March 11, 2005

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Attention: Meena Khanna

Subject: Project No. 704 – BWRVIP-117NP: BWR Vessel and Internals Project, RAMA Fluence Methodology Plant Application – Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1-5

Reference: Letter from Carl Terry (BWRVIP) to Document Control Desk (NRC), "Project 704 – BWRVIP-117: BWR Vessel and Internals Project, RAMA Fluence Methodology Plant Application – Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1-5," dated August 1, 2003.

Enclosed are two (2) copies of the report "BWRVIP-117NP: BWR Vessel and Internals Project, RAMA Fluence Methodology Plant Application – Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1-5," EPRI Technical Report 1008065NP, March 2005. This is a non-proprietary version of the proprietary document submitted to the NRC by the letter referenced above.

If you have any questions on this subject please call Robin Dyle (Southern Nuclear, BWRVIP Integration Committee Technical Chairman) at 205.992.5882.

Sincerely,

William A Eaton

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

Copy forwarded to Meena Khanna

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EPRI

BWRVIP-117NP: BWR Vessel and Internals Project RAMA Fluence Methodology Plant Application—Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1–5

Technical Report

BWRVIP-117NP: BWR Vessel and Internals Project

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

1008065NP

Final Report, March 2005

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TransWare Enterprises Inc.

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This report describes research sponsored by EPRI and its BWRVIP participating members.

The report is a corporate document that should be cited in the literature in the following manner:

BWRVIP-117NP: 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. 1008065NP.

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 that was performed to qualify the RAMA Fluence Methodology for use in the evaluation of neutron fluence in BWRs.

Results & Findings

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

Challenges & 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 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, Values & 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 reactor pressure vessel 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 reactor pressure vessel (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 at other internal components for structural integrity assessments or to evaluate repair technologies

The RAMA Fluence Methodology is a state-of-the-art and 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

Fluence
Embrittlement
Boiling water reactor
Vessel and internals
Reactor pressure vessel

ABSTRACT

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.

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

ACKNOWLEDGMENTS

The undersigned wish to acknowledge EPRI and the Boiling Water Reactor Vessel and Internals Project (BWRVIP) for their support of the RAMA Fluence Methodology project. Special thanks is given to Robert Carter of EPRI for his guidance, comments, and overall support in completing this document.

Mr. Ken Knoll and Mr. Bruce Swoyer of PPL Susquehanna, LLC are acknowledged for providing the operating history, fuels data and the mechanical design data for this evaluation. Steven Baker of TransWare Enterprises Inc. is acknowledged for his effort in assisting PPL Susquehanna in the collection and formatting of the data. Charlotte Potze deserves special recognition for her concentrated effort in the performance of the evaluation documented in this report.

The undersigned also wish to acknowledge Dr. Mark Williams of Louisiana State University for providing a thorough technical review of the information provided in this report.

Dean B. Jones, TransWare Enterprises Inc.

Kenneth E. Watkins, TransWare Enterprises Inc.

CONTENTS

LIST OF FIGURES

LIST OF TABLES

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.

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.

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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.

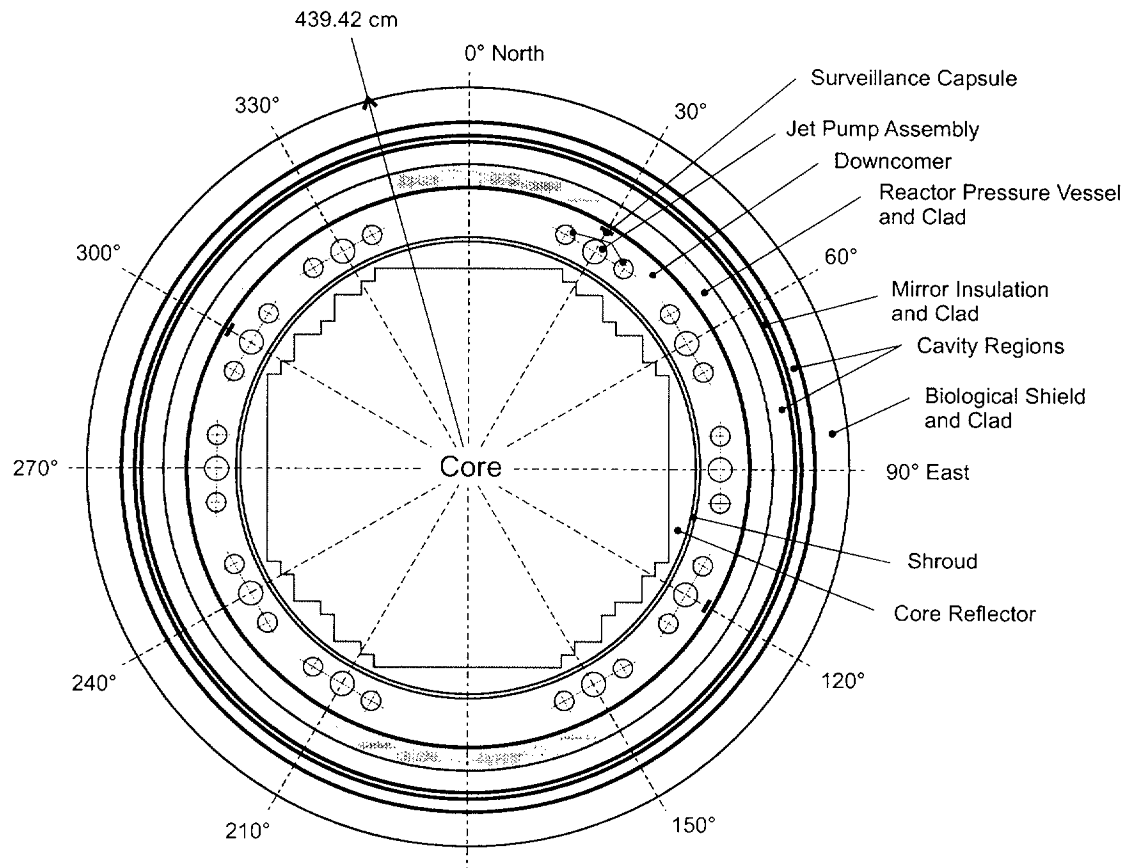

Figure 3-1
Planar View of the Susquehanna Unit 2 Reactor

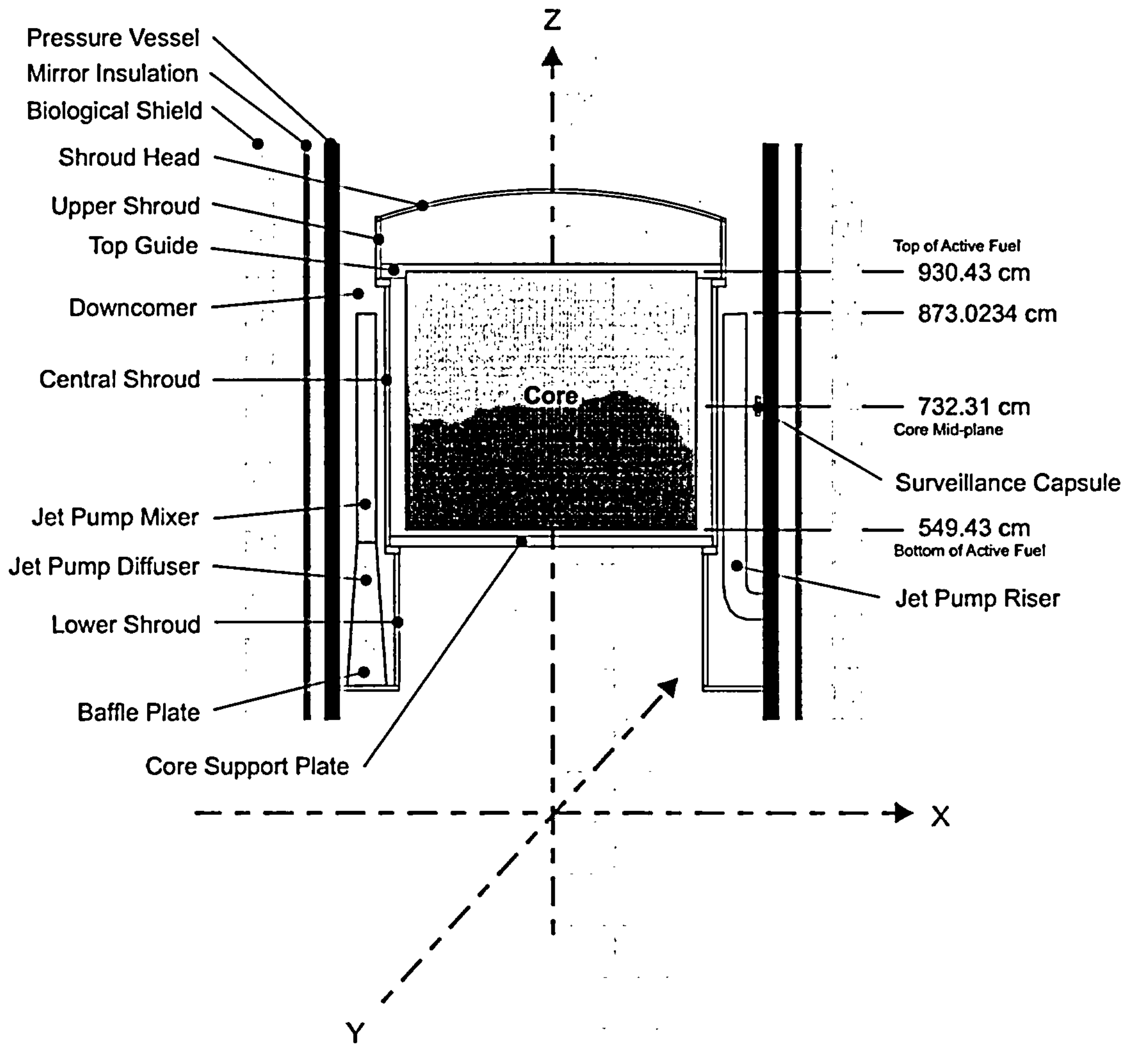

Figure 3-2
Elevation View of the Susquehanna Unit 2 Reactor

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.

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.

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.

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.

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Figure 3-3
Susquehanna Unit 2 Cycle 1 Relative Power History

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Figure 3-4
Susquehanna Unit 2 Cycle 2 Relative Power History

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Figure 3-5
Susquehanna Unit 2 Cycle 3 Relative Power History

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Figure 3-6
Susquehanna Unit 2 Cycle 4 Relative Power History

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Figure 3-7
Susquehanna Unit 2 Cycle 5 Relative Power History

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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Figure 4-1
Planar View of the Susquehanna Unit 2 RAMA Model

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Figure 4-2
Axial View of the Susquehanna Unit 2 RAMA Downcomer Model

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Figure 4-3
Planar View of the Susquehanna Unit 2 Jet Pump Assembly Design

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Figure 4-4
Planar View of the Susquehanna Unit 2 Surveillance Capsule

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4.3 RAMA Calculation Parameters

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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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Figure 4-5
2D Sensitivity to Distance Between Planar Parallel Rays

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Figure 4-6
3D Sensitivity to Distance Between Planar Parallel Rays

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Figure 4-7
3D Sensitivity to Distance Between Axial Parallel Rays

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Figure 4-8
Sensitivity to Flux Convergence Criterion

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Figure 4-9
Sensitivity to the Angular Quadrature Order

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**Figure 4-10
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
Calculated Neutron Fluence and Rated Power Flux for Susquehanna Unit 2 Capsule

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5.2 Comparison of Predicted Activation to Measurements

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Table 5-2
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 5-3
Capsule Analytic Uncertainty

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Table 5-4
Combined Capsule Uncertainty

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5.4 Best Estimate Neutron Fluence and Flux

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

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