore, no cavity benchmark results were included in the BUGLE-80 comparisons to the other libraries. The Np-237 cross section in the BUGLE-93 library, was clearly biased in comparison to the CASK and BUGLE-80 results. Therefore, no Np-237 dosimetry was included in the BUGLE-93 comparisons to the results from other libraries. With the biases appropriately treated for each library, the benchmark standard deviations were evaluated. In addition, the benchmark results between libraries were compared and the standard deviations evaluated.

The unbiased uncertainty evaluation indicated that each library had an uncertainty that was statistically indistinguishable from the uncertainties in the other libraries. Furthermore, the evaluation indicated that the standard deviations between libraries were statistically insignificant compared to the standard deviations of each library.

The additional benchmark comparisons of the results from one library, to those of the other libraries, established a cross-reference relating the uncertainties between libraries. Thus, it is possible to estimate the differences in the results between calculations of capsule or cavity

dosimetry using the BUGLE-93 library, relative to calculations using CASK. The probability that the BUGLE-93 results will bound the CASK results is well-defined, with a high level of confidence.

If calculations using the CASK library are benchmarked to a set of measurements, and a benchmark uncertainty is estimated, then the BUGLE-93 benchmark uncertainty may be estimated without performing the calculations. The common benchmark of calculations using BUGLE-93 and CASK provides the means of combining the two benchmarks to estimate the standard deviation in the BUGLE-93 benchmark.

In conclusion, calculations using the CASK, BUGLE-80, and BUGLE-93 cross section libraries to estimate a benchmark uncertainty, provide consistency by including a cross-reference where the libraries are appropriately benchmarked to one another. The cross comparison of benchmark results, and the statistical assessment of the significance of any differences, provides the means of estimating an uncertainty with the appropriate level of confidence.

Meeting Question 3

The bias removal function within the energy range greater than 1.0 MeV, needs to be explained.

Discussion

As discussed during the meeting between FTI, the B & W Owners, and the NRC, the FTI calculational methodology has a bias as a function of energy. The bias, the bias removal function that is used to eliminate the bias, and the application of the bias

removal function to obtain best-estimate fluences, was previously presented to the NRC. The presentation was in a letter dated March 4, 1997, from Arkansas Nuclear One, Unit 1. This letter was in response to a set of Request for Additional Information regarding the RCS Pressure and Temperature Limit Technical Specification Change Request. The information discussed below is an update of that previously presented.

Before explaining the development of the bias removal function, the definition of the key terms is presented.

Definitions

(A) The bias removal function can be expressed as either a continuous function (f) of energy (E),

$$h = f(E),$$

or a discrete constant by energy group (g),

$$h_g = \text{Constant}_g .$$

The discrete form is used in practice.

(B) The h_g's were determined during the evaluations and analyses of the Cavity Dosimetry Experiment, as discussed below. The numerical values of the h_g's are given in Table D-2.

(C) The h_g is independent of (1) any specific plant, (2) spatial locations throughout the core, reactor internals, vessel, and cavity, within the belt-line region, and (3) the dosimeter material type.

(D) Application of the bias removal function (h_g) to the DORT - calculated fluence, produces the best-estimate (unbiased) fluence. Typically, this amounts to less than a 5 percent change in the magnitude of the calculated fluence. (The h_g is not applied to the measurements.)

Development

Introduction

One of the primary goals of the B & W Owners Cavity Dosimetry Program was to develop a calculational-based methodology that could be used to accurately determine the neutron fluence in the surveillance capsule, reactor vessel, and reactor vessel cavity structure. An accurate methodology already existed for the capsule and vessel, however, it was necessary to extend and modify the methodology to accurately calculate the energy-dependent dosimeter responses in the cavity. The measurement results from the Cavity Dosimetry Program were used in a statistical analysis to identify, and quantify an energy dependent bias in the calculated fluence. This calculational bias is a function of the methodology. As such, it is general, not specific to the Cavity Dosimetry Experiment, and therefore applies to all analyses that use the Semi - Analytical methodology described in the topical. The discrete form of the bias removal function, is applied on a group-by-group basis, using a set of constant "bias factors" (h_g) which remove the bias in the calculated fluence in each specific energy group.

The Concept of the Bias Removal Function

This section describes the general concept associated with the bias removal function. The true value of some arbitrary physical quantity, Q, is defined as Q^{TRUE}. The value

of the same quantity, determined by some analytical process, is defined as C. Likewise, the value of the same quantity, determined by some experimental process, is defined as M. In general,

$$C \neq Q^{TRUE},$$
$$M \neq Q^{TRUE}, \text{ and}$$
$$M \neq C.$$

The goal is to determine the best-estimate of the true value, Q^{BEST}, which in a calculational-based methodology is defined by:

$$Q^{BEST} = C_{UNBIASED} \tag{D.20}$$

(D.21)

(D.22)

$$C_{UNBIASED} = C\left(h^{-1}\right) \tag{D.23}$$

where $h =$ (D.24)

Q^{BEST} will of course differ from Q^{TRUE}, however, Q^{TRUE} is bracketed by Q^{BEST} over a range that is defined by either, the sum of Q^{BEST} and the uncertainty in Q^{BEST}, such as,

$$\left[Q^{BEST} - U(Q^{BEST}) \right] \leq Q^{TRUE} \leq \left[Q^{BEST} + U(Q^{BEST}) \right] \qquad (D.25)$$

or, by the product.

$$\left[C_{UNBIASED} \left(1 - U\right) \right] \leq Q^{TRUE} \leq \left[C_{UNBIASED} \left(1 + U\right) \right] \qquad (D.26)$$

Combining Equations D.23 and D.26 yields

$$\left[C \left(h^{-1}\right) \left(1 - U\right) \right] \leq Q^{TRUE} \leq \left[C \left(h^{-1}\right) \left(1 + U\right) \right] \qquad (D.27)$$

(D.28)

The Bias Removal Factor

The preceding generalized discussion expresses the theory upon which the determination of the bias in the fluence is based. In moving from the general to the

specific, however, there are a number of significant differences, which will now be discussed.

The neutron fluence, which is the quantity of interest, is not (and cannot be) measured directly. Instead, a quantity that is related to the flux in a known way, - the dosimeter response - is measured. Consequently, M, C, and C/M would not have the same relationship to the neutron flux (ϕ) that they would have had in the previous theoretical discussion, but the fundamental idea still applies.

The flux of interest is integrated over the energy range, E > 1.0 MeV. The measured quantity is a dosimeter response. This response is related to the flux through energy dependent cross section - response functions. The dosimeter measurement represents an integration over the energy range of the dosimeter response. With four or more dosimeter measurements, each representing an integration over different energy ranges, a bias, which is a function of energy, can be uniquely identified. Since the energy dependent bias can be uniquely identified, an energy-dependent bias removal function can be derived to remove the bias from the calculated flux. While the bias removal is a function of energy, it is a constant related to the calculational methodology. It is not related to a plant-specific calculation, but rather to all calculations for every plant. Expressed discretely, the bias removal function would have the following form,

$$h_g = \qquad \text{(D.29)}$$

and it would be used to determine the best-estimate flux as follows:

$$\phi^{BEST} = \sum_g \left(\phi_g^{calc}\right)\left(h_g^{-1}\right) \qquad \text{(D.30)}$$

where g = energy index

ϕ_g^{calc} = calculated neutron flux in group "g"

h_g = bias removal factor for group "g"

The bias removal methodology must be able to determine the fluence at numerous locations in the reactor vessel. Given the fact that the geometrical configuration of the core, and internals structure is very complex, it would be reasonable to think that the bias would be spatially dependent as well as energy dependent. If the energy-dependent bias was also a function of space, the best-estimate fluence at the vessel inside surface would have to be obtained using multiple sets of bias removal factors.

The possibility of a spatially dependent bias in the calculational methodology was one of the fundamental issues that the Cavity Dosimetry Experiment was designed to address.

Table D-2 Bias Removal Factors (E > 1 MeV)

Energy Group	Upper Energy, MeV	h_g
1	17.33	
2	14.19	
3	12.21	
4	10.00	
5	8.607	
6	7.108	
7	6.065	
8	4.966	
9	3.679	
10	3.012	
11	2.725	
12	2.466	
13	2.365	
14	2.346	
15	2.231	
16	1.921	
17	1.653	
18	1.353	
19	1.003	

Meeting Question 4

The measurement uncertainty computed with Equation 7.6, does not clearly represent the sensitivity of the response function relation between the database, and each of the 728 dosimeter measurements listed in Table A-1. The NRC would like an explanation describing the consistency between the individual dosimeter measurement uncertainties and the overall measurement uncertainty for the dosimetry database.

Discussion

RAIs 1, 6 and 17 from Set 2, illustrate the range of meeting discussions that concerned the uncertainties in the measurements. The range varied from discussions concerning, (a) what is actually being evaluated in the benchmark of calculations to measurements, and consequently, what specifically is related to the benchmark uncertainty, to (b) what is the meaning of the correlation coefficients

The NRC processed the dosimetry database in Table A-1 and confirmed that there was no bias in the FTI calculational methodology, for neutron reactions with energies above 0.1 MeV. From this result, a reasonable conclusion was that the benchmark uncertainty could be determined by statistically processing the individual dosimeters as outlined above in Section D.2.3, discussing the "Statistical Processing of Table A-1 Data".

As the discussions during the meeting provided the additional information and explanations for the RAIs, it became clear that individual dosimeter benchmarks of calculated activities to measured values did not provide a sufficient benchmark for the calculational methodology, and thereby did not provide a sufficient benchmark uncertainty. Thus, even though the NRC processing of Table A-1 confirmed that FTI's calculations of greater than 0.1 MeV fluences and activities have no bias, the conclusion that the statistical processing of the individual dosimeters provides an estimate of the benchmark uncertainty is not valid.

When the dosimetry is sufficient to provide two or more energy - dependent responses in the range above 0.1 MeV, the measurements are combined by weighting the respective materials. The measurement of the specific activity from neutron reactions with energies greater than 0.1 MeV is unbiased. Corresponding to the measured specific activity, there is a single calculation of the fluence as a function of space, energy, and the integrated time period for the dosimetry exposure. The dosimetry specific activities are calculated from the fluence spectral results, with energy group constants for the cross section - activity - response functions, and time dependent decay and operational effects represented analytically. The dosimetry in the capsules and cavity show negligible spatial - spectral effects (with the exception of the Owners Cavity Dosimetry Experiment) because they are in such close proximity to one another.

The overall objective of the "Uncertainty Methodology" in Section 7.0 of the topical is to be able to have an appropriately high degree of confidence that the results of the calculated fluence, plus or minus an estimated uncertainty, have a known probability of bounding the true fluence. The fluence of interest has neutron energies greater than 1.0 MeV. The true fluence is defined in terms of measured specific activities, and the measurement techniques are calibrated to National Institute of Standards and Technology certified standards.

While the measurements are unbiased, they have an uncertainty associated with them due to random deviations. Consequently, in using the measurements as a reference for the benchmark of the calculations, it is necessary to know an estimate of the standard deviation and confidence level in the experimental methodology. The estimate of the standard deviation in the measured specific activity begins with Table 7-1 (page 7 - 12) in the topical. The random deviations in the table are combined in Equations 7.1 through 7.5 with examples of

details shown in Equations D.14 and D.16. The result is a relative standard deviation for each dosimeter.

On page D - 42, in the paragraph above Equation D.17, the combination of dosimeter measurements for a single capsule or cavity evaluation is discussed in relation to the uncertainty determined with Equation 7.6. The calculations, and the calculational uncertainty evaluations, indicate that the capsule or cavity dosimetry have the same fluence. Therefore, all dosimeters of the same material type, such as Fe-54 foils, should have the same specific activity. Since the measurements have no biases (Section 7.1.1 of the topical, pages 7 - 9 and 7 – 10), a single mean measured specific activity is obtained by averaging the measured results.

The standard deviation in the mean specific activity for all dosimeters of the same material type could be estimated from the deviations in specific activity between pairs of the individual dosimeters. The individual deviations would be independent of one another. Accordingly, the cross terms in the covariance matrix would be zero, and the standard deviation would be estimated by the root mean square of the sum of the individual deviations, squared. This would include the statistical approximation that the degrees of freedom in the denominator, would be the total number of dosimeters of the respective material type, minus one.

While the above procedure would be acceptable, the preferable procedure (a) recommended by the draft regulatory guide, DG-1053, and (b) the one historically used in the fluence arena, is to use the components of the experimental methodology as

discussed in the topical and represented by Equations 7.1 and 7.5. The standard deviation in the mean specific activity for all dosimeters of the same material type, would be estimated as the root mean square of the covariance matrix, as expressed by Equation 7.6.

The resulting response function is represented by Equation D.18, modified by the statistical approximation for the degrees of freedom. If the correlation coefficients for cross product dependency represent a null set, then the mean measurement uncertainty for all dosimeters of the same material, is the square root of the sum of the individual standard deviations, squared. However, as explained on page D - 43, following Equation D.18, the individual standard deviation for each dosimeter is dependent on a set of constant parameters. Accordingly, the values in the set of correlation coefficients are unity. Thus, the mean measurement uncertainty is the square root of the sum of the products and cross products of the individual dosimeter standard deviations.

The uncertainty that has been estimated in the above discussion is related to one material type in a single capsule or cavity evaluation. While this is an interesting value, and is suitable for benchmark comparisons of the calculated dosimeter material specific activity, the objective is to determine the uncertainty in the specific activities resulting from neutron reactions greater than 1.0 MeV. The uncertainty in a single material type of dosimeter measurements that principally respond to a unique portion of the energy range above 1.0 MeV, does not represent the uncertainty in the measurement of the entire range. Consequently, the uncertainties in several material types of dosimeters

that are sensitive to different ranges of the neutron spectrum above 1.0 MeV, are required.

In capsule or cavity evaluations of the greater than 1.0 MeV fluence, the dosimetry consists of several material types, with several dosimeters of each type. The measurements of the specific activity resulting from neutron reactions with energies greater than 0.1 MeV show no functional relation to the neutron energy. Thus, the Cu-63 reaction to produce Co-60 in the energy range above 5.0 MeV, and the Co-59 reaction to produce Co-60 in the energy range below 10.0 KeV, show no significant differences. Accordingly, the uncertainty in the combined measurements of the specific activity, incorporates a response function that provides each material uncertainty with an equal weight. Thus, in Equation 7.6, each of the four or more materials that are combined to represent the uncertainty in the greater than 0.1 MeV specific activity, has an equal weight (usually one-fourth). In addition, each dosimeter of that material in the capsule or cavity, has an equal weight relative to the inverse of the total number of dosimeters of that material type.

It has been explained above, and in the "Statistical Methods" section, that the correlation coefficients are represented by two sets

The result is the uncertainty in a capsule or cavity measurement of the specific activities greater than 0.1 MeV. The example of using Equation 7.6 in the topical, is associated with the B & W Owners Cavity Dosimetry Experiment. The topical discussion of measurement uncertainty (page 7 - 21, below Equation 7.7, and page 7 - 22) states that the standard deviation from Equation 7.6 is percent. "While this is a reasonable estimate for the

dosimeters in the cavity benchmark experiment, FTI considers a reasonable estimate for the entire database to be 7.0 percent or less".

The discussion in the topical continues, and explains how the database uncertainty is estimated to be 7.0 percent or less. While there are additional explanations of the statistical process, which include such pertinent details as the fact that at least 4 dosimeter - materials are grouped per set, which means that there are 143 sets in the database (with the Cavity Dosimetry Experiment consisting of 24 sets), there are no further developments of statistical equations showing the combination of the Cavity Dosimetry Experiment uncertainty with the rest of the capsule and cavity uncertainties in the database. During the meeting, it became clear that ending the development of statistical equations with Equation 7.6 in the topical caused confusion. The explanations in the topical which explained that there are 143 sets of measurements, with at least 4 dosimeter - materials per set, seemed to fit the format of Equation 7.6. Consequently, it appeared that all capsule and cavity dosimetry in the database was combined with Equation 7.6.

As noted in the discussion beginning with the second paragraph on page D - 45 in this appendix, and continuing through Equation D.19 on page D - 46, the measurement uncertainties for the capsules and cavities in the database are combined using Equation D.19. The weighting in Equation D.19 follows the same type of relations as expressed by Equations D.17 and D.18. Accordingly, the weight represents the response function. The correlation coefficient in the covariance matrix for cross product dependency parameters is represented by a null set. Thus, the uncertainty in the measurements for the entire FTI dosimetry database is determined by the square root, of the sum of the squares, of the uncertainties in each capsule and cavity measurement.

To summarize, the sensitivity of the response function relations, between the database uncertainty and each of the 728 dosimeter measurement uncertainties, are grouped into three weighting functions. The reason for the three response function weights is that the measured specific activity for neutron reactions above 0.1 MeV is determined by a combination of dosimeter - materials. The dosimeters of the same materials in a capsule or cavity are grouped into individual material uncertainties by equally weighting each dosimeter. The various dosimeter materials within a capsule or cavity are grouped into sets of four dosimeter - materials to estimate the measurement uncertainty. Each dosimeter - material set uncertainty is combined using equal material - set weighting to estimate the uncertainty in a capsule or cavity analysis. The measurement uncertainty for the entire database is estimated by combining each capsule and cavity analysis. However, if one analysis represents 24 sets of measurements of the greater than 0.1 MeV specific activity, and another analysis represents just one set, then an appropriate weighting by set is needed. Therefore, each capsule and cavity is weighted by the respective sets of dosimeter - materials that provide measurements of the greater than 0.1 MeV specific activities.

Meeting Question 5

The measurement uncertainty computed with Equation 7.6, and the benchmark uncertainty computed with Equation 7.15, do not appear to be consistent. This is particularly apparent considering that the benchmark uncertainty from Table A-1 is percent, when all dosimeters are treated independently. Explain how the statistical processing to determine the uncertainties is consistent.

Discussion

RAIs 6, 11 and 14 from Set 2, along with "Meeting Question 4" from this section (D.2.4), illustrate the range of discussions during the meeting that focused on estimating the uncertainty in the calculational methodology. As noted in the response to RAI 14 (Set 2), it is not possible to infer a vessel fluence uncertainty (including an appropriate level of confidence) without performing an analytical uncertainty evaluation. Therefore, the uncertainties in the calculations are analytically determined with a series of sensitivity evaluations that propagate design, operational and fabrication uncertainties into fluence uncertainties. The fluence uncertainties are relative values representing the deviations in the fluence relative to the unbiased nominal fluence. The unbiased nominal fluence is determined assuming a reference design, with nominal operating conditions, and fabrication values for the various parameters.

The problem with the analytically estimated uncertainties, is assessing what confidence level and probability distribution that the root mean square deviations represent. The design, operational, and fabrication uncertainties are frequently defined as limiting or bounding values, and the confirmation of their validity rarely involves more than one measurement. Consequently, while it is possible to estimate a bounding uncertainty for the fluence at the vessel, and throughout the internals, and vessel-cavity structure, it is generally not possible to specifically define the level of confidence in the bounding uncertainty, or the probability that the combination of calculated results and uncertainties bound the truth.

The fact that it is not possible to infer a vessel fluence uncertainty without an analytical uncertainty evaluation, and the fact that it is not possible for the analytical fluence uncertainty to have a well defined level of confidence, means that there must be a second statistical technique to define the level of confidence in the calculational

uncertainty.

While the meeting question referred to Equation 7.6, and it is now apparent that Equation D.19 represents the database measurement uncertainty, this does not change the point of the question. Equations D.19 and 7.15 should be consistent, but it is not apparent that they are consistent.

A large part of the questionable consistency between the measurement uncertainty (Equations 7.6 and D.19) and the benchmark uncertainty (Equation 7.15) was related to Equation 7.6. It was not clear what the measurement, and measurement uncertainty, actually represented. In addition, the cross product dependency between the individual dosimeter uncertainties being represented by a set of unity correlation coefficients, increased the ambiguity of what the uncertainty represented. The previous explanations in the sections on "Statistical Methods" (D.2.1) and "Additional Explanations" (D.2.4) for "Meeting Question 4", have described how the measurement uncertainty represents the standard deviation (7.0 percent) in the measurement of specific activities from neutron reactions with energies greater than 0.1 MeV. The topical included discussions explaining that the distribution of deviations could be represented by Student's central "t" with 142 degrees of freedom.

The previous explanations in this appendix, have cleared up quite a bit of the confusion related to Equation 7.6, and the measurement uncertainty. Thus, some of the apparent inconsistency between the measurement uncertainty and benchmark uncertainty (Equation 7.15) has also been cleared up. However, there are two important areas of

consistency that need to be explained. The first is the combination of C/M values represented by Equations 7.12 and 7.13. The second is that Equation 7.9 implies that the confidence factor and distribution of deviations are the same.

Concerning the combination of C/M values,

The product of energy dependent fluences, and constant energy group response function - cross sections, provide the specific activities for the benchmark comparison to the measurements. To obtain the measured values, at least four different dosimeter materials are combined. For the capsule or cavity dosimetry analyses, the dosimeter C/M values for each material are combined as expressed by Equation 7.12. Since the calculation represents one unique fluence analysis, and the calculated activity for each dosimeter of the same material is generally represented by one value, (even though there may be multiple dosimeters of that material), the material C/M is represented by one value.

The C/M values for each material in a capsule or cavity fluence analyses are combined as expressed by Equation 7.13. The weight, and cross product dependency for the different materials, could reflect the fluence spectrum that affects each material, and the amount of spectral overlap between the reactions in each material. However, the evaluation of the energy dependent bias function is based on the results of Equation 7.13.

Consequently, the calculated material activities are dependent on one fluence result, and are thereby dependent on one another. Likewise, with the energy dependent bias function evaluated from the results of Equation 7.13,

each material uncertainty is not unique in relation to the total uncertainty. Thus, one equal weight combines the material dependent C/M results.

While Equations 7.12, 7.13 and 7.15 represent a reasonable statistical technique for propagating errors, the benchmark standard deviation of percent, as shown by Equation 7.16, on page 7 - 33 of the topical, and computed from Equation 7.15, causes concerns. The concerns are related to the fact that the standard deviation in M/C from the combination of 728 statistically independent dosimeters is percent. Consequently, during the meeting, the NRC raised the question whether the difference is not a result of inconsistency in the statistical techniques in Equations 7.12 and 7.13.

Removing the bias from the standard deviations in the reformulated form of Equations 7.12 and 7.13, and computing the benchmark uncertainty for the database using Equation 7.15, results in an unbiased uncertainty of percent. Therefore, the form of Equations 7.12, 7.13 and 7.15 appear to be appropriate for estimating the benchmark uncertainty in the Table A-1 database of greater than 0.1 MeV specific activities.

The second area of consistency that needs to be explained concerning Equations 7.6 and D.19, and Equation 7.15, is the confidence factor and distribution of deviations. In Equation 7.15, the denominator represents 38 degrees of freedom. (This is also true of the reformulated expression discussed above in the evaluation of an percent uncertainty.) In Equation D.19, the weight function denominator represents 142 degrees of freedom. The topical suggests that both the measurement and benchmark uncertainties can be represented by Student's central "t" distribution. Accordingly, the two different degrees of freedom are inconsistent with the formulation of Equation 7.9. However, the confidence factor differences for the 38 and 142 degrees of freedom, at a 95 percent confidence level, were applied to lower the measurement uncertainty when Equation 7.9 was used

Therefore, there is consistency between the uncertainties

References for *Appendix* D

D1. United States Nuclear Regulatory Commission letter to J.J. Kelly, Manager, B & W Owners Group Services, **Request for Additional Information for Topical BAW-2241P**, from Joseph L. Birmingham, Project Manager, Office of Nuclear Reactor Regulation, January 30, 1998.

D2. United States Nuclear Regulatory Commission letter to J.J. Kelly, Manager, B & W Owners Group Services, **Request for Additional Information for Topical BAW-2241P**, from Joseph L. Birmingham, Project Manager, Office of Nuclear Reactor Regulation, April 8, 1998.

D3. Office of Nuclear Regulatory Research, **"Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence",** Draft Regulatory Guide, DG-1053, (page 1) United States Nuclear Regulatory Commission, June, 1996.

D4. Office of Nuclear Regulatory Research, **"Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence",** Draft Regulatory Guide, DG-1053, (page 4) United States Nuclear Regulatory Commission, June, 1996.

D5. T. G. Williamson, "Evaluation of the Photofission Effect in Pressure Vessel Dosimetry", UVA/532886/NEEP88/101CN, University of Virginia School of Engineering and Applied Science, Charlottesville, Va., 22901, 1988.

D6. J. T. Caldwell, et al, "Photonuclear Measurements on Fissionable Isotopes using Monoenergetic Photons", LA-UR 76-161J, Los Alamos Scientific Laboratory, 1976.

D7. L. Petrusha, "Photofission Effects in B&W 177 FA Reactor Vessel Surveillance Capsule Dosimeters", *Proceedings of the Seventh ASTM-EURATOM Symposium on Reactor Dosimetry*, Kluwer Academic Publishers, P. O. Box 17, 3300 AA Dordrecht, the Netherlands.

D8. Karel Rektorys, Editor, *Survey of Applicable Mathematics*, The M.I.T. Press, Massachusetts Institute of Technology, Cambridge, Massachusetts, 1969.

D9. Paul L. Meyer, *Introductory Probability And Statistical Applications*, Second Edition, Addison-Wesley Publishing Company, Reading, Massachusetts, 1970.

Appendix E Generic PWR Uncertainties

The purpose of this appendix is to update the uncertainties in this topical, BAW-2241P-A, "Fluence and Uncertainty Methodologies", that are associated with Westinghouse and Combustion Engineering (CE) fluence calculations. The update consists of reevaluating the benchmarks in FTI's dosimetry database. Equal weights are applied to each Pressurized Water Reactor (PWR) type: Westinghouse, CE, PCA test - reactor, and B & W.

This update was developed after the safety evaluation to the original publication of the topical was issued. The updated documentation is therefore presented as Revision 1 to the topical. The format for this revision is the original publication (now Volume 1, Revision 1) followed by this volume (2). The updates to the original publication only include the "Title" page, "Record of Revisions" page, and the pages with the "Table of Contents". Because the document is quite lengthy, the topical has been published in two volumes. Volume 1 contains the documentation from the original topical. This volume (2) is focused on responses to the NRC questions and contains the generic PWR update to the uncertainties. The two volumes together represent Revision 1 to the topical.

The reason for Revision 1 is to extend the application of uncertainties to all reactors of the pressurized water type. When the United States Nuclear Regulatory Commission (NRC) published the safety evaluation for the original version of this topical, they noted that the application of the methodology was limited to B & W (a McDermott company) designed reactors. As explained in the following section (Introduction and Background), the focus of the NRC's limitation was a concern with the industry's

database of benchmark uncertainties from non - B & W designed PWRs, such as those designed by Westinghouse and CE.

The NRC explained that, if fluence analysts expect to apply their results to Westinghouse, CE, and B & W designed PWRs, then they need to have an adequate database for each respective reactor type. The adequate database consists of multiple benchmark comparisons of the results from the calculational methodology to appropriate dosimetry results from the measurement methodology. FTI agrees with the concept that analysts need multiple benchmark comparisons to each PWR type that they intend to analyze for fluence - embrittlement evaluations. The FTI database in this topical consists of 728 dosimeters responding to neutron reactions above 0.1 MeV. These 728 dosimeters come from 39 capsules and cavities. These 39 capsules and cavities are from 5 Westinghouse, 5 CE, 2 PCA, and 23 B & W capsule evaluations and 4 B & W cavity evaluations.

The uncertainty evaluation of the calculational methodology has indicated that the functional and correlated dependencies of the biases have been appropriately assessed. The result of the evaluation is that the best-estimate fluence from FTI's calculational methodology is unbiased throughout the beltline region, including the reactor internals, vessel, and vessel cavity structure. The uncertainty evaluation of the calculational methodology has also indicated that the precision in the best-estimate fluences is consistent with the embrittlement "Margin" terms from (a) the Pressurized Thermal Shock (PTS) Safety Analyses,[3,4,5] and (b) Regulatory Guide 1.99, Revision 2.[17] Statistical evaluations of the fluence uncertainties ensure that there is a 95 percent probability that the embrittlement "Margin" term will appropriately bound the vessel embrittlement evaluations with a value of ± 2.000 for the confidence factor. Table 7-6,

on page 7 - 39 of this topical, gives the respective fluence uncertainties. This table is repeated below and noted as Table E-1.

Table E-1

Calculational Fluence Uncertainties

For B & W Designed PWRs

	Uncertainty %	
Type of Calculation	Standard Deviation	95 % / 95 % Confidence
Dosimetry (Capsule)	7.00	
Pressure Vessel		
Pressure Vessel (Extrapolated in Time)		

Sections E.1 through E.3.2 explain that the data set samples from Westinghouse and CE plants can be represented by the population of the FTI benchmark database. Thus, Table E-1 above would provide appropriate uncertainties for all PWRs.

In Section E.4 however, it is noted that even though the evaluations clearly indicate that (a) the Table E-1 uncertainties are applicable to Westinghouse and CE reactors, and (b) the benchmark standard deviation is percent for any PWR, there is the possibility that the uncertainties in the calculations are mostly dependent on plant uncertainties The plant data may be too sparse for statistical evaluations to adequately detect this possibility. Therefore, the margin of safety for generic PWR fluence uncertainties has been reevaluated on a plant basis. The statistical results are shown in Table E-2. These fluence uncertainties are applicable to any PWR.

Table E-2

Calculational Fluence Uncertainties

For All PWRs

	Uncertainty %	
Type of Calculation	Standard Deviation	95 % / 95 % Confidence
Dosimetry (Capsule)		
Pressure Vessel		
Pressure Vessel (Extrapolated in Time)		

E.1 Introduction and Background

In February of 1999, the NRC staff published the safety evaluation for the original version of this topical. The "Summary and Limitations" section of the safety evaluation concluded that the methodology is acceptable for determining the pressure vessel fluence of B & W designed reactors. Also noted, was the specific limitation that the methodology is applicable only to B & W designed reactors.

The topical presents two methodologies, one for determining the fluence and the other for estimating the uncertainty in the methodology for determining the fluence. The fluence and the uncertainty methodologies developed in the topical are fundamentally theoretical methods, combined with procedural and modeling approximations. The theoretical methods are generic, and the procedures and models are generic to PWR designs. Thus, the methodologies are applicable to any PWR. Consequently, the limitation of the methodology to B & W reactors in the "Summary and Limitations" section was confusing.

To clarify the confusion, discussions were held with the NRC staff. The discussions began by reviewing the methodologies that the NRC contractors and the industry have used for fluence and uncertainty evaluations. The methodologies fall into one of two categories; (1) those based on unfolding a measured fluence with a measurement-based uncertainty, and (2) those based on calculating the fluence with a calculational-based uncertainty. FTI used a measurement-based methodology for 20 years, and the other industry vendors continue to use it today. This methodology provides excellent techniques for determining the fluence values at capsule and cavity dosimetry locations. However, in 1993 the NRC held a meeting with the industry that focused on the

consistency between vessel fluence uncertainties and the fluence uncertainties associated with the PTS rule, 10 CFR 50.61.[6] When the PTS safety analyses[3, 4, 5] are reviewed, it is clear that the fluence uncertainty must be consistent with a 95 percent probability that the vessel fluence value bounds the true value. In the months following the meeting, the NRC published draft regulatory guide DG-1025[8] (updated in 1996 to DG-1053[19]) describing "Calculational And Dosimetry Methods For Determining Pressure Vessel Neutron Fluence". The draft regulatory guide notes that measurement-based fluence predictions are not consistent with the PTS safety analyses; only calculational-based fluence predictions are consistent.

FTI has explained to numerous utilities that without vessel dosimetry measurements, it is very difficult to show that there is a 95 percent probability that the "measured" vessel fluence bounds the true value. Consequently, FTI tailored the fluence and uncertainty methodologies in this topical to closely follow the draft regulatory guide.[19] Thus, the topical presents a calculational-based methodology that is consistent with the uncertainty "Margin" assumed in the PTS safety analyses.[3, 4, 5]

FTI has utility customers with Westinghouse designed reactors, and with Combustion Engineering (CE) designed reactors. These utilities have agreed that when evaluating vessel embrittlement for either the PTS rule (10 CFR 50.61),[6] or the (Regulatory Guide 1.99, Revision 2)[17] technical specification limits for pressure - temperature values during heat-ups and cool-downs, it is important for the fluence to be consistent with the embrittlement "Margin" term uncertainties.

The NRC staff has agreed that it would be preferable to utilize the BAW-2241P-A calculational-based methodologies on all PWRs, including those designed by Westinghouse and CE. However, they have noted that while FTI's "Fluence and

Uncertainty Methodologies" are applicable to any PWR, there are technical issues associated with industry analyses of plant-specific uncertainties that need to be addressed. One company, that performs a significant number of non - B & W fluence analyses using a measurement-based methodology, consistently produces biases, with uncertainties between 10 and 25 percent. The FTI best-estimate fluence methodology produces unbiased results with an uncertainty of 9.9 percent as explained in the topical. Therefore, the NRC staff requested that the Westinghouse and CE analyses, that are part of the FTI dosimetry database, be specifically evaluated as a function of plant type to determine if consistent biases or large random uncertainties are evident. The staff noted that the FTI database is weighted with more B & W plants (27 capsules and cavities out of 39, or 69 percent B & W analyses). Furthermore, the B & W plants are weighted with more Crystal River, Unit-3, and Davis Besse, Unit-1 analyses (20 out of 27, or 74 percent). Thus, they requested that the statistical evaluation of the database be reviewed to verify that the data set samples from Westinghouse and CE plants are appropriately represented by the population of 728 dosimetry benchmarks in 39 capsules and cavities.

As part of the verification process, the NRC staff requested that the review of the data by plant type include:

1 - A description of the important physical parameters and characteristics affecting the uncertainties, with discussions explaining why differences between plant types do not result in the data representing different populations.

2 - An evaluation of the data, with discussions explaining why it represents an adequate set for estimating statistical properties.

3 - An evaluation substantiating that the statistical treatment of the data with the uncertainty methodology is appropriate to estimate the uncertainties.

The focus of the NRC's question concerning the application of Table E-1 to Westinghouse, CE, and other non - B & W PWRs, is associated with the benchmark uncertainties of calculations (C) to measurements (M). Thus, the focus of this appendix is the reevaluation of the benchmark uncertainties from Table A-2 (page A - 25 in this topical). As shown by Equations 7.8 and 7.9 on page 7 - 26, the benchmark bias ($B_{C/M}$) and relative variance ($\sigma^2_{C/M}$) are determined from the

The reevaluation of the measurement and benchmark, biases and standard deviations, is based on the Table A-2 database. As previously noted, this database includes 5 capsules from 5 Westinghouse plants, and 5 capsules from 4 CE plants. The data samples from Westinghouse and CE plants have been independently evaluated. This independent evaluation addresses the crux of the NRC's concern with the uncertainties in Table E-1 being applied to other PWRs. In conversations with the

staff, they noted that there is a large inconsistency between Westinghouse benchmark uncertainties and FTI benchmark uncertainties for Westinghouse designed reactors.

E.2 Measurement Uncertainties

Westinghouse uses a measurement-based unfolding methodology to evaluate capsule fluences. In 1994, they updated the benchmark evaluation of their entire capsule dosimetry database.[E1] Their reported overall uncertainty is 22.4 percent, with 12.1 percent in the form of a mean bias, and 10.3 percent in the form of a mean standard deviation. Several of the plants reported in the reference[E1] are also in the FTI dosimetry database. The overall FTI benchmark uncertainty is 9.9 percent, with no bias, and the total uncertainty in the form of a root mean square standard deviation. The large difference in uncertainties could be the result of the weighting of B & W plants in the FTI database. This section examines the measurement uncertainties for Westinghouse and CE plants, and discusses the three issues in the verification process that the NRC requested (page E - 7).

The measurement-based methodology that Westinghouse uses includes dosimeter activities in the same manner as CE and FTI. The activities measure the fluence - dosimeter reaction rate effects that are related to the fluence, but there is no measure of the fluence. Westinghouse, CE and FTI use the same techniques to evaluate biases in the measured activities (see pages 7 - 9 and 7 - 10 in this topical). While CE and FTI use the combination of unbiased activities and cross sections to assess any biased measurements of the fluence, Westinghouse includes unfolding techniques to actually evaluate the fluence. As noted in the FTI paper on "Biased Fluences In The Charpy Embrittlement Database" (given at the same conference as the Westinghouse update of capsule fluence evaluations[E1]), unfolding methodologies, such as FERRET-SAND,

have previously introduced biases into the measured fluences. The following discussion reviews FTI's evaluation of "Measurement Biases".

E.2.1 Measurement Biases

When FTI developed the calculational-based uncertainty methodology, an important step in the development was the evaluation of the uncertainty differences between the measurement-based methodology and the new calculational-based methodology. The reason for the evaluation is that the correlations of embrittlement are from a database that is based on measured specimen fluences. While it was clear that calculated vessel fluences would be a significant improvement over "measured" vessel fluences, it was not clear how the new calculational methodology would be consistent with the measured fluences in the existing capsule embrittlement database.

Since embrittlement evaluations (PTS[6] and Regulatory Guide 1.99, Revision 2[17]) are based on correlations of the change in the material specimen properties to the specimen measured fluences, it is apparent that calculated fluences must be equivalent to the measured ones as expressed below.

Capsule Embrittlement Database Criterion –

$$\begin{aligned} &\textit{Measured Fluence} - \sigma_M(\textit{Fluence}) \leq \\ &\textit{Calculated Fluence} \leq \textit{Measured Fluence} + \sigma_M(\textit{Fluence}) \end{aligned} \tag{E.1}$$

where

σ_M is the standard deviation of the uncertainties (random deviations) in the measurements.

Calculations of capsule specimen fluences, using a methodology consistent with the draft regulatory guide,[19] must equal the measured specimen fluences determined for the embrittlement database (in the 1970's), within an uncertainty range that is equal or less than the uncertainty in the measurements. From Equation E.1, the standard deviation (σ) in the capsule fluences predicted by calculations (C) must be equal or less than the standard deviation of the measurement (M) predictions as expressed by Equation E.2 .

$$\sigma_C \leq \sigma_M \tag{E.2}$$

In the 1970's, FTI (then Babcock & Wilcox {B & W}) provided embrittlement and fluence measurements from capsule specimens to NRC contractors Simons[15] and Guthrie.[16] This data help establish the database for correlations of embrittlement to fluence. Guthrie performed the correlations of embrittlement properties, and Simons used FERRET-SAND to adjust the fluence values from Westinghouse, CE, and B & W capsules to provide Guthrie with fluences that were consistent with one another. Therefore, when FTI performed evaluations to determine if the new calculational methodology would provide fluence values equal to those of the 1970's, it was Simons' values that were used for the measured fluence comparisons.

The comparisons of the calculated fluences to Simons' measured ones gave very disappointing results. The differences between the values were much larger than anticipated, and consistently outside the range of the appropriate uncertainty. As indicated by Equation E.3, if the calculated fluence values were increased by a multiplicative bias factor (of approximately 12 percent), the differences were reduced and were within the acceptable range of the measurement uncertainty (σ_M).

$$\textit{Calculated Fluence}\left(1 + \textit{Bias}\right) = \textit{Measured Fluence} \pm \sigma_M(\textit{Fluence}) \tag{E.3}$$

To better understand the calculational bias, the calculated and measured activities were compared. Surprisingly, the calculated and measured activities compared very well, with no evidence of a bias, as shown by Equation E.4 .

$$\textit{Calculated Activities} = \textit{Measured Activities} \pm \sigma_M(\textit{Activities}) \tag{E.4}$$

Pursuing the explanation for the bias, the calculated and original FTI measured fluence values were found to be in agreement and showed no bias. Reviewing Simons' FERRET-SAND adjusted fluence results as shown below, the adjustments were found to produce biases relative to the original predictions from all capsules {Westinghouse, CE, and B & W}.[15]

$$\textit{FERRET - SAND Fluence Biases} = \begin{Bmatrix} 1.35 & \textit{Westinghouse} \\ 1.23 & \textit{CE} \\ 1.12 & \textit{B \& W} \end{Bmatrix} = 1.30\ \textit{Average} \tag{E.5}$$

Capsules — Database

The B & W calculated and measured capsule fluence values for the embrittlement database had a 12 percent bias compared to Simons' measurement predictions. In addition to the B & W bias, it was found that the ABB-CE capsules had a 23 percent bias, the Westinghouse ones had a 35 percent bias, and the weighting of the biases produced an overall 30 percent increase in the fluence values that Guthrie used for the embrittlement correlation. FTI found that the biases in the embrittlement database fluences were caused by the FERRET-SAND adjustment techniques. These biases are

not real with respect to the theoretical models that are the bases for the calculations, nor are they real with respect to the experimental techniques that are the bases for the measured dosimeter activities. They are simply associated with biased unfolding methods and procedures in FERRET-SAND.

If the FERRET-SAND results are biased, there should be others who have also observed the biases. Reference 37 (in Section 8 of this topical) provides a comparison of the "PCA Blind Test" results from FERRET-SAND with those from the LSL-M2 predecessor. As indicated by Equation E.5, the FERRET-SAND fluence results should have been, and were higher than those determined by the LSL-M2 unfolding methods and procedures. The FERRET-SAND bias was confirmed by the Hanford laboratory, three years after the publication of Reference 37, when they revised the FERRET-SAND fluences to agree with the Oak Ridge laboratory LSL-M2 values.

In Reference E2, three senior scientists from Germany presented a paper that evaluates "Neutron Fluence Determination at Reactor Filters by ^{3}He Proportional Counters: Comparison of Unfolding". They stated that an unknown neutron spectrum in an iron filtered reactor beam was unfolded using the SAND-II iteration algorithm, and the appropriate response functions and covariance matrix. However, the scientists noted that as a consequence of the solution technique, the results reached by the SAND-II iteration may not be unique. Biases (systematic uncertainties) may arise in the spectrum. If the solution is not unique, then the SAND-II solution process is not valid. There is only one unique and valid flux spectrum at the reactor beam detector location.

These three unrelated incidences:

(1) the FTI review of the FERRET-SAND adjustments to the industry fluences,

(2) the comparison of PCA results between Hanford using FERRET-SAND, and Oak Ridge using LSL-M2, and

(3) the German scientists finding that the SAND-II iteration may not be unique;

indicate that measured fluence results are frequently biased due to unfolding methods and procedures. As noted by Equation E.5, FERRET-SAND increases the measured fluence. Consequently, when calculated fluences are compared to measured values, the resulting mean C/M benchmark value would be less than unity. The C/M benchmark values in Reference E1 are noted to be less than unity.

When FTI processes Westinghouse and CE dosimeters to measure the activities, the experimental methodology is the same as that used for the B & W dosimetry, the NIST reference field dosimetry,[E3] and any other dosimetry. There are 141 dosimeters from Westinghouse and CE reactor capsules in the FTI database (Table A-1, pages A - 3 through A - 19). The five components of the measurement uncertainties listed in Table 7-1 on page 7 - 12 are exactly the same, regardless of where the dosimeters were irradiated. The evaluation of measurement biases uses the same National Institute of Standards and Technology (NIST) traceable standards and laboratory calibration procedures for all dosimetry, as described on pages 7 - 9 and 7 - 10. Consequently, neither the Westinghouse, nor CE, nor any dosimetry measurements of the specific activities are biased. No measured fluence values are evaluated in the topical database.

Concerning the three issues that the NRC requested be reviewed:

1. The important physical parameters and characteristics affecting the measurement biases are the NIST traceable calibration standards and the experimental

procedures. As discussed above, Westinghouse measures the fluence with dosimeter reaction rates and unfolding techniques. CE measures the fluence with dosimeter activities and a normalization of calculations to the mean measured specific activity. FTI does not measure the fluence, only the dosimeter specific activities are measured.

The "measurement" of the fluence requires processing measured dosimetry results with an analytical technique. Due to differences in the operational, fabrication, and design characteristics of each type of plant, the analytical technique to determine measured fluences can be a function of the plant type. The NRC's draft regulatory guide[19] recommends testing the fluence measurement methodology with a reference field standard. However, in 1994, when scientists and engineers from Westinghouse, CE, FTI, and the industry met to discuss the implications of the draft guide, no one had used reference field fluence standards for calibrating their fluence measurements. Consequently, the fluence measurements from Westinghouse, CE, FTI, and the industry may be biased.

As noted in the original version of this topical, the basis for evaluating uncertainties must be benchmark comparisons of the results from the experimental methodology to a reference standard that is known to be unbiased. The principal technique used by experimentalist to ensure that their measured results have no biases, is calibrating the methodology to certified standards referable to NIST. Without a NIST reference field fluence standard, the industry's measured fluences cannot be certified to be unbiased.

Unlike fluence measurements, which can be dependent on plant type, and for which there are no calibrations to NIST standards, dosimetry measurements have no functional dependency on reactor plant type, and there are dosimeter activity standards directly referable to NIST. The topical explains how these standards are used in the experimental calibration process to ensure that the specific activities from each dosimeter measurement are not biased. The FTI database of dosimeter activities from Westinghouse and CE plants is part of a population of unbiased measurements.

2. The evaluations of 141 dosimeters from Westinghouse and CE plants that are used to determine the measurement biases, represents a sufficiently adequate data set. As noted above, the physical parameters and characteristics of the dosimeters, the irradiation source, and the experimental process have no related dependencies. Therefore, the data represents an independent set of 141 samples. Such a set is adequate for estimating the biases and standard deviations in the data, and is adequate as an independent sample for estimating the FTI dosimetry database population biases and standard deviations.

3. The experimental methodology that is used to evaluate the dosimetry measurements in the FTI database has been validated by NIST.[E3] This validation substantiated the statistical treatment of the bias with the calibration procedures. The experimental methodology is generic and is not dependent on the dosimetry parameters or characteristics. Thus, there are no measurement biases in the dosimetry database, and the Westinghouse and CE data represent samples from the FTI database population.

E.2.2 Measurement Standard Deviation

As discussed in the "Uncertainty Methodology" section in this topical, the uncertainties in the measurements (and calculations) arise from two types of deviations, systematic and random. The systematic deviations are caused by some fundamental problem with the predictive methodology and are thereby functionally related to some variable or parameter. If there is a single functional relation between the systematic deviations and some variable, then there is a single bias. If there are two or more functional relations, then there may be two biases, or multiple biases. If there are two or more biases associated with the data, then it is not appropriate to use the techniques of mathematical statistics to estimate the standard deviation in the data. If there are no biases, or only one bias, then the techniques of mathematical statistics are appropriate.

Since there are no biases in the dosimetry measurements of the FTI database, the random deviations in the experimental process are determined from the component uncertainties listed in Table 7-1 on page 7 - 12 of the topical. These deviations are only dependent on the random variables in the experimental process and are therefore independent of where the dosimeters were irradiated. Consequently, Westinghouse and CE data samples from the FTI dosimeter database have the same standard deviation as the general population, 7.0 percent.

Concerning the three issues that the NRC requested be reviewed:

1 - The physical parameters and characteristics affecting the measurement standard deviation are the five parameters and experimental procedures listed in Table 7-1 on page 7 - 12 of the topical. The experimental procedures and parameters are independent of the plant where the dosimeters were irradiated.

Therefore, the standard deviation estimated for each dosimeter measurement is independent of the sample and is based on the database population.

2 -

The standard deviations estimated for the 141 dosimeters have no unique properties that distinguish them from any other sampling of dosimetry in the FTI database population. Consequently, the Westinghouse and CE dosimeter measurements represent an adequate data set for statistically estimating the standard deviation.

3 - The measurement uncertainty methodology that is used to statistically evaluate the dosimetry data from Westinghouse and CE plants is the same as that used for any dosimeter measurement. This methodology has been validated by NIST.[E3] NIST concluded that the accuracy and precision that the B & W laboratory has estimated for the measurement uncertainties are valid values. Thus, the measurements have no statistically significant biases, and the methodology for estimating the standard deviation from the component uncertainties in the experimental process is valid. The NIST validation thereby substantiates the statistical treatment of the Westinghouse and CE dosimeter measurements.

E.3 Benchmark Uncertainties

As discussed previously, the crux of the NRC's concern with the uncertainties in Table E-1 being applied to PWRs, other than ones designed by B & W, is that Westinghouse has reported benchmark uncertainties with biases of 25.2 percent, a mean bias of 12.1 percent, and a mean standard deviation of 10.3 percent.[E1] However, 5 of the plants that are in the Westinghouse benchmarks of Westinghouse designed plants, are also in the FTI benchmark database. While the Westinghouse benchmark suggests that the Prairie Island plant has a C/M bias of 25.2 percent, the FTI benchmark indicates no bias. Furthermore, the FTI benchmark database indicates that all FTI fluence analyses of Westinghouse plant capsules have no bias, and have an uncertainty represented by a root mean square standard deviation

It would be expected that the biases and standard deviations in the benchmark comparison of calculations and measurements would be similar between Westinghouse, CE, and FTI. However, as noted in the discussion above, the Westinghouse mean benchmark bias of 12.1 percent, indicates an inconsistency with the FTI calculational and measurement methodology, which has no bias in the benchmark database. The bias inconsistency accentuates the inconsistency in the overall uncertainty estimated by Westinghouse and FTI. FTI's overall benchmark uncertainty is just the root mean square standard deviation from the capsule and cavity analyses, percent. Since Westinghouse uses the calculated fluence spectrum as the "a priori" spectrum for unfolding the measured fluence, and they do not incorporate a bias removal function, the overall uncertainty is the statistical combination of the bias (12.1 percent[E1]) and the standard deviation (10.3 percent[E1]) with correlation coefficients of unity. Thus, the overall uncertainty from the Westinghouse capsule benchmark database is 22.4 percent.

The NRC would like FTI to isolate the Westinghouse and CE plants, and statistically process the FTI benchmark data by plant type. The statistical processing is to ensure that no biases are associated with Westinghouse or CE plants, and that the fluence uncertainty can be appropriately represented by a root mean square standard deviation. As explained in the "Introduction and Background" section, with 69 percent of the FTI benchmark database weighted with B & W plants, the NRC wants to know whether the large B & W weighting disguises differences in the Westinghouse and CE plant uncertainties? In statistical terms, the question is whether the Westinghouse and CE data is unique or does it represent samples from the same population? There is a corollary to the question: If the uncertainties are plant dependent, does the statistical evaluation associated with the uncertainty methodology sufficiently estimate the standard deviation? The following discussions address the benchmark biases and

standard deviations associated with Westinghouse and CE plants in relation to the FTI benchmark database.

E.3.1 Benchmark Biases

The benchmark biases for the FTI database are discussed in Section 7.2.1 on pages 7 - 27 and 7 - 28. The expression used to estimate the bias is Equation 7.10. To determine the benchmark C/M value for each PWR plant type, the C/M value is determined for each dosimeter within a capsule.

Table E-3 gives the C/M values for the 5 Westinghouse plants in the FTI dosimetry database, and Table E-4 gives the values for the 4 CE plants. Using Equation 7.10 to compute the respective biases for the Westinghouse and CE plants, and statistically estimating the values, shows that there are no statistically significant biases associated with either the Westinghouse plant samples or the CE plant samples.

Concerning the three issues that the NRC requested be reviewed:

1 - The physical parameters and characteristics that affect the benchmark bias are those associated with the calculations since the measurements have previously been reviewed in Section E.2.1. The calculational methodology is discussed in Section 3 of the topical. There is nothing associated with the BUGLE-93 cross

sections, or the DORT model that would bias the results with respect to plant type. The methods and procedures used in the modeling of the various plant types are not unique or sensitive to any physical parameter or characteristic that differentiates one plant from another. Therefore, as noted in Tables E-3 and E-4, the mean C/M benchmark values for Westinghouse and CE plants show the same statistical traits as the mean C/M benchmark value for the FTI database. The relation between (a) the unbiased standard deviation in the data, and (b) the difference between the mean C/M values and unity, indicates that the benchmark deviations are of a random nature. No statistically significant biases are evident.

The C/M benchmark results reported in Reference E1 show large biases. The 12.1 percent mean bias[E1] is inconsistent with the FTI results in Table E-3, which have a 0.0 mean bias.

As the NRC staff knows, the Virginia Power Corporation has an expert in the field of fluence analyses. Virginia Power developed their own independent analytical methodologies for discrete ordinates, and Monte Carlo modeling of the North Anna (Units 1 and 2) and Surry (Units 1 and 2) reactors. To help

substantiate the fact that calculations of Westinghouse reactors are not inherently biased, FTI requested Virginia Power send their discrete ordinates dosimetry calculations and the measurements. (The measurements only involved the activation of the dosimetry, no measured fluences were predicted from unfolding techniques.)

It was also requested that FTI be allowed to statistically process the dosimetry C/M benchmarks using the uncertainty methodology in the topical. The Virginia Power benchmark comparisons of dosimetry calculations to measurements consisted of 42 dosimeters in 9 capsules from both North Anna units, and both Surry units. The evaluation of the mean bias and standard deviation are shown in Table E-5 (page E - 28). The Virginia Power benchmark results are consistent with those from FTI. They indicate that no statistically significant bias can be observed in the benchmark data. This of course is inconsistent with the Westinghouse results in Reference E1.

2 - The benchmark data for the Westinghouse and CE plants includes a combination of 141 dosimeter comparisons. The Westinghouse data consists of 63 dosimeters, and the CE data, 78 dosimeters. For the data to represent adequate sets for estimating the biases, it must be sufficiently normal.

The distribution of deviations in the FTI dosimetry benchmark database were shown to adequately fit within William Sealy Gosset's (Student's) central "t" distribution (pages 7 – 33 and 7 - 34). The key criterion was that 95 percent of

Thus, the CE plant C/M benchmark deviations appropriately fit within the central "t" distribution.

Thus, the Westinghouse C/M benchmark deviations appropriately fit within the central "t" distribution.

the data needs to be separated by physical parameters and characteristics to evaluate the functional and correlative dependencies on the respective variables.

3 -

The fact

substantiates the conclusion that neither data sample has a statistically significant bias.

Table E-3

Westinghouse Plant Benchmarks

Plant	C/M
Prairie Island, Unit 1	
North Anna, Unit 1	
North Anna, Unit 2	
Shearon Harris, Unit 1	
Zion, Unit 1	

Sample Statistics

Parameter	Value
Mean C/M	
Bias	0.0
$\overline{\sigma_{C/M}}$ (Sample)	
$\overline{\sigma_{C/M}}$ (Population)	

Table E-4

C E Plant Benchmarks

Plant	C/M
Calvert Cliffs, Unit 2	
Millstone, Unit 2	
St. Lucie, Unit 2	
Waterford, Unit 3	

Sample Statistics

Parameter	Value
Mean C/M	
Bias	0.0
$\overline{\sigma_{C/M}}$ (Sample)	
$\overline{\sigma_{C/M}}$ (Population)	

Table E-5

Virginia Power Combined Statistics
From North Anna, Units 1 & 2
And Surry, Units 1 & 2

Parameter	Value
Mean C/M	
Bias	0.0
$\overline{\sigma_{C/M}}$	7.46 %

E.3.2 **Benchmark Standard Deviations**

The benchmark standard deviation for the FTI database is discussed in Section 7.2.2 on pages 7 - 32 and 7 - 33. One expression that is used to estimate the standard deviation is Equation 7.15. Without any biases in the data, this expression is appropriate for estimating the standard deviation in the database population listed in Table A-2 (pages A - 25 and A - 26). It is also appropriate for estimating the standard deviation in the isolated samples from Westinghouse and CE plants. Table E-3 (page E - 26) and Table E-4 (page E - 27) provide the respective sample statistics for the Westinghouse and CE plants.

The weighting of the capsule benchmark data by plant significantly reduces the number of data points by plant type. The 63 Westinghouse dosimeter benchmarks are reduced to 5 independent plants as shown in Table E-3. Thus, the estimate of the benchmark standard deviation is based on 4 degrees of freedom (DF).

The 78 CE dosimeter benchmarks are reduced to 4 independent plants as shown in Table E-4. Thus, the estimate of the benchmark standard deviation is based on 3 degrees of freedom (DF).

$$\chi^2 = \sum_i^k \frac{\left(Observed_i - Expected_i\right)^2}{Expected_i} \tag{E.6}$$

$$\frac{\sigma_W}{\chi_{\{.9|4\}}} \leq \sigma_{DD} \leq \frac{\sigma_W}{\chi_{\{.1|4\}}} \tag{E.7}$$

Concerning the three issues that the NRC requested be reviewed:

1 - The physical parameters and characteristics that affect the benchmark standard deviations are those associated with the calculations. The measurements used by FTI have no biases and the standard deviations have been validated by NIST as noted in Section E.2.2. The calculational modeling of Westinghouse, CE, B & W, and other PWRs is affected by the uncertainties associated with (1) the fuel rod fission sources, (2) the design and fabrication tolerances for the fuel, internals, and vessel, and (3) the operational characteristics of the reactor. These uncertainties could increase the calculational uncertainties in one plant type versus another. If the calculational uncertainties are increased, the calculation to measurement benchmark uncertainties will be increased in direct proportion.

FTI is the fabricator of many Westinghouse vessels. Deviations in fabrication specifications are thereby very consistent. The significant operational characteristics that affect fluence predictions are the downcomer inlet temperatures and the former region temperatures. FTI has performed the conversion work on several Westinghouse plants for former region flow. In addition, the sensitivities of the control system with respect to inlet temperatures in the Westinghouse plants for which FTI is responsible for the reload licensing have been reviewed.

2 - Evaluations of the Westinghouse and CE plant data, which show that it represents adequate sets for estimating standard deviations, was discussed in Section E.3.1 for the benchmark bias evaluation. As noted in that discussion, both the Westinghouse and CE plant deviations in the benchmark data appropriately fit within the central "t" distribution. The discussion above, concerning the Westinghouse and CE standard deviations being representative of a sampling from the FTI benchmark database population, also indicates that the plant data represents adequate sets for estimating statistical properties.

3 -

Thus, the statistical and physical evidence substantiates the treatment of Westinghouse and CE uncertainties with the statistical properties that have been estimated for the FTI database.

E.4 Plant Dependent Benchmark Uncertainties

The above assessment in Sections E.1 through E.3.2, is sufficient to present the conclusion to this appendix. The uncertainty methodology and statistical evaluation of the benchmark database have been reviewed. The review verifies that data set samples from Westinghouse and CE plants can be represented by the population of the FTI benchmark database. The large inconsistency between Westinghouse dosimetry benchmark uncertainties, and FTI benchmark uncertainties for Westinghouse designed reactors, has been explained. The key difference is the biased results that Westinghouse shows in the "measured" fluences in Reference E1. Biases in unfolding techniques, such as FERRET-SAND, have been observed by (1) Oak Ridge unfolding with LSL-M2, (2) German scientist finding that the SAND-II iteration process may not be unique, and (3) Virginia Power benchmark uncertainties. However a NIST reference field was used to validate that the FTI measurement uncertainties are unbiased. Consequently, the unbiased FTI uncertainties in Table E-1 are applicable to Westinghouse, CE, and B & W reactor plants, and any similar PWR or test - reactor.

The problem with the above conclusion is the combination of statistical inference and safety analyses. Even though the evaluations clearly indicate that, (a) the Table E-1 uncertainties are applicable to Westinghouse and CE reactors, and (b) the benchmark standard deviation is percent for any plant, there is the possibility that the uncertainties in the calculations, and thereby the benchmark uncertainties, are mostly dependent on plant uncertainties associated with the fuel, internals, vessel, and operation. If the uncertainties are unbiased random variables, but are plant dependent, then the statistical properties need to be evaluated on a plant basis.

The fluence uncertainties in Table E-1 are associated with the Table A-2 benchmark database of capsule and cavity uncertainties. To increase the margin of safety associated with any PWR fluence calculation, FTI has reevaluated the benchmark database uncertainties using a plant basis. The statistical results are shown in Table E-6 on page E - 35.

Table E-6

Statistical Combination of Plants

Plants	C/M	$\sigma_{C/M}$
Westinghouse		
CE		
PCA		
B & W		

Combined Statistics

Parameter	Value
Mean C/M	
Bias	0.0
$\overline{\sigma_{C/M}}$	

$$B_C\,(Fluence) \;=\; 0.0 \tag{E.8}$$

$$\sigma_C\,(Dosimetry\ \ Fluence) \;\leq \tag{E.9}$$

$$\overline{\sigma_{C/M}}\;\;(Plant\ \ Benchmark\ \ Database) \;\leq \tag{E.10}$$

These confidence factors are appropriate for a 95 percent confidence level.

$$\sigma_C\left(\textit{Vessel Fluence}\right) \leq \qquad \text{(E.11)}$$

Section 7.2 on pages 7 - 36 through 7 - 41 explains that there are additional sets of analytical uncertainties associated with the vessel fluence. The first set is related to the analytical evaluations of the source, design, fabrication, and operational uncertainties. Combining these uncertainties with Equation E.9, gives the vessel fluence uncertainty as shown by Equation E.11. The second set of additional uncertainties is associated with the source uncertainties when extrapolated over time. Combining these source - time related uncertainties with Equation E.11, gives the EOL vessel fluence uncertainty as shown by Equation E.12. While Equation E.12 defines an EOL uncertainty, this value is only valid with appropriate fluence monitoring evaluations.

$$\sigma_C\left(\textit{EOL Vessel Fluence}\right) \leq \qquad \text{(E.12)}$$

These results are summarized in Table E-2 on page E - 4.

Appendix E References

E1. E.P. Lippincott and S.L. Anderson, "Systematic Evaluation of Surveillance Capsule Data", Proceedings of the 9'th International Symposium on *Reactor Dosimetry*, Czech Republic, September, 1996, World Scientific Publishing Co., River Edge, NJ.

E2. M. Matzke, W.G. Alberts, and E. Dietz, "Neutron Fluence Determination at Reactor Filters by ^{3}He Proportional Counters: Comparison of Unfolding Algorithms", *Reactor Dosimetry*, ASTM STP 1228, Harry Farrar IV, et alia, editors, American Society for Testing and Materials, Philadelphia, 1994.

E3. J.R. Worsham III, "Standard and Reference Field Validation", FTI document # 51-5003585-00, released, 4/26/99.

E4. R.E. Maerker, "LEPRICON Analysis of Pressure Vessel Surveillance Dosimetry Inserted into H.B. Robinson-2 During Cycle 9", *Nuclear Science And Engineering*, # 96, (pages 263-289), 1987.

Appendix F FTI Responses to the -

Request for Additional Information* on
Topical BAW-2241P, Revision 1
Fluence and Uncertainty Methodologies

Question 1

Were the calculations and measurements (including the processing required to convert the measured activities to reaction rates) used in determining the Westinghouse Power Company (W) and Combustion Engineering (CE) data base of calculated to measured ratios (C/Ms) performed by FTI using the methods described in the topical report? If not, provide justification for assuming this data constitutes a single population and can be combined to determine an overall C/M bias and calculational uncertainty.

Response

The calculations and measurements used in determining the Westinghouse Power Company (W) and Combustion Engineering (CE) data-base of calculated to measured ratios (C/M s) were performed using the methods described in the topical (BAW-2241P, Revision 1). The measured activities for the 141 Westinghouse and CE dosimeters came from the B & W Nuclear Environmental Services laboratory (a McDermott Company) as described in Reference 33, Section 8. Since the measurements were performed using the B & W laboratory procedures, there were no

* This *Appendix* contains its own Reference section. Reference F1 refers to the NRC requests for additional information.

conversions to other forms. The calculated activities of the radioactive product isotopes in the dosimeters came from the calculated reaction rates in the target isotopes. As shown in Table A-1, on pages A - 3 through A - 19, the measured and calculated activities from Westinghouse and CE dosimetry are treated the same as dosimetry from B & W reactors and the PCA test reactor.

Since the measured and calculated evaluations of the Westinghouse and CE dosimetry are the same as that for all the dosimetry in the FTI data-base, the measurement (M) and benchmark (C/M) uncertainties should not be unique. For this reason, and others discussed in *Appendix E*, the Westinghouse and CE dosimetry data appear to represent a sample from the same population, which is the FTI dosimetry data-base.

Question 2

> Were any FTI evaluations of W or CE dosimetry excluded from the BAW-2241P data base and, if so, provide justification for excluding this data.

Response

No evaluations of W or CE dosimetry were excluded from the BAW-2241P data-base. The original release of the topical occurred in April of 1997. The processing of the data-base was completed by December of 1996. At that time, the 5 Westinghouse and 5 CE capsules represented all of those in the FTI data-base. The last Westinghouse capsule analysis was from the Prairie Island plant; it was completed in June of 1996 (Reference A14, *Appendix A*). The last CE capsule analyses was from the Calvert Cliffs plant, it was completed in February of 1994 (Reference A4, *Appendix A*). Since 1996, there have been other Westinghouse capsule analyses. (They will be included in the FTI data-base when it is updated.)

Question 3

Provide the method and basis used for determining the values of $\sigma_{C/M}$ (Population | DF = 38) in Tables E-3 and E-4. What is the basis for assuming the W data is one sample out of the 39 plants in the FTI data base?

Response

The method and basis used for determining the values of the benchmark standard deviation ($\sigma_{C/M}$) in Tables E-3 (page E - 26) and E-4 (page E - 27), follows the same concepts of mathematical statistics as those discussed on pages D - 30 through D - 33 (*Appendix D*). To explain the method and basis, the following discussion reviews examples of estimating the standard deviation with the probability distribution function defined to be either Gauss's, or (Student's) William Sealy Gosset's central "t".

Equation 7.15 on page 7 - 32 of the topical is appropriate for estimating the benchmark standard deviation for a set of C/M data. If one set of central "t" data has a total of four deviations $\{\sigma_{C/M}(\mathrm{DF} = 3)\}$ (where DF is the degrees of freedom), and another set has essentially an infinite number $\{\sigma_{C/M}(\mathrm{DF} = \infty)\}$, then the comparison, or combination of the statistical properties is somewhat complex.

$$\begin{aligned} P\left\{ \pm\, 1.0\ \sigma_{C/M}(\mathrm{DF} = 3) \right\} &= 61\,\% \\ \neq\ 68\,\% &= P\left\{ \pm\, 1.0\ \sigma_{C/M}(\mathrm{DF} = \infty) \right\} \end{aligned} \tag{F.1}$$

Equation F.1 shows that ± 1.0 standard deviation, with 3 degrees of freedom (DF = 3), gives a 61 % probability (P) of representing the deviations in the data set,

while $\pm$ 1.0 standard deviation, with an infinite degree of freedom (DF = ∞), gives a 68 % probability (P). Comparing, or combining the two standard deviations $\{\sigma_{C/M}(\mathrm{DF} = 3),\ \sigma_{C/M}(\mathrm{DF} = \infty)\}$ requires an equivalent probability, or level of confidence.

Equations D.2 through D.5, on pages D - 30 through D - 33, show that to combine standard deviations at the same level of confidence requires combinations of the product of the confidence factor and the standard deviation. Equation F.2 shows the appropriate confidence factor from the central "t" distribution to have an equivalent 95 % confidence in the comparison of the two standard deviations $\{\sigma_{C/M}(\mathrm{DF} = 3),\ \sigma_{C/M}(\mathrm{DF} = \infty)\}$.

$$P\left\{\pm\ 3.331\ \sigma_{C/M}(\mathrm{DF} = 3)\right\} = 95\ \% \qquad (F.2)$$

$$= P\left\{\pm\ 2.0\ \sigma_{C/M}(\mathrm{DF} = \infty)\right\}$$

This example is analogous to the situation that we have in *Appendix E*, where the FTI data-base population has 38 degrees of freedom, and we want to know if the 5 Westinghouse plant samples and 4 CE plant samples have comparable statistical properties.

The example assumes a complete data-base population of random deviations that are known to exactly fit Gauss's probability distribution function. The sum of (the first moment of) all the deviations is 0.0. The mean value of the sum of the square of (the second moment of) all the deviations (the variance, $\overline{\sigma^2}$) is 2.0. This gives a standard deviation of $\sqrt{2.0}$, or 1.414. A sample of 4 deviations is taken from the population. If the sample is a statistically valid one, it will have the same properties as the population. This means that the sum of the first moment of sample deviations is 0.0, and the mean value of the variance is 2.0.

$$\begin{matrix}\textit{Mean}\\ \textit{Variance}\\ \textit{Estimate}\end{matrix} = \left(\begin{matrix}\textit{Variance For A}\\ \textit{Statisticaly Known}\\ \textit{Data - Base}\end{matrix}\right)_{Gauss} \times \left\{\begin{matrix}\textit{Central "t" Statistical}\\ \textit{Function For Estimating}\\ \textit{A Finite Data - Base}\end{matrix}\right\}_{Gosset} \tag{F.3}$$

$$\overline{\sigma^2} =$$

For most evaluations, Gauss's distribution of the data is not attainable.

Thus, the degrees of freedom (DF) is $N' - 1$, or DF = 3.

As noted in the worded expression for Equation F.3, the estimate of the mean variance ($\overline{\sigma^2}$) may be defined by the product of the two terms. The term in parenthesis $\left(\quad \right)$ is the expression for Gauss's data-base population. This means that the deviations $\left(\Delta X_n\right)$ fit a Gaussian probability distribution function. The term in braces $\left\{ \quad \right\}$ is the expression for estimating the mean variance assuming that, due to the finite number of data points, there is some uncertainty associated with the sample of data being part of the Gaussian population.

When the Westinghouse and CE benchmark data samples were selected from the FTI data-base population to independently evaluate the statistical properties, two methods of estimating the (standard deviation) variance were used. These methods are the ones just described, based on Equation F.3. Thus, Tables E-3 and E-4 in *Appendix E* have two values for the estimated standard deviation for the samples. Reviewing the

Westinghouse plant benchmarks in Table E-3, the first value of the root mean square standard deviation is based on Equation F.3 with the number of plants (N) being 5, The second value of the root mean square standard deviation is also based on Equation F.3, with N equal 5.

The assumption for the above evaluation is not that the Westinghouse data is one sample out of 39 plants, it is that the 5 capsules from the Westinghouse plants are not unique relative to the 39 capsules in the FTI data-base. The basis for assuming that the Westinghouse capsules are not unique comes from the NRC request that the review of the measured and calculated data by plant type include (page E - 7):

1 - A description of the important physical parameters and characteristics affecting the uncertainties, with discussions explaining why differences between plant types do not result in the data representing different populations.

There is no difference in the dosimetry measurements for Westinghouse plants, nor is there a difference in the analytical methods to calculate the dosimetry activities. Therefore, the sample of Westinghouse deviations should have the same central "t" probability distribution function as the deviations from the data-base population.

Question 4

How do the C/M values of the five selected W plants compare with the C/M values for the other plants in the W data base of Reference-1 in the submittal? In view of the C/M difference between the five selected plants and the W data base average, provide justification for using the C/M value based on the five plants.

Response

Table F-1 compares the C/M values from the five Westinghouse (W) plants analyzed by FTI, with the values that Westinghouse notes in Reference E1 from their data-base.

Table F-1 FTI & W C/M Comparison

Plant	W[E1]	FTI
Prairie Island Unit 1	.748	
North Anna, Unit 1	1.017	
North Anna, Unit 2	1.017	
Shearon Harris, Unit 1	.927	
Zion, Unit 1	.780	
Mean C/M	.898	1.039
FTI Data-Base Mean C/M (39 Capsules & Cavities)		1.026

Table F-1 also includes the mean C/M value from the FTI benchmark data-base,

which is 1.026 This data-base value is very close to the 1.039 value for the Westinghouse plant sample. The 0.898 mean C/M value for the Westinghouse analyses is close to the 0.879 value in their data-base (Reference E1).

The mean C/M value for the Westinghouse analyses is 10.2 percent less than unity (.898), while the mean value for the FTI analyses is 3.9 percent greater than unity. The differences between the Westinghouse mean C/M value and the FTI one (-10.2 and $+3.9$ percent) result in a 14.1 percent absolute difference. These differences are not a concern because they have been previously explained. The explanation is discussed on pages E - 11 through E - 13, and the first paragraph on page E - 14 of *Appendix E*. The FTI C/M comparison is the actual measured specific activity, while the Westinghouse comparison is the unfolded flux (fluence rate). On page E - 12, in Equations E.3 through E.5, it is explained that the FERRET-SAND methods have caused a 12.0 percent bias in the C/M comparisons of unfolded fluence values relative to FTI results. Subtracting the 12.0 percent expected difference, from the 14.1 percent difference, gives a 2.1 percent residual.

Question 5

What is the effect on the bias and uncertainty calculation of eliminating the seven (of twenty-seven) B & W capsule / cavity measurements from the FTI uncertainty analysis (p. E-34, paragraph-3)? How is this effect accommodated in the methodology?

Response

The effect on the calculation to measurement benchmark bias and

the (unbiased) uncertainty ($\overline{\sigma_{C/M}}$) caused by eliminating seven of the twenty-seven B & W capsule - cavity measurements from the FTI uncertainty analysis (page E - 34, paragraph 3) is shown below in Table F-2.

Table F-2 Uncertainty Comparison

Number of B & W Plants	Bias	$\overline{\sigma_{C/M}}$
20		
27		

Tables E-6 and F-2 also show that the mean standard deviation for 20 B & W capsules and cavities is %. If the other 7 capsules and cavities from the B & W plants were to be added to the 20, the mean standard deviation for the 27 capsules and cavities would decrease to %. Thus, the effect on the standard deviation caused by eliminating the 7 capsules and cavities is to increase the estimated value.

The effect of these increases on the methodology is related to (1) the overall bias for all Pressurized Water Reaction (PWR) plants, (2) the overall standard deviation for all PWR plants, and (3) the confidence factor for all PWR plants.

the overall bias for all PWR plants in the data-base continues to be statistically insignificant, as shown in Table E-6.

The standard deviation increase in the B & W data, for 20 capsules and cavities, produces an increase in the overall PWR standard deviation. The combination of 39 capsule and cavity benchmarks in the FTI data-base (including the 27 B & W capsules and cavities) produced a standard deviation of % (page 7 - 33, Equation 7.16). Eliminating the 7 B & W capsules and cavities, and combining the benchmarks in the FTI data-base with equal plant weights, increases the standard deviation from % to % (assuming 38 degrees of freedom). Thus, the eight percent increase shown in Table F-2, results in a five percent increase in the overall standard deviation for all PWR plants.

The elimination of the 7 B & W capsules and cavities from the data-base allowed the methodology for estimating the standard deviation to include response function weights of the data by plant type. The assumption of plant dependent response functions reduced the degrees of freedom to eleven. With eleven degrees of freedom, versus thirty-eight, the confidence factor to achieve a 95 percent confidence level increased from An increased confidence factor results in an increase in the uncertainties in the benchmarks and the calculations.

Question 6

Was the energy-dependent bias used in the FTI methodology applied to the Virginia Power calculations of Table E-5 and, if not, discuss the applicability of these results to the FTI methodology.

Response

The energy dependent bias used in the FTI methodology manifests itself as

The energy dependant bias, observed in the FTI methodology, was evident in the Virginia Power benchmark of calculations to measurements shown in Table E-5 (page E - 28). However, the Virginia Power analyst did not use the FTI bias removal function described on page D - 80. Nor did the analyst develop an energy dependant bias. The application of the energy dependent bias removal to the benchmark of the Virginia Power calculations, shown in Table E-5, was through the combination of uncertainties with Equation 7.13.

Thus, both the Virginia Power and FTI methodology have no bias in the greater than 1.0 MeV dosimeter reactions. The fact that the Virginia Power methodology for the calculations shows no bias, supports the fact that the FTI methodology can produce unbiased calculations of Westinghouse plants. This is in contrast with the fact that the Westinghouse methodology produces biased calculations of their plants.

Question 7

In view of the substantially reduced calculational uncertainty associated with the CE plants, provide justification for including this data in the FTI data base.

How is it assured that the inclusion of the CE plants in the FTI data base does not result in a reduction in the calculational uncertainty applied to the W and B&W plants?

Response

The benchmarks of the calculational to measurement uncertainty for the CE plants is %, as shown in Table E-6, (page E - 35). This is substantially less than the FTI data-base uncertainty for plant benchmarks, which results in a root mean square standard deviation of %, as shown by Equation E.10 (page E - 36). With the CE plant standard deviations combined with Westinghouse and B & W plants, the data-base standard deviation is reduced from % to % with eleven degrees of freedom. The assurance that the CE plants may be included with the population of the twelve plants in the data-base, comes from testing the population with (Student's) William Sealy Gosset's central "t" probability distribution function.

Reviewing Table E-3 (page E - 26) for Westinghouse plants, and Tables E-6 and A-2 (page A - 25) for the B & W plants, shows that indeed, no plant has a mean deviation greater than

The fact that the following three conditions are true, is assurance that the CE plant deviations are not biasing the uncertainty for Westinghouse and B&W plants. (1) The deviations from Westinghouse, CE, PCA, and B & W plants fit within (Student's) William Sealy Gosset's central "t" distribution. (2) The fit is based on the conditional probabilities related to twelve plants. (3) The product of the central "t" confidence factors, for the appropriate conditional probabilities, and the standard deviation of %, bounds the plant deviations. Thus, the CE plant data appears to be an appropriate part of the FTI plant data-base population. The low CE plant standard deviation is merely a fortuitous random occurrence.

Question 8

Why are the $\sigma_{C/M}$ values for the W and CE plants of Tables E-3 and E-4 different than the values given in Table E-6?

Response

The $\sigma_{C/M}$ (standard deviation) values for the Westinghouse and CE plants in Tables E-3 (page E - 26) and E-4 (page E - 27) are different than the values given in Table E-6 (page E - 35) because

This concept was explained when addressing Question 3, on pages F - 3 through F - 7. Rather than list the mean deviations of each of the twelve plants in Table E-6, to define the plant weighted standard deviation (%) of the FTI benchmark data-base, the mean deviation for each grouping of plant types is given. The reason for giving the mean deviation by plant type is to address the possibility that the uncertainties in the

calculations, and thereby the benchmarks, are dependant on plant type. As discussed in topical Section E.4, "Plant Dependent Benchmark Uncertainties", on pages E – 33 and E – 34, the statistical evaluation of the data set samples from the Westinghouse and CE plants could be represented by the statistical properties of the FTI data-base population. However, it was noted that this conclusion could be simply due to a fortuitous combination of the estimated properties. Thereby, the statistical inference of the conclusion would not be appropriate to ensure safe conditions.

Equation F.3 (page F - 5) was used to define a root mean square standard deviation for each plant type in Table E-6. Using the Westinghouse plant data in Table E-3 as an example,

If the mean deviation ($\overline{\Delta X}$) is unbiased, then its value is zero. Squaring the mean deviations from the five Westinghouse plants, with N equal 5, gives a root mean square standard deviation of %, as shown in Table E-6.

the standard deviations can be useful when evaluating the differences between data sets that are not statistically equivalent. As the NRC noted in the previous question (Question 7 on page F - 12), the CE plant data appears questionable relative to the comparable mean standard deviations for the Westinghouse, PCA, and B & W plants. The reason that the CE data appears questionable is due to the estimates of the mean standard deviations by plant type. Each plant type mean standard deviation is based on its unique degrees of freedom (N).

Question 9

There are certain plant features (e.g., vessel thickness, presence of a thermal shield and capsule location) that can have a unique effect on the C/M ratios and require a separate uncertainty analysis. Provide justification for concluding that plants with these types of features do not have to be analyzed separately.

Response

There are various plant features, including the ones that the NRC specifically noted in the above question, that may effect the C/M ratios, and be outside the bounds of the uncertainty analysis presented in this topical. There is no justification for concluding that plants that have features that were not part of the overall uncertainty evaluations may be included under the uncertainty results of this topical. In fact, for each plant-specific fluence analysis, there must be an evaluation of (1) the dosimetry measurements, (2) the C/M ratios, and (3) the analytical uncertainties, to justify the application of the uncertainties in Tables E-1 and E-2 (on pages E - 3 and E - 4 respectively).

In the topical, on pages 7 - 16 and 7 - 17, in Tables 7-2 and 7-3, there are lists of all the dosimeter types that were qualified, or requalified, to be used in conjunction with fluence monitoring. The qualification assessment focused on each laboratory's experimental methodology, to ensure an uncertainty methodology that was appropriately associated with the experimental results. The uncertainty methodology demonstrated that the experimental methodology produced unbiased measurements, or statistically insignificant biases. Moreover, the measurements were determined to have well-defined statistical uncertainties. The statistical uncertainties were defined in terms of (a) standard deviations, (b) levels of confidence consistent with embrittlement

uncertainties, and (c) the central "t" probability distribution function. Any plant-specific fluence evaluation may only use the dosimeter types qualified in the topical. Furthermore, the results of the plant-specific measurements must include an uncertainty evaluation for every dosimeter. The mean standard deviation in the dosimeter activation - reaction measurements must be consistent with the dosimetry qualification outlined in the topical.

The measurement qualification in the topical evaluated more dosimeter types than listed in Tables 7-2 and 7-3. However, the unlisted dosimeter types, such as the Solid State Track Recorders (SSTRs), were disqualified because they could not meet the qualification requirements. For example, as discussed on page 7 - 9, the SSTRs do not have a sufficient mass standard for determining biases in the thin-film deposits.

No new dosimeter types, or new locations of the dosimetry, may be implemented in plant-specific fluence evaluations without a comprehensive measurement uncertainty evaluation, such as that discussed in the topical.

In addition to the disqualified dosimeter types noted in the topical, two of the types that are qualified for measurement uncertainties in Tables 7-2 and 7-3 are disqualified later. As noted on page 7 - 18, in Table 7-4, the dosimeters are disqualified when assessing the uncertainties in the greater than 0.1 MeV activation - reactions and fluence values. The type of dosimeter is sufficient for the spectrum that it covers. However, this dosimeter type has statistical properties that are inconsistent with the other dosimeters covering other portions of the greater than 0.1 MeV spectrum. Thus, it is insufficient for dosimeters to be combined with other dosimeters to estimate the statistical uncertainties in the greater than 0.1 MeV fluence. (At a later date, the dosimeters could be qualified to have consistent statistical properties, and thereby be incorporated into the list of qualified dosimetry.)

The other dosimeter type that is disqualified is . As noted on page 7 - 29, when discussing the application of Equation 7.12, the dosimeters are disqualified for evaluating fluence uncertainties because the reactions are inconsistent with previous calculational benchmark uncertainties. This inconsistent behavior is observed when benchmark ratios of calculations to measurements are compared for the various qualified dosimetry. Since the dosimetry is disqualified for C/M benchmark evaluations, it would be inconsistent to have the statistical properties of the measurements partially based on this dosimeter type. Thus, it is disqualified from evaluations where it would be combined with other dosimeters to estimate the statistical uncertainties in the greater than 0.1 MeV fluence. (Like the dosimetry, the dosimetry could be qualified to have consistent statistical properties at a later date.)

Once the plant-specific dosimetry measurements have been shown to be consistent with the FTI dosimetry measurement data-base, the plant-specific dosimetry benchmark ratio (C/M) must also be shown to be consistent. On page 7 - 34, following the Equation 7.19 estimate of the standard deviation in the calculations of dosimetry activation - reactions,

produces a benchmark uncertainty of percent.

Each plant-specific C/M ratio must be statistically consistent with the FTI benchmark data-base. This does not imply that each C/M ratio must be within percent of unity. Rather, when the plant-specific evaluation becomes part of the data-base (at a later time, during a data-base update) the distribution of deviations must fit within William Sealy Gosset's central "t" (Student's "t"). For example, three of the data-base plants, and a plant-specific evaluation, could have a C/M deviation as large as

percent. However, if a plant has a feature that is unique to the data-base, then the plant-specific deviation must be within the percent standard deviation of the data-base. Thus, if a plant has a unique feature, the only justification for applying the FTI data-base calculational uncertainty to the plant, is if there is a high probability that the uncertainty is applicable. The means of achieving the high probability is to reduce the acceptable C/M deviation.

The third evaluation that must be performed for a plant-specific evaluation, is the verification that the analytical uncertainties remain valid. There are two parts of the analytical uncertainty verification. The first is associated with the C/M verification discussed above. The second is associated with verification of the uncertainties in the parameters and variables that are part of the analytical modeling and computational procedures (Section 7.2, "Dosimetry Calculational Biases and Standard Deviations", pages 7 - 23 through 7 - 27).

The C/M verification discussed above is based on the assumption that the unique feature associated with a specific plant has been evaluated with respect to the physical parameters and characteristics. The basis for the physical evaluation is the same as that noted by the NRC in item 1, on page E - 7. If the evaluation indicates that the calculational methodology has sensitivities to the uncertainties associated with the unique feature that are similar to other uncertainties, then the C/M evaluation is adequate. However, if the evaluation indicates that the calculational methodology has greater uncertainty sensitivities, then an additional analytical uncertainty must be evaluated, and the result added to the existing FTI calculational uncertainties.

The second part of the analytical uncertainty verification is reviewing the uncertainties in the parameters and variables that comprise the uncertainties in the analytical

modeling and computational procedures. The unique feature in a specific plant evaluation may be such that the C/M results are not sufficient to validate the uncertainty. An example of such a situation occurred in the FTI evaluation of the Virginia Power, Surry plant, Unit 1, Capsule X analyses.[F2]

The Surry, Capsule X analyses included partial length poison rods, of two different lengths, in two peripheral fuel assemblies. Moreover, the three-dimensional neutron source distribution originated from Virginia Power calculations. The unique features of the Surry, plant-specific analyses, were partial length rods, of two different lengths, and the Virginia Power source calculation. While Virginia Power source calculations are part of the FTI benchmark data-base (North Anna, Unit 1, Capsule V, and North Anna, Unit 2, also Capsule V, page A - 25), the depressed peripheral powers created a second degree of uniqueness. The locations and operational history of Capsule X could provide only a marginal verification that the FTI calculational uncertainties would be applicable.

To verify that the FTI calculational uncertainties from the data-base would be applicable to the Surry fluence analysis, the analytical source uncertainty evaluated for the topical, needed to be revalidated. In addition, the three-dimensional, multi-channel synthesis needed to be validated. The benchmark calculations in the data-base incorporated a function over the axial length of the problem (Section 3.3.1, "Three-dimensional Synthesis of Results", on pages 3 - 24 through 3 - 29 in the topical).

To validate the Virginia Power three-dimensional source distribution, with particular emphasis on the depressed peripheral powers, the statistical properties of the FTI analytical uncertainty evaluation for the source were reviewed. Virginia Power evaluated their (true) three-dimensional results (no synthesis) with respect to their in-

core instrumentation. The statistical properties were shown to be consistent with the bases for the FTI uncertainties. Thus, the uncertainties in the source distribution for the partial length poison rods were validated.

To validate the three-dimensional, multi-channel syntheses

The evaluation reviewed the deviations in the relative fluence distribution between the FTI synthesis and the Virginia Power benchmarked results. The deviations should be consistent with the FTI uncertainties. If they were, then no additional uncertainty would be needed for the calculations. The uncertainties in the multi-channel synthesis analysis of the partial length poison rods were consistent with the FTI uncertainties for single channel synthesis. Thus, the uncertainties in the calculations were validated for the unique feature of partial length poison rods.

The above discussion notes that there is no general justification for assuming that specific plants, with unique features, that were not a part of the FTI benchmark data-base, would have calculational uncertainties associated with the data-base. The data-base uncertainties include: (1) the dosimetry measurements, (2) the ratio relating the comparison of the calculations to measurements, C/M, and (3) the components of the analytical modeling and computational procedures

If a plant-specific feature is found, that is unique in relation to the three types of uncertainties evaluated in the topical, then the validity of the uncertainties must be verified. If the uncertainties associated with a unique feature in a specific plant cannot be shown to be statistically consistent with the FTI uncertainty data-base, then the calculated fluence uncertainty must be appropriately increased.

Appendix F References

F1. United States Nuclear Regulatory Commission letter to J.J. Kelly, Manager, B & W Owners Group Services, **Request for Additional Information - Framatome Topical Report BAW-2241P, Revision 1**, from Stewart Bailey, Project Manager, Office of Nuclear Reactor Regulation, October 26, 1999.

F2. M.J. DeVan and S.Q. King, **Analysis of Capsule X, Virginia Power Surry Unit No. 1, Reactor Vessel Material Surveillance Program**, BAW-2324, Framatome Technologies Inc., April, 1998.