April 28, 2006

Mr. Ronnie L. Gardner, Manager
Site Operations and Regulatory Affairs
AREVA NP Inc.
3315 Old Forest Road
Lynchburg, VA 24501

SUBJECT: FINAL SAFETY EVALUATION FOR REVISION 1 OF APPENDIX G TO BAW-2241(P) REVISION 2, "FLUENCE AND UNCERTAINTY METHODOLOGIES" (TAC NO. MC6631)

Dear Mr. Gardner:

By letter dated March 31, 2005, and its supplement dated November 8, 2005, Framatome ANP (FANP), now known as AREVA NP (AREVA), submitted Topical Report (TR) Revision 1 of Appendix G to BAW-2241(P) Revision 2, "Fluence and Uncertainty Methodologies," to the U.S. Nuclear Regulatory Commission (NRC) staff for review and approval. By letter dated March 20, 2006, an NRC draft safety evaluation (SE) regarding our approval of Revision 1 of Appendix G to BAW-2241(P) Revision 2, was provided for your review and comments. By letter dated March 30, 2006, AREVA commented on the draft SE. These comments were discussed in a teleconference between AREVA and the NRC staff on April 18, 2006. The NRC staff's disposition of AREVA's comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter.

The NRC staff has found that Revision 1 of Appendix G to BAW-2241(P) Revision 2, is acceptable for referencing in licensing applications to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for our acceptance of the TR.

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

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

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

Sincerely,

/RA/
Ho K. Nieh, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 728

Enclosure: Final SE

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

Sincerely,

/RA/
Ho K. Nieh, Deputy Director
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Enclosure: Final SE

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FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REVISION 1 OF APPENDIX G TO BAW-2241(P) REVISION 2,

"FLUENCE AND UNCERTAINTY METHODOLOGIES"

AREVA NP

PROJECT NO. 728

1.0 INTRODUCTION AND BACKGROUND

By letter dated March 31, 2005, and its supplement dated November 8, 2005, Framatome ANP (FANP), now known as AREVA NP, submitted Revision 1 of Appendix G to BAW-2241(P) Revision 2, "Fluence and Uncertainty Methodologies," to the U.S. Nuclear Regulatory Commission (NRC) staff for review and approval (Ref. 1). The proposed methodology is intended for application to boiling water reactors (BWRs). Appendix G constitutes an extension of the BAW-2241(P) pressurized water reactor (PWR) pressure vessel fluence methods and uncertainties to account for the differences introduced by its application to BWRs. The Appendix G approach for BWRs 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 (rather than by an extrapolation of the measurements). The dosimeter measurements are only used to determine the calculational bias and uncertainty.

Appendix G provides a description of the extension of the BAW-2241-P PWR calculational methodology for application to BWRs. This includes the treatment of the BWR jet pump/riser geometrical configuration in the numerical transport calculation, determination of the core water number densities (void fractions) and the accuracy assessment for BWRs. BAW-2241(P) and Appendix G to BAW-2241(P) adhere to General Design Criteria (GDC) 30, 31, and the guidance in Regulatory Guide (RG) 1.190 (Ref. 2). As part of the qualification for BWR application, Appendix G presents benchmark comparisons for the Pool Critical Assembly (PCA) dosimetry experiment (Ref. 3), the BNL BWR (BNL-6115) calculational benchmark problem (Ref. 4), and a Browns Ferry-2 (BF-2) surveillance capsule dosimetry measurement (Ref. 5). The Appendix G fluence calculation and uncertainty methodology is summarized in Section 2 of this safety evaluation (SE). 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 of this SE.

2.0 SUMMARY OF THE APPENDIX G FLUENCE CALCULATIONAL METHODS AND BENCHMARKING COMPARISONS

2.1 Semi-Analytic Fluence Calculational Methodology

The basic FANP methodology for calculating BWR fluence is the same semi-analytic methodology used for PWRs. The fluence 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 6 and 7. 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 cross sections, based on the evaluated nuclear data file B-VI (ENDF/B-VI) multi-group nuclear data set (Ref. 8). The fluence calculations are performed with the DORT discrete ordinates transport code (Ref. 9). As in the case of PWRs, the BWR core neutron source term is determined using core-follow data which has been matched to in-core measurements of the three-dimensional power distribution. The prediction of the best-estimate fluence is based on a direct calculation rather than a measurement extrapolated to the vessel inner-wall. The BAW-2241(P) approach incorporates the provisions of RG 1.190 for predicting both the vessel fluence and the dosimeter response.

The extension of the semi-analytic method for BWR applications includes a detailed modeling of the neutron transport through the jet pumps. The Appendix G procedure for constructing the DORT model provides a fine (r, θ) planar mesh for representing the BWR jet pump cylindrical geometry. Using analytic expressions for the model region thickness and area, a detailed description of both (a) the flux attenuation through the jet pump structures and (b) the neutron collision densities in the jet pump material regions is provided.

The calculation of the BWR vessel fluence is further complicated (compared to the PWR analysis) by the coolant voiding in the fuel bundles. The reduced water density in the fuel bundles reduces the neutron flux radial attenuation and increases the leakage from the core. The FANP calculational method includes a special treatment of the increased core leakage due to fuel bundle coolant voiding. The FANP method is based on an accurate matching of the DORT transport calculations and the core-follow calculations of the core leakage in the presence of reduced coolant density in the fuel bundles. The core-follow calculations provide an accurate simulation of the core operating power history.

In the FANP semi-analytic method for PWRs, axial synthesis is used to determine the vessel three-dimensional fluence distribution. However, FANP has extended this method for application to BWRs to account for the increased number of axial shapes due to control rod insertion and non-uniform axial voiding. This extension allows for an increased number and a non-uniform distribution of axial planes in the synthesis. The detailed input for the synthesis and multi-channel planar model calculations is provided by time-dependent three-dimensional core-follow calculations.

2.2 Fluence Measurement and Calculational Benchmarks

Appendix G provides an extensive description of the benchmarking of the FANP vessel fluence calculational methodology. The Appendix G benchmarks include: (a) the Oak Ridge National Laboratory PCA Benchmark Experiment, (b) the Brookhaven National Laboratory BWR pressure vessel benchmark calculation (BNL-6115) and (c) BF-2 pressure vessel surveillance capsule dosimetry measurement. The ratio of the FANP calculation-to-benchmark result provides a quantitative indication of the FANP calculation uncertainty.

The PCA is a well documented vessel mock-up experiment including high accuracy dosimetry measurements. The PCA core includes twenty-five Material Test Reactor curved-plate type fuel elements and the simulator geometry includes a thermal shield, pressure vessel and void box outside the vessel. The PCA dosimetry measurements were made at positions in front and behind the thermal shield, at locations in front and behind the vessel and at vessel internal locations. The PCA dosimetry measurements include the Np-237(n, f), U-238(n, f), In-115(n, n’), Ni-58(n, p) and Al-27(n, α) reactions. Detailed comparisons presented for both the thermal shield and vessel locations indicate good agreement with the dosimetry measurements.

NUREG/CR-6115, "Pressure Vessel Fluence Calculation Benchmark Problems and Solutions,” (Ref. 4) provides the detailed specification and corresponding numerical solutions for a BWR pressure vessel fluence benchmark problem. The benchmark problem provides a reference calculation for a configuration typical of an operating BWR including downcomer and vessel fluences and the dosimeter response at an in-vessel surveillance capsule. The surveillance capsule dosimetry includes the Np-237(n, f), U-238(n, f), Ni-58(n, p), Fe-54(n, p), Ti-46(n, p) and Cu-63(n, α) reaction rates. The FANP model provides a detailed representation of an octant of the problem geometry and includes a radial region which extends from the center of the core out to the outer surface of the vessel. Detailed FANP/BNL-6115 comparisons are presented for both (a) the azimuthal fluence through the vessel and (b) the dosimetry reaction rates. The vessel fluence and surveillance capsule dosimetry comparisons indicate good agreement.

The BF-2 capsule dosimetry measurement provides a benchmark that includes the full as-built BWR material/geometry configuration and an operational core neutron source. The BF-2 capsule (E > 1.0 MeV) flux was determined by General Electric Nuclear Energy (GENE) using the measured iron, nickel and copper reaction rates. The FANP prediction of the BF-2 capsule flux measurement indicated that the fluence calculations are accurate and consistent with the random uncertainty of the FANP data base.

3.0 TECHNICAL EVALUATION

Appendix G of the topical report BAW-2241(P) provides the FANP methodology for performing BWR pressure vessel fluence calculations and determining the associated calculational uncertainty. The review of the FANP 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 FANP. The request for Additional Information (RAI) was transmitted in Reference 10. The information requested was provided by FANP in the responses included in Reference 11. This evaluation is based on the material presented in the topical report and in Reference 11. The evaluation of the major issues raised during the review is summarized below.

3.1 Semi-Analytic Fluence Calculational Methodology

The FANP 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 benchmark measurements generally follow the approach described in the RG 1.190.

RG 1.190 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 Response 8 of Reference 11, FANP 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 the incore detector data. In Response 8, FANP 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.

Because of the strong exponential fluence attenuation, the calculation of the fluence is especially sensitive to both the distance separating the core and the vessel and the barrel thickness. In order to insure an accurate prediction of BWR vessel fluence, consistent with the uncertainty analysis of Appendix G, a reliable estimate of the vessel diameter and barrel thickness are required for input to the DORT transport calculation. To insure the vessel internals geometry is accurately represented, FANP has indicated (Response 5, Reference 11) that a quality assurance review of the drawings is performed as part of the determination of the dimensions used in the DORT transport models.

The fluence analysis of the Davis-Besse benchmark experiment is presented in Section 3 of the BAW-2241(P) topical report to illustrate the application of the semi-analytical methodology. In this analysis, a 45-degree sector of the configuration geometry determined by the symmetry of the PWR fuel loading pattern is modeled. In BWR fluence calculations, the configuration geometry also includes the jet-pumps, risers and surveillance dosimetry which must also be considered in the determination of the azimuthal sector to be modeled. In Response 2 of Reference 11, FANP has indicated that, if the BWR plant core/vessel/dosimetry geometry does not have sufficient symmetry to allow the use of a 45-degree sector, the model will be expanded to an appropriate angular representation (e.g., as a 90-degree sector).

In applications of earlier versions of the semi-analytic methodology, benchmark calculations were performed in the cavity region for the nozzles and seal plate. The calculational modeling in this region, several hundred centimeters above the beltline, was limited and resulted in negative neutron fluxes. The negative fluxes are of concern since they are unphysical and indicate large per cent errors in the calculation. FANP has indicated in Response 1 of Reference 11, that the negative fluxes were due to the large spatial and angular mesh used in the earlier models due to limited computer memory. Because of advances in computer technology which allow fine mesh spatial representations, negative fluxes have not been obtained using the current DORT fluence calculational models.

In the semi-analytic methodology, the fluence accumulated at the vessel at end-of-life (EOL) is determined in two steps. The current fluence is determined first based on the actual operating power history of the plant. The additional fluence accumulated during the remaining plant life (i.e., at EOL) is determined based on a projected core power history. The PWR power history projection and resulting fluence uncertainty are described in Section 7 of BAW-2241(P). In Response 6 of Reference 11, FANP has indicated that the BWR power projection uncertainty has been determined and the BWR EOL fluence standard deviation is less than twenty percent.

The MELLLA$^+$ expansion of the operating range has been implemented at several BWR plants. This expansion can result in a change in the thermal-hydraulic conditions in the downcomer that can affect the attenuation of the neutron flux. In response to RAI-7, FANP has indicated in Reference 11 that the downcomer water properties determined by the core-follow calculations are exactly duplicated in the DORT fluence calculations.

The reduced water density in the fuel bundles (compared to PWRs) introduces an additional complication in the determination of the BWR vessel fluence. The reduced water density (i.e., coolant voiding) reduces the radial attenuation of the neutron flux and increases the leakage from the core. The extension of the semi-analytical method for BWR application includes a new method described by Equation-G.7 of Section G.3. No quantitative validation or verification has been provided to justify the application of this new method in either Appendix G or in Responses 9, 10, 11, 13 and 16. In view of the many approximations implicit in this method and the lack of supporting qualification, this method is not acceptable to be used in applications of the FANP fluence methodology.

3.2 Fluence Measurement and Calculational Benchmarks

The comparison of the semi-analytic fluence predictions with measurement and calculational benchmarks is a necessary and critical part of the qualification of the FANP methodology. The calculation and measurement benchmarks provide an independent assessment of the accuracy of the Appendix G fluence predictions. The calculation-to-measurement (C/M) values resulting from the measurement benchmarking are used to determine the calculation bias and uncertainty (i.e., standard deviation).

In the measurement benchmarks, the methods used to convert the dosimeter response to fluence are complex typically involving adjustments for power history, reaction product half-lives, photo-fission contributions to the fission dosimeters, local perturbation factors for the surveillance capsule and/or instrumentation and dosimeter impurities. In addition, to ensure an accurate prediction of the dosimeter response, a detailed spatial representation of the dosimeter holder tube/surveillance capsule geometry must be included in the DORT model. In Response 3 of Reference 11, FANP has indicated that the differences in the dosimetry introduced by the BWR application (viz., dosimetry wires/foils, holder tubes, encapsulation, etc.) are treated explicitly rather than by modeling approximations. FANP states further in Response 4 of Reference 11 that the procedures for determining the fluence from the dosimeter response conform to the applicable ASTM standards.

The FANP calculational procedure includes the application of a bias removal function to the calculated ($E > 1.0$ MeV) fluence. The bias removal function is based on PWR data taken as part of the Davis Besse Unit-1 Cavity Dosimetry Measurement Program. No BWR data has been provided to justify application of the function in BWR applications. Since this correction can result in a nonconservative reduction in the ($E > 1.0$ MeV) fluence, the bias removal function is not acceptable to be used in BWR applications.

The uncertainty in the vessel fluence calculation depends on the plant-to-plant variation in the as-built core/internals/vessel geometry, core power and exposure distributions, and the plant power history. Because of the limited number of BWR operating reactor measurement benchmarks included in Appendix G and to insure a reliable assessment of the fluence calculational uncertainty, additional measurement qualification must be provided in plant-specific applications of the fluence methodology. In the initial four (4) applications of the FANP BWR methodology, the fluence predictions of the Appendix G methodology must be compared with surveillance capsule or cavity fluence measurements for the vessel being analyzed. If the results of the C/M comparisons for these measurements are not consistent with the BAW-2241(P) uncertainty analysis (recognizing the uncertainty of a limited sample size), the uncertainty analysis must be updated or the deviations explained. In addition, after the initial four applications of the fluence methodology, the uncertainty analysis must be updated with at least four (4) additional BWR dosimetry measurement comparisons to confirm, and update if necessary, the Appendix G fluence calculational bias and uncertainty. As required by RG 1.190, this confirmation/update must also be performed as subsequent measurements become available.

4.0 CONCLUSION, LIMITATIONS AND CONDITIONS

Appendix G of the Topical Report BAW-2241(P), “Fluence and Uncertainty Methodologies,” and the supporting documentation provided in Reference 11 have been reviewed in detail. Based on this review, it is concluded that the proposed methodology is acceptable for determining the pressure vessel fluence of BWRs under the following conditions:

1. In view of the many approximations in the method described by Equation-G.7 of Section G.3 and the lack of supporting qualification, this method is not acceptable to be used in applications of the FANP fluence methodology. However, in conjunction with Condition No. 3 below, if additional BWR benchmark comparisons show biases that are directly related to calculations without Equation-G.7, then the Equation-G.7 results would be acceptable for a single plant-specific application. For each and every plant-specific application, the NRC staff must be notified and the dosimetry benchmark results, with and without Equation-G.7, presented in either the surveillance report or some other appropriate report. If the results from the eight (8) additional dosimetry benchmark comparisons to measurements required by Condition No. 3 below validate Equation-G.7, then FANP may submit the combined data to the NRC staff and request a revision of this condition.

2. The bias correction is based on PWR data and no qualification data is available for justifying BWR application. Since this correction can result in a nonconservative reduction in the > 1-MeV fluence, the bias removal function is not acceptable to be used in BWR applications. However, in conjunction with Condition No. 3 below, if additional BWR benchmark comparisons confirm that BWR dosimeter biases are the same as the FANP benchmark database biases, then the bias removal function would be acceptable for a single plant-specific application. For each and every plant-specific application, the NRC staff must be notified and the dosimetry benchmark results, with and without the application of the bias removal function, presented in either the surveillance report or some other appropriate report. If the results from the eight (8) additional dosimetry benchmark comparisons to measurements required by Condition No. 3 below validate the bias removal function for BWRs, then FANP may submit the combined data to the NRC staff and request a revision of this condition.

3. Because of the limited number of BWR benchmark calculations to operating data, an additional qualification must be provided in plant-specific applications of the Appendix G fluence methodology. When measured data is available, this must include: (1) in the

initial four (4) applications, a comparison of the Appendix G fluence prediction with measurements for the vessel being analyzed and an update of the uncertainty analysis if necessary and (2) after the four initial applications of the methodology, the uncertainty analysis must be updated with at least four (4) additional BWR dosimetry measurement comparisons to confirm, and update if necessary, the Appendix G fluence calculational bias and uncertainty. As required by RG 1.190, this confirmation/update must also be performed as subsequent measurements become available. When measured data is not available, the plant-specific application must include an analytic sensitivity evaluation of the calculational uncertainties between the plant without measured data and a comparable plant that has an appropriate benchmark of the calculations to dosimetry measurements. The plant-specific evaluation, without an appropriate calculational benchmark, must incorporate a larger uncertainty and a positive bias in the fluence predictions for the structural materials.

5.0 REFERENCES

1. Letter from Jerald S. Holm, Framatome ANP to Document Control Desk (NRC), “Request for Review and Approval of Revision-1 of Appendix G to BAW-2241(P), Revision 2, ‘Fluence and Uncertainty Methodologies’,” dated March 31, 2005.

2. Office of Nuclear Regulatory Research, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” Regulatory Guide 1.190, U.S. Nuclear Regulatory Commission, March 2001.

3. W.N. McElroy, Editor, "LWR Pressure Vessel Surveillance Dosimetry Improvement Program: PCA Experiments and Blind Test," NUREG/CR-1861 (Hanford Engineering Development Laboratory, HEDL-TME 80-87), July 1981.

4. J. F. Carew, K. Hu, A. Aronson, A. Prince, and G. Zamonsky, “Pressure Vessel Fluence Calculation Benchmark Problems and Solutions,” NUREG/CR-6115, BNL-NUREG-52395, September 2001.

5. L. J. Tilly, B. D. Frew, B. J. Branlund, “Pressure-Temperature Curves for TVA Browns Ferry Unit 3,” GE Nuclear Energy, GE-NE-0000-0013-3193-02a-R1, Revision 1, August 2003.

6. King, S. Q., et al., “Pressure Vessel Fluence Analysis for 177-FA Reactors,” BAW-1485P, Rev. 1, April 1998.

7. Whitmarsh, C. L., “Pressure Vessel Fluence Analysis for 177-FA Reactors,” BAW-1485, June 1978.

8. Radiation Shielding Information Center (RSIC), Oak Ridge National Laboratory (ORNL), “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,” DLC-175, April 1994.

9. Mark A. Rutherford, et. al., “DORT, Two Dimensional Discrete Ordinates Transport Code, (BWNT Version of RISC/ORNL Code DORT),” FANP Document # BWNT-TM-107,
May 1995.

10. “Request for Additional Information for Appendix G of Topical Report BAW-2241-P,” E-Mail, Michelle C. Honcharik (NRC) to Gayle Elliot (FANP), dated September 8, 2005.

11. “Response to a Request for Additional Information Regarding BAW-2241(P), Appendix G, ‘Fluence and Uncertainty Methodologies’,” Letter, Ronnie L. Gardner to Document Control Desk, U. S. Nuclear Regulatory Commission, dated November 8, 2005.

Principal Contributor: L. Lois

Date: April 28, 2006

RESOLUTION OF AREVA'S COMMENTS

ON DRAFT SAFETY EVALUATION FOR TOPICAL REPORT REVISION 1 OF APPENDIX G

TO BAW-2241(P) REVISION 2, "FLUENCE AND UNCERTAINTY METHODOLOGIES”

By letter dated March 30, 2006, AREVA commented on the NRC draft SE for Revision 1 of Appendix G to BAW-2241(P) Revision 2, "Fluence and Uncertainty Methodologies.” These comments were discussed in a teleconference between AREVA and the NRC staff on April 18, 2006. The NRC staff agrees with AREVA's comments and the modifications as discussed with AREVA in the teleconference have been made to the final SE, as described below.

No.	Draft SE Reference	AREVA's comments	NRC Staff Resolution
1.	Page No. 4, Line No. 26-27	The omitted statement is misleading. Since the Davis Besse experiment had dosimeters extending to the 'seal plate" in the cavity region, AREVA NP performed benchmark comparisons of the calculations to the measurements. However, in this region which is several hundred centimeters above the beltline, the model was insufficient and gave negative fluxes.	Adopted
2.	Page No. 5, Line No. 4-10	Proprietary information	Adopted
3.	Page No. 5, Line No. 37-42	Proprietary information	Adopted
4.	Page No. 6, Line No. 22-24	Condition number "1" on page 6 is proposed because the method represented by Equation G.7 in Section G.3 involves approximations that lack statistically sufficient qualification with appropriate benchmarks. However, it is noted (page 2, lines 23-25) that the AREVA NP method is based on an accurate matching of the DORT transport calculations and the core-follow calculations with respect to the core leakage. Moreover on page 4, line 7, the strong exponential fluence attenuation is noted. AREVA NP agrees that the current benchmarks have not required using Equation G.7 and thereby the equation should not be utilized in predictions of the vessel fluence. However, condition number “3" requires additional benchmark information. If AREVA NP finds that	Adopted as discussed in the teleconference

No.	Draft SE Reference	AREVA's comments	NRC Staff Resolution
		Equation G.7 is required to have unbiased benchmark results, then AREVA NP should have the option of using Equation G.7 and showing the NRC the benchmark comparison. Therefore, condition number "1" should be modified to delete proprietary information (see Section 6.0 of Affidavit dated March 31, 2005) and be expanded to include an additional statement as shown in Insert B.	
5.	Page No. 6, Line No. 26-29	Condition number "2" on page 6 is proposed because the bias removal function lacks statistically sufficient qualification with the appropriate BWR benchmarks. Moreover, the bias removal function increases the fluence in the energy groups above 3.0 MeV and decreases the fluence in the lower energy groups. Therefore, it is possible for the function to decrease the total fluence above 1.0 MeV as well as increase it. AREVA NP agrees that the bias removal function lacks statistically sufficient BWR benchmarks. However, the function appears to be generic; it was first identified by R.E. Mearker of the Oak Ridge National Laboratory. Mearker's database included many test reactors that are unrelated to either PWRs or BWRs. Therefore, it is more likely that the same critical spectrum bias that is evident in test reactors and PWRs will also be evident in BWRs. Therefore, condition number "2" should be modified to delete proprietary information (see Section 6.0 of Affidavit dated March 31, 2005) and be expanded to include an additional statement as shown in Insert C.	Adopted as discussed in the teleconference
6.	Page No. 6, Line No. 31-40	Condition number "3" on page 6 is proposed because the AREVA NP benchmark database is too sparsely populated with BWR data. Therefore, AREVA NP agrees that additional data would be appropriate to support the statistical confidence levels. However, as the NRC is aware from the BWRVIP program, the BWR owners have combined their surveillance into an	Adopted as discussed in the teleconference

No.	Draft SE Reference	AREVA's comments	NRC Staff Resolution
		integrated program. Thus, there are several reactors that will not actually have measured data. In a plant-specific application of the AREVA NP fluence analysis, if the appropriate measured data is not available, then there needs to be an additional benchmark requirement. AREVA NP has expanded the condition number "3" statements to include an analytic sensitivity of the uncertainties in a plant-specific analysis to increase the fluence values in the structural materials. The analytic uncertainty for a BWR that is in the AREVA NP benchmark database will be reviewed in comparison to a specific BWR analysis that does not have the appropriate measured data. The differences in the uncertainty values between the BWR with benchmark data and the one without it will be appropriately applied to increase the structural material fluence values. The increase will be based on the analytic uncertainties and will increase the fluence independent of whether the sensitivity uncertainties increased or decreased the fluence values.	