FRAMATOME ANP

BAW-2241NP
Revision 2
Appendix G

Fluence and Uncertainty Methodologies

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Framatome ANP, Inc.

Non-Proprietary

G.1 Introduction

A number of utilities that own and operate boiling water reactors (BWRs) have indicated that they need a cost-effective alternative to the fluence methodology offered by the competition. Because of this need, Framatome ANP (Advanced Nuclear Power) developed a fluence methodology for specific application to BWRs. The methodology is based on the PWR methodology that was accepted for licensing applications in Revision 1 of this topical report (BAW-2241P-A). The BWR methodology described in this appendix constitutes Revision 2 to BAW-2241P. The purpose of this appendix is to gain the NRC's acceptance of the BWR fluence methodology for license applications.

Before developing this appendix, Framatome ANP (FANP) held discussions with NRC to confirm the content required on which the NRC could base its acceptance of the methodology. It was agreed that the basic methodology for the fluence calculations, dosimetry measurements, and uncertainty evaluations would be the same for PWRs and BWRs. However, the BWR fluence methodology needed to include water density effects on the neutron leakage rate and the calculational uncertainty needed to be consistent with a set of BWR benchmarks.

Section 3.0 of this topical report provides a discussion of the neutron leakage function. It is explained that the fluence at specimen and vessel locations is directly dependent on the neutron leakage from the core. The core neutron leakage rate is a function of the fuel region material properties, specifically the neutron mean free path within each fuel assembly. The greater the mean free path, the greater the neutron leakage rate. This is a key consideration for BWR fluence analysis.

If the neutron collision density in the fuel region is decreased, the mean free path increases and the neutron leakage will increase. Due to the boiling water within the channel of a BWR fuel assembly, the collision density continuously decreases as the flow proceeds from inlet to outlet. Since high-energy neutrons have a high probability of colliding with the hydrogen atoms in water molecules, the decreasing water density significantly reduces the collision density and increases the neutron leakage.

When operating at rated power, the water density within an axial segment of a PWR fuel assembly is nearly constant. Thus, the neutron leakage from the PWR fuel region is not significantly affected by water density variations. Depending on the operating condition, the water density within a BWR fuel assembly can vary significantly between inlet and outlet axial segments. The BWR water density decreases as a function of (a) the axially integrated assembly power and (b) the controlled decrease in core flow rate with full power operation. Because of the potential for large variations in BWR water density, the calculational methods used to evaluate the neutron leakage from each fuel assembly must be modified [

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A second key consideration associated with fluence analysis for BWR s is the validation of the uncertainties. As noted in Section 7 of this topical report, there is a set of independent uncertainties associated with the measurement methodology and another independent set associated with the calculational methodology. Since the measurement methodology is the same for PWR and BWR dosimetry, only the uncertainties associated with the calculations require adjustment.

The basis of the theoretical methodology for calculating the fluence in a BWR is the same as that for a PWR. The DORT[G1] computer code is used in the same manner for both, and both use the BUGLE-93[7] cross section library (Section 3 describes these methods). The source term for PWRs and BWRs is developed from core-follow data that matches the in-core operational measurements of the three-dimensional power. Since the theoretical methodology for BWRs is the same as that for PWRs, the uncertainty methodology would also be the same. Moreover, the estimated value of the uncertainty for BWR and PWR results should be the same. However, the complexity associated with varying water densities in the axial segments of BWR fuel assemblies introduces an additional uncertainty into the analytical modeling. Thus, the BWR uncertainties need to be validated using the methods explained in Section 7.

As noted in *Appendix E*, the technologies used in modeling the various reactor components can have a significant effect on the bias and random deviations in the calculations. In accepting Revision 1 of this topical report, the NRC concluded that Framatome ANP's calculations of PWR dosimetry obtained from Westinghouse-designed reactors are exceptionally accurate. Westinghouse reports a bias of 12.1 percent[E1] in its benchmark of dosimetry calculations in addition to a standard deviation of 10.3 percent.[E1] This gives a total uncertainty of 22.4 percent. Framatome ANP reports no bias and a standard deviation of 10.0 percent.

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This appendix presents the extended calculational methodology that has been developed for BWR fluence evaluations. Following this introductory section, are Sections *G*.2, *G*.3, and *G*.4. Section *G*.2 presents the "Background," Section *G*.3 the "Extension of Fluence Methods," and Section *G*.4 the "Uncertainty Update." The "Background" section discusses the issues that have influenced the development of the methods and the benchmark validation of dosimetry. [

] The "Extension of Fluence Methods" section discusses the calculational methods and procedures used to model the core and internal structures [

] The last section, "Uncertainty Update," discusses the integration of the BWR dosimetry with the Framatome ANP dosimetry database.

G.2 Background

As noted in the introductory discussion, the utilities that own and operate BWRs have stated a need for cost-effective fluence methodology. The methodology described in this appendix was developed to meet that need. Although BWR reactor vessels are not expected to receive enough irradiation damage to require a comprehensive fluence - embrittlement evaluation, the fluence calculations are required to meet the standards of Regulatory Guide 1.190.[G3]

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The following text in Section G.2.1 describes the industry's historical development of uncertainties and what has to be done to establish new uncertainties for BWRs based on FANP's techniques, which are already accepted for PWR applications.

G.2.1 Historical Perspective

In the 1970s, the accuracy of PWR fluence calculations in comparison to measurements showed large and inconsistent deviations. Thus, uncertainties were difficult to validate. In 1977, the NRC established the "Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program." The purpose of the program was to establish an appropriate level of confidence in the bias and random uncertainty that was associated with each vendor's calculational methods. To establish a high level of confidence in the results, the program included a requirement to evaluate a "blind test." The "blind test" ensured that the participants would not know the measurement results until everyone had submitted their respective calculational results.

The Pool Critical Assembly (PCA) blind test was supervised by the Oak Ridge National Laboratory (ORNL).[37] Framatome ANP and the other industry participants modeled the PCA reactor and predicted dosimetry activities in the vessel and internals structure. Framatome ANP and the others submitted their calculations to ORNL. ORNL compared the calculated results (C) with their measurements (M) and sent Framatome ANP the C/M results along with the assessment of the measurement uncertainty. The C/M results indicated a mean deviation of 9.3 percent, which was extraordinarily precise. The ORNL measurement uncertainty was between 6.0 percent and 10.0 percent. The Framatome ANP results were the most accurate of all participants.[37]

GE did not participate in the original "blind test" from the 1970s. To achieve the appropriate level of confidence in *GE*'s methods, the NRC stated in the "Summary and Limitations" section (4.0) of the "Safety Evaluation Report":[G2]

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

Between September 2001 (the approval date of the *GE* topical report) and September 2004, *GE* must conduct four "blind test" benchmark comparisons of calculated to measured dosimetry results. Since GE has conditionally approved fluence methods for

BWRs, the acceptance of the methodology presented in this appendix is necessary to provide a feasible alternative so that evaluations of BWR specimens can be performed by FANP.

This appendix describes the methods that FANP developed to perform BWR fluence calculations. It also reviews the fluence uncertainty methods. [

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G.2.2 **Benchmarks**

Since FANP has performed the NRC "blind test"[37] and was the only participant to provide results that were within the defined uncertainty band, [

] (These are benchmarks; not the four additional "blind tests" required of *GE.*)

Since Framatome ANP has dosimetry measurement methods with certified uncertainties from reference field validation by the National Institute of Standards and Technology, the benchmarks will include a certified uncertainty for the calculated results as well as the benchmark and measurement uncertainties. (This benchmark evaluation goes beyond the information presented in the *GE* topical report that shows no measurement validation.)

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G.3 Extension of Fluence Methods

Revision 1 of the topical report presents two methodologies, one for determining the fluence and the other for estimating the uncertainty in the methodology for determining the fluence. This section (*G*.3, Revision 2) focuses on extending the methodology for determining the fluence throughout the BWR "beltline" region. The following section (*G*.4) focuses on the "Uncertainty Update".

There are two major parts of Framatome ANP's methodology for determining the fluence. The first part is the evaluation of dosimetry measurements. The second is the calculation of the fluence throughout the reactor internal structures, vessel, and reactor shield-support structure within the "beltline" region.

The theoretical and experimental methods used to determine the calculated and measured results for the fluence and dosimetry activities are not dependent on the reactor design. Thus, the theoretical and experimental methods (DORT, BUGLE-93, etc.) for BWRs are the same as those for PWRs. While the approximations used to obtain solutions to the theoretical methods for PWRs need to be extended when applied to BWRs, the measurement process requires no extension of the techniques or procedures. Consequently, the experimental methodology is not discussed in this section. The BWR experimental methods are the same as those discussed in Section 5 of this topical report. This section addresses the BWR calculational models and procedures used in the solution of the theoretical methods (Section *G*.3.1). [

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G.3.1 Solution of BWR Fluence Methods

This section explains how the demonstrated accuracy achieved for PWRs can be applied to BWRs.

The fluence methodology presented in this report describes theoretical methods, with procedural and modeling approximations that provide accurate and reliable predictions of the greater than 0.1 MeV fluence values. These methods are generic to any water-moderated reactor. Consequently, the PWR calculational models and procedures are utilized as the basis for calculating the fluence throughout the internal components and vessel of BWR s. While the PWR approximations are generic to any water-moderated reactor, there are three areas where the approximations must be expanded to provide accurate and reliable predictions of the greater than 0.1 MeV fluence values for BWR s. The following discussion explains the development process that led to the identification of the three areas. The discussion continues by explaining the development of the expanded models and procedures to ensure the same accuracy in the results as previously shown by the benchmark database.

The development of the procedural and modeling approximations for BWR s is based on evaluations utilizing PWR models. The PWR fluence analyses gave satisfactory results for BWR fluence values even though it did not include precise analytical modeling of the BWR core and "down-comer" regions. In 2001, when Framatome ANP formed a joint venture with Siemens which possessed BWR technology, valuable expertise became available for applying PWR fluence methods to BWR designs.

The three areas that require an extension of the PWR models and procedures to accurately analyze BWR vessel fluence values are: (1) the transport of neutrons from the core through the internal structures associated with the jet pumps, (2) the integrated core leakage function from the fuel, and (3) the three-dimensional synthesis of the core leakage function. The details of the development in each of the three areas are described in Sections G.3.1.1, G.3.1.2, and G.3.1.3 respectively.

G.3.1.1 **Neutron Transport Through Jet Pumps**

Neutron transport from the core region through the internals and other reactor structures is envisioned as being divided into two parts for the purpose of this discussion. The first part involves the leakage of neutrons from the core. The second involves the transport of the neutrons through the internal components and vessel to the concrete shield and support structure. The Framatome ANP models and procedures used to obtain a solution to the transport process in the second part are equally applicable in PWRs and BWRs. However, if BWR dosimetry is located within the radiation shadow area of the jet pumps, the modeling of the pump structures must include the same type of procedures for volume and surface accuracy as those used in PWRs for the core-baffle-plate regions and surveillance capsules.

Figure 3-2, on page 3 - 6, shows a schematic of the radial plane of a PWR. To model this geometry, a cylindrical coordinate system (r, θ) is used. However, there is a problem with using cylindrical coordinates to accurately model a square volume and straight surface area. This problem is accentuated further by the baffle plate structure that forms part of the connection between the fuel assemblies and barrel. The baffle plates are rectangular, with a thin width along the radial coordinate (r), and a long length along the angular coordinate (θ). Accurately modeling the volume and surface area of the baffle region requires a detailed mesh when using cylindrical coordinates.

In addition to the core and baffle regions, Figure 2-5, on page 2 - 40, of Reference 9, shows a schematic of the radial plane of a surveillance capsule. The surveillance capsule is cylindrical. It is located along a radial line at some r, θ point. A detailed r, θ coordinate mesh is also required to accurately model the volume and surface area of the "off-centered" surveillance capsule.

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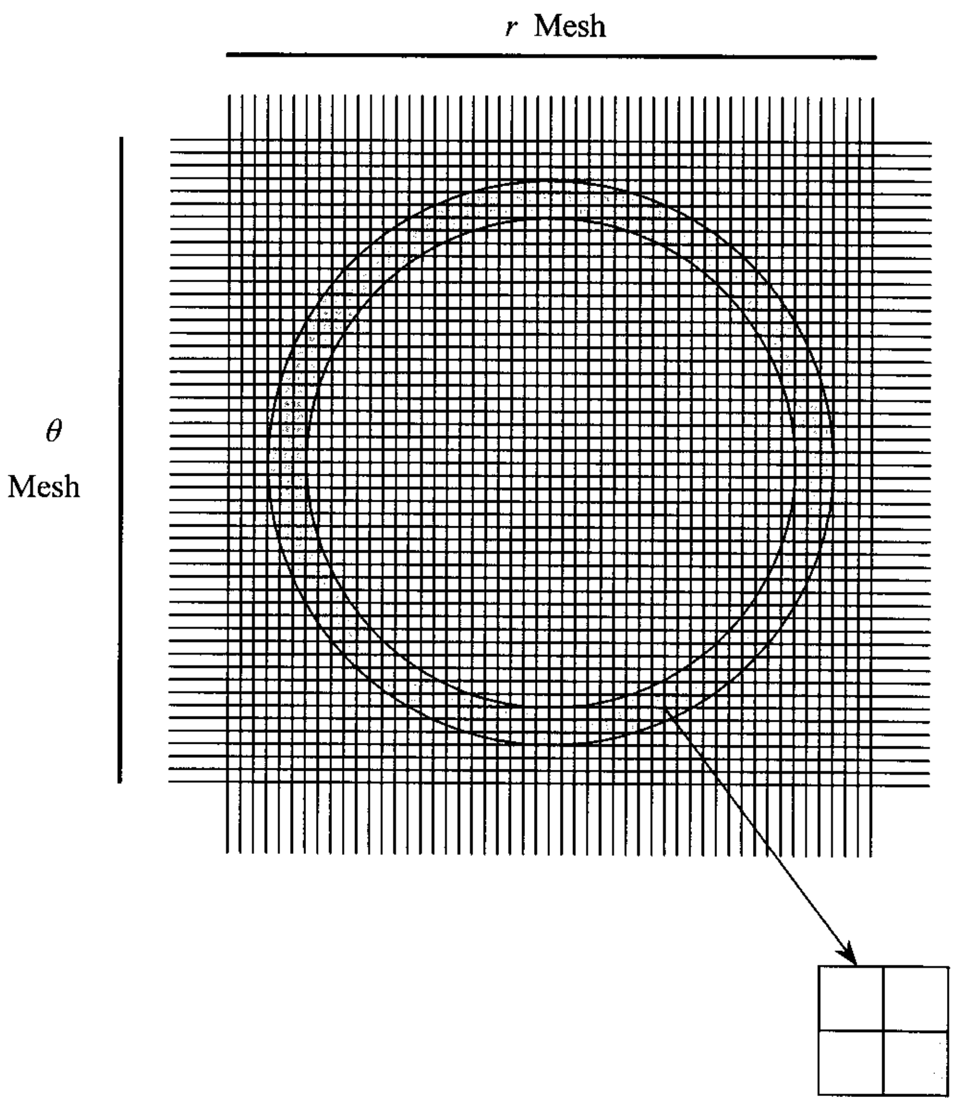
Figure G-1 on the following page (G - 15) shows a schematic of a cylindrical section of a jet pump component in the radial plane. The vessel fluence rate (flux) is shielded from the neutrons leaking from the core by the internal structures, such as the jet pumps. Therefore, the maximum vessel flux does not occur in the shadow behind the jet pump structures. So, the evaluation of the maximum flux does not need an accurate pump model. However, the evaluation of the dosimetry in the shadowed area behind the jet pump structures is affected by the pump modeling. To achieve accurate dosimetry calculations, the r, θ modeling, and the procedures for representing the materials must be "accurate." The following discussion reviews the models and procedures used to attain the needed accuracy in the flux calculations.

Reviewing Figure G-1, the jet pump structure is schematically shown in the radial plane as the "shaded" tubular region. The coordinates, noted by the square cross-hatch of grid lines, are cylindrical.

Figure G-1 Schematic of r, θ Modeling
For Jet Pump Tubular Structure



The abscissa (horizontal line) is noted as the radial (r) coordinate, and the ordinate (vertical line) is noted as the angular (θ) coordinate. While the grid lines are equally spaced and appear to be centered within the jet pump tubular structure, the center is to the left, at the center of the core. The radial mesh is noted with an index of (i), r_i, and the angular mesh with an index of (j), θ_j. Considering a radial line that is centered along the angular mesh, this line would go through the middle of the tube and thereby include the diameter. The thickness of the radial mesh $(r_{i+1} - r_i)$ is the outer diameter of the tube minus the inner diameter, or some incremental fraction of this thickness. This radial mesh is constant throughout the tube, from the left outer diameter to the right outer diameter. The thickness of the angular mesh $(\theta_{j+1} - \theta_j)$ is the same as the radial. Like the radial mesh, the angular mesh is constant throughout the tube, from the top outer diameter to the bottom outer diameter. Thus, the r, θ mesh is represented by a square array.

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] A two-by-two r, θ mesh schematic in Figure G-1 shows pump material in two of the mesh-blocks and water in the other two. [

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The extension of the Framatome ANP models and procedures to BWR jet pumps provides a means of accurately evaluating dosimetry reactions in the vicinity of the pumps. [

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G.3.1.2 Core Leakage Function

An important consideration in the flux solution for vessel fluence evaluations is the leakage of neutrons from the core. The Framatome ANP methods in Section 3 of this topical are based on the solution of the three-dimensional (**r**) fission rates integrated over the energy (E) and angular variables (**Ω**) of the velocity groups (g), and time (t). The accuracy of the process begins with the core-follow simulation of the measured fission rates for power production. The core-follow results match the measurements within the uncertainty criteria for the power level and distribution.

The measurements of core operation are taken at periodic intervals. The core-follow simulation of the operation utilizes the measured data from each period to follow the power production. Given the close relationship between the calculated three-dimensional power distribution and the comparable measurements of axial segments for each assembly, the core-follow time-steps provide a numerical means of integrating the fission rates over the operational cycles. The average time-weighted source parameters are those given in Equations 3.1 and 3.2 on pages 3 - 11 and 3 - 12. As shown by the equations, the neutron source terms are represented by three-dimensional (**r**) values for each fuel rod and axial rod segment. These sources are processed for the cylindrical coordinate system used in the DORT modeling.

Since the discrete source eigenfunctions represent a solution to the three-dimensional neutron transport equation, these source eigenfunctions may be returned to a three-dimensional neutron transport model to serve as a "fixed" source term. The neutron transport theory expression with a "fixed" source eigenfunction $\{S(\mathbf{r}, \mathrm{E}, \mathbf{\Omega})\}$ is represented by Equation G.4.

$$\mathbf{\Omega} \bullet \nabla\phi(\mathbf{r}, \mathrm{E}, \mathbf{\Omega}) \quad + \mathbf{\Sigma}_{\mathrm{T}}(\mathbf{r}, \mathrm{E})\,\phi(\mathbf{r}, \mathrm{E}, \mathbf{\Omega}) \quad = \quad S(\mathbf{r}, \mathrm{E}, \mathbf{\Omega}) \qquad \text{(G.4)}$$

The average time-weighted collision density parameters $\{\mathbf{\Sigma}_{\mathrm{T}}(\mathbf{r}, \mathrm{E})\,\phi(\mathbf{r}, \mathrm{E}, \mathbf{\Omega})\}$ from the three-dimensional core-follow calculations are evaluated using the same procedures as those used for the source parameters. Assuming that there is no average time-weighted effect on the leakage function $\{\mathbf{\Omega} \bullet \nabla\phi(\mathbf{r}, \mathrm{E}, \mathbf{\Omega})\}$, the collision density parameters and source parameters in Equation G.4 produce the same flux (fluence rate) values as those from the average time-weighted core-follow calculations.

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Using the models and procedures discussed in Section 3 of this topical to compute BWR leakage rates from the core periphery indicates that the approximations in the modeling and procedures must be extended. The average time-weighted "fixed" source eigenfunctions and collision density parameters do not produce accurate peripheral flux (fluence rate) values. To understand the failure in the approximations, the solution of Equation G.4 needs to be reviewed. DORT provides a general numerical solution of Equation G.4, but it is not useful to specifically evaluate the relationship between the leakage rate, collision density, and source density. [

](G.5)

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The neutrons crossing the boundary between $\mathbf{r}'$ and $\mathbf{r}$ represent the leakage of source neutrons from $\mathbf{r}'$. If we consider a fuel region defined by an array of $\mathbf{r}'$ mesh positions, the leakage from the $\mathbf{r}'$ region is evaluated by integrating the current density at the surface of the fuel, the boundary of $\mathbf{r}'$. The leakage of the greater than 1.0 MeV flux from the surface of the fuel region is expressed by the Equation G.6 integrals over energy (E), angle $(\mathbf{\Omega})$, and the surface area (A) perpendicular $(\perp)$ to a unit of the vector "$\mathbf{r}'_{\perp A}$" in the direction of the neutron current from the region.

$$
\begin{aligned}
Leakage\left(\overline{\mathbf{r}'}, \overline{\mathrm{E}}, \overline{\mathbf{\Omega}}\right) &= \int_{A} \int_{\mathrm{E}} \int_{\mathbf{\Omega}} \mathbf{\Omega}\ \phi\left(\mathbf{r}', \mathrm{E}, \mathbf{\Omega}\right) \left(\frac{\mathbf{r}'_{\perp A}}{|\mathbf{r}'|}\right) \bullet dA\ d\mathrm{E}\ d\mathbf{\Omega} \\
&= \phi\left(\overline{\mathbf{r}'}, \overline{\mathrm{E}}, \overline{\mathbf{\Omega}}\right) f_{leakage}\left(\mathrm{N}^{\mathrm{Water}}, Constants\right)
\end{aligned}
\tag{G.6}
$$

Substituting the equivalent Equation G.5 solution into the integral part of Equation G.6 gives the leakage in terms [of the scalar region flux $\{\phi(\overline{\mathbf{r}'}, \overline{\mathrm{E}}, \overline{\mathbf{\Omega}})\}$] with energies greater than 1.0 MeV and an exponential integral function ($f_{leakage}$). Since (a) the current density is determined by the angular integral of the vector flux density $\{\mathbf{\Omega}\ \phi(\mathbf{r}, \mathrm{E}, \mathbf{\Omega})\}$, and (b) the source density produces the flux from the leakage, and scattering reaction $\{\mathbf{\Sigma}_s(\mathbf{r}', \mathrm{E})\ \phi(\mathbf{r}', \mathrm{E}, \mathbf{\Omega})\}$ rate densities,

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These approximations in the PWR models and procedures produce accurate flux results within the core region of the peripheral fuel assemblies, and for dosimetry reactions. However, in a BWR model, the core and dosimetry calculational accuracy is insufficient. The problem is that the water concentration (N^{Water}) in an axial segment of a fuel assembly varies during a cycle, and may vary from cycle to cycle for the assemblies located in the same position on the periphery of the core. Consequently, the total cross section $\{\boldsymbol{\Sigma}_T(\mathbf{r}', g)\}$ varies with time during the operation of the various cycles.

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G.3.1.3 **Three-Dimensional Synthesis**

The fluence calculational methodology discussed in the previous sections (3.1.2 and 3.3.1) of this topical report begins with "exact" three-dimensional ($\mathbf{r}$; x, y, z) core-follow analyses (no synthesis approximation) for the core region. Reviewing the results from any PWR model shows that all cores that operate without control rods or non-uniform poison shields have only one unique axial (z) power shape. Moreover, those cores that operate with axial power shaping rods (B & W plants) can be modeled using only one unique axial power shape for fluence (rate) calculations. Thus, collapsing the "exact" three-dimensional model to two-dimensional models (x, y) or (r, θ) is a straightforward integral process. It is also straightforward to integrate the r, θ model over the θ direction and incorporate the z - source distribution when developing the r, z model.

When the peripheral fuel of a PWR core has axially segmented fuel assembly components to shield a critical weld location, multichannel - planar models, with piece-wise continuous axial shape functions, are necessary for calculating the three-dimensional effects. However, the models and procedures continue to be clear-cut. The number of discrete axial channels is generally no greater than four.

BWR fluence analyses, like the PWR analyses discussed in Sections 3.1.2 and 3.3.1, begin with "exact" three-dimensional core-follow models in the core region. Reviewing the results from BWR analyses shows that there are many unique axial (z) power shapes associated with normal operation. Not only does the inserted position of the control rods contribute to various distinctive axial shapes, but the degree of boiling also creates unique axial power shapes.

The degree of boiling is a function of the axially integrated power in the channel of each assembly. Each assembly in the core with a different "assembly" power will have a different axial power shape. Due to the many unique "assembly" powers and axial power shapes in the BWR core, collapsing the "exact" three-dimensional model to two-dimensional models for fluence analysis is more complex than discussed in Section 3.3.1. In addition, the coupling of the boiling water density and the axial power shape, along with the control rod position and the axial power shape does not provide an accurate means of axially integrating the water density and control rod effects for a r, θ model. [

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Due to the complexity of collapsing the "exact" three-dimensional model of the core to two-dimensional models for three-dimensional synthesis analysis of the vessel fluence, the best method for calculating the three-dimensional flux (fluence rate) would appear to be an "exact" three-dimensional model (TORT). However, the accuracy of three-dimensional TORT models is very poor. The problem is not the calculational methods; it is numerical limitations associated with the computer. For each of the 67 - BUGLE energy groups, and each of the 1 - million mesh points used in the three-dimensional modeling, there are on the order of 100 - directional flux values. This results in 6 - billion, 700 - million values for the flux (fluence rate) solution that must be iteratively evaluated. Without significant upgrades to commercial workstations, three-dimensional TORT models cannot be effectively utilized for vessel fluence calculations at this time. Therefore, the three-dimensional synthesis model discussed in Section 3.3.1 of the topical needs to be extended for BWR analysis.

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The axial spacing of the planar regions is developed from the "exact" three-dimensional core-follow model. The core-follow model results are used to identify the axial shape functions that best represent the effects of the control rod positions and the degree of channel boiling. The axial spacing of the planar regions is not uniform since inflections in the shape functions do not generally occur in equal increments.

Figure G-2 Schematic of Three-Dimensional Synthesis For BWR Fuel Assemblies

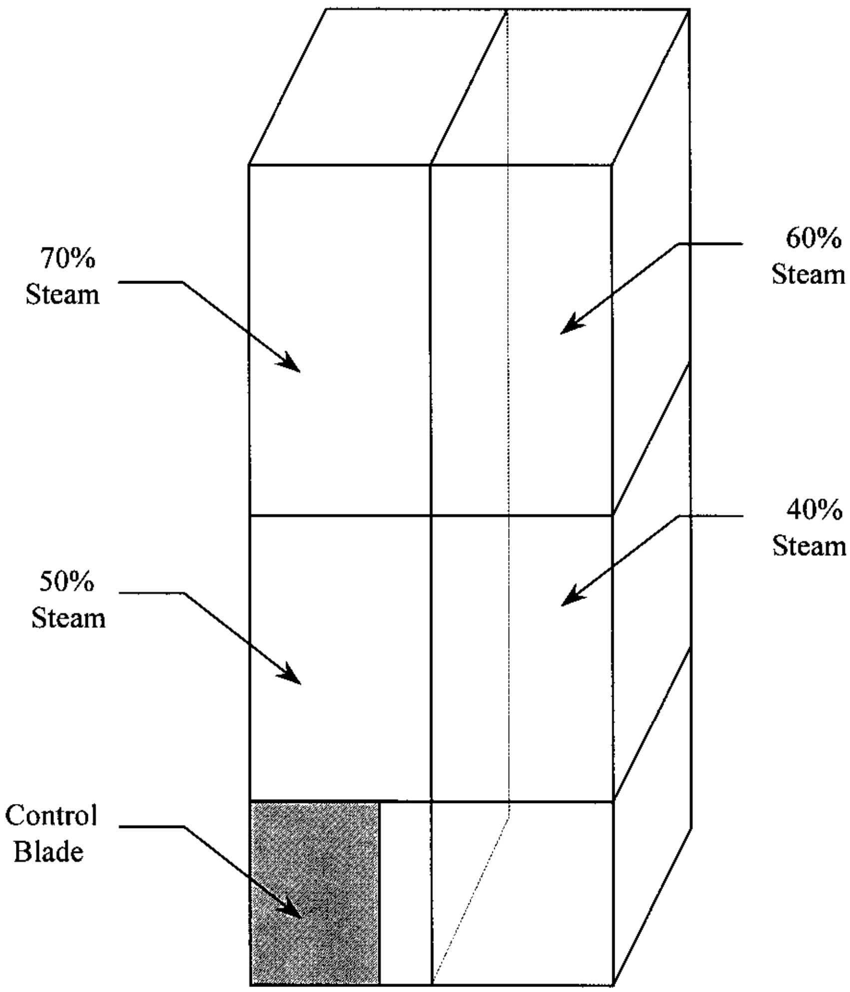

Figure G-2 on the previous page gives a schematic of two boiling water fuel assemblies. The purpose of the schematic is to help explain the extension of the synthesis methods. The schematic is not as detailed as the synthesis model; it only shows three unequally spaced planar regions for the axial mesh spacing rather than seven or more. Each synthesis and schematic planar region represents a combination of x, y or r, θ planar regions from the core-follow model. Viewed from the top of the figure, looking down, the x, y assembly pitch of the radial plane of the core region would be obvious. The combination of axial segments from two or more planar regions in the core-follow model would give one of the assembly segments that is shown in Figure G-2.

$$S^{3D}\left(x, y, \overline{\delta z}, \overline{\mathrm{E}}, \overline{\boldsymbol{\Omega}}\right) = \frac{\int_{\delta z} S^{3D}\left(x, y, z, \overline{\mathrm{E}}, \overline{\boldsymbol{\Omega}}\right) dz}{\int_{\delta z} dz} \tag{G.8}$$

Equation G.8 expresses the integration of the three-dimensional ($3D$) source function (S) for each planar region segment of a fuel rod modeled in the synthesis calculation. In Figure G-2, the Equation G.8 "$3D$" source function is schematically associated with the axial segment of one assembly. To develop the source function for a two-dimensional "$R\theta$" synthesis calculation, a z-dependent multichannel source function (S_C^z) is used as shown by Equation G.9.

[ ]] (G.9)

The x, y or r, θ planar regions in the "$R\theta$" synthesis calculation not only include the Equation G.9 source functions in each axial segment, but the functional weighting of the collision reactions is also included. As discussed above, in Section G.3.1.2 for the "Core Leakage Function", the core-follow time-steps provide a numerical means of integrating the source and collision parameters over the operational periods of interest for the fluence evaluations. [

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To summarize, the Framatome ANP synthesis models and procedures described in Section 3.3.1 of this topical are appropriate for BWR calculations. However, the multichannel - planar models used previously for PWRs need to be expanded for BWRs. The reason for the modeling-procedure extension is the multiple time-dependent, non-separable, axial power shapes, which result from control rod insertion and channel boiling during operation. The shapes from each core-follow time-step are integrated into average time-weighted axial shapes for each assembly. These time-weighted, average assembly shapes provide the basis for the BWR multichannel modeling. The extension of Framatome ANP's models and procedures for BWR synthesis calculations involves more channels than previously evaluated and thereby more calculations to obtain the integrated coupling of the "$R\theta$" planes with piece-wise axial shape functions. [

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The extended models and procedures for synthesis calculations of BWR s is validated in the same manner as the core leakage function (Section G.3.1.2) and the transport of neutrons through the jet pumps (Section G.3.1.1). The DORT results, based on Equation G.4, must agree with the results from calculations that have a defined degree of accuracy, such as those from the core-follow model. If the approximations used in the DORT modeling and analysis procedures are valid, the results from the DORT synthesis will be accurate in comparison to the reference three-dimensional (core-follow) calculations. [

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The methodology presented in Section 3 of this topical has been extended as explained above in Section G.3.1 of this Appendix. The extension includes a more accurate treatment of (1) the transport of neutrons from the core through the internal structures associated with the jet pumps, (2) the integrated core leakage function from the fuel, and (3) the three-dimensional synthesis of the core flux function. With the more accurate treatment, the methodology presented in this Appendix is appropriate for calculating the fluence throughout the internal structures and vessel of BWR s.

G.3.2 Unfolding the Measured Fluence

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The uncertainty evaluations of the Framatome ANP fluence methods identified an energy dependent bias. To provide accurate calculations of the fluence rate (time-averaged flux), a bias removal function was developed in each discrete energy group. This means that when the appropriate modeling approximations are utilized, the fluence results have no bias uncertainty. Thus, the calculated results from an appropriately evaluated methodology are accurate and simply contain a degree of randomness that is quantified by a well-defined statistical uncertainty.

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G.4 Uncertainty Update

This topical report 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 uncertainty methodologies are fundamentally theoretical methods combined with procedural and modeling approximations. The theoretical methods are generic to BWR s and PWR s. While the models and procedures discussed in Section 7 are generic, the results in *Appendix A* are weighted with more B & W plants. The statistical evaluation of the models and procedures was expanded in *Appendix E* to equally weight all PWR plants. This section (*G*.4) extends the discussion of the uncertainty evaluation to all BWR plants.

The information in Section 7 explained the uncertainty evaluations in terms of the (1) measurements, (2) calculations, and (3) benchmark comparisons of the calculations to the measurements. Two types of deviations, systematic and random, characterize these uncertainties. The systematic deviations are caused by inaccurate results with one or more unique biases producing the errors. The random deviations have no specific cause, but a "normal" distribution function and a standard deviation, that are estimated using mathematical statistics, can represent the precision of the overall random uncertainty. The mathematical statistics processing of the distribution of random deviations provides a level of confidence in the precision of the results.

One essential part of the methodology is that all uncertainties must be defined in terms of reference standards that are known to be "true" values. The reference standard for the calculations, and benchmark comparisons of the calculations to the measurements is the measured dosimetry results. However, as explained in the regulatory guide for determining the vessel fluence,[G3] the measured results are not "true" values unless they have been validated by a National Institute of Standards and Technology (NIST) reference field. The NIST reference field validation is more than the usual calibration standards for the experimental equipment. It is an actual quality assurance validation of the measured dosimetry results by a NIST team. The NIST team independently performs the measurements and compares their results to those of the B & W laboratory. Moreover, the NIST team reviews each part of the experimental process. By reviewing each part they determine if any small biases exist, and whether any biases essentially cancelled each other. As explained in Framatome ANP's "Standard and Reference Field Validation" document,[E3] NIST certified the laboratory results to have no statistically significant biases. Thus, the mean value of the measured results is accurate and only varies randomly about the "true" value. NIST also confirmed that the laboratory's estimate of the standard deviation in the random uncertainties provided the appropriate level of confidence in the variation of the mean measurement about the "true" value.

The dosimeters associated with BWR specimen coupons are of the same type and form as those validated by NIST for the B & W laboratory measurements. Consequently, the Framatome ANP evaluation of BWR dosimetry measurements is valid. In fact, based on the information in the *GE* topical,[G2] Framatome ANP has the only "reference field" validated measurement uncertainties for BWR dosimetry.

The Framatome ANP dosimetry measurements have no statistically identifiable bias and have a standard deviation that is not greater than 7.0 %. Equation G.11 represents confirmation that Equation 7.7 (page 7 - 21) and the information in Section 7.1 (page 7 - 7) of the topical applies to BWR s.

$$\boxed{\textit{Mean Measurement Uncertainty} \ \leq \ 7.0\,\%} \qquad (G.11)$$

The uncertainties in the calculational methodology are determined from two evaluations, an analytic sensitivity of parameters affecting the calculations, and a benchmark of the calculated dosimetry results to the measurements. The parameters affecting the solution of Equation G.4 are evaluated using analytic sensitivity calculations of the neutron source, geometry, material composition, and modeling. The statistical combination of the fluence deviations provides an estimate of the standard deviation in the dosimetry reactions and greater than 1.0 MeV vessel fluence. The DORT results from Equation G.4 are compared to the Framatome ANP dosimetry database to statistically evaluate biases, and to iteratively determine a consistent level of confidence in the unbiased calculational uncertainties.

As noted in the "Background" discussion (Section *G*.2), utilities consider an approved topical an important requirement before awarding a fluence contract. However, an approved topical for BWR fluence evaluations should include benchmark comparisons of the calculations to the dosimetry measurements. To be able to have benchmark comparisons of the calculations to the dosimetry measurements, Framatome ANP has developed this topical.

Even without specific benchmarks of BWR dosimetry from specimen coupons, the uncertainty associated with the extended BWR methodology in this Appendix can be estimated. The estimated uncertainty update is based on analytical sensitivity assessments and a review of the benchmarks from Framatome ANP's dosimetry database.

The benchmark of the dosimetry database provides the means of evaluating biases in the calculational methodology and iteratively determining a consistent level of confidence in the unbiased calculational uncertainties. The extended calculational methodology for BWRs involves (1) the transport of neutrons from the core through the internal structures associated with the jet pumps, (2) the integrated core leakage function from the fuel, and (3) the three-dimensional synthesis of the core flux function. Each extension in the methodology was evaluated to insure that no bias in the approximations to the models and procedures could cause inaccurate calculations. The evaluations were based on reference analyses that were known to be accurate in comparison to the methods that were discussed in Section 3 of this topical. Consequently, the extended methods for the models and procedures presented in this Appendix (*G*) are not biased.

Previous bias evaluations associated with the calculations are discussed in Section 7.2.1 of this topical (pages 7 - 27 through 7 - 31). It is noted that not only are the measurements unbiased and highly accurate (Equation G.11), but the mean value of the calculated neutron fluence values is also unbiased. The benchmark comparisons of the calculations to the dosimetry measurements indicate that there are no statistically significant biases associated with the fluence reactions with energies greater than 0.1 MeV. Even though the mean value of the fluence is unbiased, the assessment of an energy dependent bias within the energy range from 0.1 MeV to 17 MeV shows a bias. The development of the bias removal function is discussed in *Appendix D* of this topical report (pages D - 73 through D - 80). The combination of the bias removal function and the information presented above indicates that the BWR calculations have no statistically significant bias (B_C). This is represented by Equation G.12 below.

$$\boxed{B_C\,(Fluence) \;=\; 0.0} \qquad \text{(G.12)}$$

The analytic sensitivity evaluation performed previously for the neutron source and geometry may be extended to the BWR modeling-procedure uncertainties. The BWR extensions for (1) the transport of neutrons from the core through the internal structures associated with the jet pumps, (2) the integrated core leakage function from the fuel, and (3) the three-dimensional synthesis of the core flux function, represent a subset of the previous evaluations. The previous calculations have been updated and extended to specifically treat the (BWR) modeling and procedures described in Sections *G*.3.1.1 through *G*.3.1.3. As noted with the previous analytical uncertainty evaluation, the results of the deviations have no well-defined level of confidence. To obtain the appropriate level of confidence in the analytic sensitivity evaluation, the statistical parameters associated with actual data sets are required. The benchmark of the dosimetry database is used to continue to provide a consistent level of confidence in the unbiased calculational uncertainties from the analytic sensitivity evaluation.

The benchmark reference for the BWR analytical sensitivity modeling is *Appendix E.* Including the *Appendix E* benchmark data, the analytical uncertainty associated with BWR dosimetry calculations is represented by a standard deviation (σ_C) of 7.0 %.

$$\boxed{\sigma_C\,(Analytic) \;=\; [\qquad]} \qquad \text{(G.13)}$$

Based on the consistency between the analytic uncertainty and the benchmark reference, a confidence factor of [ ] provides a 95 % level of confidence in the uncertainty with [ ] degrees of freedom.

[

]

$$\boxed{\sigma_C\,(Dosimetry) \;=\; [\qquad]} \tag{G.14}$$

Using the *Appendix E* evaluation of the dosimetry database for consistency with the statistical parameters, there is a 95% level of confidence that the mean BWR benchmark uncertainty ($\overline{\sigma_{C/M}}$) would not be greater than the statistical combination of the calculational and measurement uncertainties. A confidence factor of [ ] with [ ] degrees of freedom represents the calculational uncertainty; and a confidence factor of [ ] degrees of freedom represents the measurement uncertainty. Equation G.15 gives the estimate of the BWR benchmark uncertainty from the combination of the calculational and measurement uncertainties.

$$\boxed{\overline{\sigma_{C/M}}\;(BWR\ Dosimetry\ Benchmark) \;=\; [\qquad]} \tag{G.15}$$

With [ ] degrees of freedom representing the calculational uncertainty, any one comparison of dosimetry calculations to measurements could have a mean random deviation in the $C/_M$ ratio of [

]

Section 7.3 (page 7 - 36) in this topical explains how the standard deviations from the analytic sensitivity evaluation were estimated to be consistent with the benchmark database and combined with the standard deviations in the calculations to estimate the vessel fluence standard deviation. Equation 7.22 forms part of the basis to insure consistency between the analytic and benchmark uncertainties, and Equation 7.23 gives the combined standard deviation for the vessel. The uncertainty in Equation G.14 is sufficient to represent the fluence uncertainty at dosimetry locations as shown by Equation 7.20. Utilizing Equations 7.22 and 7.23, the vessel fluence uncertainty is that shown by Equation G.16.

$$\boxed{\sigma_C\,(BWR\ Vessel\ Fluence) \ = \ [\qquad]} \qquad (G.16)$$

The vessel fluence uncertainty, represented by the Equation G.16 standard deviation, is consistent with [ ] providing a 95% level of confidence that vessel fluence - embrittlement predictions will be within the uncertainty of the embrittlement database.

The Framatome ANP uncertainties associated with BWR dosimetry measurements and calculations are unbiased (Equation G.12) and have well-defined standard deviations for the appropriate levels of confidence. The Framatome ANP results from the laboratory measurements appear to be the only ones for a BWR with NIST reference field validation. The measured standard deviation has been validated to be less than 7.0%. The extended models and procedures discussed in Sections *G*.3.1.1 through *G*.3.1.3 have an estimated dosimetry uncertainty from analytic sensitivity evaluations that is not greater than [ ] The combination of this uncertainty and the calculational uncertainty from the *Appendix E* dosimetry benchmark evaluation gives a dosimetry standard deviation of [ ] The combination of calculational and measurement uncertainties gives a benchmark standard deviation of [ ] This benchmark standard deviation needs confirmation with independent benchmark comparisons of BWR dosimetry from specimen coupons.

The analytic sensitivity evaluation for the vessel uncertainty is not greater than [ ] Combining the analytic vessel standard deviation in a consistent manner with the uncertainty in the calculations for the dosimetry benchmarks indicates the vessel standard deviation is not greater than [ ] The uncertainty in the vessel fluence calculations needs to be less than 20.0% to be consistent with vessel embrittlement evaluations. Clearly the vessel value of [ ] meets the criterion. Therefore, the

Framatome ANP BWR fluence methods and corresponding uncertainties are sufficient for BWR fluence - embrittlement analyses.

Appendix G References

G1. 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."

G2. S. Sitaraman, et al, "Licensing Topical Report, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations" *GE* Nuclear Energy, Document # NEDO-32983-A, Revision 0, December, 2001.

G3. Office of Nuclear Regulatory Research, "Calculational And Dosimetry Methods For Determining Pressure Vessel Neutron Fluence", U.S. Nuclear Regulatory Commission, Regulatory Guide 1.190, March, 2001.