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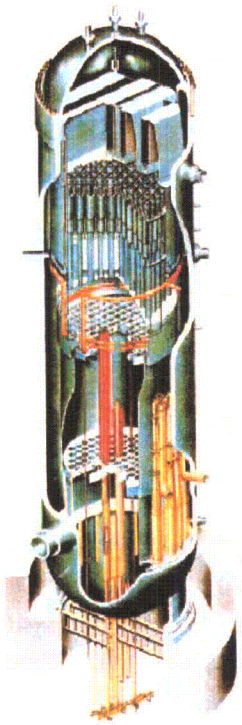
BWRVIP-117NP-A: BWR Vessel and Internals Project

RAMA Fluence Methodology Plant Application-Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1-5



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BWRVIP-117-A: BWR Vessel and Internals Project

RAMA Fluence Methodology Plant Application—Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1–5

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Final Report, December 2009

EPRI Project Manager
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NRC SAFETY EVALUATION

In accordance with an NRC request, the NRC Safety Evaluation immediately follows this page. Other NRC and BWRVIP correspondence on this subject are included in appendices.

Note: The changes proposed by the NRC in this Safety Evaluation as well those proposed by the BWRVIP in response to NRC Requests for Information have been incorporated into the current version of the report (BWRVIP-117-A).

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

May 13, 2005

Bill Eaton, BWRVIP Chairman
Entergy Operations, Inc.
Echelon One
1340 Echelon Parkway
Jackson, MS 39213-8202

SUBJECT: SAFETY EVALUATION OF PROPRIETARY EPRI REPORTS, "BWR VESSEL AND INTERNALS PROJECT, RAMA FLUENCE METHODOLOGY MANUAL (BWRVIP-114)," "RAMA FLUENCE METHODOLOGY BENCHMARK MANUAL-EVALUATION OF REGULATORY GUIDE 1.190 BENCHMARK PROBLEMS (BWRVIP-115)," "RAMA FLUENCE METHODOLOGY-SUSQUEHANNA UNIT 2 SURVEILLANCE CAPSULE FLUENCE EVALUATION FOR CYCLES 1-5 (BWRVIP-117)," AND "RAMA FLUENCE METHODOLOGY PROCEDURES MANUAL (BWRVIP-121)," AND "HOPE CREEK FLUX WIRE DOSIMETER ACTIVATION EVALUATION FOR CYCLE 1 (TWE-PSE-001-R-001)" (TAC NO. MB9765)

Dear Mr. Eaton:

By letters dated June 11, 2003, June 26, 2003, August 5, 2003, October 29, 2003, and March 24, 2004, respectively, the Boiling Water Reactor Vessel and Internals Project (BWRVIP) submitted the following Electric Power Research Institute (EPRI) proprietary reports for staff review and approval, "BWR Vessel and Internals Project, RAMA Fluence Methodology Manual (BWRVIP-114)," "RAMA Fluence Methodology Benchmark Manual-Evaluation of Regulatory Guide 1.190 Benchmark Problems (BWRVIP-115)," "RAMA Fluence Methodology-Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1-5 (BWRVIP-117)," "RAMA Fluence Methodology Procedures Manual (BWRVIP-121)," and "Hope Creek Flux Wire Dosimeter Activation Evaluation for Cycle 1 (TWE-PSE-001-R-001)."

The reports listed above provide and support a methodology which is a new approach to neutron transport that has been developed by the BWRVIP for determining neutron fluence to the reactor pressure vessel (RPV) and internal components of BWR plants. The Radiation Analysis Modeling Application (RAMA) code will be applied in the reactor beltline region defined by the top and bottom planes of the active fuel and the inner wall of the biological shield. The methodology employs the RAMA computer code for evaluating the neutron flux from the core through the downcomer, vessel internals, and through the RPV wall.

The staff has completed its review of the proposed methodology and finds that the methodology performs as described; however, the BWRVIP did not quantify the bias and uncertainty required for the qualification of the methodology, as stated in RG 1.190, "Radiation Embrittlement of Reactor Vessel Materials." Therefore, the staff's approval is conditional based on the following criteria: (1) for plants that are similar in core, shroud and downcomer-vessel geometry to that of the Susquehanna and Hope Creek plants, the RAMA methodology can be applied without a bias for the calculation of vessel neutron fluence, (2) for plants (or plant groups) with a different geometry than that of the Susquehanna or Hope Creek plants, a plant-specific application for RPV neutron fluence is required to establish the value of a bias, and (3) relevant benchmarking will be required for shroud and reactor internals applications.

The staff evaluation of the proposed RAMA methodology is attached. Please contact Meena Khanna of my staff at 301-415-2150 if you have any further questions regarding this subject.

Sincerely,

Martha A. Mitchell for

William H. Bateman, Chief
Materials and Chemical Engineering Branch
Division of Engineering
Office of Nuclear Reactor Regulation

Enclosure: As stated

cc: BWRVIP Service List

U.S. NUCLEAR REGULATORY COMMISSION OFFICE OF NUCLEAR REACTOR REGULATION SAFETY EVALUATION OF BWR VESSEL AND INTERNALS PROJECT, SAFETY EVALUATION OF PROPRIETARY EPRI REPORTS, "BWR VESSEL AND INTERNALS PROJECT, RAMA FLUENCE METHODOLOGY MANUAL (BWRVIP-114)," "RAMA FLUENCE METHODOLOGY BENCHMARK MANUAL-EVALUATION OF REGULATORY GUIDE 1.190 BENCHMARK PROBLEMS (BWRVIP-115)," "RAMA FLUENCE METHODOLOGY-SUSQUEHANNA UNIT 2 SURVEILLANCE CAPSULE FLUENCE EVALUATION FOR CYCLES 1-5 (BWRVIP-117)," "RAMA FLUENCE METHODOLOGY PROCEDURES MANUAL (BWRVIP-121)," AND "HOPE CREEK FLUX WIRE DOSIMETER ACTIVATION EVALUATION FOR CYCLE 1 (TWE-PSE-001-R-001)"

1.0 INTRODUCTION

1.1 Background

By letters dated June 11, 2003, June 26, 2003, August 5, 2003, October 29, 2003, and March 23, 2004, respectively, the Boiling Water Reactor Vessel and Internals Project (BWRVIP) submitted the following Electric Power Research Institute (EPRI) proprietary reports for staff review and approval, "BWR Vessel and Internals Project, RAMA Fluence Methodology Manual (BWRVIP-114)," "RAMA Fluence Methodology Benchmark Manual-Evaluation of Regulatory Guide 1.190 Benchmark Problems (BWRVIP-115)," "RAMA Fluence Methodology-Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1-5 (BWRVIP-117)," "RAMA Fluence Methodology Procedures Manual (BWRVIP-121)," and "Hope Creek Flux Wire Dosimeter Activation Evaluation for Cycle 1 (TWE-PSE-001-R-001)." These reports were supplemented by letter dated September 20, 2004, in response to the staff's request for additional information (RAI) dated April 20, 2004.

The BWRVIP-114 report describes the theory of the neutron transport calculation methodology and the uncertainty analysis. The BWRVIP-115 report documents benchmarking of the neutron fluence calculation methodology against two reactor pressure vessel (RPV) simulator measurements, a PWR surveillance capsule measurement and a calculational benchmark. The BWRVIP-117 and TWE-PSE-001-R-001 reports present plant-specific surveillance capsule neutron fluence benchmark comparisons for the Susquehanna and Hope Creek plants, respectively. The BWRVIP-121 report provides the standard procedures for carrying out neutron fluence calculations using this methodology.

The proposed methodology is essentially a new approach that has been developed by the BWRVIP for determining the fast ($E \geq 1.0$ MeV) neutron fluence accumulated by the RPV and internal components of BWR plants. The methodology employs the RAMA computer code for evaluating the neutron flux from the core through the downcomer, vessel internals and through the RPV wall. An important feature of the methodology is that the neutron transport calculation is 3-dimensional, rather than a synthesis of two 2-dimensional calculations that is used in the finite differences method on which presently approved methodologies are based. An additional feature of this approach is that the computer modeling of the physical geometry is represented without approximation. The RAMA code will be applied in the reactor beltline region defined by the top and bottom planes of the active fuel and the inside surface of the biological shield. The methodology employs the most recent BUGLE-96 nuclear transport and reaction-specific

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measured activity cross section data. The BWRVIP calculation and uncertainty methodology is summarized in Section 2. The technical evaluation is presented in Section 3, and the limitations and conclusions are provided in Section 4.

1.2 Purpose

The staff reviewed the reports discussed above to determine whether the BWRVIP's proposed methodology will provide an acceptable method for determining the fast ($E \geq 1.0$ MeV) neutron fluence accumulated by the RPV and internal components of BWR plants.

1.3 Regulatory Evaluation

The basis for this review is Regulatory Guide (RG) 1.190, "Radiation Embrittlement of Reactor Vessel Materials." RG 1.190 is based on General Design Criterion (GDC) 14, 30 and 31, and describes the attributes of neutron transport methodologies which are acceptable to the staff. The basic feature of an acceptable methodology is that the code is benchmarked by acquiring and evaluating a statistically significant database of measurement-to-calculation ratios and the resulting bias and uncertainty are within certain limits.

2.0 SUMMARY OF THE EPRI BWRVIP VESSEL NEUTRON FLUENCE METHODOLOGY

2.1 RPV Neutron Fluence Calculation Methodology

The BWRVIP neutron fluence calculational methodology employs the RAMA code to evaluate the neutron flux through the core, vessel internals, and vessel geometry. The code uses the BUGLE-96 cross-section library to calculate the neutron transport and to determine the reaction-specific measured activities. The RAMA code employs a combinatorial geometry method which allows an exact representation of geometrically complex components. This is accomplished by building the desired internal component using various primitive geometry elements (Ref. 8).

The neutron transport calculation is based on the following: (1) the three-dimensional transport equation is integrated by attenuating the neutron fluence along discrete rays according to the macroscopic cross-section and optical path in the intersected region, (2) a set of parallel rays are chosen in both a radial and axial plane and the neutron fluence is determined on this grid, (3) to account for the various possible directions of particle transport, rays are defined on a discrete set of angular quadratures, and (4) anisotropic scattering is treated using a Legendre expansion of the neutron scattering cross-section.

The neutron source is determined based on the core power density and the region-wise power distribution. The RAMA source accounts for the exposure dependence of the core neutron source and allows for a detailed pin power description of the source distribution. Typically, reflective boundary conditions are applied on the planes that define the angular sector of the geometry being calculated (typically, a core octant or quadrant), and vacuum boundary conditions are applied at the outer radial boundary (e.g., the outside wall of the RPV) and on upper and lower axial boundaries.

In order to facilitate comparisons of measurements to calculated values (as instructed by RG 1.190), RAMA calculates the corresponding quantities for the measured reaction rates. RAMA

determines the time-dependent neutron flux and tracks the target and reaction product nuclides.

The RAMA methodology includes a detailed neutron fluence uncertainty analysis. The parameters making a significant contribution to the neutron fluence calculation uncertainty are identified and RAMA is used to determine numerical sensitivity coefficients for these parameters. The uncertainty contribution from these parameters is determined by combining the numerical sensitivities with the estimates of the input parameter uncertainties. When making comparisons to benchmark measurements, the calculation-to-measurement (C/M) differences are combined using a covariance matrix to determine the uncertainty contribution from the measurements. The overall calculation uncertainty and bias are determined based on the C/M differences and the calculation input parameter uncertainties.

2.2 Calculation of the RPV Benchmarks

In validating the RAMA methodology, comparisons of RAMA predictions were performed for the following four benchmarks: (1) the Oak Ridge National Laboratory (ORNL) Pool Critical Assembly (PCA) benchmark experiment (Ref. 9), (2) the VENUS-3 engineering benchmark experiment (Ref. 10), (3) the H. B. Robinson-2 (HBR-2) RPV benchmark measurement (Ref. 11), and (4) the Brookhaven National Laboratory (BNL) RPV calculation benchmark of NUREG-6115 (Ref. 12). The PCA and VENUS-3 experiments are well-documented RPV mock-ups, including high accuracy dosimetry measurements. The PCA core includes twenty-five material test reactor (MTR) curved-plate type fuel assemblies and the simulator geometry includes a thermal shield, RPV, and void box outside the RPV. The PCA dosimetry measurements were made at positions in front and behind the thermal shield, at locations in front and behind the RPV, and at RPV internals locations. The PCA dosimetry measurements include the Np-237 (n, f), U-238 (n, f), In-115 (n, n'), Ni-58 (n, p) Co-58 and Al-27 (n, α) Na-24 reactions. The RAMA model is 3-dimensional and includes a radial quadrant of the PCA geometry, the full height of the core and the regions above and below the core. Detailed comparisons presented for both the thermal shield (or core shroud) and RPV locations indicate good agreement with the dosimetry measurements.

The VENUS-3 core consists of twelve 15x15 pressurized water reactor (PWR) fuel assemblies and the simulator geometry includes the baffle, core barrel, neutron pad and RPV simulator. The VENUS-3 dosimetry measurements include the Ni-58 (n, p) Co-58, In-115 (n, n'), and Al-27 (n, α) Na-24 reactions. The RAMA model is 3-dimensional and includes a radial quadrant of the simulator geometry, the full height of the core, and the regions above and below the core. Detailed comparisons are presented for the core, baffle, and core barrel and indicate good agreement with the measurements.

The HBR-2 benchmark experiment provides a well-documented set of dosimetry measurements for a full-height operating PWR, including core barrel, thermal shield and RPV. The HBR-2 dosimetry measurements include Np-237 (n, f), U-238 (n, f), Ni-58 (n, p) Co-58, Fe-54 (n, p) Mn-54, Ti-46 (n, p) Sc-46 and Cu-63 (n, α) Co-60. The measurements were made at an in-vessel capsule and at a cavity location. The HBR-2 RAMA model is 3-dimensional and provides a detailed representation of an octant of the problem geometry for a centrally-located axial region of the core. The model extends from the center of the core out to the outer surface of the biological shield. Detailed comparisons are presented for both the in-vessel surveillance capsule and the cavity measurements, and indicate good agreement with the measured data.

BNL NUREG-6115 provides the detailed specification and corresponding numerical solutions for a BWR RPV neutron fluence benchmark problem. The benchmark problem provides a reference calculation for a configuration that is typical of an operating BWR which includes the downcomer and RPV neutron 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) Co-58, Fe-54 (n, p) Mn-54, Ti-46 (n, p) Sc-46, and Cu-63 (n, α) Co-60 reaction rates. The RAMA model is 3-dimensional and provides a detailed representation of an octant of the problem geometry over an axial region that includes the core as well as the regions above and below the core. The model extends from the center of the core out to the outer surface of the biological shield. Detailed comparisons are presented for both the RPV neutron fluences and the dosimetry reaction rates. The surveillance capsule comparisons indicate good agreement for all reaction rates. The downcomer and RPV neutron fluence comparisons indicate that RAMA is conservative relative to the reference solution.

2.3 Calculation of the Susquehanna Neutron Fluence Measurements

As part of the RAMA plant-specific qualification, RAMA transport calculations have been performed for the Susquehanna Unit 2 surveillance capsule that was removed at the end of Cycle 5. In order to validate the fast ($E \geq 1.0$ MeV) neutron fluence evaluations of the Susquehanna RPV, comparisons of the calculated and measured neutron fluence have been made to determine the neutron fluence calculational uncertainty and to identify any systematic bias in the neutron fluence predictions. The Cycle 5 surveillance capsule was located in the downcomer, radially at a position close to the innerwall of the RPV, and azimuthally 30° from the core flats. The surveillance capsule included three each of the following dosimeter wires: copper, nickel, and iron. The measured activities included the Cu-63 (n, α) Co-60, Ni-58 (n, p) Co-58, and Fe-54 (n, p) Mn-54 dosimetry reactions. The measurements were of high quality and were reported to have uncertainties on the order of a few percent.

The RAMA calculational model was based on detailed plant data provided by the Pennsylvania Power and Light (PPL) Company. The geometry data were taken from plant drawings and used to model the surveillance capsule and various core, core shroud, jet pump/riser and RPV components. RAMA provided a geometry model of high accuracy in which both the Cartesian geometry of the core boundary and the cylindrical geometry of the jet pump/riser components were represented without approximation. The RAMA model included a one-eighth (45°) azimuthal sector and the radial geometry from the center of the core out to the inner wall of the biological shield.

The core neutron source was based on the Susquehanna Cycles 1-5 operating history. Three-dimensional power, void and exposure distributions were constructed from the plant operating history files. The pin-wise gradient and exposure dependence of the neutron source for the fuel assemblies on the core periphery were included. Each cycle was described by a representative set of operating state-points. The neutron fluence accumulated by the capsule dosimeters was

determined by an appropriate weighting of the RAMA state-point calculations. An extensive set of sensitivity calculations was also performed to ensure the stability and convergence of the numerical solution.

RAMA calculations of the dosimeter activities were performed and compared with the measurements (dps/g). The average C/M overall measurement was found to be very close to unity indicating that there is no significant bias in the RAMA neutron fluence predictions. The standard deviation of all C/M values was less than 20% as recommended in RG 1.190 (Section 1.4.3). In order to provide an independent assessment of the accuracy of the RAMA neutron fluence prediction, a detailed analytic uncertainty analysis was also performed. The important input parameter uncertainties were identified and an estimate of the uncertainty in each parameter was determined. The uncertainty in each parameter was propagated through the RAMA calculation using numerical sensitivity calculations. The resultant analytic estimate of the RAMA neutron fluence calculation uncertainty, corresponding to the observed C/M standard deviation, was also shown to be less than 20%.

2.4 Calculation of the Hope Creek Neutron Fluence Measurements

RAMA transport calculations were performed for the surveillance capsule removed from the Hope Creek RPV at the end of the first cycle. In order to validate the fast ($E \geq 1.0$ MeV) neutron fluence evaluations of the RPV, comparisons of the calculated and measured neutron fluence have been made to determine the neutron fluence calculational uncertainty and to identify any systematic bias in the neutron fluence predictions. The first cycle surveillance capsule was located in the downcomer, radially at a position close to the innerwall of the RPV, and azimuthally at 33° from the core flats. It is noted that two additional capsules are located at 121° and 299°. The surveillance capsule included three copper and three iron flux wires. The measured activities included the Cu-63 (n, α) Co-60 and Fe-54 (n, p) Mn-54 dosimetry reactions. The measurements were reported to have uncertainties on the order of a few percent. The copper activity was corrected for the presence of Co-59 impurity of about 0.25 parts per million (ppm).

The RAMA calculational model was based on detailed plant data. The geometry data were taken from plant drawings and used to model the surveillance capsule, the core, core shroud, jet pump/riser, and RPV components. RAMA provided a geometry model of high accuracy in which both the Cartesian geometry of the core boundary and the cylindrical geometry of the jet pump/riser components were represented without approximation. The RAMA model included a one-eighth (45°) azimuthal sector and the radial geometry from the center of the core to the biological shield.

The core neutron source was based on the first cycle's operating history. Three-dimensional power, void, and exposure distributions were constructed from the plant operating history files. The pin-wise gradient and exposure dependence of the neutron source for the fuel assemblies on the core periphery were included. The neutron fluence accumulated by the capsule dosimeters was determined by an appropriate weighting of the RAMA state-point calculations. An extensive set of sensitivity calculations was also performed to ensure the stability and convergence of the numerical solution.

RAMA calculations of the dosimeter activities were performed and compared with the measurements (dps/gm). The average C/M overall measurement was found to be very close to

unity indicating that there is no significant bias in the RAMA neutron fluence predictions. The standard deviation of all C/M values was less than 20% as recommended in RG 1.190 (Section 1.4.3). In order to provide an independent assessment of the accuracy of the RAMA neutron fluence prediction, a detailed analytical uncertainty analysis was also performed. The important input parameter uncertainties were identified and an estimate of the uncertainty in each parameter was determined. The uncertainty in each parameter was propagated through the RAMA calculation using numerical sensitivity calculations. The resultant analytical estimate of the RAMA neutron fluence calculation uncertainty, corresponding to the observed C/M standard deviation, was also shown to be less than 20%.

3.0 TECHNICAL EVALUATION

The staff's review of the BWRVIP neutron fluence methodology focused on the details of the application of the neutron fluence calculation methodology and the qualification of the methodology provided by the benchmark comparisons and the plant-specific C/M database.

3.1 RPV Neutron Fluence Calculation Methodology

In the RAMA transport calculation, the neutron flux is determined by summing the contributions from a set of particle ray tracings through the problem geometry. The accuracy of this technique depends on the specific problem geometry, as well as the number and distribution of the rays used to track the neutrons through the geometry. In addition, the components that are associated with the problem geometry are represented with a discrete set of spatial regions (i.e., a spatial mesh). Because the neutron flux is averaged over these regions, a mesh-related uncertainty is introduced into the calculation. Since both of these numerical uncertainties are sensitive to the problem geometry, they require an evaluation that accounts for the geometry.

By letter dated April 20, 2004, the staff requested that the BWRVIP address the specific tests and criteria used to assure the adequacy of the number of rays and volumes used in the RAMA neutron fluence calculations for plant-specific applications. By letter dated September 29, 2004, the BWRVIP indicated that in plant-specific model applications of the RAMA fluence methodology, numerical sensitivity calculations will be performed to assure the adequacy of the number of particle tracking rays and the number of volumes used to represent component geometry in the RAMA neutron fluence evaluations. The staff found this approach acceptable.

The RAMA geometry model represents the individual components and regions of the problem geometry using a library of pre-calculated geometry elements. The modeling of the reflector region surrounding the core is particularly complicated in that it involves geometry elements that have both planar and cylindrical side boundaries. However, RAMA provides an exact representation of the true geometry (i.e., preserves the exact location, orientation and shape of all surfaces defining the physical geometry). For example, in the case of these reflector regions, the BWRVIP indicated in its letter dated September 29, 2004, that the geometry model allows for complex geometries, including the transition between the rectangular core and the cylindrical core shroud, to be precisely represented.

The RAMA code has the necessary mechanisms for geometrical representation, neutron scattering and neutron transport approximations. Therefore, the staff finds the RAMA code acceptable, based on its structural features.

3.2 Calculation of the RPV Benchmarks

The RPV benchmark calculations are performed to evaluate the accuracy of RAMA and to identify any systematic bias in the proposed licensing methodology. In order for the benchmark comparisons to reflect the difference between the benchmark and the proposed methodology, the methods used in the benchmark calculations must be the same as the proposed licensing methods. By letter dated April 20, 2004, the staff requested that the BWRVIP identify the differences between the methods used in performing the RAMA benchmark analyses in the BWRVIP-115 report and the methods that will be used in performing the calculations of the RPV and core shroud neutron fluence. By letter dated September 29, 2004, the BWRVIP indicated that the methods used in performing the RAMA benchmark analyses are the same as the methods that will be used in performing BWR RPV and core shroud neutron fluence calculations. The staff found this acceptable in that there would be no inconsistencies in the methods used.

The BWRVIP-115, BWRVIP-117, and TWE-PSE-001-R-001 reports present the RAMA analysis of a set of simulator calculations and operating reactor benchmarks which provide the basis of the Susquehanna and Hope Creek applications of the RAMA neutron fluence methodology. However, it is expected that as additional surveillance capsules are removed, new benchmark C/M data will become available. RG 1.190 requires that as new measurements become available, they shall be incorporated into the C/M database and the neutron fluence calculational bias and uncertainty estimates shall be updated as necessary.

By letter dated April 20, 2004, the staff requested that the BWRVIP address how it will ensure that new measurements are incorporated in the C/M database and that the neutron fluence bias and uncertainty will be updated in a timely manner. In its response by letter dated September 29, 2004, the BWRVIP stated that comparisons to measured surveillance capsule and benchmark dosimetry are maintained in a database that is updated as additional plant capsule evaluations are performed using the RAMA methodology. In addition, the BWRVIP stated that currently, TransWare Enterprises, Inc. (a primary contractor to the BWRVIP) maintains a surveillance capsule and benchmark dosimetry measurement database. The BWRVIP further stated that it would consider options of establishing a mechanism to collect and evaluate new C/M data. Based on the above, the staff found the BWRVIP's response acceptable.

The staff's review of this section established that the RAMA methodology is applied to the benchmarks in the same manner (approximations, cross-sections, etc.) as applied in plant-specific applications, therefore, the staff is in agreement that if a bias exists in the proposed code, it should appear in the benchmarks.

3.3 Results of the Susquehanna Dosimetry Measurements

The Susquehanna, Unit 2 surveillance capsule contained three of each of the following dosimeter wires; copper, iron and nickel. The RAMA calculated ratios and the corresponding measured specific activity (dps/g) C/M ratios are close to unity and display very good agreement. The individual ratios are well within the 20% limit specified in RG 1.190. In addition, the standard deviation is just a few percent.

In accordance with the guidance in RG 1.190, the BWRVIP-117 report includes an analytical neutron fluence uncertainty analysis. This analysis is important since it provides an independent estimate of the plant-specific Susquehanna RAMA neutron fluence calculational uncertainty. The uncertainty analysis requires that estimates of the major components of the uncertainty be determined and the uncertainties be propagated through the RAMA neutron fluence calculation. The uncertainty propagation is performed using numerical component sensitivity as calculated by RAMA. The important uncertainty components have been identified and include the following: (1) capsule and flux wire locations, (2) RPV inner radius, (3) core void fraction, (4) peripheral bundle power, and (5) iron cross-sections. In order to make an accurate determination of the RAMA uncertainty, reliable estimates of the component uncertainties are required.

By letter dated April 20, 2004, the staff requested that the BWRVIP discuss the basis for the parameter uncertainty for the components/locations listed above. In its letter dated September 29, 2004, the BWRVIP indicated that the uncertainty estimates for these components/locations is based on the following: (1) as-built measurements, (2) design drawing tolerances,
(3) experience estimates of ±5% variation in computed void fraction, (4) reported accuracy of core simulation analysis, and (5) experience estimates of ±5% in the cross section, respectively. In addition, the staff noted that Table 5-3 of the BWRVIP-117 report provided the values of the calculated bias and total uncertainty. The BWRVIP also displayed the calculation of the total uncertainty and bias from the C/M and the analytic uncertainty with weighting factors inversely proportional to the analytic and C/M variances in the BWRVIP-117 report. The staff finds the BWRVIP's response to the staff's request for additional information and the values of the bias and uncertainty, as provided in the BWRVIP-117 report, acceptable because the values are well within the limits set forth in RG 1.190.

3.4 Results of the Hope Creek Dosimetry Measurements

The Hope Creek surveillance capsule contained three copper dosimeter wires and three iron dosimeter wires. The surveillance capsule was irradiated during the first cycle for 377.9 effective full power days. The RAMA code calculated the specific dosimeter activity to the corresponding measured specific activity (dps/g). The C/M ratios are close to unity and displayed very good agreement. The individual dosimeter ratios are well within the 20% limit, as specified in RG 1.190, and the standard deviation is just a few percent. However, it was noted that unlike the Susquehanna case, the Hope Creek calculation does not include an analytical uncertainty and bias calculation.

4.0 CONCLUSION

4.1 BWR RPV Neutron Fluence

Based on the staff's review of the BWRVIP-114, -115, -117, and -121 reports, the TWE-PSE-001-R-001 report, and the supporting documentation, the staff concludes that the BWRVIP methodology, as described in these reports, provides an acceptable best-estimate plant-specific prediction of the fast ($E \geq 1.0$ MeV) neutron fluence for BWR RPVs. This acceptance is limited to the axial region defined by the core active fuel height. The best-estimate RPV neutron fluence prediction is determined using the RAMA transport code, detailed plant-specific geometry, core operating history, and the BUGLE-96 nuclear data library with a minimum of a P_3 Legendre polynomial approximation in the iron inelastic scattering.

With respect to the calculation of BWR RPV neutron fluence, the staff concludes that based on the plant-specific benchmark data presently available, no calculational bias is required for the application of the methodology to plants of similar geometrical design to Susquehanna and Hope Creek, i.e., BWR-IV plants. However, in order to provide continued confidence in the proposed neutron fluence methodology for the BWR RPVs, the acceptance of this methodology is subject to the following conditions for plants which do not have geometries similar to the cited BWR-IV's:

- To apply the RAMA methodology to plant groups which have geometries that are different than the cited BWR-IV's, at least one plant-specific capsule dosimetry analysis must be provided to quantify the potential presence of a bias and assure that the uncertainty is within the RG 1.190 limits

  and

- Justification is necessary for a specific application based on geometrical similarity to an analyzed core, core shroud, and RPV geometry. That is, a licensee who wishes to apply the RAMA methodology for the calculation of RPV neutron fluence must reference, or provide, an analysis of at least one surveillance capsule from a RPV with a similar geometry.

4.2 Reactor Internals

EPRI's stated objective for this submittal included neutron fluence calculations for reactor internals. Neutron fluence values for reactor internal components are used to either quantify irradiation assisted stress corrosion cracking (IASCC) susceptibility, or to quantify helium formation which could affect the weldability of reactor internals components. IASCC depends on fast ($E \geq 1.0$ MeV) neutron fluence, while helium formation is a function of thermal, epithermal, and fast neutron fluence. The calc ulational accuracy requirements for reactor internals are not the same as those for the RPV, and are not covered by the guidance in RG 1.190. In addition, the submittal does not include any benchmarking for reactor internals' neutron fluence calculations. Therefore, the staff will review qualification of RAMA for reactor internals applications on a case-by-case basis, based on consideration of C/M values and the associated accuracy requirements.

Licensees who wish to use the RAMA methodology for the calculation of neutron fluence at reactor internals locations must reference, or provide, an analysis which adequately benchmarks the use of the RAMA methodology for uncertainty and calculational bias based on the consideration of: (1) the location at which the neutron fluence is being calculated, (2) the geometry of the reactor, and, (3) the accuracy required for the application. In addition, if a licensee qualifies RAMA for calculating, for example, helium generation at one location (e.g., the core shroud), this qualifies RAMA for the same reactor and purpose at other reactor internals locations (e.g., at the location of the jet pumps).

4.3 Assembling a Statistically Significant Database

EPRI stated that efforts are underway to assemble a database which will enable the staff to remove any limitations placed on the use of the RAMA methodology. For such an effort to be successful, the staff expects that the neutron fluence uncertainty analysis and determination of the calculational bias for the relevant fleet of plants will be updated, as additional measurements are taken and as additional data become available. The results of the updated analysis, including the C/M ratios, should be submitted to the staff for review and approval.

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CITATIONS

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BWRVIP-117: BWR Vessel and Internals Project, RAMA Fluence Methodology Plant Application—Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1–5. EPRI, Palo Alto, CA: 2003. *Technical Report 1008065, prepared by TransWare Enterprises Inc., principal investigator Ken Watkins.*

PRODUCT DESCRIPTION

This report describes the results of a surveillance capsule fluence evaluation performed for the Susquehanna Unit 2 reactor at the end of cycle 5. This evaluation was performed to qualify the Radiation Analysis Modeling Application (RAMA) Fluence Methodology for use in the evaluation of neutron fluence in boiling water reactors (BWRs). A previous version of this report was published as BWRVIP-117 (1008065). This report (BWRVIP-117-A) incorporates changes proposed by the BWR Vessel and Internals Project (BWRVIP) in response to the U.S. Nuclear Regulatory Commission (NRC) Requests for Additional Information, recommendations in the NRC Safety Evaluation (SE), and other necessary revisions identified since the previous publication of the report. All changes to the report except corrections to typographical errors are marked with margin bars. In accordance with an NRC request, the report number includes an "A," indicating the version of the report accepted by the NRC staff.

Results and Findings

The total average calculated-to-measured (C/M) results of specific activities for all flux wires was determined to be 0.98 with a standard deviation of ±8%. These C/M ratios are in very good agreement, indicating that the RAMA Fluence Methodology is accurately predicting fluence and flux.

The total capsule neutron fluence analytic uncertainty is 15.0% for energy >1.0 MeV and 15.0% for energy >0.1 MeV. The largest source of the capsule neutron fluence analytic uncertainty is attributable to the geometry parameters, with the reactor pressure vessel (RPV) inner radius dimension having the single highest uncertainty of 10.0% for energy >1.0 MeV. By combining the measurement uncertainty and analytic uncertainty, the combined capsule fluence uncertainty is determined to be 17.2% for energy >1.0 MeV and 17.2% for energy >0.1 MeV. Therefore, the RAMA Fluence Methodology produces accurate results that compare very well with measured data.

Challenges and Objectives

A key aspect of this work was to ensure that the RAMA methodology adheres to the requirements set forth in NRC Regulatory Guide 1.190 for the determination of neutron fluence in a BWR. To accomplish this, the RAMA methodology was applied to compare and validate the accumulated fluence of an actual surveillance capsule in an operating BWR.

Applications, Value, and Use

The RAMA Fluence Methodology software package determines neutron fluence in BWR components in compliance with the requirements and guidelines provided in NRC Regulatory Guide 1.190. It has been demonstrated that RAMA, Version 1.0, can calculate the fluence for surveillance capsules, the RPV within the active fuel height, and the core shroud within the active fuel height. Future versions of RAMA will be developed to extend the methodology to other internal components that are beyond the active fuel height.

EPRI Perspective

Accurate neutron fluence determinations for BWRs are required for a number of reasons:

- To determine neutron fluence within the RPV and at surveillance capsule locations to address vessel embrittlement issues
- To determine neutron fluence on the core shroud in order to determine fracture toughness and crack growth rate for use in flaw evaluation calculations
- To determine neutron fluence in other internal components for structural integrity assessments or to evaluate repair technologies

The RAMA Fluence Methodology is a state-of-the-art, versatile tool for calculating the fluence of the BWR pressure vessel and internals.

Approach

A surveillance capsule containing flux wires and Charpy specimens was extracted from the Susquehanna Unit 2 reactor in 1992 at the end of cycle 5, and testing of the surveillance materials was performed. Activation measurements were performed on the flux wires, and impact testing was performed on the Charpy specimens. The RAMA Fluence Methodology was used to calculate the capsule flux wire activities and fluence at the end of cycle 5. A comparative analysis of the calculated and measured activities was performed. The neutron fluence and uncertainty for the capsule were also determined.

Keywords

Boiling water reactor
Embrittlement
Fluence
Reactor pressure vessel
Vessel and internals

ABSTRACT

This document reports the results of a surveillance capsule fluence evaluation performed for the Susquehanna Unit 2 reactor at the end of cycle 5. The capsule evaluation was performed using the RAMA Fluence Methodology software.

The RAMA Fluence Methodology is a system of software components that include a transport code, parts model builder code, state-point model builder code, fluence calculator, and nuclear data library. The RAMA transport code couples a three-dimensional deterministic transport solver with an arbitrary geometry modeling capability to provide a flexible and accurate tool for determining fluxes in any light water reactor design. The model builder codes use reactor design inputs and operating data to generate geometry and material inputs for the transport solver. The fluence calculator uses isotopic activation and decay information with reactor operating history to provide an accurate estimate of component fluence. The nuclear data library contains nuclear cross section data and response functions that are used in the transport and fluence calculations. The nuclear data library is based upon the ENDF/B-VI data file and the BUGLE-96 nuclear data library.

A total average calculated-to-measured (C/M) result of specific activities for all flux wires was determined to be 0.98 with a standard deviation of ±8%. These C/M ratios are in very good agreement indicating the RAMA Fluence Methodology is accurately predicting fluence and flux.

The total capsule neutron fluence analytic uncertainty is 15.0% for energy >1.0 MeV and 15.0% for energy >0.1 MeV. The largest source of the capsule neutron fluence analytic uncertainty is attributable to the geometry parameters with the reactor pressure vessel inner radius dimension having the single highest uncertainty of 10.0% for energy >1.0 MeV. By combining the measurement uncertainty and analytic uncertainty, the combined capsule fluence uncertainty is determined to be 17.2% for energy >1.0 MeV and 17.2% for energy >0.1 MeV.

Therefore, the RAMA Fluence Methodology produces accurate results that compare very well with measured data.

RECORD OF REVISIONS

Revision Number	Revisions
BWRVIP-117	Original Report (1008065)
BWRVIP-117-A	This report is based on a previous report published as BWRVIP-117 (1008065) that was reviewed by the U.S. Nuclear Regulatory Commission (NRC). This report (BWRVIP-117-A) incorporates changes proposed by the BWRVIP in response to the NRC Requests for Additional Information, recommendations in the NRC Safety Evaluation (SE), and other necessary revisions identified since the last issuance of the report. All changes to the report except corrections to typographical errors are marked with margin bars. In accordance with a NRC request, the report number includes an "A" indicating the version of the report accepted by the NRC staff. NRC Safety Evaluation added to Front matter Appendices A-B added: NRC correspondence Details of the revisions can be found in Appendix C

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1
INTRODUCTION

This report presents the results of the capsule fluence evaluation performed for the Susquehanna Unit 2 reactor at the end of cycle 5. A surveillance capsule containing flux wires and Charpy specimens was extracted from the Susquehanna Unit 2 reactor at the end of cycle 5 and testing of the surveillance materials was performed [1]. Activation measurements were performed on the flux wires and impact testing was performed on the Charpy specimens. This report evaluates the activity measurements for the flux wires.

The RAMA Fluence Methodology was used to calculate the capsule flux wire activities and fluence at the end of cycle 5. A comparative analysis of the calculated and measured activities was performed and the results are presented in this report. The neutron fluence and uncertainty for the capsule were also determined and these results are provided in this report.

The RAMA Fluence Methodology (hereinafter referred to as the Methodology) has been developed for EPRI and the BWR Vessel and Internals Project (BWRVIP) for the purpose of calculating neutron fluence in Boiling Water Reactor (BWR) components. The Methodology includes a transport code, model builder codes, a fluence calculator code, an uncertainty methodology, and a nuclear data library. The transport code, fluence calculator, and nuclear data library are the primary software components for calculating the neutron flux and fluence. The transport code uses a deterministic, three-dimensional, multigroup nuclear particle transport theory to perform the neutron flux calculations. The transport code couples the nuclear transport method with a general geometry modeling capability to provide a flexible and accurate tool for calculating fluxes in light water reactors. The fluence calculator uses reactor operating history information with isotopic production and decay data to estimate activation and fluence in the reactor components over the operating life of the reactor. The nuclear data library contains nuclear cross-section data and response functions that are needed in the flux, fluence, and reaction rate calculations. The cross sections and response functions are based on the BUGLE-96 nuclear data library [2]. The Methodology and procedures for its use are described in the following reports: Theory Manual [3], User's Manual [4], and Procedures Manual [5].

The Methodology has been benchmarked using experimental and numerical problems specified in U.S. NRC Regulatory Guide 1.190 [6]. The results of the benchmark cases are documented in the EPRI report entitled "RAMA Fluence Methodology – Benchmark Manual Evaluation of Regulatory Guide 1.190 Benchmark Problems" [7]. This report provides further validation of the Methodology by evaluating the flux wire measurements for the Susquehanna Unit 2 boiling water reactor using utility-generated design inputs and actual operating history data.

The information and associated evaluations provided in this report have been performed in accordance with the requirements of 10CFR50 Appendix B.

1.1 Implementation Requirements

This report is provided for information only. Therefore, the implementation requirements of Nuclear Energy Institute (NEI) 03-08, Guideline for the Management of Materials Issues, are not applicable.

2
SUMMARY AND CONCLUSIONS

This section provides a summary of the results of the surveillance capsule fluence evaluation for Susquehanna Unit 2 cycles 1 through 5. Detailed tables of all results are presented in Section 5 of this report. The primary purpose of this evaluation is to determine the capsule fluence and rated power neutron flux for energy >1.0 MeV and for energy >0.1 MeV.

For energy >1.0 MeV, the following results are obtained for the capsule during this evaluation:

- Rated power neutron flux = 7.930×10^{8} n/cm^2-sec
- Neutron fluence after 6.22 EFPY = 1.555×10^{17} n/cm^2

For energy >0.1 MeV, the following results are generated during this evaluation:

- Rated power neutron flux = 1.428×10^{9} n/cm^2-sec
- Neutron fluence after 6.22 EFPY = 2.801×10^{17} n/cm^2

In addition to the determination of neutron fluence and flux values, specific activities are predicted for the copper, iron, and nickel flux wires and compared to measurements. The total average calculated-to-measured (C/M) result of specific activities for all flux wires is determined to be 0.98 with a standard deviation of ±8%. These C/M ratios are in very good agreement indicating the RAMA Fluence Methodology is accurately predicting fluence and flux. On the average, the C/M values are lower (0.88) for the copper flux wire, identical (1.00) for the iron, and slightly higher (1.05) for the nickel values.

Another result from this evaluation is the calculated capsule fluence analytic uncertainty and combined uncertainty values. The total capsule neutron fluence analytic uncertainty is 15.0% for both energy >1.0 MeV and energy >0.1 MeV. The largest source of the capsule neutron fluence analytic uncertainty is attributable to the geometry parameters with the reactor pressure vessel inner radius dimension having the single highest uncertainty of 10.0% for energy >1.0 MeV. By combining the measurement uncertainty and analytic uncertainty, the combined capsule fluence uncertainty is determined to be 17.2% for both energy >1.0 MeV and energy >0.1 MeV.

In conclusion, the RAMA Fluence Methodology produces accurate results that compare very well with measured data. The Methodology for determining the best estimate capsule neutron fluence has been performed in accordance with the guidelines presented in Regulatory Guide 1.190 and is determined to be acceptable in accordance with the guidelines.

3
DESCRIPTION OF THE REACTOR SYSTEM

This section describes the design inputs for the Susquehanna Unit 2 reactor that were used in the surveillance capsule fluence evaluation presented in this report. The basic design inputs include mechanical design drawings, material compositions, and reactor operating history. The design inputs were provided for this project by the utility support staff of PPL Susquehanna [8,9].

3.1 Reactor System Mechanical Design Inputs

The RAMA Fluence Methodology employs a three-dimensional modeling technique to describe the reactor geometry for the neutron transport calculations. Detailed mechanical design information is needed in order to build an accurate three-dimensional RAMA computer model of the reactor system. The mechanical design information for Susquehanna Unit 2 was generated by PPL Susquehanna in accordance with project data specifications [10]. A summary of the important design inputs is presented in this subsection.

Susquehanna Unit 2 is a General Electric BWR/4 class reactor with a rated thermal power output of 3293 MWt. Figure 3-1 shows a planar view of the reactor at an axial elevation near the core mid-plane. The primary radial components and regions are shown, including the core region, core reflector, shroud, downcomer, jet pumps, pressure vessel, mirror insulation, cavity regions, and biological shield (concrete wall). The reactor core region has a core loading of 764 fuel assemblies. There are 10 jet pump assemblies in the downcomer region that are positioned azimuthally at 30, 60, 90, 120, 150, 210, 240, 270, 300, and 330 degrees. Three surveillance capsules were initially loaded in the reactor and were positioned azimuthally at 30, 120, and 300 degrees. The capsules reside in the downcomer region at a radial position near the inside surface of the reactor pressure vessel wall. The capsule at azimuth 30 degrees was pulled at the end of cycle 5 and is analyzed in this report.

Figure 3-2 shows a partial elevation view of the Susquehanna Unit 2 reactor. The elevation of interest for the capsule fluence evaluation is near the core mid-plane where the surveillance capsules are loaded. The capsules are situated axially such that half of the capsule extends above and half below the designated mid-plane elevation mark. For the purpose of evaluating the surveillance capsule measurements, only the axial elevations within the active fuel height (i.e., the axial height between the bottom of active fuel and top of active fuel) are required for the analysis.

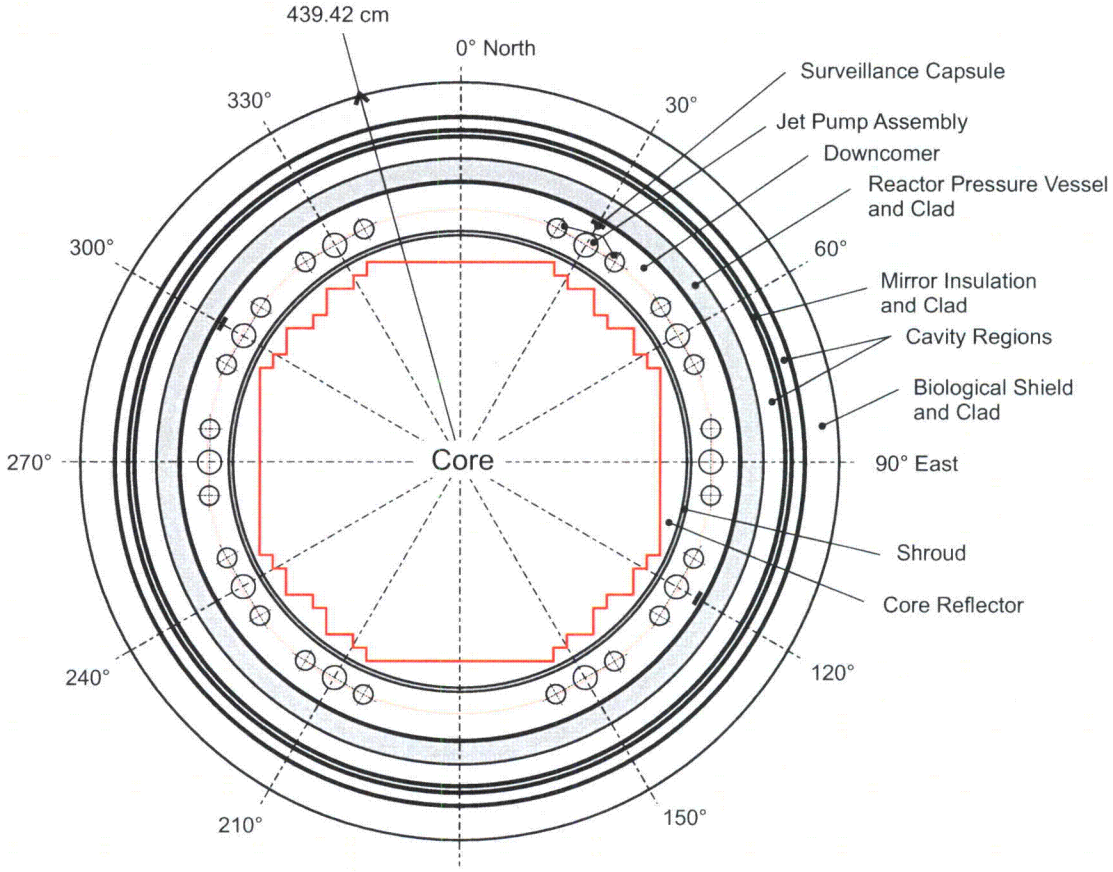

Figure 3-1
Planar view of the susquehanna unit 2 reactor

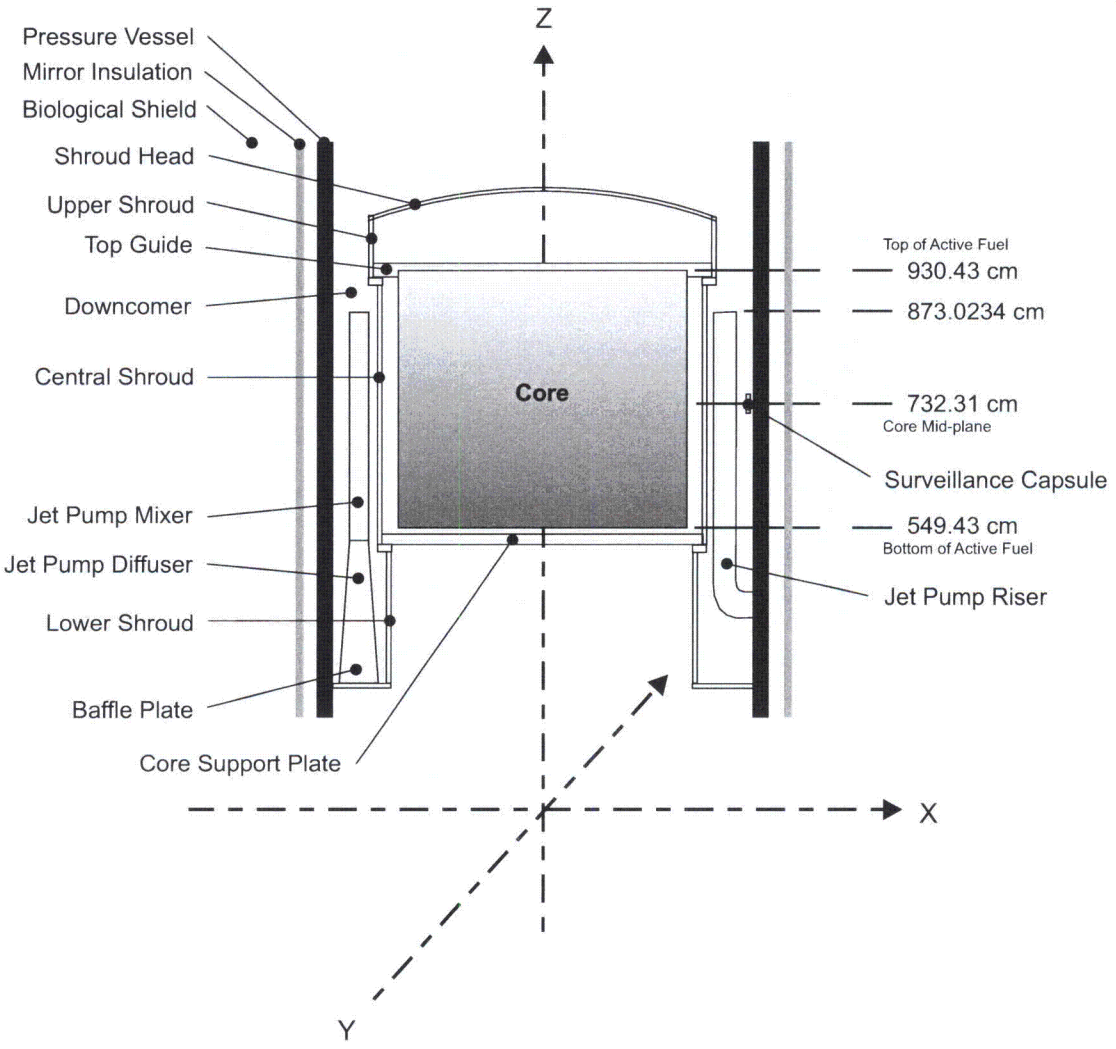

Figure 3-2
Elevation view of the susquehanna unit 2 reactor

3.2 Reactor System Material Compositions

Each region of the reactor is comprised of materials that include reactor fuel, steel, water, insulation, and air. Accurate material information is essential for the fluence evaluation as the material compositions determine the scattering and absorption of neutrons throughout the reactor system and, thus, affect the determination of neutron fluence in the reactor components.

Table 3-1 provides a summary of the material compositions in the various components and regions of the Susquehanna Unit 2 reactor. The attributes for the steel, insulation, and air compositions (i.e., material densities and isotopic concentrations) are assumed to remain constant for the operating life of the reactor. The attributes for the water compositions will vary with the operation of the reactor, but are generally represented at nominal hot operating conditions and assumed to be constant for an operating cycle. The attributes of the fuel compositions in the

reactor core region change continuously during an operating cycle due to changes in power level, fuel burnup, control rod movements, and changing moderator density levels (voids). Because of the dynamics of the fuel attributes with reactor operation, one to several data sets describing the operating state of the reactor core are required for each operating cycle.

Table 3-1
Summary of material compositions by region for susquehanna unit 2

Region	Material Composition
Reactor Core Region (Fuel)	^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, O, Zr, Water
Core Reflector Region	Water
Shroud	Stainless Steel SS-304L
Downcomer Region	Water
Jet Pump Riser and Mixer Flow Area	Water
Jet Pump Riser and Mixer Metal	Stainless Steel SS-304
Surveillance Capsule	Stainless Steel SS-304
Reactor Pressure Vessel Clad	Stainless Steel SS-304
Reactor Pressure Vessel Wall	Carbon Steel CS-A533B
Cavity Regions	Air (Oxygen)
Insulation Clad	Stainless Steel SS-304
Insulation	Glass Wool
Biological Shield Clad	Carbon Steel CS-A533B
Biological Shield	Concrete

3.3 Reactor Operating Data Inputs

An accurate evaluation of fluence in the reactor requires an accurate accounting of the reactor operating history. The primary reactor operating parameters that affect neutron fluence evaluations for BWR's include the reactor power level, core relative power distribution, core void fraction distribution (or equivalently, water density distribution), and fuel material distribution.

3.3.1 Power History Data

The reactor power history used in the Susquehanna Unit 2 capsule fluence evaluation was obtained from daily power history edits provided by PPL Susquehanna for the five operating cycles in which the capsule was loaded in the reactor [9]. The daily power values represent step changes in power on a daily basis and the power is assumed to be representative of the power over the entire day. The fluence evaluation for Susquehanna Unit 2 considered the complete daily operating history of the reactor over the evaluation period.

Figures 3-3 through 3-7 show the relative power history of Susquehanna Unit 2 for operating cycles 1 through 5, respectively. Also accounted for in the analysis are the shutdown periods. The shutdowns were primarily due to the refueling outages between cycles.

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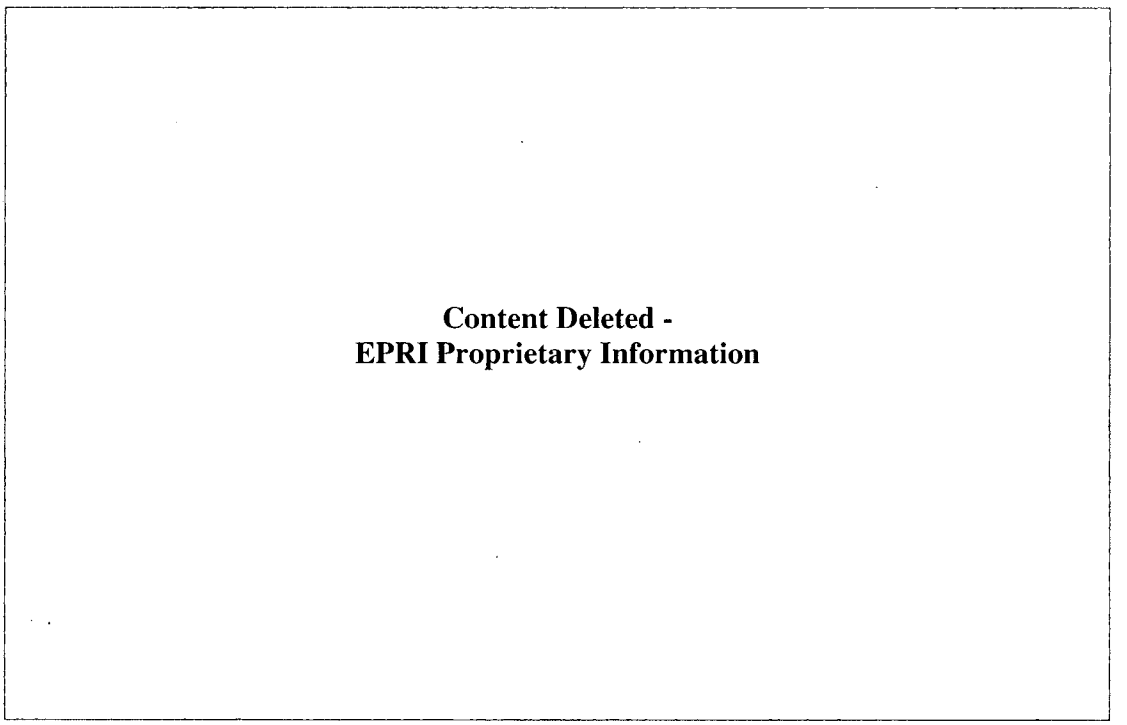

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Figure 3-3
Susquehanna unit 2 cycle 1 relative power history

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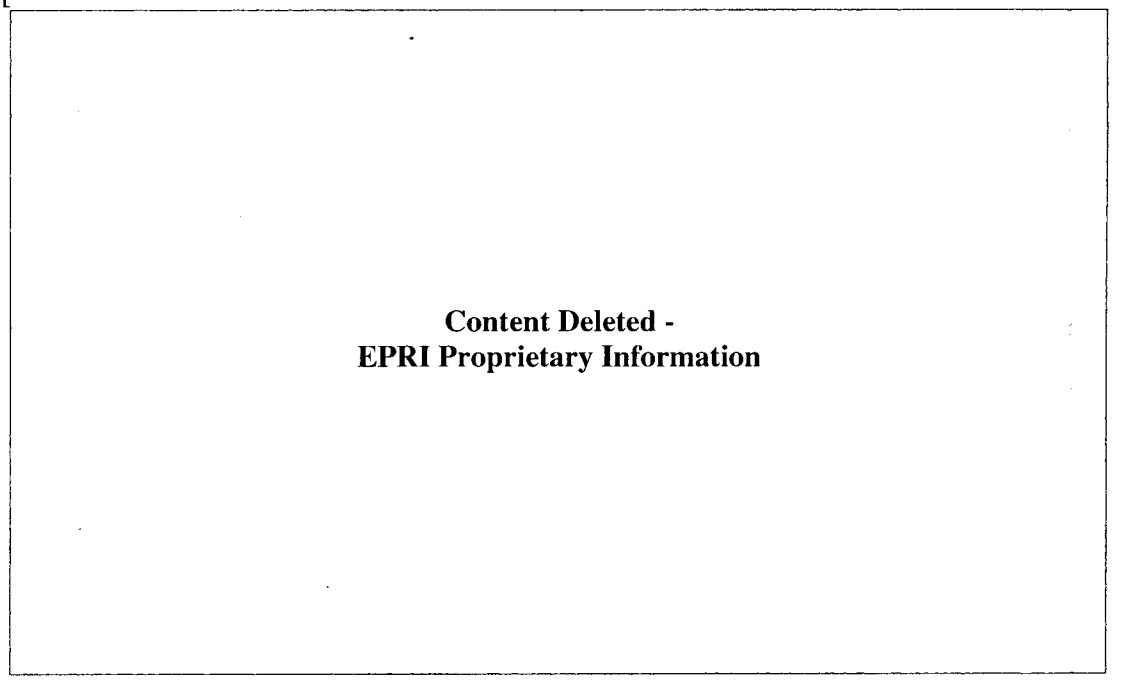

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Figure 3-4
Susquehanna unit 2 cycle 2 relative power history

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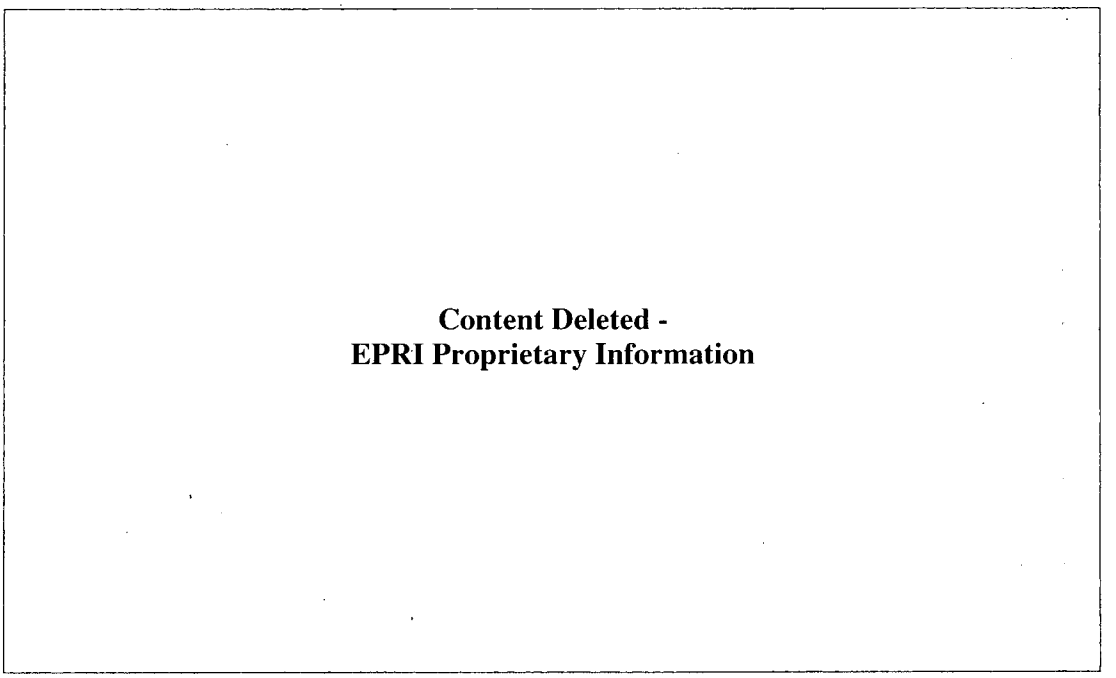

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Figure 3-5
Susquehanna unit 2 cycle 3 relative power history

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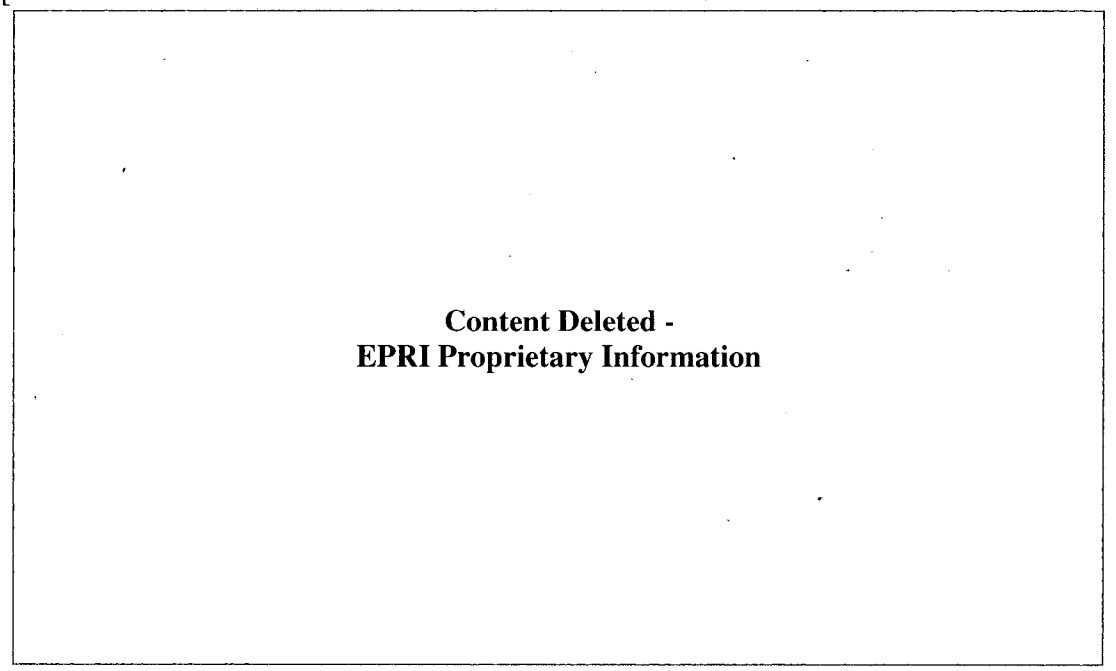

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Figure 3-6
Susquehanna unit 2 cycle 4 relative power history

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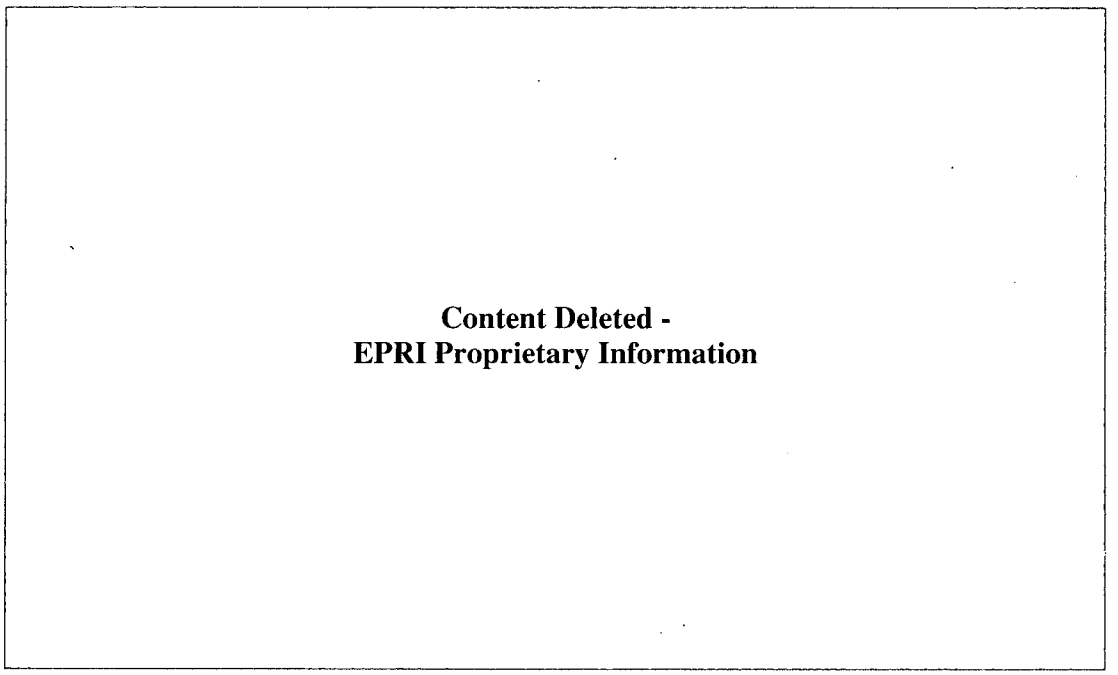

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Figure 3-7
Susquehanna unit 2 cycle 5 relative power history

3.3.2 Reactor State Point Data

Reactor operating data for the Susquehanna Unit 2 capsule fluence evaluation was provided as state point data files by PPL Susquehanna [9]. Each state point file represents the operating conditions of the unit at a specified moment in time. The data files include three-dimensional data arrays that describe the fuel materials, moderator materials, and the relative power distribution in the core region. The data files represent core simulation evaluations that are performed as a part of routine plant performance tracking.

Sixty-three state point data files were provided for Susquehanna Unit 2. These data files represent the operating states of the reactor for cycles 1 through 5. Of the sixty-three state point data files, eighteen data files are provided for cycle one, thirteen data files for cycle two, eleven data files for cycle three, nine data files for cycle four, and twelve data files for cycle five. The guidelines and criteria for selecting the state points for use in RAMA fluence evaluations are described in [5].

A separate neutron transport calculation was performed for each state point. The calculated neutron flux for each state point was combined with the appropriate power history data described in Section 3.3.1 to predict the neutron fluence in the surveillance capsule at the end of cycle 5.

3.3.3 Core Loading Pattern

It is common in BWRs that more than one fuel assembly design will be loaded in the reactor core in any given operating cycle. For fluence evaluations, it is important to account for the fuel assembly designs that are loaded in the core peripheral locations in order to accurately represent the neutron source distribution at the core boundary.

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Table 3-2
Summary of the susquehanna unit 2 core loading pattern

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4
CALCULATION METHODOLOGY

The Susquehanna Unit 2 capsule fluence evaluation was performed using the RAMA Fluence Methodology software package. The Methodology and the application of the Methodology to the Susquehanna Unit 2 reactor are described in this section.

4.1 Description of the RAMA Fluence Methodology

The RAMA Fluence Methodology software package is a system of codes that is used to perform fluence evaluations in light water reactor components. The significance of the Methodology is the integration of a three-dimensional arbitrary geometry modeling technique with a deterministic transport method to provide a flexible and accurate platform for determining neutron fluence in light water reactor systems. The Methodology is complemented with model building codes to prepare the three-dimensional models for the transport calculation and a post-processing code to calculate fluence from the neutron flux calculated by the transport code.

The primary software components in the software package are: the Parts Model Builder (PMB) code for constructing reactor geometry models; the State-point Model Builder (SMB) code for processing material data for the geometry model; the RAMA transport code for calculating the neutron flux distribution throughout the model; the fluence calculator (RAFTER) code that calculates activations and fluence for component regions of the model; and the RAMA nuclear data library. The codes and nuclear data library are tightly integrated to facilitate the effort of building computer models and performing component fluence analysis. Each software component of the RAMA Fluence Methodology is implemented as a stand-alone module to further provide flexibility in the analysis effort.

The primary inputs for the RAMA Fluence Methodology are mechanical design parameters and reactor operating history data. The mechanical design inputs are obtained from reactor design drawings (or vendor drawings) of the plant. The reactor operating history data is obtained from reactor core simulation calculations, system heat balance calculations, and daily operating logs that describe the operating conditions of the reactor.

The primary outputs from the RAMA Fluence Methodology calculations are neutron flux, neutron fluence, and uncertainty determinations. The RAMA transport code calculates the neutron flux distributions that are used in the determination of neutron fluence. Several transport calculations are typically performed over the operating life of the reactor in order to calculate neutron flux distributions that accurately characterize the operating history of the reactor. The RAFTER code is then used to calculate component fluence and nuclide activations using the neutron flux solutions from the transport calculations and daily operating history data for the plant. If desired, the fluence calculated by RAFTER may then be adjusted in accordance with the calculational bias to determine the best estimate fluence and uncertainty to complete the evaluation.

4.2 The RAMA Geometry Model for Susquehanna Unit 2

The RAMA Fluence Methodology uses a flexible three-dimensional modeling technique to describe the reactor geometry. The geometry modeling technique is based on the Cartesian coordinate system in which the (x,y) plane describes radial-azimuthal configuration of the reactor and the z-axis describes the elevations in the reactor.

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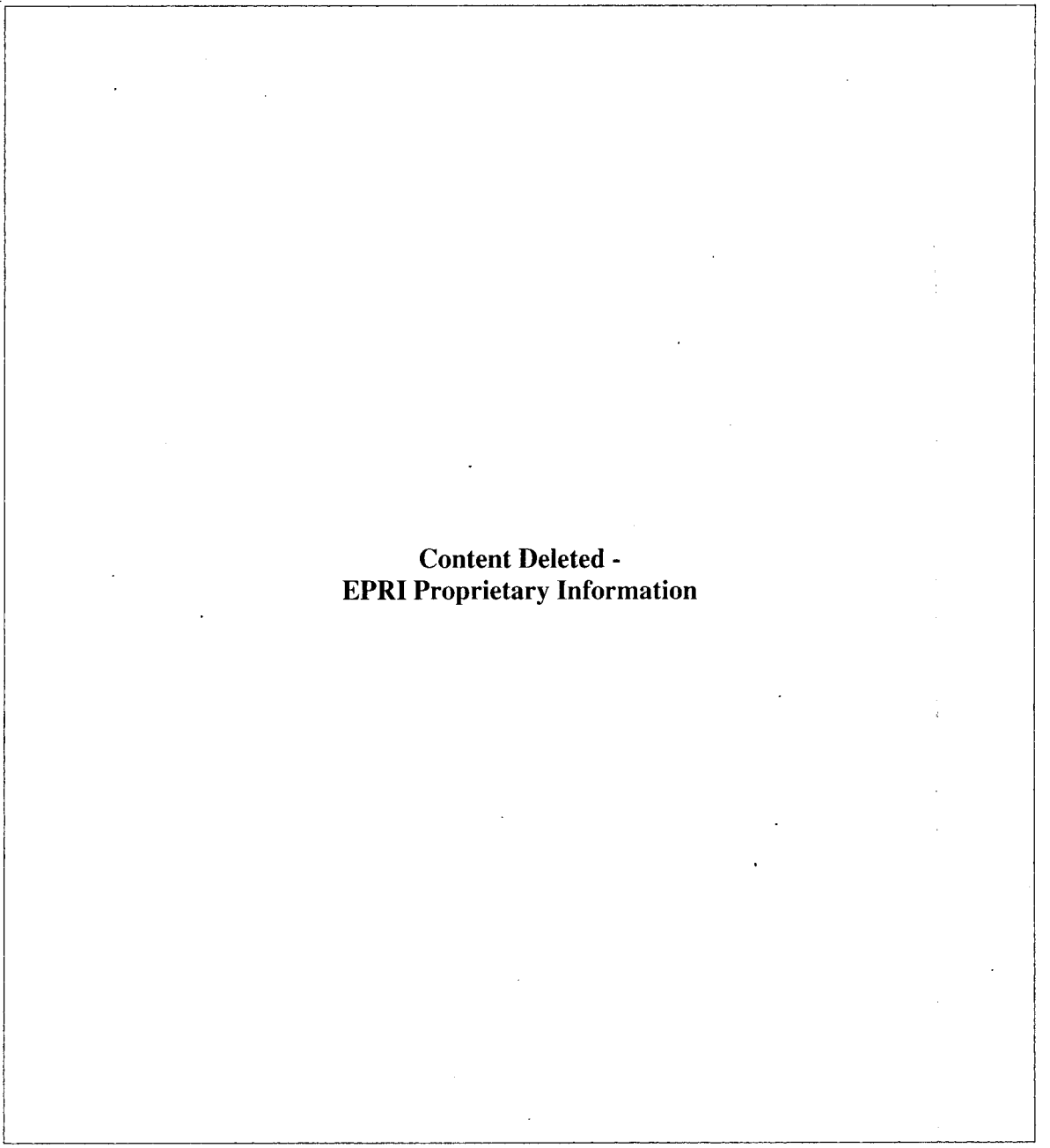

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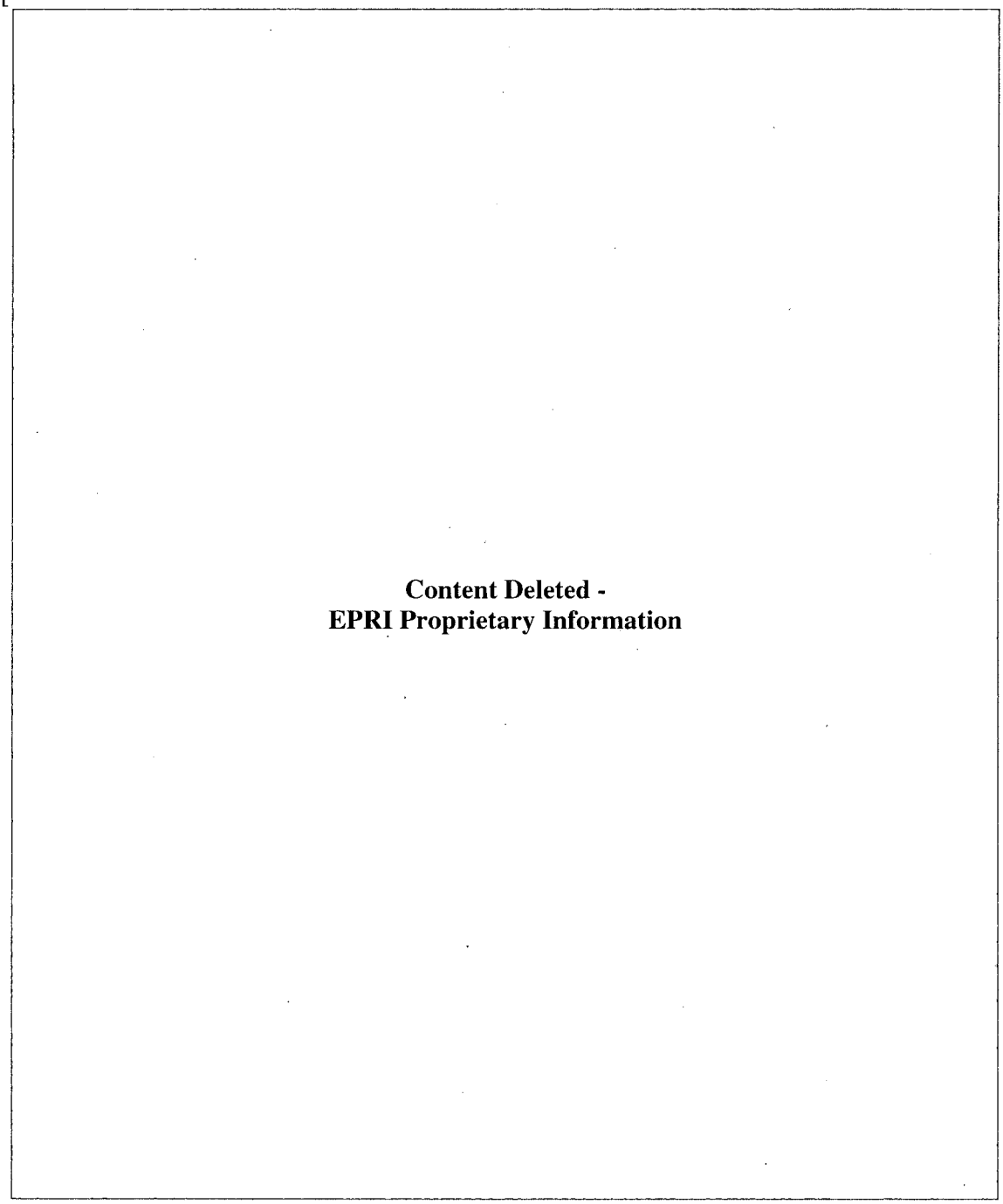

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Figure 4-1
Planar view of the susquehanna unit 2 RAMA model

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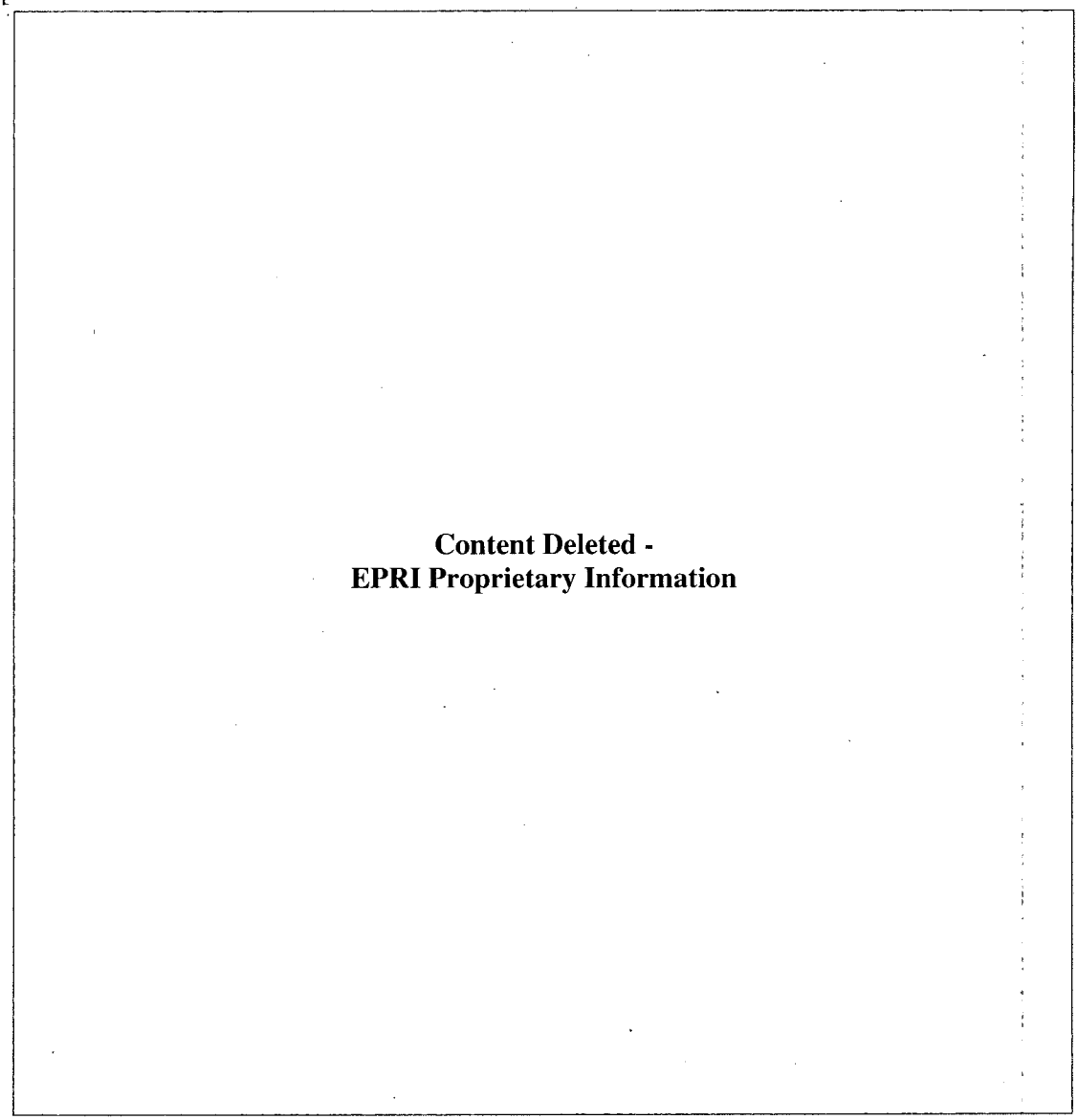

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Figure 4-2
Axial view of the susquehanna unit 2 RAMA downcomer model

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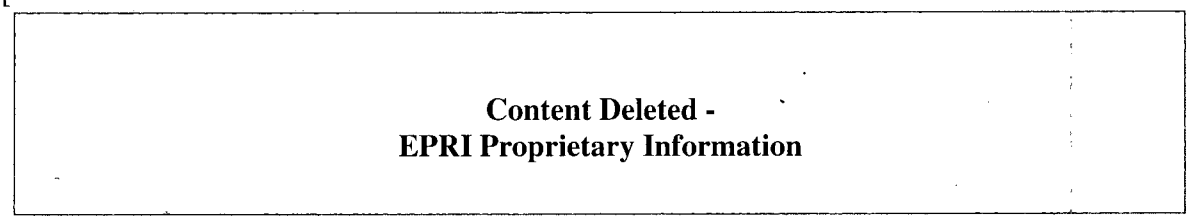

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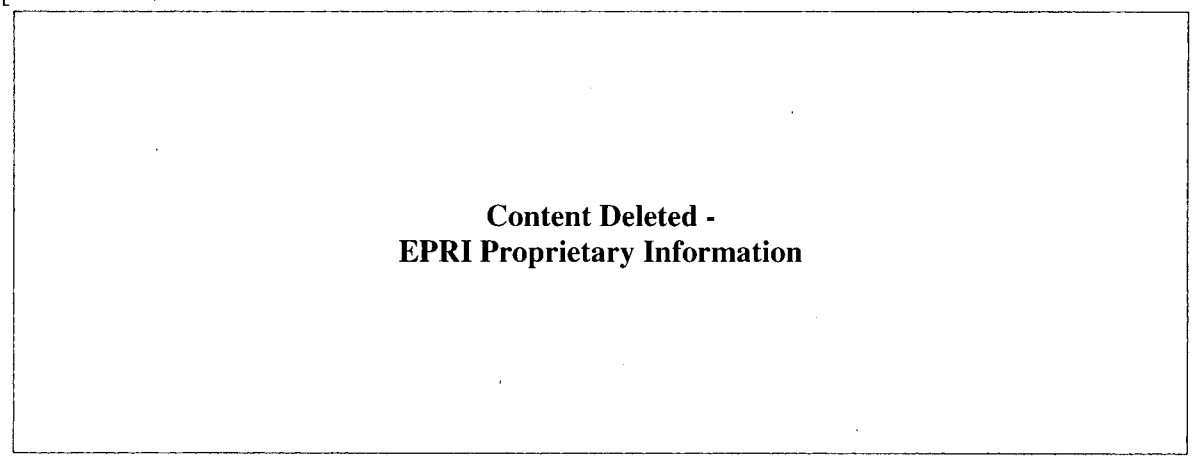

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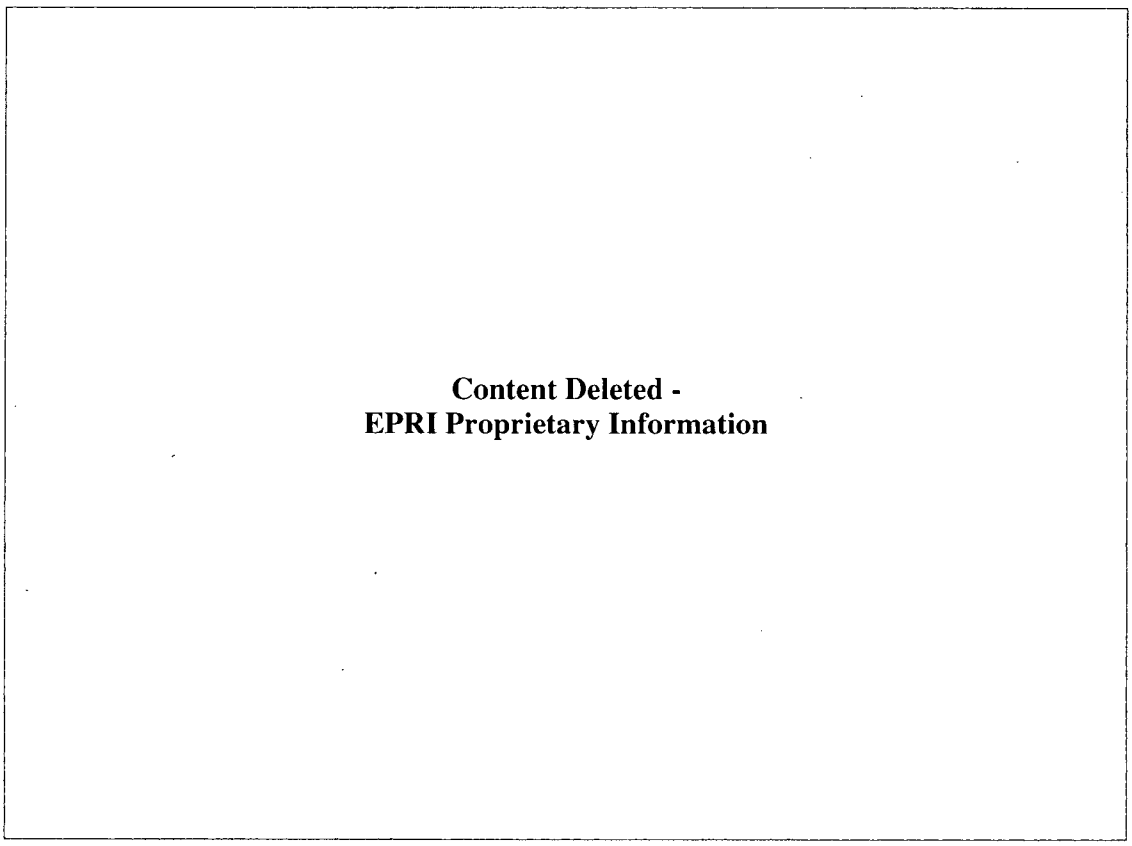

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Figure 4-3
Planar view of the susquehanna unit 2 surveillance capsule

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4.4 Parametric Sensitivity Analyses

Several sensitivity analyses were performed to evaluate the stability and accuracy of the RAMA transport calculation for the Susquehanna Unit 2 model. Several parameters were evaluated including mesh size and the integration parameters discussed in Section 4.3. A summary of the analyses is presented in Table 4-1.

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Table 4-1
Sensitivity analyses

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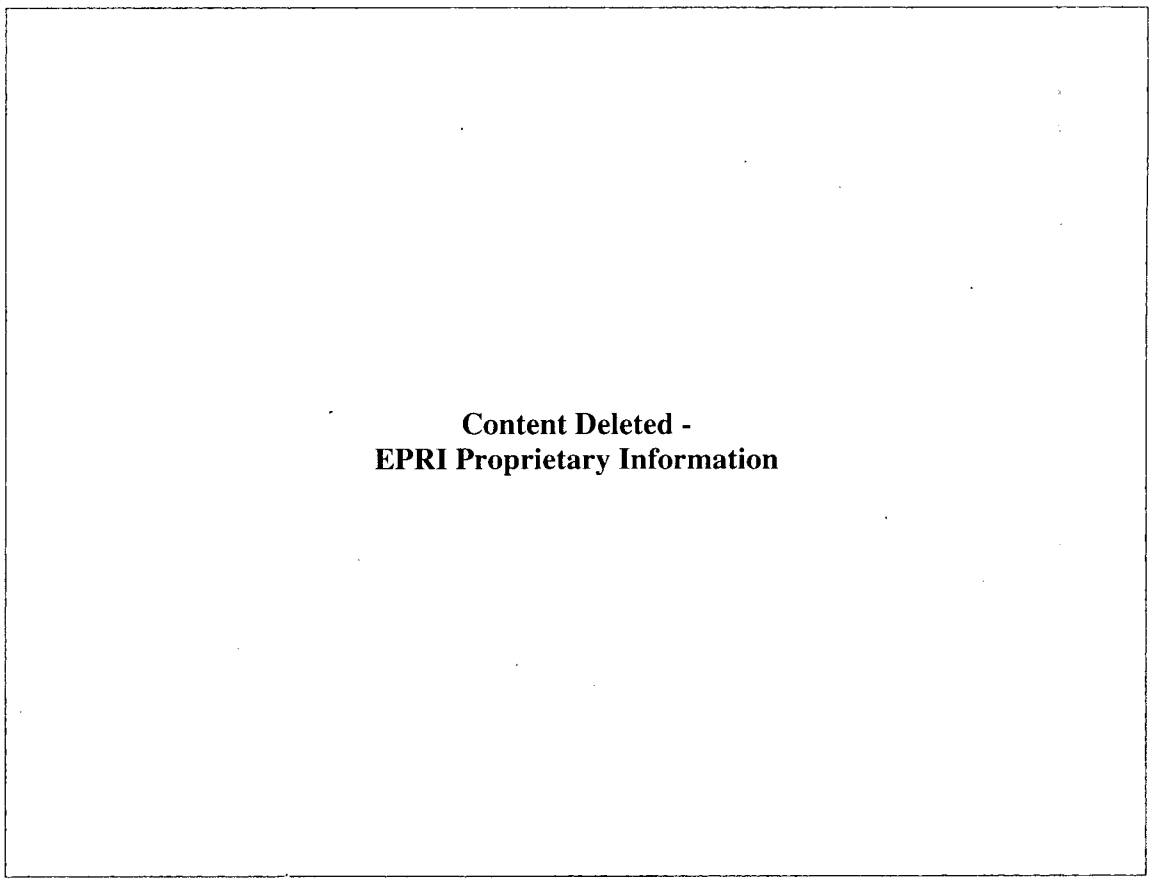

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Figure 4-4
2D sensitivity to distance between planar parallel rays

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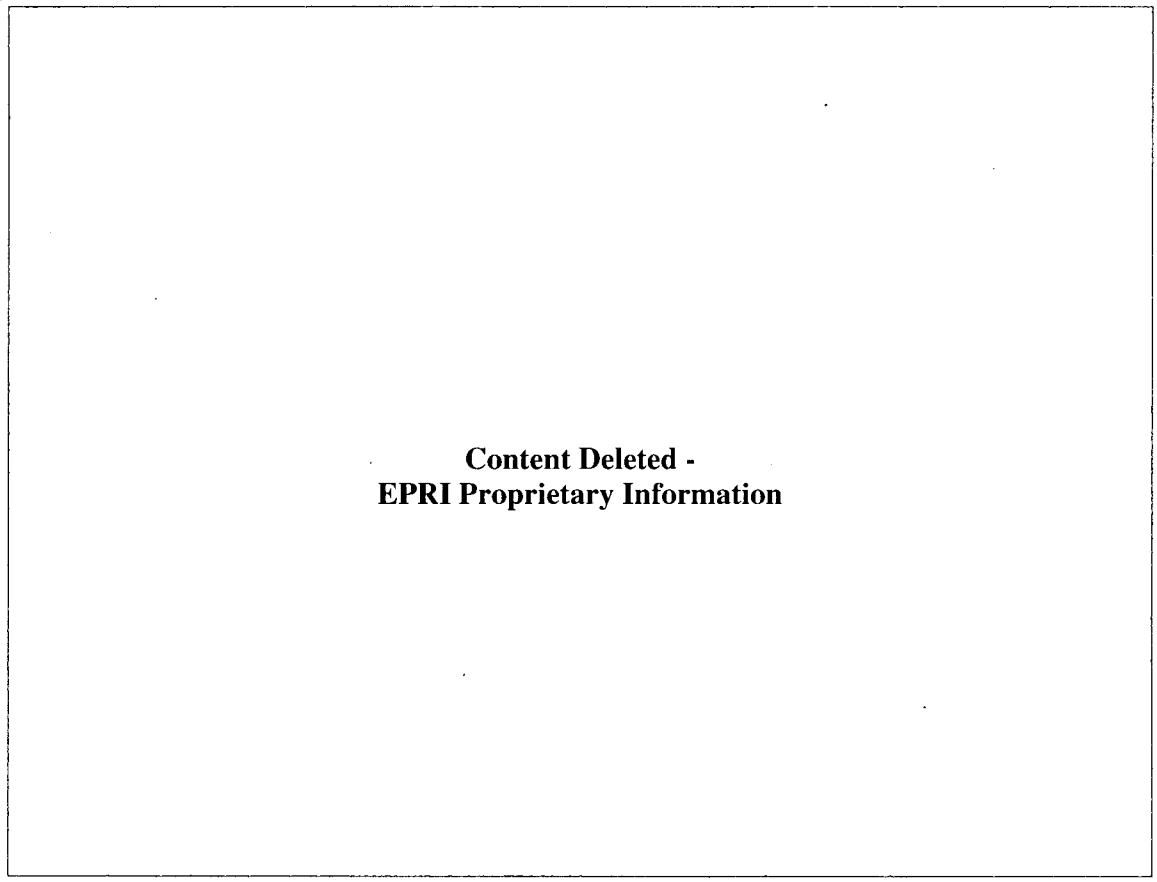

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Figure 4-5
3D sensitivity to distance between planar parallel rays

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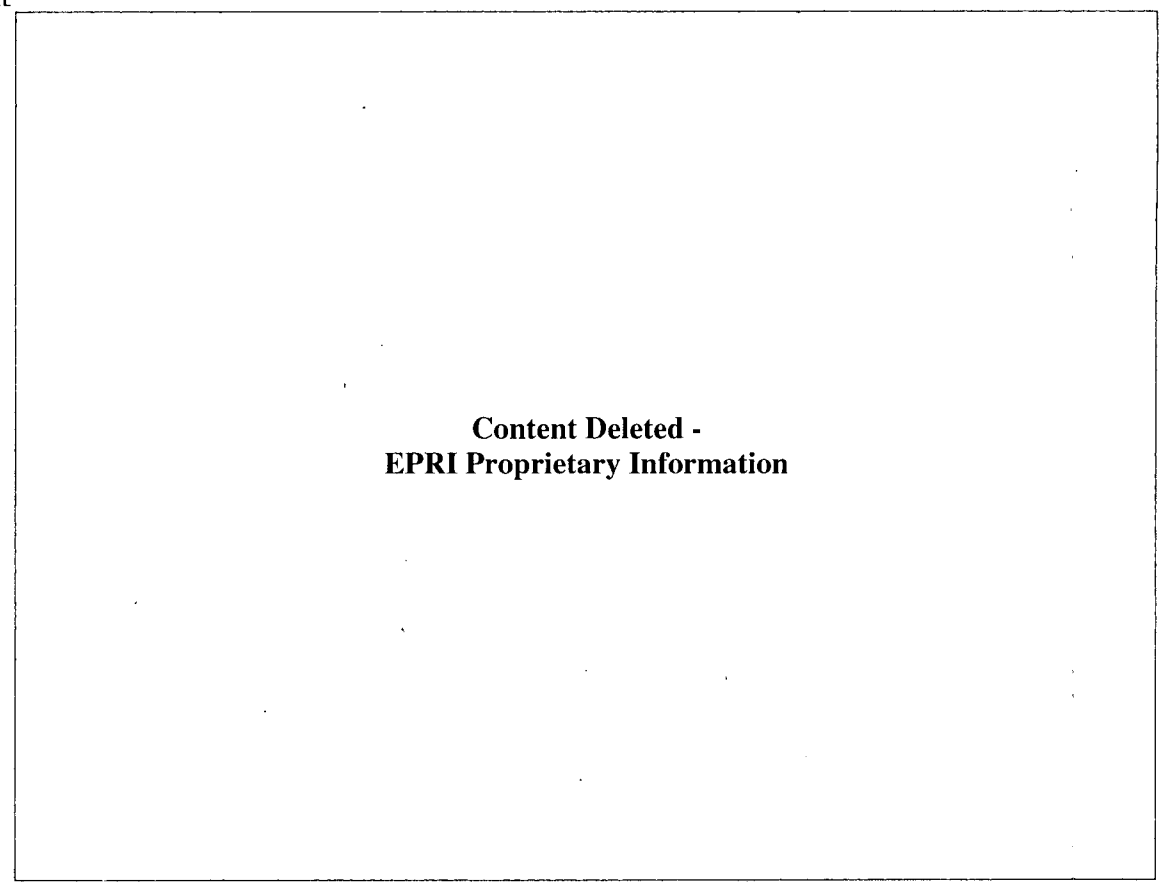

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Figure 4-6
3D sensitivity to distance between axial parallel rays

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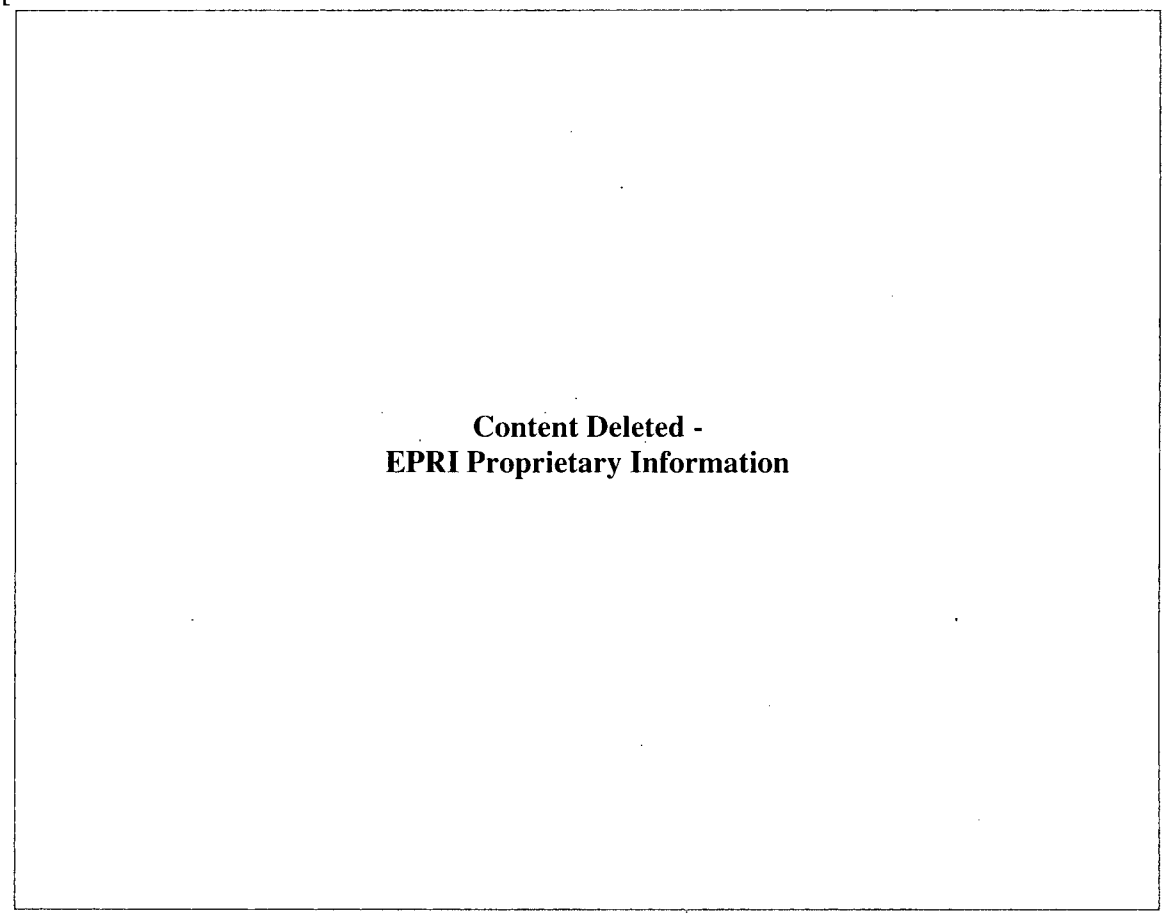

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Figure 4-7
Sensitivity to flux convergence criterion

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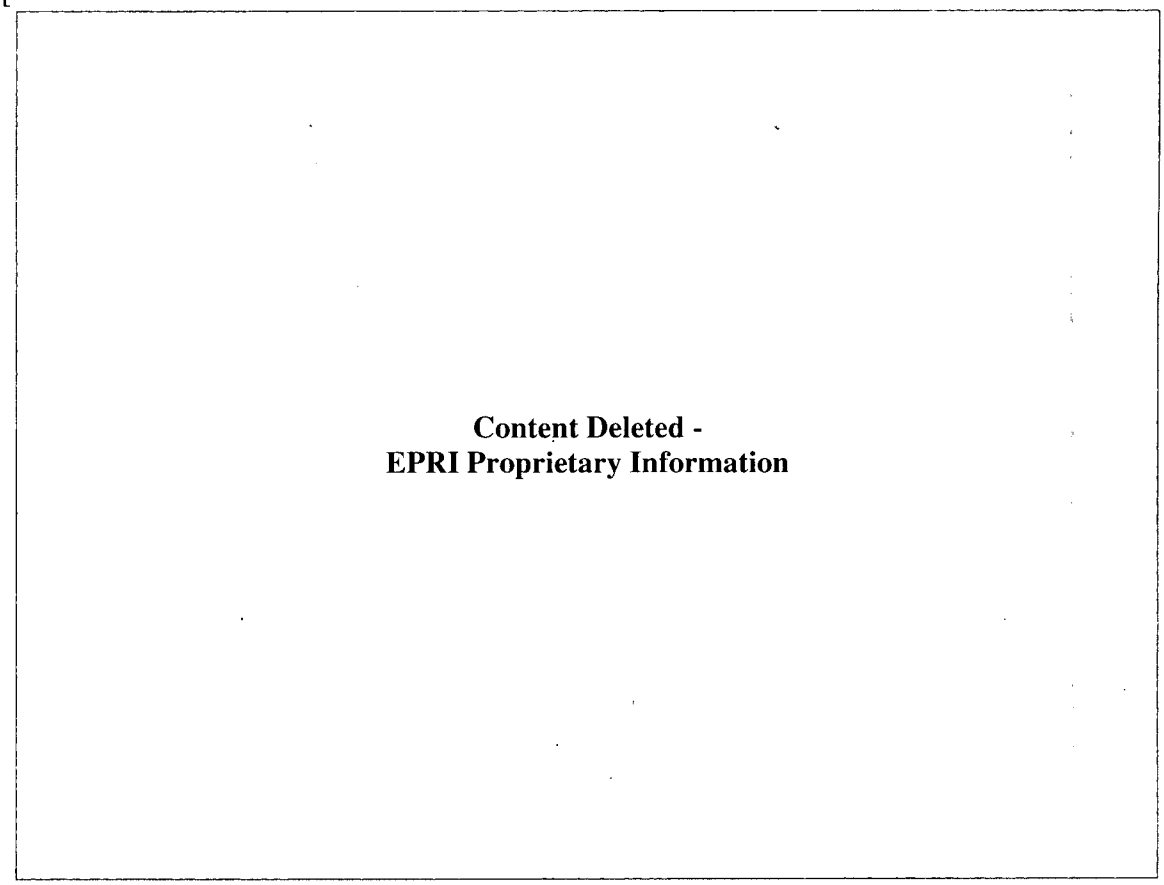

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Figure 4-8
Sensitivity to the angular quadrature order

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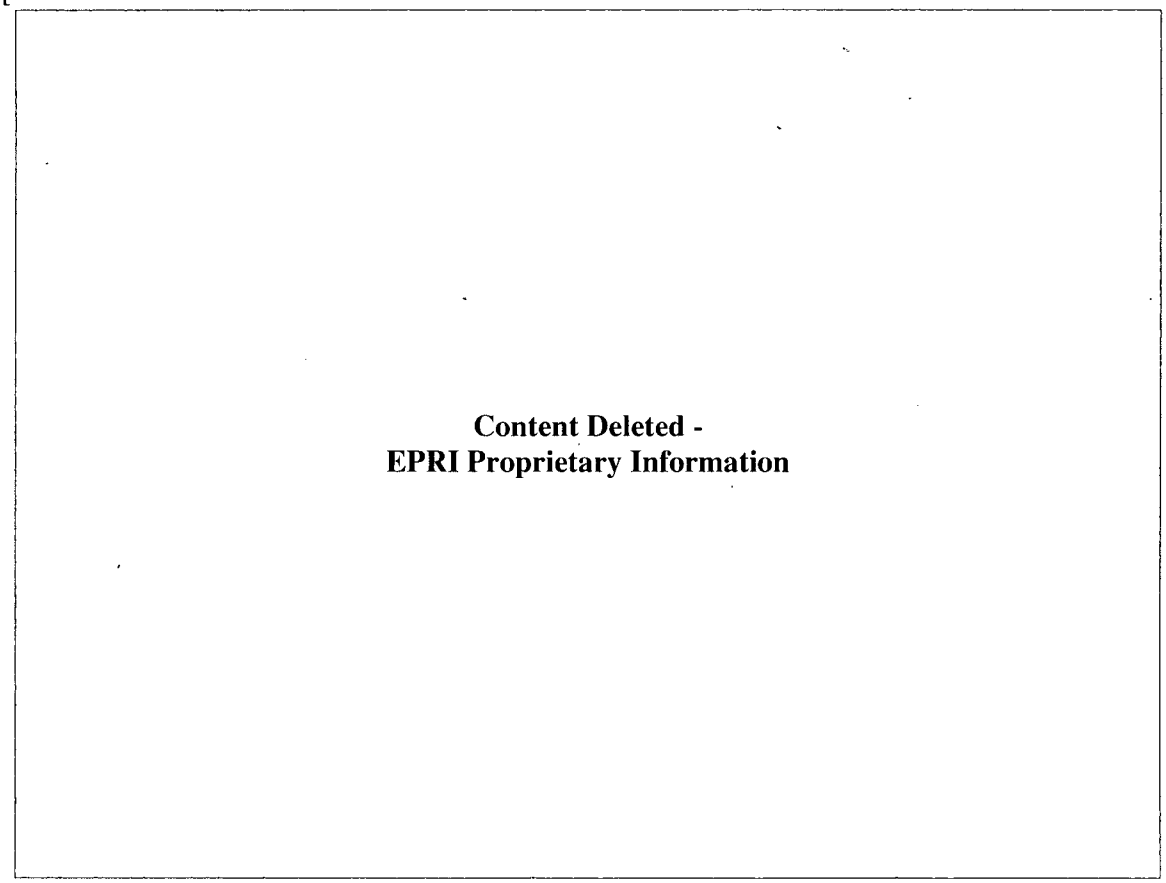

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Figure 4-9
3D sensitivity to the number of axial planes

5
SURVEILLANCE CAPSULE FLUENCE EVALUATION RESULTS

This section contains the results from the Susquehanna Unit 2 surveillance capsule fluence evaluation. Predicted neutron fluence, neutron flux for energy >1.0 MeV and >0.1 MeV, and comparison of the predicted activation (i.e., specific activities) to the activation measurements for the capsule are provided. The Susquehanna Unit 2 surveillance capsule was removed at the end of cycle 5 after being irradiated from initial reactor start-up on August 1, 1984 through September 12, 1992 for a total of 6.22 effective full power years (EFPY).

5.1 Calculated Neutron Fluence and Flux

Table 5-1 provides the RAMA calculated values for the neutron fluence and rated power flux in the Susquehanna Unit 2 capsule for energy >1.0 MeV and energy >0.1 MeV. [[

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Table 5-1
Comparison of specific activities (in dps/g) for surveillance capsule flux wires (C/M)

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5.3 Surveillance Capsule Uncertainty Evaluation

The sources of the capsule uncertainty include analytic uncertainty and comparison uncertainty These are combined to provide an estimate of the overall fluence bias and uncertainty (1σ). This subsection describes the parameters that were considered for the analytic uncertainty, the calculated comparison uncertainty, and the calculated combined uncertainty for the capsule fluence evaluation. The calculated combined uncertainty is used in Section 5.4 to calculate the capsule best estimate fluence and rated power flux.

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Table 5-2
Capsule analytic uncertainty

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1. *Susquehanna Steam Electric Station Unit 2 Vessel Surveillance Materials Testing and Fracture Toughness Analysis*, GE Nuclear Energy, GE-NE-523-107-0893, DRF 137-0010-6, October 1993, Revision 1.

2. "BUGLE-96: Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," RSICC Data Library Collection, DLC-185, March 1996.

3. *BWRVIP-114-A: BWR Vessel Internals Project, RAMA Fluence Methodology Theory Manual.* EPRI, Palo Alto, CA: 2009 1019049.

4. D. B. Jones et al.,"RAMA Fluence Methodology User's Manual," EPRI, Palo Alto, CA, 2003.

5. *BWRVIP-121-A: BWR Vessel Internals Project, RAMA Fluence Methodology Procedures Manual.* EPRI, Palo Alto, CA: 2009 1019052.

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

7. *BWRVIP-115-A: BWR Vessel and Internals Project, RAMA Fluence Methodology Benchmark Manual – Evaluation of Regulatory Guide 1.190 Benchmark Problems.* EPRI, Palo Alto, CA: 2009 1019050.

8. Letter from Bruce Swoyer of PPL Susquehanna, LLC to Dean Jones of TransWare Enterprises Inc., PLE-23053, "Susquehanna Steam Electric Station Transmittal of Calculation Package," December 5, 2002.

9. Letter from Kenneth Knoll of PPL Susquehanna, LLC to Dean Jones of TransWare Enterprises Inc., PLE-0023118, "Susquehanna Steam Electric Station BWRVIP/RAMA Benchmark Data for SSES Unit 2 Cycles 1 Through 5 CCN741071," February 3, 2003.

10. D. B. Jones, "RAMA Fluence Methodology Data Requirements Specification," TransWare Enterprises Inc., EPR-VIP-002-S-003, Revision 0, September 30, 2002.

11. Personal email communication from Dr. Horace Pops of Superior Essex to Mr. William Black of the Copper Development Association, "Cobalt Impurity in Copper Wires," April 14, 2003.

A
NRC REQUEST FOR ADDITIONAL INFORMATION

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

April 20, 2004

Bill Eaton, BWRVIP Chairman
Entergy Operations, Inc.
Echelon One
1340 Echelon Parkway
Jackson, MS 39213-8202

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - REVIEW OF BWR VESSEL AND INTERNALS PROJECT REPORTS, BWRVIP-114, BWRVIP-115, BWRVIP-117, AND BWRVIP-121, AND TRANSWARE ENTERPRISES INC. REPORT TWE-PSE-001-R-001, REVISION 0 (TAC NO. MB9765)

Dear Mr. Eaton:

By applications dated August 1, August 5, October 23, and October 29, 2003, respectively, you submitted for NRC staff review, four Electric Power Research Institute (EPRI) proprietary reports, BWRVIP-114, "RAMA Fluence Methodology Theory Manual," BWRVIP-115, "RAMA Fluence Methodology Benchmark Manual-Evaluation of Regulatory Guide 1.190 Benchmark Problems," BWRVIP-117, "RAMA Fluence Methodology Plant Application-Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1-5," and BWRVIP-121, "RAMA Fluence Methodology Procedures Manual." In addition, by application dated March 23, 2004, you submitted for NRC staff review, TransWare Enterprises, Inc. Report, TWE-PSE-001-R-001, Revision 0, "Hope Creek Flux Wire Dosimeter Activation Evaluation for Cycle 1 Using the RAMA Fluence Methodology." These reports were submitted to the NRC as a means of exchanging information with the NRC for the purpose of supporting generic regulatory improvements related to methodologies to determine neutron fluence in BWR internal components.

The NRC staff has completed its initial review of the BWRVIP-114, BWRVIP-115, BWRVIP-117, and BWRVIP-121 reports, and the TransWare Enterprises, Inc. Report, TWE-PSE-001-R-001, Revision 0. As indicated in the attached request for additional

B. Eaton - 2 -

information (RAI), the NRC staff has determined that additional information is needed to complete the review. If you have any questions, please contact Meena Khanna at (301) 415-2150.

Sincerely,

Stephanie M. Coffin, Chief
Vessels & Internals Integrity and Welding Section
Materials and Chemical Engineering Branch
Division of Engineering
Office of Nuclear Reactor Regulation

Project No. 704

Enclosure: As stated

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REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE ELECTRIC POWER RESEARCH INSTITUTE (EPRI) RAMA METHODOLOGY FOR REACTOR PRESSURE VESSEL FLUENCE EVALUATION

BWRVIP-114: "BWR Vessel and Internals Project, RAMA Fluence Methodology Theory Manual"

RAI 114-1 In the plant-specific applications, what specific tests and criteria are used to assure the adequacy of the number of rays and the number of volumes used in the RAMA fluence calculations?

RAI 114-2 It is not evident that the RAMA geometry model described in Ref. 1 provides a correct representation of the true geometry (i.e., preserves the location, orientation and shape of all surfaces defining the physical geometry). For example, the modeling of the reflector region, surrounding the core, involves geometry elements that have both planar and cylindrical side boundaries. Since the geometry elements described in Ref. 1, Section 3.2, do not include bodies of this type, does RAMA introduce any distortion of the physical geometry in modeling the reflector and, if so, how is this distortion controlled to ensure acceptable accuracy?

RAI 114-3 The equation provided in Ref. 1, (Equation 7-38) for determining the M/C bias for the benchmark database requires an additional 1/M multiplicative normalization factor.

RAI 114-4 Equation 7-40 of Ref. 1 combines the analytical bias (B_a) and the benchmark bias (B_{bi}) to determine the overall calculational bias. The analytical bias (B_a), defined in Equation 7-34, provides the effect of not using the optimum asymptotic calculational input in the RAMA fluence calculation. Since the benchmark biases include the effect of the approximate calculational input used in the benchmark calculations (i.e., use of the standard input parameters rather than the asymptotic parameters), the analytical bias is only required when there is an inconsistency between the input used in the vessel fluence calculations and the benchmark calculations; e.g., when the calculations of the benchmark measurements are made with the asymptotic input values and the vessel fluence calculations are made with the standard input values. The staff requests that the BWRVIP clearly address the determination of the bias.

RAI 114-5 The weights defined in Equation 7-41 are not normalized (i.e., sum to unity), as required. Also, the weights should reflect the reliability of the bias estimates. If, for example, a weight of $1/\sigma^2$ is used, the σ should represent the standard deviation of the bias estimate, not the standard deviation of the M/C data about the mean.

ATTACHMENT

RAI 114-6 The values of σ_a, σ_{b1} and σ_{b2} of Equation (7-43) represent the (one standard deviation) uncertainty in the RAMA calculated fluence, based on the analytical estimate of the uncertainties, comparisons with simulator benchmarks, and comparisons with operating plant data, respectively. These three uncertainty values represent independent estimates of the RAMA calculational uncertainty.

Therefore, the staff requests that the BWRVIP, in calculating the final estimate of the RAMA calculational uncertainty, σ_c, use an appropriately weighted combination of these three values, where each weight reflects the reliability of the uncertainty estimate, and then normalize the weights. The staff requests that the BWRVIP address this issue and provide a justification.

BWRVIP-115, "BWR Vessel and Internals Project, RAMA Fluence Methodology Benchmark Manual - Evaluation of Regulatory Guide 1.190 Benchmark Problems"

RAI 115-1 Identify all differences between the methods used in performing the RAMA benchmark analyses of Reference 2 and the methods that will be used in performing the calculations of the vessel and shroud fluence. Also, address how the effects of these inconsistencies will be accounted for in determining the RAMA calculational bias and uncertainty.

RAI 115-2 (a) Regulatory Guide 1.190 requires that, as they become available, new measurements are to be incorporated into the M/C database and the fluence calculational bias and uncertainty estimates are to be updated, as necessary. The staff requests that the BWRVIP address how it will ensure that new measurements are incorporated in the M/C database and that the fluence bias and uncertainty will be updated in a timely manner.

(b) How many BWR samples (measurements) are currently available and when is it anticipated that a statistically significant set of measurements will be available to evaluate the overall bias?

RAI 115-3 In the calculation of the VENUS-3 benchmark, it is stated that the source is normalized to the experimental results. If the experimental results used for this normalization are the fluence measurements (which would erroneously reduce the M/C uncertainty), rather than the measurements of the core source distribution, discuss the effect that this simplification has on the calculational bias and uncertainty inferred from this benchmark comparison.

RAI 115-4 In Table 2-24, the sensitivity of the RAMA calculation of the NUREG-6115 benchmark problem to the axial distance between parallel rays has not been included (as in Table 2-16 for the H3R-2 calculation). Please discuss the sensitivity of the RAMA calculation to the axial distance between parallel rays. Please present your results on the same (or a similar) graph as Figures 5.4.6 or 5.4.8 of NUREG-6115.

BWRVIP-117, "BWR Vessel and Internals Project, RAMA Fluence Methodology Plant Application - Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1-5"

RAI 117-1 In Ref. 3, what criteria was used to select the sixty-three state points used to represent the Cycle 1-5 core operating history and what determination criteria was used in the weighing assignments of each state point calculation?

RAI 117-2 Was the Susquehanna Cycle 1-5 power, void and exposure distribution data based on calculational results or plant process computer data? If this data was the result of recent calculations, rather than the original historical calculations, discuss why new calculations were required and what differences were introduced in the calculations. Also, discuss the effect of any approximations used in representing the state-point dependence of the pin-wise source distribution of the peripheral fuel bundles.

RAI 117-3 Discuss the basis for the Table 5-3 parameter uncertainty for the following locations: (1) capsule and flux wire locations, (2) vessel inner radius, (3) core void fraction, (4) peripheral bundle power, and the (5) iron cross section.

RAI 117-4 Describe the spatial mesh used to represent the capsule and the capsule/vessel water gap.

RAI 117-5 What fluence uncertainty is introduced by the uncertainty in the Cu-63(n, α)Co-60, Fe-54(n, p)Mn-54 and Ni-58(n, p)Co-58 dosimetry cross sections?

RAI 117-6 Provide a discussion of the method used to determine the analytical modeling input bias and the associated uncertainty provided in Table 5-3.

RAI 117-7 In view of the fact that the uncertainty in the bias, inferred from the measurements of Table 5-4, is larger than the bias itself, provide justification for applying this bias to the RAMA calculated fluence.

RAI 117-8 In view of the fact that the RAMA calculation of the benchmark measurements used the "standard" fluence input parameters and the C/M comparisons (and the inferred C/M bias), address the effect of these parameters and provide justification for applying the analytical bias to the RAMA fluence calculation.

RAI 117-9 Discuss the methods used to measure the flux wire activations and conformance to ASTM E-263-93 (Ref. 4), ASTM E-263-93 (Ref. 5) and ASTM E-264-92 (Ref. 6). Also, discuss the basis for the 2.5% measurement accuracy.

BWRVIP-121, "BWR Vessel and Internals Project RAMA Fluence Methodology Procedures Manual"

RAI 121-1 Ref. 7 states that the BWR shroud is a "priority 1 component." However, no mention or attempt was made to demonstrate how RAMA performs in the evaluation of the shroud. Provide benchmarking data and calculations for the core shroud.

RAI 121-2 The staff requests that the BWRVIP provide a justification of the statement in the BWRVIP-121 report, "The nature of the guidelines is applicable to BWR plants without jet pumps..." In most BWRs, the dosimeters are placed behind the jet pump, which introduces spectral distortions, particularly for Fe and Ni dosimeters. If the BWRVIP report is indicating that the RAMA bias and uncertainties, based on jet pump plants, are applicable to plants without jet pumps, then the staff requests that the BWRVIP justify this statement.

TWE-PSE-001-R-001, "Hope Creek Flux Wire Dosimeter Activation Evaluation for Cycle 1"

1. The surveillance capsule is situated directly behind the jet pump. Given the "window" in the inelastic scattering of Fe in the 1.0 to 2.5 MeV range, what is the effect of the spectrum on the Fe, Ni, and Cu activation?

2. There is no mention of the estimation of the neutron spectrum in these calculations. The report states that there are 12 segments in the cycle, with different material compositions. It seems that the major differences in these segments are the decreasing concentration of U-235, the increasing concentration of Pu-239, and the increasing concentration of fission products. How do these changes affect the spectrum and how is it calculated?

3. What were the findings/results from the sensitivity study? Are the parameter default settings optimized?

4. Given the systematic underestimation of the Cu dosimeters, address whether an investigation shall be launched to determine if a dosimeter-specific bias exists?

5. The report states that the Cu discrepancy could be due to Co-59 impurity. The staff requests that the BWRVIP address that dosimeters supposed to be chemically and isotopically pure?

REFERENCES

1. BWRVIP-114, "BWR Vessel and Internals Project, RAMA Fluence Methodology Theory Manual," EPRI, Palo Alto, CA 2003 1003660.

2. "BWRVIP-115, "BWR Vessel and Internals Project, RAMA Fluence Methodology Benchmark Manual - Evaluation of Regulatory Guide 1.190 Benchmark Problems," EPRI, Palo Alto, CA 2003 1008063.

3. "BWRVIP-117, "BWR Vessel and Internals Project, RAMA Fluence Methodology Plant Application - Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1-5," EPRI, Palo Alto, CA 2003 1008065.

4. ASTM E-263-93, "Standard Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Iron," ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.

5. ASTM E-523-92, "Standard Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Copper," ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.

6. ASTM E-264-92, "Standard Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Nickel," ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.

7. BWRVIP-121, "BWR Vessel and Internals Project RAMA Fluence Methodology Procedures Manual," EPRI, Palo Alto, CA 2003 1008062.

B
BWRVIP RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

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BWRVIP BWR Vessel & Internals Project 2004-420

September 29, 2004

Document Control Desk
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852

Attention: Meena Khanna

Subject: Project No. 704 – BWRVIP Response to NRC Request for Additional Information on BWRVIP-114, -115, -117 and -121

References:
1. Letter from Meena Khanna (NRC) to Bill Eaton (BWRVIP Chairman), "Request for Additional Information – Review of BWR Vessel and Internals Project Reports, BWRVIP-114, BWRVIP-115, BWRVIP-117, and BWRVIP-121, and Transware Enterprises Inc. Report TWE-PSE-001-R-001, Revision 0 (TAC NO. MB9765)," dated April 20, 2004.
2. Letter from Carl Terry (BWRVIP Chairman) to Document Control Desk (NRC), "Project 704 – BWRVIP-114: BWR Vessel and Internals Project, RAMA Fluence Methodology Theory Manual," dated June 11, 2003.

Enclosed are ten (10) copies of the BWRVIP response to the NRC Request for Additional Information (RAI) on the BWRVIP-114, -115, -117, -121 reports on the RAMA fluence methodology and a Transware Enterprises report on a Hope Creek flux wire dosimeter evaluation that was transmitted to the BWRVIP by the Reference 1 NRC letter identified above. The enclosure repeats each of the items from the NRC RAI verbatim followed by the BWRVIP response to that item.

Please note that the enclosed document contains proprietary information. Therefore, the request to withhold the BWRVIP-114 report from public disclosure transmitted to the NRC by the Reference 2 letter identified above also applies to the enclosed document.

If you have any questions on this subject, please contact George Inch (Constellation Energy, BWRVIP Assessment Committee Technical Chairman) by telephone at 315.349.2441.

Sincerely,

William A Eaton

William A. Eaton
Entergy Operations
Chairman, BWR Vessel and Internals Project

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REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE ELECTRIC POWER RESEARCH INSTITUTE (EPRI) RAMA METHODOLOGY FOR REACTOR PRESSURE VESSEL FLUENCE EVALUATION

The U. S. Nuclear Regulatory Commission (NRC) has reviewed the RAMA Fluence Methodology documents submitted by the Boiling Water Reactor Vessel and Internals Project (BWRVIP) to qualify the application of the methodology for use in determining neutron fluence in BWR components. As a result of the review, twenty-seven Requests for Additional Information (RAIs) were identified in a letter transmitted to BWRVIP dated April 20, 2004. This report documents the response to these RAIs.

RAI 114-1 **Comment:** In the plant-specific applications, what specific tests and criteria are used to assure the adequacy of the number of rays and the number of volumes used in the RAMA fluence calculations?

Response: The adequacy of the RAMA fluence model parameters is assured by means of model sensitivity evaluations that are performed for each reactor model. A combination of 2-dimensional and 3-dimensional geometry and transport integration sensitivity evaluations are performed to ensure consistent results throughout the fluence model. Sections 4.6 and 4.7 of Ref. 7 describe the specific parametric cases and methodology for applying the 2-dimensional and 3-dimensional sensitivity evaluations, respectively, that are performed as a part of BWR vessel fluence calculations.

RAI 114-2 **Comment:** It is not evident that the RAMA geometry model described in Ref. 1 provides a correct representation of the true geometry (i.e., preserves the location, orientation and shape of all surfaces defining the physical geometry). For example, the modeling of the reflector region, surrounding the core, involves geometry elements that have both planar and cylindrical side boundaries. Since the geometry elements described in Ref. 1, Section 3.2, do not include bodies of this type, does RAMA introduce any distortion of the physical geometry in modeling the reflector and, if so, how is this distortion controlled to ensure acceptable accuracy?

Response: The solution regions in a RAMA geometry model are formed by combinations (i.e., intersections and differences) of the bodies described in Section 3.2 of Ref. 1. This allows complex geometries, including the transition between the rectangular core and the cylindrical shroud, to be precisely represented in a RAMA model. As

an example, a solution region can be formed by intersecting a right circular cylinder body with a rectangular parallelepiped body which results in a solution region that is cylindrical on one face and planar on the other faces. The use of these types of solution regions to transition between the planar core surfaces and the cylindrical shroud surface is illustrated in Figure 6-4 of Ref. 7.

RAI 114-3 **Comment:** The equation provided in Ref. 1, (Equation 7-38) for determining the M/C bias for the benchmark database requires an additional 1/M multiplicative normalization factor.

Response: The 1/M multiplicative factor was inadvertently omitted from the definition of the average value presented in Equation 7-38 of Ref. 1. The correct average value was used in the uncertainty evaluation presented in Ref. 3. Attachment 1 to this document contains a revised Page 7-16 from Ref. 1 illustrating the correct equation 7-38.

RAI 114-4 **Comment:** Equation 7-40 of Ref. 1 combines the analytical bias (B_a) and the benchmark bias (B_{b1}) to determine the overall calculational bias. The analytical bias (B_a), defined in Equation 7-34, provides the effect of not using the optimum asymptotic calculational input in the RAMA fluence calculation. Since the benchmark biases include the effect of the approximate calculational input used in the benchmark calculations (i.e., use of the standard input parameters rather than the asymptotic parameters), the analytical bias is only required when there is an inconsistency between the input used in the vessel fluence calculations and the benchmark calculations; e.g., when the calculations of the benchmark measurements are made with the asymptotic input values and the vessel fluence calculations are made with the standard input values. The staff requests that the BWRVIP clearly address the determination of the bias.

Response: It is acknowledged that the analytical bias that is determined from vessel fluence sensitivity evaluations is implicitly included in the benchmark and operating plant measurement bias. The theoretical basis for determining the analytical bias is included in the RAMA fluence methodology for completeness. In general practice, the analytical bias can be omitted from the uncertainty evaluation, but will be available if an analytical bias adjustment to the calculated fluence is required.

RAI 114-5 **Comment:** The weights defined in Equation 7-41 are not normalized (i.e., sum to unity), as required. Also, the weights should reflect the reliability of the bias estimates. If, for example, a weight of $1/\sigma^2$ is used,

the σ should represent the standard deviation of the bias estimate, not the standard deviation of the M/C data about the mean.

Response: An error existed in the definition of the weighting factor in Equation 7-42 in the original Ref. 1 document. A revision to the weighting factor definition was issued as: Errata for "BWRVIP-114: BWR Vessel and Internals Project, RAMA Fluence Methodology Theory Manual," 1003660 May 2003 and was transmitted to the NRC with a letter from Carl Terry, Chairman of the BWRVIP, dated August 21, 2003. The revision provides for weights that are normalized (i.e., sum to unity), as expected. Since the measurement bias estimate is based on the mean of the M/C data, using the standard deviation of the measurement data should provide a reasonable estimate of the standard deviation of the bias estimate. The revised equation is shown in Attachment 2.

RAI 114-6 **Comment:** The values of σ_a, σ_{b1} and σ_{b2} of Equation (7-43) represent the (one standard deviation) uncertainty in the RAMA calculated fluence, based on the analytical estimate of the uncertainties, comparisons with simulator benchmarks, and comparisons with operating plant data, respectively. These three uncertainty values represent independent estimates of the RAMA calculational uncertainty.

Therefore, the staff requests that the BWRVIP, in calculating the final estimate of the RAMA calculational uncertainty, σ_c, use an appropriately weighted combination of these three values, where each weight reflects the reliability of the uncertainty estimate, and then normalize the weights. The staff requests that the BWRVIP address this issue and provide a justification.

Response: It is correct that each of the three uncertainty values represents independent estimates of the RAMA calculational uncertainty. Using the unweighted contribution of the individual uncertainty values, as proposed in Ref. 1, is conservative in that it leads to an overestimate of the uncertainty. However, it is appropriate to estimate the overall uncertainty using a weighted mean of each of the three uncertainty estimates. Therefore, the BWRVIP intends to revise the computational process for determining the calculational uncertainty to incorporate a weighted treatment of the individual uncertainty components as shown in Equation 7-43 of Attachment 2. The weight factors of Equation 7-41 (w_a, w_b and w_c) are now multiplied by their respective variances to obtain a weighted mean.

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The application of the revised uncertainty treatment will be documented in BWRVIP-117 (Ref. 3). Attachment 3 to this document contains revised Page 5-5 of Ref. 3 that illustrates the application of the revised uncertainty treatment.

RAI 115-1 **Comment:** Identify all differences between the methods used in performing the RAMA benchmark analyses of Ref. 2 and the methods that will be used in performing calculations of the vessel and shroud fluence. Also, address how the effects of these inconsistencies will be accounted for in determining the RAMA calculational bias and uncertainty.

Response: The methods used in performing the RAMA benchmark analyses in Ref. 2 are the same as the methods that will be used in performing BWR vessel and shroud fluence calculations. The methods are described in Ref. 7. The application of the methods to operating BWRs is described in Refs. 3 and 9.

RAI 115-2(a) **Comment:** Regulatory Guide 1.190 requires that, as they become available, new measurements are to be incorporated into the M/C database and the fluence calculational bias and uncertainty estimates are to be updated, as necessary. The staff requests that the BWRVIP address how it will ensure that new measurements are incorporated in the M/C database and that the fluence bias and uncertainty will be updated in a timely manner.

Response: The comparisons to measured surveillance capsule and benchmark dosimetry are maintained in a database that is updated as additional plant capsule evaluations are performed using the RAMA methodology. The fluence bias and uncertainty are re-evaluated as new comparison data is added to the database. At present, TransWare Enterprises Inc., a primary contractor to EPRI and the BWRVIP, is performing fluence calculations using RAMA. TransWare also maintains a surveillance capsule and benchmark dosimetry measurement database. However, it is envisioned that in the future other organizations may choose to perform the fluence calculations and contribute to the database. Therefore, the BWRVIP will consider options for establishing a mechanism to collect and evaluate new M/C data and disseminate the information to all users of RAMA.

RAI 115-2(b) **Comment:** How many BWR samples (measurements) are currently available and when is it anticipated that a statistically significant set of measurements will be available to evaluate the overall bias?

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Response: The current RAMA comparison database includes comparisons to 15 measurement samples from two BWR-4 reactors and 237 measurement samples from three capsules in a BWR-2 reactor with no jet pumps. Work currently being performed includes comparison to measurements from three different BWR-4 reactors with jet pumps for the following measurements: 1) three additional surveillance capsules; 2) scrapings from various axial locations in the core shroud and top guide; and 3) samples from shroud head bolts. This work and other anticipated comparisons will provide a statistically significant set of measurements for both jet pump and non-jet pump BWRs when this work is completed (estimated to be within two years). This work will also demonstrate RAMA's capability to determine fluence for additional reactor system components.

RAI 115-3 **Comment:** In the calculation of the VENUS-3 benchmark, it is stated that the source is normalized to the experimental results. If the experimental results used for this normalization are the fluence measurements (which would erroneously reduce the M/C uncertainty), rather than the measurements of the core source distribution, discuss the effect that this simplification has on the calculational bias and uncertainty inferred from this benchmark comparison.

Response: The VENUS-3 measurement results reported by the experimenters included a normalization to an arbitrary source magnitude. The intent of the statement regarding the normalized source is to indicate that the same source magnitude used by the VENUS-3 experimenters was also used in the RAMA benchmark calculation. There was no normalization of the RAMA predicted activation to measured values.

RAI 115-4 **Comment:** In Table 2-24, the sensitivity of the RAMA calculation of the NUREG-6115 benchmark problem to the axial distance between parallel rays has not been included (as in Table 2-16 for the HBR-2 calculation). Please discuss the sensitivity of the RAMA calculation to the axial distance between parallel rays. Please present your results on the same (or a similar) graph as Figures 5.4.6 or 5.4.8 of NUREG-6115.

Response: The sensitivity of the RAMA calculation of the NUREG-6115 benchmark problem to the axial distance between parallel rays is determined by evaluating the >1.0 MeV neutron flux at the capsule location for various values of the parallel ray axial distance. The axial distance between parallel rays was varied over a range of 2 cm to 16 cm. Over the range of 2 cm to 9 cm the maximum observed deviation was ≤1%. Thus, the default value of 5 cm was conservatively used in

the RAMA calculation. Attachment 4 contains revised Pages 2-46 through 2-48 of Ref. 2. The sensitivity of the RAMA calculation of the NUREG-6115 benchmark problem to the axial distance between parallel rays is included in Table 2-24 of Attachment 4 and the plot that illustrates the sensitivity is provided in Figure 2-20 of Attachment 4.

RAI 117-1 **Comment:** In Ref. 3, what criteria was used to select the sixty-three state points used to represent the Cycle 1-5 core operating history and what determination criteria was used in the weighting assignments of each state point calculation?

Response: The guidelines and criteria for selecting the state points that are to be used in RAMA fluence evaluations are described in Section 5.2.1 of Ref. 7. Daily reactor power for the period over which a state point is deemed representative is used as the weighting assignment for each state point calculation.

RAI 117-2 **Comment:** Was the Susquehanna Cycle 1-5 power, void and exposure distribution data based on calculational results or plant process computer data? If this data was the result of recent calculations, rather than the original historical calculations, discuss why new calculations were required and what differences were introduced in the calculations. Also, discuss the effect of any approximations used in representing the state-point dependence of the pin-wise source distribution of the peripheral fuel bundles.

Response: The Susquehanna power, void, and exposure distribution data were based upon "core follow" calculations that were performed during the five cycles of operation. Restart edit cases were executed to retrieve the required data from the previous calculations, however, no recalculation of data was performed. The core calculations provide pin-wise power distributions for each bundle in the core for each state point that was used in the analysis. Thus no approximations were needed to represent the state-point dependence of the pin-wise source distribution of the peripheral fuel bundles.

RAI 117-3 **Comment:** Discuss the basis for the Table 5-3 parameter uncertainty for the following locations: (1) capsule and flux wire locations, (2) vessel inner radius, (3) core void fraction, (4) peripheral bundle power, and the (5) iron cross section.

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Response: (1) The uncertainty in radial and axial locations of the capsule is based upon the design drawing tolerances. The uncertainty in capsule azimuthal location is based upon as-built measurements from a similar BWR. The uncertainty in the location of the flux wires is based upon the assumption that the flux wires can be located anywhere within the surveillance capsule. (2) The uncertainty in RPV inner radius is based upon design drawing tolerances. (3) The uncertainty in void fraction is based upon experience estimates of ±5% variation in computed void fraction. (4) The uncertainty in peripheral bundle power is based upon the reported accuracy of the core simulation analysis computer code. (5) The uncertainty in the iron cross section is based upon experience estimates of ±10% uncertainty in the cross section.

RAI 117-4 **Comment:** Describe the spatial mesh used to represent the capsule and the capsule/vessel water gap.

Response: Figures 4-1, 4-2, and 4-4 of Ref. 3 illustrate the location and size of the capsule in the Susquehanna fluence model. The capsule is positioned in the radial plane to provide for a water gap between the capsule and pressure vessel wall. The capsule geometry is represented with 12 mesh volumes of the following configuration: 3 azimuthal sectors, 2 radial annuli, and 2 axial planes. The water gap between the capsule and the pressure vessel wall is represented with 6 mesh volumes of similar configuration to the capsule with the exception that 1 annulus is used to represent the radial thickness of the gap.

RAI 117-5 **Comment:** What fluence uncertainty is introduced by the uncertainty in the Cu-63(n, α)Co-60, Fe-54(n, p) Mn-54 and Ni-58(n, p)Co-58 dosimetry cross sections?

Response: The dosimetry cross sections are used in the comparison of calculated activations to measurements so that the uncertainty introduced by the activation cross sections is inherently included in the comparison of calculations to measurements for the respective dosimetry reactions. As a result, no separate estimate of the uncertainty associated with activation cross sections is required.

RAI 117-6 **Comment:** Provide a discussion of the method used to determine the analytical modeling input bias and the associated uncertainty provided in Table 5-3.

Response: The method used to determine the analytical modeling uncertainty and bias estimation is described in Section 7.3.1 of Ref. 1 and in Section 8 of Ref. 7.

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RAI 117-7 **Comment:** In view of the fact that the uncertainty in the bias, inferred from the measurements of Table 5-4, is larger than the bias itself, provide justification for applying this bias to the RAMA calculated fluence.

Response: The application of the bias in the case of the Susquehanna fluence evaluation is provided as an example of the bias application process. As described in Section 8.3.1 of Ref. 7, the application of a computed bias to the fluence evaluation should only be done when the bias is statistically significant. Section 5.4 of the Susquehanna fluence evaluation presented in Ref. 3 will be revised to be consistent with the anticipated application of the analytic (and overall) bias treatment in practice. Attachment 3 to this document provides a revised Page 5-5 that clarifies the intended treatment.

RAI 117-8 **Comment:** In view of the fact that the RAMA calculation of the benchmark measurements used the "standard" fluence input parameters and the C/M comparisons (and the inferred C/M bias), address the effect of these parameters and provide justification for applying the analytical bias to the RAMA fluence calculation.

Response: As noted in the response to RAI 114-4, the analytical bias is generally implicitly included in the measurement comparisons. The application of an analytical bias in the case of the Susquehanna fluence evaluation was carried out to demonstrate the application of an analytical bias should there be inconsistencies between the methodology used for the measurement comparisons and the fluence evaluation. In addition, any combined bias should be applied only if it is statistically significant (Section 8.3.1 of Ref. 7), which is not the case for the Susquehanna evaluation. Section 5.4 of the Susquehanna fluence evaluation presented in Ref. 3 will be revised to be consistent with the anticipated application of the analytic (and overall) bias treatment in practice. Attachment 3 to this document provides a revised Page 5-5 that clarifies the intended treatment.

RAI 117-9 **Comment:** Discuss the methods used to measure the flux wire activations and conformance to ASTM E-263-93 (Ref. 4), ASTM E-263-93 (Ref. 5) and ASTM E-264-92 (Ref. 6). Also, discuss the basis for the 2.5% measurement accuracy.

Response: The flux wire measurements were performed by GE. The methods used to measure the flux wire activations, measurement results, and measurement accuracy are described in Ref. 8.

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RAI 121-1

Comment: Ref. 7 states that the BWR shroud is a "priority 1 component." However, no mention or attempt was made to demonstrate how RAMA performs in the evaluation of the shroud. Provide benchmarking data and calculations for the core shroud.

Response: The purpose of the Ref. 7 document is to provide general modeling guidelines that can be used to assist users in the application of RAMA to BWR component fluence evaluations. Application of the RAMA methodology to RPV vessel and surveillance capsule fluence evaluations, including comparison of calculated values to measurements, is described in Refs. 1, 2, and 3. Application of the RAMA Fluence Methodology to the core shroud in the active fuel region is straightforward since this region is modeled to obtain the RPV fluence. In Ref. 7 the shroud is evaluated using the same criteria as the RPV in the geometry meshing sensitivity studies. A benchmark evaluation is currently underway to demonstrate the adequacy of the RAMA Fluence Methodology for determining the fluence of the core shroud and the top guide.

RAI 121-2

Comment: The staff requests that the BWRVIP provide a justification of the statement in the BWRVIP-121 report, "The nature of the guidelines is applicable to BWR plants without jet pumps...". In most BWRs, the dosimeters are placed behind the jet pump which introduces spectral distortions, particularly for Fe and Ni dosimeters. If the BWRVIP report is indicating that the RAMA bias and uncertainties, based on jet pump plants, are applicable to plants without jet pumps, then the staff requests that the BWRVIP justify this statement.

Response: The intent of the statement is to indicate that the general modeling guidelines and process for evaluating the adequacy of the RAMA methodology described in Ref. 7 are valid for BWR plants with and without jet pumps. There is no intent to imply that the results obtained from evaluations performed in accordance with the methodology described in Ref. 7 are the same for BWR plants with and without jet pumps. Paragraph 4 on Page 1-1 of Ref. 7 has been revised to clarify this matter. The revised Page 1-1 is provided in Attachment 5 to this document.

RAI HC-1

Comment: The surveillance capsule is situated directly behind the jet pump. Given the "window" in the inelastic scattering of Fe in the 1.0 to 2.5 MeV range, what is the effect of the spectrum on the Fe, Ni, and Cu activation?

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Response: The RAMA Fluence Methodology has the capability to accurately represent jet pumps in the transport model. As a result, the spectral effects associated with the presence of the jet pumps is implicitly included in the transport calculation. Comparative studies show that the calculated activities for Fe, Ni, and Cu are consistently predicted (Refs. 2, 3, and 9) for jet pump and non-jet pump plants. Relative to each isotope, Cu activities have shown a consistent ~5% negative bias relative to Fe and Ni. Because jet pump and non-jet pump plants show the same trend, it is suggested that the difference in the calculated Cu activities is attributable to either the Cu cross sections or unaccounted for impurities in the metal (see RAI HC-4 and RAI HC-5).

RAI HC-2

Comment: There is no mention of the estimation of the neutron spectrum in these calculations. The report states that there are 12 segments in the cycle, with different material compositions. It seems that the major differences in these segments are the decreasing concentration of U-235, the increasing concentration of Pu-239, and the increasing concentration of fission products. How do these changes affect the spectrum and how is it calculated?

Response: Each segment (or state point) represents an exposure interval of the reactor cycle. The intervals for the analysis were selected in accordance with the criteria presented in Section 5.2.1 of Ref. 7. The state point data for each state point includes fuel isotopics (i.e., the number densities for the uranium and plutonium nuclides) corresponding to the exposure of the state point. The spectrum is calculated in RAMA using a weighting based upon the contribution of the various uranium and plutonium nuclides, as described in Equation 4-25 of Ref. 1.

RAI HC-3

Comment: What were the findings/results from the sensitivity study? Are the parameter default settings optimized?

Response: The results of the sensitivity study for Hope Creek are reported in Section 4.4 of Ref. 9 and are consistent with the results observed for the other operating plants (BWR and PWR) reported in Refs. 2 and 3. All of the parameters except the mesh size and angular quadrature selection are optimized. These latter two parameters can have significant computational penalties, thus both are evaluated to provide an acceptable balance between accuracy and computational performance. The mesh size results in <3% deviation from asymptotic value and the angular quadrature selection results in <7% deviation from the asymptotic value. The parameter set used in the fluence

evaluation provides acceptable accuracy and computational performance.

RAI HC-4 **Comment:** Given the systematic underestimation of the Cu dosimeters, address whether an investigation shall be launched to determine if a dosimeter-specific bias exists?

Response: It is observed from the benchmarks that the underestimation of Cu activities is consistent and on the order of about 5%. It is noted in the H. B. Robinson benchmark report (Ref. 10) that impurities in the Cu metal, specifically cobalt, can account for about 2% of the difference. Predicated on this statement and the response provided for RAI HC-5, it is not clear whether the observed bias is material or cross section related. Further investigation would need to include the full compositional characterization of the Cu metal. The BWRVIP has no plans to investigate this matter.

RAI HC-5 **Comment:** The report states that the Cu discrepancy could be due to Co-59 impurity. The staff requests that the BWRVIP address that dosimeters supposed to be chemically and isotopically pure?

Response: The possibility of trace (on the order of <0.25 ppm) cobalt impurity in pure copper has been acknowledged by copper industry experts (Ref. 11). Due to the large thermal neutron reaction rate of cobalt-59, this level of impurity can lead to a few percent of additional cobalt-60 in the dosimeter due to the activation of cobalt-59. A correction of approximately 2% for cobalt impurity in the copper dosimetry was provided for in the H. B. Robinson Unit 2 Cycle 9 benchmark results reported in Ref. 10.

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REFERENCES

1. BWRVIP-114, "BWR Vessel and Internals Project, RAMA Fluence Methodology Theory Manual," EPRI, Palo Alto, CA 2003. 1003660.

2. BWRVIP-115, "BWR Vessel and Internals Project, RAMA Fluence Methodology Benchmark Manual - Evaluation of Regulatory Guide 1.190 Benchmark Problems," EPRI, Palo Alto, CA 2003. 1008063.

3. BWRVIP-117, "BWR Vessel and Internals Project, RAMA Fluence Methodology Plant Application - Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1-5," EPRI, Palo Alto, CA 2003. 1008065.

4. ASTM E-263-93, "Standard Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Iron," ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.

5. ASTM E-523-92, "Standard Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Copper," ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.

6. ASTM E-264-92, "Standard Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Nickel," ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1995.

7. BWRVIP-121, "BWR Vessel and Internals Project RAMA Fluence Methodology Procedures Manual," EPRI, Palo Alto, CA 2003 1008062.

8. *Susquehanna Steam Electric Station Unit 2 Vessel Surveillance Materials Testing and Fracture Toughness Analysis*, GE Nuclear Energy, GE-NE-523-107-0893, DRF 137-0010-6, October 1993, Rev. 1.

9. "Hope Creek Flux Wire Dosimeter Activation Evaluation for Cycle 1," TransWare Enterprises Inc., TWE-PSE-001-R-001, Rev. 0, October 2003.

10. I. Remec and F. B. K. Kam, "H. B. Robinson-2 Pressure Vessel Benchmark," NUREG/CR-6453, Oak Ridge National Laboratory, ORNL/TM-13204, February 1998.

11. Personal email communication from Dr. Horace Pops of Superior Essex to Mr. William Black of the Copper Development Association, "Cobalt Impurity in Copper Wires," April 14, 2003.

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Attachment 1

BWRVIP 114: BWR Vessel and Internals Project RAMA Fluence Methodology Theory Manual, Revised Page 7-16

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Activation, Fluence, and Uncertainty Methods

The bias, based upon comparison of calculated to measured dosimeter results, is:

$$B_b = \frac{1}{M}\sum_{i=1}^{M}\frac{m_i - c_i}{c_i} = \frac{1}{M}\sum_{i=1}^{M}\left(\frac{m_i}{c_i} - 1\right) \qquad (7\text{-}38)$$

where m_i is the i-th measured activation value in the database and c is the i-th calculated activation value. Note that an implicit assumption in Eg. (7-38) is that the relative bias based upon comparison to measured values applies to RPV locations as well.

The elements contributing to the comparison uncertainty analysis are generally quite different for the vessel simulator benchmark evaluations as opposed to operating light water reactor dosimetry evaluations. As a result, the bias and uncertainty (standard deviation) are determined using the above methodology for two different measurement databases: (1) the vessel simulator benchmark database consisting of comparison results for the PCA and VENUS-3 benchmark problems, and (2) the operating system database consisting of dosimetry measurement data from operating light water reactor plants.

The comparison databases must be evaluated to confirm their statistical validity for use in determining the RPV "best estimate" bias. Statistical valid databases must meet three criteria: (1) the database should provide a representative sample over the range of operating states for which the fluence evaluation methodology is to be applied, (2) the uncertainty in the database comparisons should be small compared to the comparison bias, and (3) the calculation and measurement errors of the comparison ratios must be uncorrelated (i.e., no systematic bias is present in the comparisons).

The method of evaluating the extent of correlated comparisons in the databases, and the method for removing the correlated bias is described in [9]. The database comparisons are expressed in a regression model of the form:

$$\left(\frac{m}{c}\right) = \mu_{m/c} + \sum_{k=1}^{K} c_k \alpha_k \qquad (7\text{-}39)$$

where $\mu_{m/c}$ is the fitted mean of the comparisons, c_k are fit coefficients, and α_k are parameters that represent various possible correlation conditions, such as the type of detector, the location of the detector (e.g., in-vessel and behind jet pumps), the energy threshold of the detector, etc. The statistics of the fit parameters are used to determine correlated parameters. The regression model of Eq. (7-39) is used to remove the systematic bias from the measurement comparisons. The measurement comparisons are used to determine an adjusted bias, as in Eq. (7-38).

7.3.3 *Combined Uncertainty*

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Attachment 2

BWRVIP 114: BWR Vessel and Internals Project RAMA Fluence Methodology Theory Manual, Revised Page 7-17

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Activation, Fluence, and Uncertainty Methods.

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7.3.4 *Best Estimate Fluence*

The combined fluence bias and standard deviation determined from Section 7.3.3 are used to compute the best estimate neutron fluence from the calculated fluence as specified in [1] using the following methodology.

If the combined standard deviation is ≤20%, the best estimate neutron fluence is

$$\varphi = \varphi_c\left(1 \div B_c\right) \tag{7-44}$$

where φ_c is the calculated neutron fluence and B_c is the combined fluence bias. If the combined standard deviation is greater than 20% but less than 30%, the best estimate neutron fluence is

$$\varphi = \varphi_c\left(1 + B_c \div \frac{\sigma_c(\%) - 20}{100}\right) \tag{7-45}$$

where σ_c is the combined fluence standard deviation from Eq. (7-43).

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Attachment 3

BWRVIP-117: RAMA Fluence Methodology Plant Application – Susquehanna Unit 2 Surveillance Capsule Evaluation, Revised Page 5-5

The combined capsule bias (and uncertainty) is the weighted sum of the analytic and comparison biases (and uncertainties) where the weighting factors are inversely proportional to the analytic and comparison variances, respectively [3]. Table 5-4 shows that the combined capsule uncertainty is determined to be 10.0% with a bias of –0.7% for both the >1.0 MeV fluence and the >0.1 MeV fluence. The combined uncertainty is less than 20 percent as recommended in Section 1.4.3 of Regulatory Guide 1.190 [6].

Table 5-4
Combined Capsule Uncertainty

Energy Range	Analytic Bias Weight Factor	Comparison Bias Weight Factor	Combined Bias %	Combined Uncertainty % (1σ)
>1.0 MeV Average	0.22	0.78	-0.7	10.0
>0.1 MeV Average	0.22	0.78	-0.7	10.0

5.4 Best Estimate Neutron Fluence and Flux

Table 5-5 provides the RAMA calculated best estimate neutron fluence and rated power flux values for the Susquehanna Unit 2 capsule for energy >1.0 MeV and for energy >0.1 MeV. Since the combined bias from Section 5.3 of this report is substantially smaller than the corresponding combined uncertainty, the computed combined bias is not statistically significant. The combined uncertainty of 10.0% is also less than 20% as specified in Regulatory Guide 1.190. Therefore, the best estimate values for flux and fluence are equivalent to the calculated values (i.e., no bias is applicable for the calculated neutron flux and fluence). The best estimate capsule neutron fluence for energy >1.0 MeV is 1.555×10^{17} n/cm^2 and for energy >0.1 MeV is 2.801×10^{17} n/cm^2. The best estimate capsule rated power neutron flux for energy >1.0 MeV is 7.930×10^{8} n/cm^2-s and for energy >0.1 MeV is 1.428×10^{9} n/cm^2-s.

Table 5-5
Best Estimate Neutron Fluence and Rated Power Flux for Susquehanna Unit 2 Capsule

Energy Range	Fluence n/cm^2	Standard Deviation n/cm^2	Rated Power Flux n/cm^2-s	Standard Deviation n/cm^2-s
>1.0 MeV Average	1.555E+17	1.555E+16	7.930E+08	7.930E+07
>0.1 MeV Average	2.801E+17	2.801E+16	1.428E+09	1.428E+08

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Attachment 4

BWRVIP 115: BWR Vessel and Internals Project RAMA Fluence Methodology Benchmark Manual — Evaluation of Regulatory Guide 1.190 Benchmark Problems, Revised Pages 2-46 through 2-48

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Attachment 5

BWRVIP 121: BWR Vessel and Internals Project RAMA Fluence Methodology Procedures Manual, Revised Page 1-1

1
INTRODUCTION

The BWR Vessel and Internals Project (BWRVIP) has developed the RAMA Fluence Methodology (hereinafter referred to as the Methodology) for use in calculating neutron fluence in boiling water reactors (BWRs). The current version of the Methodology is applicable for calculations at the surveillance capsule location as well as on the core shroud and within the reactor vessel over the active fuel height. The Methodology is designed to meet the requirements of the U. S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.190 [1].

The Methodology includes computerized analysis tools that perform neutron fluence calculations, modeling guidelines that describe the use of the methodology, and benchmark reports that document the capability of the Methodology to accurately predict neutron fluence. The benchmark problems that have been used to demonstrate the capability of the Methodology include the analysis of specific benchmark problems identified in the NRC Regulatory Guide 1.190 and analyses of surveillance capsule measurements for commercial BWRs.

Accurate neutron fluence determinations are required for a number of reasons: 1) to determine neutron fluence in the reactor pressure vessel (RPV) and at surveillance capsule locations to address vessel embrittlement issues; 2) to determine neutron fluence in the core shroud in order to determine fracture toughness and crack growth rate for use in flaw evaluation calculations; and 3) to determine neutron fluence in other internal components above and below the active core for structural integrity assessments or to evaluate repair technologies. Fluence predictions are potentially required in other parts and locations within the reactor pressure vessel. However, the near term need for fluence calculations includes mainly the internals such as the pressure vessel, core shroud, surveillance capsule locations, and jet pumps, at elevations within the height of the active fuel.

This manual is intended to provide guidelines for the user of the Methodology to assist in ascertaining the fluence evaluation to be performed, collecting the data needed for the evaluation, building the geometry models for the reactor and components of interest, processing material data, evaluating the flux and fluence results generated by the Methodology, and performing an uncertainty analysis of the results. The discussions and examples in this manual describe the modeling and analysis process for typical BWR plants with jet pumps. However, the basic process presented in the guidelines is applicable to BWR plants without jet pumps as well.

A summary of the remaining sections of this manual is presented in the following paragraphs. Section 2 of this manual presents an overview of the Methodology software package. The individual software components that comprise the Methodology are presented along with a brief discussion of the calculational flow and overview of the entire modeling process.

C
RECORD OF REVISIONS

BWRVIP-117-A	Information from the following documents was used in preparing the changes included in this revision of the report: 1. BWRVIP-117: BWR Vessel and Internals Project, RAMA Fluence Methodology Plant Application—Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1–5, EPRI, Palo Alto, CA: 2003. 1008065 2. Letter from Stephanie M. Coffin (NRC) to William Eaton (BWRVIP Chairman), Request for Additional Information – Review of BWR Vessel and Internals Project Reports, BWRVIP-114, BWRVIP-115, BWRVIP-117 and BWRVIP-121 and TransWare Enterprises Inc. Report TWE-PSE-001-R-001, Revision 0 (TAC NO. MB9765) dated April 20, 2004 (BWRVIP Correspondence File Number 2004-159). 3. Letter from Carl Terry (BWRVIP Chairman) to Meena Khanna (NRC), "Project NO. 704 – BWRVIP Response to NRC Request for Additional Information on BWRVIP-114, -115, -117 and -121" dated September 29, 2004 (BWRVIP Correspondence File Number 2004-420). 4. Letter from William H. Bateman (NRC) to Bill Eaton (BWRVIP Chairman), Safety Evaluation of Proprietary EPRI Reports, "BWR Vessel and Internals Project, RAMA Fluence Methodology Manual (BWRVIP-114)," "RAMA Fluence Methodology Benchmark Manual - Evaluation of Regulatory Guide 1.190 Benchmark Problems (BWRVIP- 115)," " RAMA Fluence Methodology-Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1 - 5 (BWRVIP-117)," and " RAMA Fluence Methodology Procedures Manual (BWRVIP-121)," and "Hope Creek Flux Wire Dosimeter Activation Evaluation for Cycle 1 (TWE-PSE-001-R-001)" (TAC NO. MB9765) dated may 13, 2005 (BWRVIP Correspondence File Number 2005-308). Details of the revisions can be found in Table C-1.

Table C-1
Revision details

Required Revision	Source of Requirement for Revision	Description of Revision Implementation
Add NRC Correspondence	NRC Request	NRC Safety Evaluation added behind report title page. Remainder or correspondence added as Appendices A through B.
Section 2, Summary and Conclusions	BWRVIP Response to RAI 117-7 and 117-8	Removed "predicted" and "Best estimate" when referring to rated power and neutron flux and fluence throughout section 2. This is for consistency with the RAI response regarding removal of bias with respect to Susquehanna.
Section 3.3.2, Reactor State Point Data	BWRVIP Response to RAI 117-2	The following sentence was added to the end of paragraph 1: "The data files represent core simulation evaluations that are performed as a part of routine plant performance tracking."
Section 3.3.2, Reactor State Point Data	BWRVIP Response to RAI 117-1	The following sentence was added to the end of paragraph 2: "The guidelines and criteria for selecting the state points for use in RAMA fluence evaluations are described in [5]."
Section 3.3.3, Core Loading Pattern	Editorial	The following sentences from Section 3.3.3, paragraph 2 were edited for clarification: Original Text: "The cycle core loading patterns provided by PPL Susquehanna were used to identify the fuel assembly designs in each cycle and to identify the dominant fuel design loaded in the core peripheral locations. For each cycle, the dominant fuel assembly design was used to build the reactor core region of the RAMA fluence model for Susquehanna Unit 2." New Text: "The cycle core loading patterns provided by PPL Susquehanna were used to identify the fuel assembly designs in each cycle and to identify the dominant fuel design loaded in the core peripheral locations, which is used to build the reactor core region of the RAMA fluence model."

Table C-1
Revision Details (continued)

Required Revision	Source of Requirement for Revision	Description of Revision Implementation
Section 4.2, The RAMA Geometry Model for Susquehanna Unit 2	BWRVIP Response to RAI 117-4	The following was added to the end of paragraph 1 in paragraph 7 of section 4.2: "The capsule geometry model consists of three azimuthal sectors, two radial annuli, and two axial planes. The water gap between the capsule and the inside surface of the reactor pressure vessel clad is modeled with similar mesh representation except that the radial thickness consists of a single annulus."
Section 5.3, Surveillance Capsule Uncertainty Evaluation	Reg. Guide 1.190; BWRVIP-189; BWRVIP Response to RAI 117-7 and 117-8	Added "calculational" to the final paragraph of Section 5.3 with respect to bias and uncertainty. This change was made to be consistent with terminology used in Reg. Guide 1.190 and BWRVIP-189. Also corrected uncertainty value in text to match new value in Table 5-4.
Section 5.4	BWRVIP Response to RAI 117-7 and 117-8	Section 5.4 of the Susquehanna fluence evaluation presented in Ref. 3 has been revised to be consistent with the anticipated application of the analytic (and overall) bias treatment in practice.
Section 6, References	Editorial	References 3, 5, and 7 updated with current report dates.
Add NEI 03-08 Implementation Requirements	BWRVIP-94, Revision 1 Requirement	Implementation Requirements Added in Section 1.1.

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