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

/***RA/***

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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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