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Subject: Project No. 704 – BWRVIP-121NP: BWR Vessel and Internals Project, RAMA Fluence Methodology Procedures Manual

Reference: Letter from Carl Terry (BWRVIP) to Document Control Desk (NRC), "Project 704 – BWRVIP-121: BWR Vessel and Internals Project, RAMA Fluence Methodology Procedures Manual," dated October 29, 2003.

Enclosed are two (2) copies of the report "BWRVIP-121NP: BWR Vessel and Internals Project, RAMA Fluence Methodology Procedures Manual," EPRI Technical Report 1008062NP, 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

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BWRVIP-121NP: BWR Vessel and Internals Project RAMA Fluence Methodology Procedures Manual

Technical Report

BWRVIP-121NP: BWR Vessel and Internals Project

RAMA Fluence Methodology Procedures Manual

1008062NP

Final Report, March 2005

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

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

PRODUCT DESCRIPTION

This report contains modeling guidelines, tips, and procedures to assist users of the RAMA Fluence Methodology software package in performing a fluence evaluation for a typical boiling water reactor (BWR).

Results & Findings

This manual describes the entire fluence evaluation process. It begins with determining the problem to be analyzed, describing computer resource requirements, and collecting the required data. Detailed information covers building geometry models for the reactor and components of interest, processing material data, evaluating flux and fluence results generated by the methodology, and performing an uncertainty analysis of the results.

Challenges & Objectives

This project's objectives were to (1) develop a state-of-the-art method for calculating fluence in a BWR, (2) adhere to the requirements of the Nuclear Regulatory Commission (NRC) Regulatory Guide 1.190, (3) validate the methodology against specific benchmark problems identified in the regulatory guide and perform plant-specific analyses, and (4) develop a system of software codes for utilities.

Applications, Values & Use

The RAMA Fluence Methodology is a system of software components used to determine neutron fluence in BWR components. The software includes a transport code, parts model builder code, state-point model builder code, fluence calculator, and nuclear data library. RAMA Version 1.0 is designed to 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 extend the methodology to other internal components that are beyond the active fuel height.

EPRI Perspective

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 verify neutron fluence in the core shroud in order to determine fracture toughness and crack growth rate for flaw evaluation calculations; and, 3) to determine neutron fluence in other internal components above and below the active core for structural integrity assessments or for evaluating 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.

The RAMA Fluence Methodology software package is a technical tool that requires an understanding of reactor physics, computer modeling techniques, nuclear plant operation, fluence determinations, and uncertainty and bias determinations. Users of the software should have sufficient knowledge and experience in these technical areas to apply the software package correctly and to interpret the results generated by the software package.

Approach

The project team developed a thorough description of individual software components that comprise the RAMA Fluence Methodology and the calculation and modeling processes. BWR designs and conventions used to describe a BWR design in the RAMA model were covered in general terms. Since accuracy of the fluence evaluation result is very dependent on the precision and detail of design inputs, the team provided detailed information on inputs used in the methodology. They also provided guidelines for determining and applying the bias and uncertainty parameters to the fluence evaluation in accordance with Regulatory Guide 1.190.

Keywords

Fluence
Embrittlement
Boiling water reactor
Vessel and internals
Reactor pressure vessel

ABSTRACT

This document contains modeling guidelines, tips, and procedures to assist the users of the RAMA Fluence Methodology software package in performing a fluence evaluation for a typical BWR reactor. It includes a description of the entire fluence evaluation process beginning with a determination of the problem to be analyzed, determination of the computer resource requirements, collecting the required data, 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 RAMA Fluence Methodology software package is used to determine neutron fluence in BWR Priority 1 components in compliance with the requirements and guidelines provided in U.S. Nuclear Regulatory Commission Regulatory Guide 1.190. The BWR Priority 1 components include surveillance capsules, the reactor pressure vessel within the active fuel height, and the core shroud within the active fuel height.

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 neutron flux from the transport calculation, isotopic activation and decay information, and 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 BUGLE-96 nuclear data library which was processed from ENDF/B-VI.

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 recognition is given to Robert Carter of EPRI for his guidance, comments, and overall support in completing this document.

The undersigned also wish to acknowledge Dr. Mark Williams of Louisiana State University and Mr. Steven Baker of TransWare for providing a thorough technical review of the document. Mrs. Virginia Jones of TransWare is recognized for her thorough quality review of the document.

Dean B. Jones, TransWare Enterprises Inc.

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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 nature of 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.

Section 3 provides general discussions about BWR designs and the conventions that are used to describe a BWR design in the RAMA model. RAMA's Cartesian coordinate system is illustrated in this section.

Section 4 presents a summary of the entire Methodology modeling process from problem description to data collection to fluence analysis. Additional details and guidelines for specific tasks of the modeling process are presented in the remaining sections of the manual.

Section 5 provides detailed information on the design inputs used in the Methodology. The inputs include mechanical design inputs and operating data inputs. The accuracy of the fluence evaluation result is very dependent upon the accuracy and detail of the design inputs, therefore, following the guidelines in this section is important to the outcome of the evaluation. Also presented in this section are the naming conventions used to describe the parts and materials in the RAMA fluence model.

Section 6 describes the RAMA geometry model building process. Included are details on the use of the Parts Model Builder code, a tool that automates the RAMA geometry model building process.

Section 7 describes the use of the RAFTER code to calculate activation and neutron fluence. The methodology used to compare the predicted activation levels to surveillance capsule dosimetry measured values is also presented in this section.

Section 8 presents guidelines for determining and applying the bias and uncertainty parameters to the fluence evaluation in accordance with the intent of Regulatory Guide 1.190.

Section 9 describes the process for calculating the best estimate neutron fluence for the reactor pressure vessel in accordance with the intent of Regulatory Guide 1.190.

Section 10 lists the references used in this document.

Other project documentation for the Methodology that provides additional information and details about particular aspects of the software includes a Theory Manual [2], a User's Manual [3], a Benchmark Manual [4], and a Plant-specific Capsule Fluence Evaluation Report [5].

2
BWR MODELING OVERVIEW

This section presents an overview of the Methodology process for calculating neutron fluence in BWRs. The computer codes that comprise the Methodology, the sequence in which the computer codes are used in a fluence analysis, the data collection process, and the documentation that describes the theoretical basis, application, and qualification of the Methodology are discussed.

2.1 RAMA Fluence Methodology

The Methodology is designed to provide a standard methodology for calculating accurate neutron fluence in BWR pressure vessels and reactor components. The Methodology is comprised of computer codes and documentation. The computer codes perform the computations for determining neutron fluence throughout the reactor system. The documentation describes the theoretical basis, application, and qualification of the software for calculating pressure vessel neutron fluence in accordance with the requirements set forth in the NRC Regulatory Guide 1.190 [1].

Figure 2-1 illustrates the calculation flow diagram for the Methodology.

The fluence evaluation process begins by specifying the scope of the analysis to be performed. For example, the components and regions of importance for pressure vessel neutron fluence determination lie in the axial elevations between the bottom of active fuel and top of active fuel as defined by the reactor core region and the radial regions that extend from the center of the core region to the outer surface of the pressure vessel wall.

Computer models for the fluence calculation are then constructed for the regions of interest using plant-specific design inputs for the reactor. Reactor design inputs are classified in two data categories: mechanical design inputs and reactor operating data inputs. Figure 2-1 shows the design inputs at the top of the calculation flow diagram indicating that the collection of this data is the first step in the RAMA Fluence Methodology calculational process. Collecting quality data for the fluence model is a key contributor to the accuracy of neutron fluence calculations.

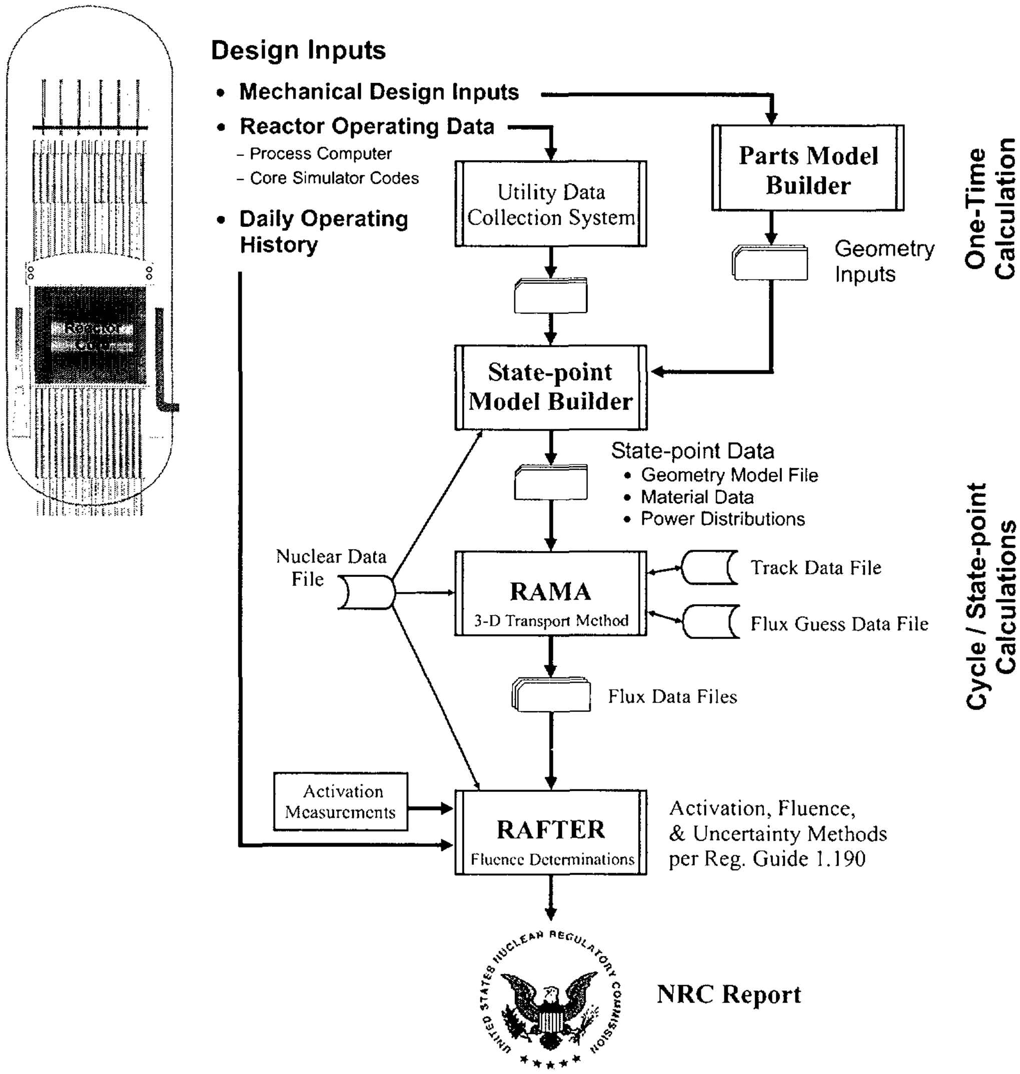

Figure 2-1
RAMA Fluence Methodology Calculation Flow Diagram

Mechanical design inputs describe the fundamental geometry of the reactor, which includes dimensional, positional, and material information about the reactor components. Accurate geometry information is needed in order to construct an accurate representation of the reactor with the Methodology. Mechanical design inputs may be available in two forms: nominal design information and as-built information. If available, as-built information should be used in the computer models.

Generally, mechanical design information is readily available for the reactor. This information is typically collected once, as the mechanical design of the reactor is not expected to change over the life of the plant. There are exceptions, however, which include fuel assembly designs that sometimes change with each operating cycle.

Reactor operating data inputs describe the operating history of the reactor. Accurate operating history information is needed in order to accurately predict the accumulated neutron fluence in reactor components and to estimate the effective full power years (EFPY) of reactor operation. Two types of reactor operating data inputs are needed: daily operating logs and reactor state-point data. Daily operating logs report the actual power levels of the reactor on a daily basis, including shutdown periods. Reactor state-point data describes the operating state of the reactor at a moment in time. State-point parameters include the reactor power level, core power distribution, core moderator (void) distribution, fuel isotopic distribution, control rod positions, and ex-core coolant density distribution.

Reactor state-point data changes continuously with the operation of the reactor. It is expected that this information will be collected frequently during an operating cycle. The frequency for collecting this data is dependent upon changes in the core operating state and, most importantly, to changes in the core power distribution.

Figure 2-1 shows three data paths for the reactor design inputs. The mechanical design inputs flow to the Parts Model Builder ("PMB") code. The reactor operating data (i.e., state point data) flows to the State-point Model Builder ("SMB") code and the daily operating history information flows to the RAFTER fluence calculation code.

The PMB code uses the mechanical design inputs to build a meshed geometry model of the reactor in a form that can be used in the RAMA transport code. The geometry meshing defines the spatial detail that is available to edit the neutron fluence for the reactor components. Typically, the PMB code is run once per operating cycle to account for any mechanical design changes that might have occurred during a refueling outage, such as the insertion of new fuel assembly designs. The outputs from the PMB code are typically provided as input to the SMB code.

The SMB code uses the reactor operating data (i.e., state-point data) to process material inputs for the geometry model built by the PMB code. The SMB code takes reactor operating data and the PMB geometry data as input. The SMB code may be run from one to several times per operating cycle, depending upon the number of operating state points that are used to characterize the operation of the reactor for the cycle.

Reactor state-point data is quite voluminous and is best handled as electronic data files. It is shown in Figure 2-1 that the reactor operating data flows into a "Utility Data Collection System" box prior to being used by the SMB code. The purpose of the data collection system is to extract data from reactor history files and process the data into a format suitable for use in the Methodology. Reactor history data can be processed electronically from process computer or core simulator code data files. The task of developing a linkage code is left to the utility, as each utility can have significantly different formats for storing and retrieving reactor operating data.

In a worst case, reactor operating data may exist only as printed material, or "hard copy" listings. Unfortunately, this is common among older plants. In this case, a more tedious effort will probably be required to transform the printed material information into electronic form. As stated earlier, the accuracy of the fluence calculations is partly dependent upon the quality of the operating data inputs. Consequently, the data collector is encouraged to obtain the highest quality data about the reactor operating history of the plant as is available and practical.

Figure 2-1 shows that the output from the SMB code is input to the RAMA transport code. The RAMA transport code performs the neutron flux calculations that are needed for the neutron fluence calculation.

The RAMA transport method is based on a three-dimensional, deterministic transport technique referred to as the Method of Characteristics [6]. The RAMA transport code includes special treatments for flexible geometry, geometry ray-tracing techniques, variable angular quadrature sets, anisotropic scattering, and vacuum and reflective boundary conditions. Several RAMA transport calculations are generally required in the performance of neutron fluence evaluations. One transport calculation is performed for each reactor operating state point processed by the SMB code. Additional transport calculations will also be required for the fluence uncertainty evaluation, if performed.

The RAMA transport code generates three data files: a track data file, a flux guess data file, and flux data file (or punch file). The track data file and the flux guess data file are automatically generated during a RAMA calculation. The data files may be saved and used in restart cases to accelerate the calculations. The use of these files in restarts is optional and is specified by the user. The primary output from a RAMA transport calculation is the flux data file. Flux data files contain the neutron flux data that is needed by the RAFTER code.

The RAFTER code calculates neutron fluence and isotopic activation for each reactor component of interest by combining the neutron flux data from the RAMA transport calculations with the reactor daily power history information. The RAFTER code also estimates the number of effective full power years of reactor operation from the operating history information provided to the code.

Figure 2-1 shows that the RAFTER code, RAMA transport code, and SMB code all access the RAMA Nuclear Data Library. The RAMA Nuclear Data Library contains all of the cross section data, activation response functions, and other nuclear constants that are needed to perform the neutron fluence calculations. The cross section data and response functions in the RAMA Nuclear Data Library are fully consistent with the 47 neutron / 20 gamma-ray energy group data described in the BUGLE-96 nuclear data library [7].

The final step in the RAMA fluence evaluation process is the calculation of reactor pressure vessel neutron fluence uncertainty. The uncertainty methodology is implemented in accordance with the requirements set forth in NRC Regulatory Guide 1.190 and includes determining analytic, comparative, and combined (or overall) uncertainties. Dozens of uncertainty parameters and biases are evaluated for the analytic uncertainty that include reactor geometry, reactor operating data, and calculational methods. The results of the uncertainty calculations are then used to determine the best estimate neutron flux and fluence values for the reactor pressure vessel or reactor components of interest.

2.2 The RAMA Fluence Methodology Package

The Methodology package is comprised of five computer software components and five code manuals, including this manual.

The software components are identified in Figure 2-1 and include four computational programs and one nuclear data library. These include: PMB, SMB, RAMA, RAFTER, and the RAMA nuclear data library. The code functions and capabilities were described in the previous subsection.

There are five code manuals that describe the software, theoretical basis, use, and qualifications of the Methodology, including this manual. Following are brief descriptions of the other four manuals that document the Methodology:

- *RAMA Fluence Methodology Theory Manual* [2] – describes the theoretical basis of the geometry modeling, flux calculation, fluence calculation, nuclear data library, and uncertainty methods.
- *RAMA Fluence Methodology User's Manual* [3] – describes the input requirements and outputs generated by the codes.
- *RAMA Fluence Methodology Benchmark Manual* [4] – describes the results of selected benchmark problems that are identified in NRC Regulatory Guide 1.190 for qualifying fluence methodologies.
- *RAMA Fluence Methodology Plant Application Manual* [5] – describes the results of a plant-specific surveillance capsule fluence evaluation for a BWR plant.

3
BWR MODEL DESCRIPTION

This section presents general information about BWR designs and how the design information is translated into a RAMA geometry model. The RAMA model incorporates several features to simplify the modeling process. Three key modeling features are:

- The RAMA geometry model employs a flexible modeling capability that allows the user to describe the reactor pressure vessel, reactor components, and coolant regions in their true geometrical form.
- The RAMA geometry model is based on a Cartesian coordinate system that allows the user to easily position reactor components and regions in three-dimensional space.
- The RAMA geometry model employs user-convenient features that include simple engineering parameters to describe the reactor geometry and descriptive mnemonic names to identify reactor components and regions in the model.

The following subsections provide further discussions on general BWR reactor designs and the use of the Methodology geometry models to describe the reactors.

3.1 BWR Reactor Designs

The BWR has several notable design features, but the most prominent feature that distinguishes it from other nuclear power plant designs is the coolant, or moderator, which is allowed to boil and create steam inside the pressure vessel. Figure 3-1 shows an axial cutaway view of a BWR assembly. This particular design is typical of BWR designs with jet pumps. The reactor pressure vessel (RPV) and major reactor components are identified. Also shown in the figure are the mirror insulation and biological shield structures that encase the pressure vessel.

There are several classes of BWR designs operating around the world. The principle differences in the designs are core loadings (i.e., the number of fuel assemblies in the reactor core region), rated core power densities, rated coolant flow rates, and shroud and pressure vessel diameters. Another difference between the reactor classes is that early designs (i.e., GE BWR/2's and earlier) do not have jet pumps while all later designs do. The Methodology is applicable to all designs.

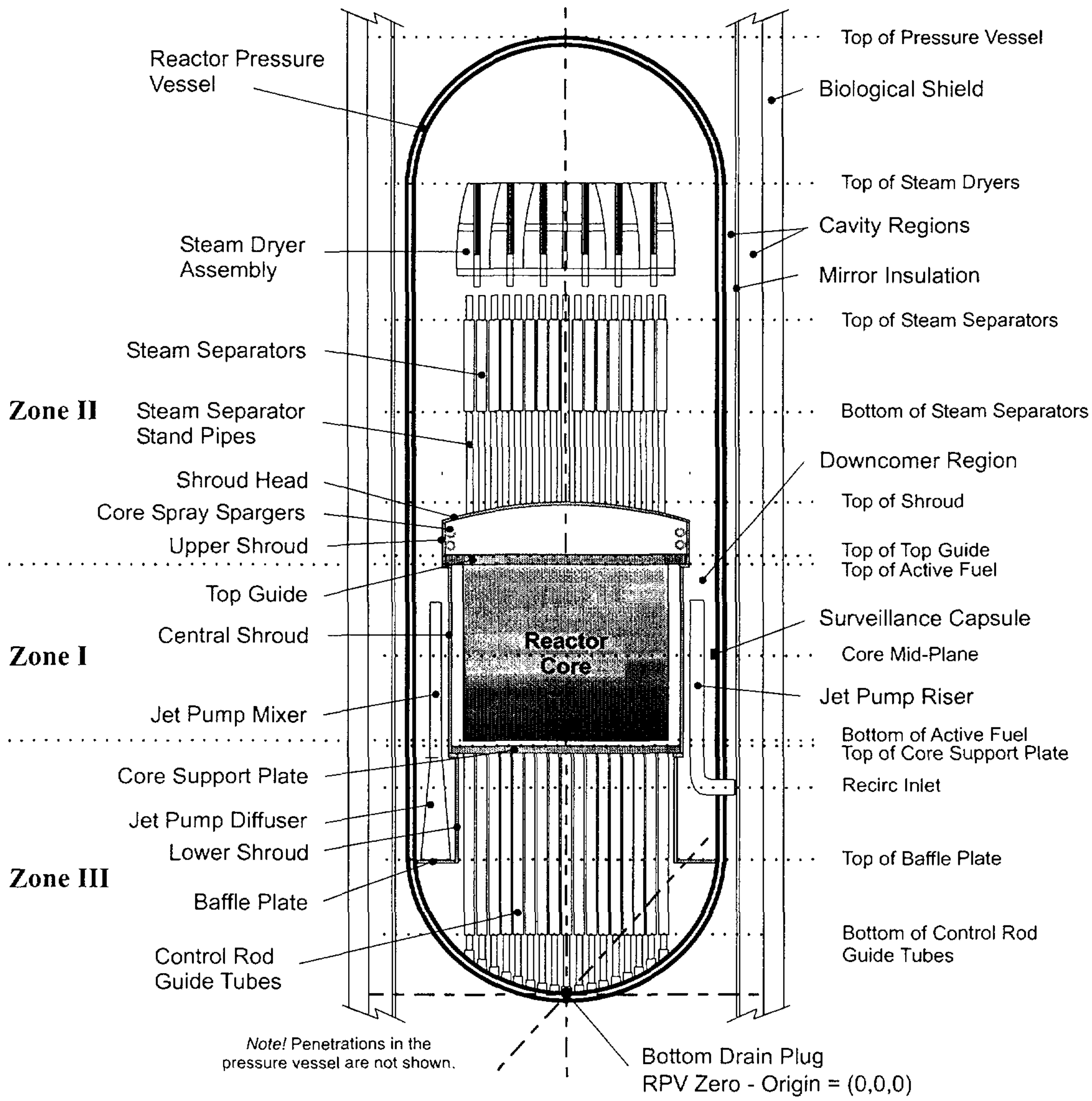

Figure 3-1
Cutaway View of a BWR Assembly

3.2 RAMA Model Coordinate System

The RAMA geometry modeling system uses the Cartesian coordinate system to describe the spatial information for the reactor model. Figures 3-2 and 3-3 illustrate the three-dimensional coordinate system for a BWR model.

For consistency with vendor reactor drawings, the origin of the model is specified at the inside surface of the pressure vessel wall at the bottom drain plug. Figure 3-2 shows that the z-axis traverses vertically through the center of the reactor core and defines the axial elevations for the reactor. The (x,y)-plane of the coordinate system is perpendicular to the z-axis and is illustrated in Figure 3-3.

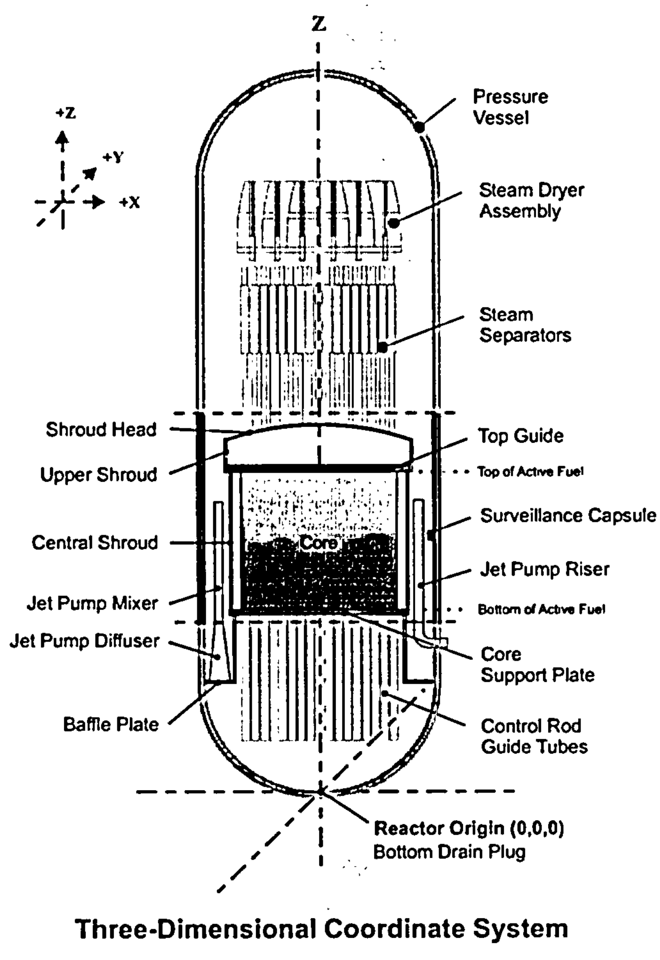

Figure 3-2
BWR Reactor Assembly 3-D Coordinate System

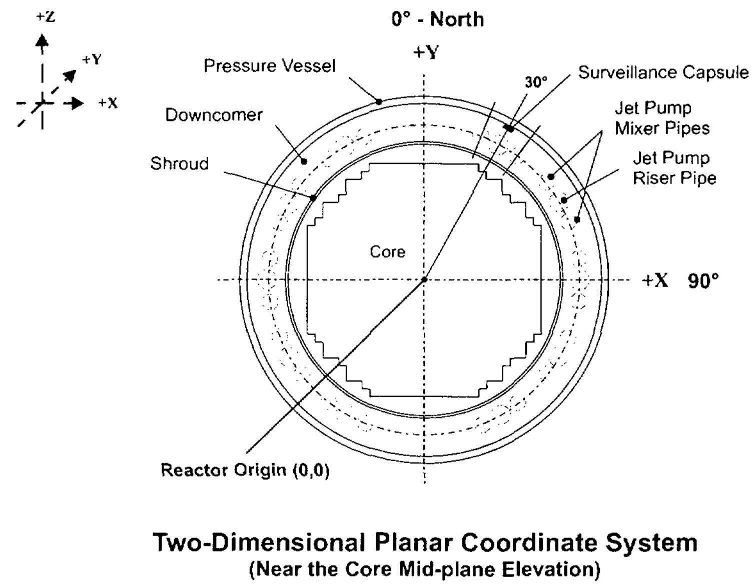

Figure 3-3
BWR Reactor Assembly 2-D Planar Coordinate System

Figure 3-3 shows that the azimuthal positioning of reactor components is relative to the north compass direction which is specified as 0 degrees north. The north compass direction also corresponds to the positive y-axis in the coordinate system. The azimuthal direction is incremented in the clockwise direction from 0 degrees to 360 degrees. The east compass direction is specified as 90 degrees clockwise from north compass direction and corresponds to the positive x-axis. The south compass direction is 180 degrees clockwise from north and corresponds to the negative y-axis. The west compass direction is 270 degrees clockwise from north and corresponds to the negative x-axis.

3.3 RAMA Geometry Modeling Concepts

The RAMA geometry modeling system is modular. This allows the user to build each part or region (e.g., core region, reflector region, shroud, etc.) of the reactor geometry model separately. The final model is then defined by combining the separately built parts to form the solution geometry. This modular geometry also allows the user to easily replace a part of one meshing description with another part having a different meshing description. Rebuilding the entire geometry model every time a part description changes, such as when fuel designs change between operating cycles, is avoided with this feature.

Figure 3-4 illustrates the modular geometry capabilities of the RAMA modeling system for a two-dimensional planar model. The illustration shows that the planar model is comprised of nine parts: the reactor core region, a core reflector region, a core shroud, a downcomer region with a jet pump assembly and surveillance capsule, a reactor pressure vessel, inner and outer cavity regions, the mirror insulation, and the biological shield. The models for each of these parts are built separately, but are then combined, as illustrated by the direction arrows, to form the RAMA solution geometry.

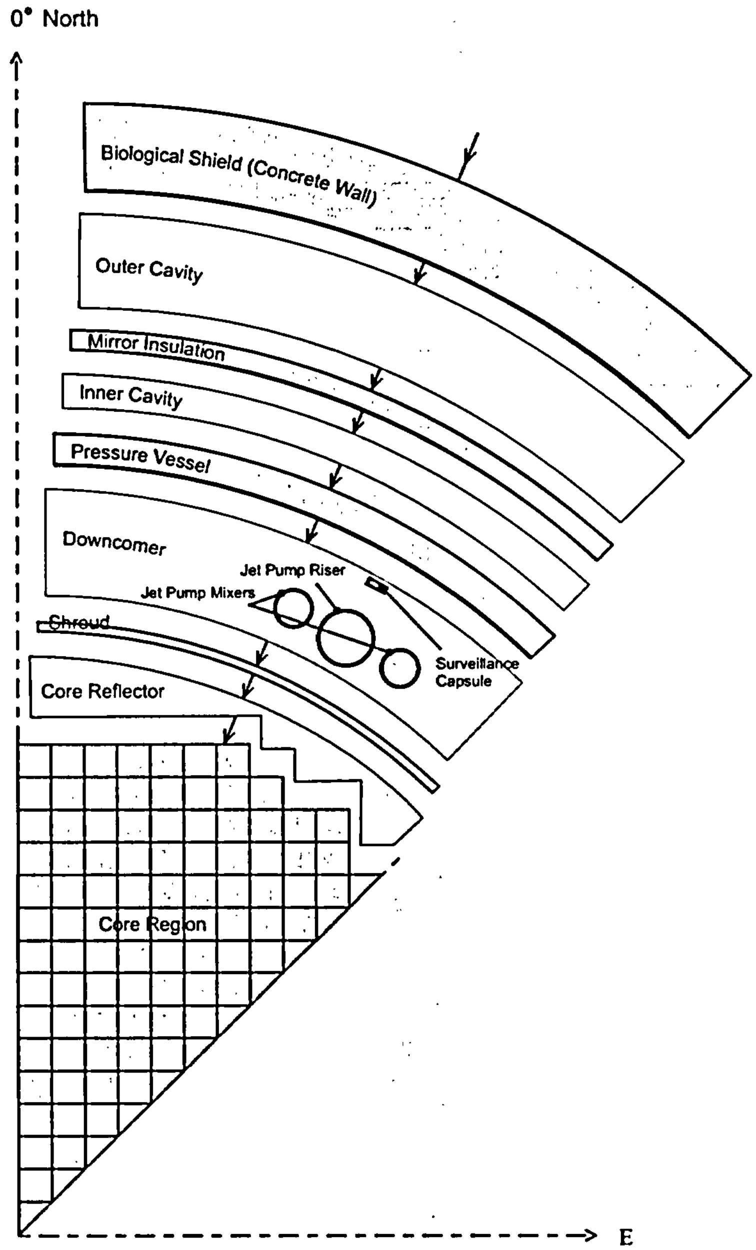

Figure 3-4
RAMA BWR Modular Planar Geometry Model

A planar geometry model, such as the one illustrated in Figure 3-4, typically has a finite height. That is, all parts in the plane start at the same axial elevation and end at the same axial elevation. A modeling practice is for each plane to have a height of 15.24 cm (6 inches). This corresponds to the axial height of a core node as defined in most core simulator codes. A single plane model of finite axial height is illustrated in Figure 3-5.

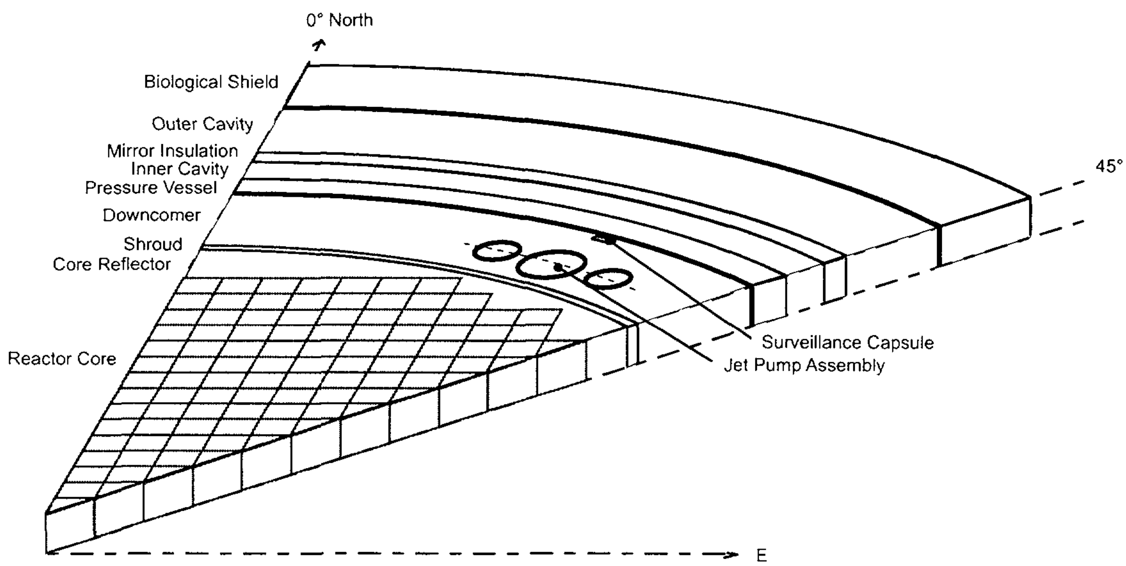

Figure 3-5
RAMA BWR Planar Geometry Model of Finite Height

A three-dimensional model of the reactor is then built by stacking a series of planar models in the axial dimension to form the solution geometry.

Modular geometry models have several benefits. Perhaps of greatest importance to the user is that smaller parts (i.e., simple parts of lesser complexity) are easier to visualize, which can facilitate the model building process. By being modular, it is relatively simple to replace one part of some detail with another part of more or less detail. Modular modeling techniques are easily applied to the practice of bootstrap modeling which is described in the next subsection.

3.4 RAMA Bootstrap Models

The RAMA geometry modeling system is capable of describing the entire reactor in a single model file. While this sounds intriguing, it is generally not practical to solve the entire reactor problem in a single transport calculation, nor is it needed in many cases. As powerful as computer systems are becoming, there remain physical limitations in computer memory, hard disk capacity, and computation time that suggest alternate modeling approaches.

To compensate for physical limitations in a computer system, the Methodology supports a modeling technique referred to as "bootstrap models". Bootstrap models allow a large problem to be broken into smaller parts that are easier to manage and quicker to solve. When the bootstrap calculations are completed, the results of the various bootstrap calculations can be combined to provide a complete solution for the entire reactor problem. Bootstrap models also allow the user to build special models that focus on specific regions of interest without the requirement to model the entire reactor system.

Figure 3-1 illustrates a BWR reactor system with three axial zones. Zone I contains the reactor core region, which is the source of the neutrons for neutron fluence calculations. This zone, or a part of this zone, must be included in all bootstrap models. Zones II and III define the components and regions that lie axially above and below the active fuel zone, respectively. The approximate scope of the axial elevations in a RAMA three-dimensional fluence model is illustrated in Figure 3-2 by the dashed horizontal lines that are shown just below the core support plate and just above the core shroud head. Fluence evaluations beyond these elevations are probably of no consequence as the neutron flux in these regions should not produce neutron fluence above the material embrittlement threshold.

3.5 RAMA Fluence Model Conventions

Three-dimensional geometry models can be quite complicated to build and understand due to the extreme number of mesh regions and materials that comprise the model. The complexity, or the perception of complexity, can be reduced to some degree by a good modeling approach and good modeling practices.

The RAMA modeling system employs an approach of identifying major regions and materials with alphanumeric names. For example, the core shroud might be given the name "CORSHR" and reactor pressure vessel wall "RPVwall". The material stainless steel 304 might be given the material identifier "SS304". The approach of using names to identify regions and materials can greatly simplify the effort of describing the model geometry as well as locating regions for editing fluence.

The RAMA Fluence Methodology package includes two model builder codes that assist the user with building a fluence model. The first code is the PMB code that generates the geometry of the reactor model. The second code is the SMB code that describes the materials for the geometry model. The model builder codes work together to build the geometry and material inputs for the RAMA transport calculation. In order for the codes to work together, naming conventions for the geometry and materials must be developed that are recognizable between the two codes.

Additional guidelines for developing geometry and material names for the RAMA model are provided in Section 5, *BWR Design Inputs*, and Section 6, *Building BWR Geometry Models*.

4

GUIDELINES FOR BWR FLUENCE EVALUATIONS

This section presents general guidelines for using the RAMA Fluence Methodology to perform BWR fluence evaluations. This section builds upon the overview of the fluence evaluation process presented in Section 2, *BWR Modeling Overview*, and the BWR design and RAMA model descriptions presented in Section 3, *BWR Model Description*. The scope of the discussions in this section includes the technical steps in performing a fluence evaluation, from describing the reactor problem to evaluating the results of the fluence calculations.

Several of the discussions presented in this section refer to RAMA input variables and data formats. These items are not defined in this manual, but are described in detail in the RAMA Fluence Methodology User's Manual. The reader is referred to the User's Manual for additional information on these items.

4.1 Problem Description

The fluence evaluation process begins by determining the desired outputs from the analysis. This involves identifying the specific regions of interest for the analysis and specifying the scope of the reactor design to be included in the fluence analysis model. Section 3 of this manual will assist in this effort as it describes typical BWR designs and how the design information is translated into a RAMA fluence model.

4.2 Reactor Design Inputs

Having defined the problem and the scope of the analysis in accordance with Section 4.1, *Problem Description*, the data needed to build a RAMA fluence model must be collected. Two types of data are needed for the RAMA model: mechanical design inputs and reactor operating data inputs.

Mechanical design inputs describe the dimensional parameters of the reactor that are needed to build the reactor geometry model. Mechanical design inputs are typically extracted from design drawings that are provided by the reactor vendor or component manufacturer.

Operating data inputs describe the power history and operating conditions of the reactor over the lifetime of the plant. Two categories of operating data are needed: daily power factors and reactor state-point data. The daily power factors provide the power level of the reactor on a daily basis, including shutdown periods. The reactor state-point data provides detailed power distributions, moderator density distributions, and fuel isotopic distributions throughout the reactor core region at specified moments in the reactor operating history. The daily power factors and state-point data are combined to produce an accurate representation of the reactor operation over the life of the plant.

The accuracy of the fluence results can be significantly influenced by the quality and detail of the mechanical design information and reactor operating data. The mechanical design inputs can be collected relatively easily. Collecting the operating data is more involved and, because of the volume of information and data to be collected, is best handled by developing computer programs to extract the data from electronic data files that are available from plant process computers or core simulator codes.

Section 5, *BWR Design Inputs*, provides detailed discussions about the design inputs that are needed to construct an accurate RAMA fluence model. The discussions include descriptions of the dimensional parameters needed for the RAMA geometry model, the evaluation process for selecting operating history data, the types of operating data, and the format in which the operating data must be prepared for input to the RAMA fluence model.

4.3 Computer Requirements

Having defined the problem and identified the amount of data to be processed in the analysis, a determination should be made for the allocation of computing resources to solve the problem. Items to consider include computational time, memory, and disk storage needs. This task should also include setting up appropriate directories on the computer hard disk to provide a logical separation between input and output data files. Proper directory and file management will ease the processing and quality assurance activities associated with handling the large amount of data that is used in fluence calculations.

The RAMA transport software performs a full three-dimensional transport calculation of the reactor model and is computationally intensive. Primary considerations when selecting a computer system to perform the calculations are computational speed, random access memory (RAM), and hard disk performance.

Many computer configurations are possible that can handle the computational demands of the RAMA fluence calculation. Example configurations of engineering workstations are included in the installation instructions distributed with the RAMA Fluence Methodology Software.

Having configured the computer hardware and operating system, a file directory system should be set up on the computer hard disk for the analysis. Because of the volume of data to be handled in the analysis, it is recommended that separate directories be created for the geometry data, material data, reactor operating data, sensitivity studies, transport calculations, and fluence analysis calculations. If multiple reactor units are to be modeled, separate base directories for each reactor unit should also be created. Table 4-1 shows an example directory structure.

Table 4-1
Example Computer File Directory Structure

Directory Structure	Type of Data
Unit Identifier	
├─ Geometry Files (PMB)	Geometry Data
├─ Material Files (Structural)	Material Data
│ ├─ Standard Materials	
├─ 2D Sensitivity Studies	Sensitivity Studies
│ ├─ Meshing	
│ └─ Integration Parameters	
├─ 3D Sensitivity Studies	Sensitivity Studies
│ ├─ Meshing	
│ └─ Integration Parameters	
├─ State-point Calculations	Transport Calculations
│ ├─ Cycle-Exposure Data	
│ │ ├─ State-point Data	
│ │ ├─ Transport Input Files	
│ │ └─ Transport Output Files	
│ ...	
├─ Transport Output Data Files	Transport Results
├─ Daily Operating History	Reactor Operating Data
├─ Dosimetry Activity Calculations	Analysis Calculations
├─ RPV Fluence Calculations	Analysis Calculations
└─ RPV Uncertainty Calculations	Analysis Calculations

The amount of disk space needed for a reactor fluence evaluation can vary significantly by plant. The amount of disk space will be determined primarily by the size of the reactor model and the number of state points selected for the calculation. There are three primary consumers of disk space: reactor state-point data files, track data files, and flux guess data files. The track data files and flux guess data files are generated during the RAMA transport calculations and are generally maintained on the computer system during a cycle analysis. Table 4-2 provides an example of the computer hard disk storage that might be used in a RAMA fluence calculation.

Table 4-2
Example Computer Hard Disk Storage Requirements

Data Item	Approximate Size	
Geometry Data Files	10	MB
Material Data Files	1	MB
2D Sensitivity Studies[1]	1-3	GB
3D Sensitivity Studies[1]	1-2	GB
Transport Calculations[2]	200	MB per state point
Track Data Files[3]	1	GB per cycle
Flux Guess Data Files[4]	2	GB per state point
Reactor Operating History	1	MB
Analysis Files	10	MB
Temporary Disk Storage	4	GB per execution

1. The computer files associated with the sensitivity studies can be removed at the conclusion of the studies.
2. The computer files, except for the RAMA output punch files, for the transport calculations can be removed at the end of the transport calculation. The RAMA output punch files must remain on the system for use in the analysis calculations. The size of an output punch file is about 30 MB per state point.
3. Track data files are generated by the RAMA transport code and should be saved for restart cases. Track files are geometry dependent and should be generated once per reactor operating cycle.
4. Flux guess data files are generated by the RAMA transport code and should be saved for restart cases. Flux guess files are also geometry dependent and should be generated once per reactor operating cycle.

4.4 Building BWR Geometry Models

Having defined the problem to be solved and collected the reactor data for the problem, the basic geometry model of the reactor problem can be built. The geometry model is built with the PMB code using the mechanical design inputs described in Section 5, *BWR Design Inputs*.

Section 6, *Building BWR Geometry Models*, contains descriptions of geometry models for several reactor components and regions that comprise a BWR geometry model. The reader is referred to this section for detailed information on the geometry model building process.

The discussions in Section 6 include illustrations of the geometries that are built for the various components and regions of the reactor model. Note that the illustrations include example geometry meshing. The meshing is shown only for discussion purposes. The actual meshing for the reactor model will be determined in accordance with the guidelines presented in Section 4.6, *BWR Planar Model Sensitivities*, and Section 4.7, *BWR Production Model Sensitivities*, once the complete model is ready for testing.

4.5 Material Inputs

Having built a basic RAMA geometry model, the material inputs for the model must be prepared. Material inputs are prepared and assigned to the material regions of the geometry model with the SMB code. The SMB code actually has two functions. In addition to preparing material inputs for the RAMA transport calculations, it also has the capability of allowing the user to select specific regions and parts to be included in the output model file. This capability is used to generate planar (two-dimensional) models and three-dimensional models of the reactor. This ensures that all applications of the RAMA code system, whether for planar studies or three-dimensional calculations, receive the same processing.

There are three fundamental types of materials that are processed by SMB for the reactor model: nuclear fuel material, structural material, and water.

Fuel material data is provided in the state-point data files described in Section 5, *BWR Design Inputs*. The state-point data files contain fuel bundle, channel, and water gap material information for all fuel locations in the core model. In addition, the state-point data files include relative power factors for the fuel regions. The SMB code has the capability of associating the fuel material data contained in the state-point data files with the core geometry description generated by the PMB code, providing that the naming conventions recommended in Section 5 are used. Thus, the input material information from the state-point data files may be passed through to the transport calculation unchanged, or homogenized in accordance with the geometry configuration output by the PMB code.

Structural material data is commonly provided in a compositions data file that is input to the SMB code. Structural materials include stainless steels, carbon steels, and steel alloys. Material properties for the different steels can be obtained from either mechanical design build sheets or standards handbooks.

Water data for the various water regions of the model can also be provided in the compositions data file or calculated by the SMB code using built-in steam tables. If the steam tables are used, the user must input the water temperature and/or pressure and a void fraction.

The SMB code also has the capability of calculating homogenized materials for the reactor regions. Using volume fractions, the SMB code can mix various combinations of steel and water compositions to fit any region of the reactor model.

4.6 BWR Planar Model Sensitivities

Having built the basic reactor model, a series of planar (i.e., two-dimensional) transport calculations should be performed to evaluate the sensitivity of the model to variations in geometry meshing and flux integration parameters. Planar sensitivity studies are performed before the three-dimensional model sensitivity studies discussed in Section 4.7, *Production Model Sensitivities*, for the following reasons.

- Planar models reduce the number of variables that must be considered at one time, thus simplifying the evaluation process.

- The planar source has a more significant contribution to the ex-core fluence than the axial source, thus, a good representation of the planar geometry can have a significant effect on the accuracy of the three-dimensional neutron fluence calculation.
- Planar models are quick to solve and easy to evaluate in sensitivity studies, which facilitates the model building process.

There are two principle studies that should be performed for the planar model. One is to determine an acceptable geometry meshing for the planar model. The other is to determine acceptable values for the numerical integration parameters that control the RAMA transport calculation. The activities for each of these studies are described in the following subsections.

The results of the planar model sensitivity studies described in this section should be applicable for all pressure vessel fluence calculations providing the fundamental geometry and material conditions of the original model do not vary significantly. It might be necessary to verify the planar model if a significant change is made in the reactor geometry, such as changes in fuel assembly designs or changes in structural materials that include adding or removing surveillance capsule materials. It might also be necessary to verify the planar model if a significant change in the reactor operating conditions is made such as a power up-rate that might affect, for example, the water density in the downcomer region.

4.6.1 Planar Meshing Sensitivity Cases

The objective for the planar meshing sensitivity studies is to determine a planar model that provides acceptable results with the fewest number of mesh regions. Acceptable results are achieved when a coarse-mesh model produces results that are comparable to a fine-mesh model throughout the reactor geometry.

For pressure vessel fluence determination, the result of interest is the >1.0 MeV neutron flux. For other fluence determinations, such as ^{237}Np dosimetry and thermal fluence, results should also be tested for other energy group ranges, i.e., >0.1 MeV and thermal groups, respectively. For discussion purposes in this manual, only the >1.0 MeV flux is considered.

In practice, the >1.0 MeV neutron flux edited from a coarse-mesh case should be compared to a corresponding result from a reference case. Because the meshing will be different between the cases, it is not practical that a one-to-one comparison can be made in all cases. It also is not practical or necessary that all mesh regions be compared. For these sensitivity studies, it is sufficient that the >1.0 MeV neutron flux be compared for selected locations in the problem.

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4.6.2 Planar Numerical Integration Parameters Studies

The second step in determining an acceptable planar model is to evaluate the sensitivity of the flux integration parameters used in the RAMA transport calculation. There are five parameters from input block CPA that should be tested for the planar model. These parameters are:

dytrk	which specifies the distance in centimeters between planar parallel lines in the ray-tracing calculation,
optrk	which specifies the distance in mean free paths that a reflected ray will be tracked in the ray-tracing calculation,
nvtrk	which specifies the angular quadrature for the transport calculation,
npord	which specifies the Legendre order of scattering for the transport calculation, and
epsin	which specifies the convergence criterion for the flux calculation.

These parameters are described in more detail in input block CPA of the RAMA transport code section of the RAMA Fluence Methodology User's Manual.

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4.7 BWR Production Model Sensitivities

The last step in the RAMA model building process involves performing sensitivity studies to determine the final geometry and flux integration parameters for the RAMA three-dimensional transport calculation. This step involves constructing a three-dimensional model of the reactor using the results of the planar model sensitivity studies in Section 4.6, *BWR Planar Model Sensitivities*.

The three-dimensional model is built to provide an accurate axial representation of the reactor geometry from the bottom of active fuel to the top of active fuel. A basic three-dimensional geometry model of the reactor was described in Section 4.4, *Building BWR Geometry Models*. Meshing configurations for a planar section of the core region was determined in Section 4.6. The meshing for the planar model should be used to build a meshed three-dimensional geometry model of the reactor.

Note that the planar meshing developed in Section 4.6 is directly applicable for all axial elevations with the same geometry configuration. The basic meshing of this planar model should also be applicable at other elevations where the geometry varies, such as above and below the surveillance capsule and above the jet pump assembly. However, the user should confirm this assumption through additional planar meshing sensitivities of the various planar geometry configurations used in the three-dimensional model.

For consistency, the same operating state-point data used in Section 4.6 should be used in these sensitivity studies.

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4.7.1 Production Model Axial Sensitivity Studies

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4.7.2 Three-dimensional Numerical Integration Parameter Studies

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4.8 RAMA Transport Calculations

The RAMA production model that was developed in Section 4.7 should now be ready to perform transport calculations for the fluence analysis.

The RAMA transport code will be executed once for each state point used in the analysis. The primary output of the transport calculation is the output punch file, which contains fluxes for every region of the geometry in the full energy group structure of the nuclear data library. The output punch file then becomes the input file for the RAFTER fluence calculation.

Perhaps the most difficult part of this task is managing the data files that are generated during the transport calculations. There will be a set of data files generated for each cycle and for each state point data set in the cycle. A description of the output files generated by the RAMA transport calculation is provided in the following paragraphs.

The RAMA transport code produces two to five output files during execution. Two files are always output: the output list file and the output summary file. In accordance with user options given in the filename inputs, an output punch file, track data file, and flux guess data file can also be generated. For fluence calculations, the output punch file must be generated.

The output list file and output summary file contain listings of the results calculated by the RAMA transport code. These files are automatically produced by RAMA during execution. The content of the list file is controlled by the user by specifying edit and print options. The output summary file contains only status information for the calculations performed during execution.

The output punch data file contains the results of a RAMA transport calculation in a format that can easily be used as input to downstream codes, such as the RAFTER fluence calculation code. Generating the punch file is optional and is controlled by the "p=" parameter in the filename inputs. The content of the punch data file is controlled by the user, however, by default, spatially-averaged neutron fluxes are always edited in the full energy group structure for each region in the problem geometry.

The output track data file is a binary file that contains the results of the ray-tracing calculation. Saving the track file is optional, although it may be used in subsequent cases to reduce computational times.

The output flux guess data file is a binary file that contains the results of the flux calculation. Saving the flux file is optional, although it may also be used in subsequent cases to reduce computational times.

The RAMA transport code treats each problem as an independent case. The concept of restart cases is supported to the extent that existing track data files and flux guess data files generated in a previous case may be used to accelerate the calculation time of the current case.

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4.9 RAMA Dosimetry Calculations

An important element in the application of the Methodology is the comparison of predicted activation levels in surveillance capsule dosimetry to measured values. Comparison of activation predictions to measurements is essential for qualifying the applicability of the methodology for use in estimating RPV fluence. The comparisons also form the basis for estimating the bias and uncertainty associated with RPV fluence predictions.

The primary source of activation measurements in BWRs is activation samples retrieved from in-vessel surveillance capsules. The activated samples may be in the form of flux wires (typically copper and iron wires) and dosimetry samples consisting of flux wires and/or foil samples.

The RAFTER module of the Methodology is used to compute the dosimetry activities for the various activation reactions that are included in the measurement data. RAFTER combines the flux results from the RAMA transport calculations with detailed operating history data to predict product nuclides in the measured reactions. The predicted activities are compared to the activity measurements to validate the reliability of the Methodology for use in determining RPV fluence and to support the RPV fluence uncertainty analysis.

Section 7, *RAMA Dosimetry Calculations*, describes the procedure for predicting the activation of in-vessel dosimetry and comparing the results to measured data.

4.10 RAMA Uncertainty Calculations

Calculation of the neutron flux distribution is sensitive to material and geometric representation of the core and reactor internals, the neutron source, the nuclear cross-section data, and the numerical scheme used in the calculation. An uncertainty analysis is incorporated into the Methodology to estimate the level of uncertainty for the vessel fluence calculation. This section describes the parameters and values that must be determined and input into the Methodology to account for the uncertainty in the fluence calculation. The uncertainty evaluation is composed of three steps: (1) analytic uncertainty analysis, (2) comparison uncertainty analysis, and (3) combined (i.e., overall) uncertainty analysis.

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Section 8, *Analytic Uncertainty Calculations*, describes the procedure for determining the various uncertainty components, including an assessment of the key analytical parameters and the corresponding impact (or sensitivity) that the parameters have on the RPV fluence.

4.11 RAMA Neutron Fluence Calculations

Generally, the evaluation of neutron fluence for RPV damage analysis is restricted to a reduced energy range since low energy neutrons are substantially less likely to inflict irradiation damage than are high-energy neutrons. Material damage assessments are usually restricted to integral fast neutron fluences with energy >1.0 MeV. The following steps form the basis for determining the RPV fast fluence to be reported in accordance with Regulatory Guide 1.190 [1].

The accumulated RPV neutron fluence with energy >1.0 MeV at any location in the RPV is determined by using flux spectrum data generated from RAMA transport calculations described in Section 4.8, *RAMA Transport Calculations*. The combination of these state-point files along with the plant power history is used in the RAFTER module to compute the accumulated RPV fluence. Section 9.1 describes this process in more detail.

The results of the RPV fluence uncertainty evaluation are then used to determine the best estimate neutron fluence (both accumulated and predicted to end-of-life). Section 9.2 describes the procedure for computing the accumulated best estimate fluence while Section 9.3 discusses the procedure for extending the fluence prediction to the RPV end-of-life.

5 BWR DESIGN INPUTS

This section describes specific mechanical design and reactor operating data inputs that are needed for the RAMA fluence model.

5.1 Mechanical Design Inputs

Mechanical design inputs describe the physical attributes of the reactor pressure vessel, reactor components, and coolant regions in the reactor. These attributes include dimensional data, material data, and the placement of the components and regions in the reactor pressure vessel. The design inputs needed to describe the fluence model are determined by the components or regions of interest in the analysis.

For determining fluence in the components within the active core height, the following major components and regions of the pressure vessel should be included in the RAMA fluence model:

- The reactor core region, including accurate descriptions of the fuel assembly designs loaded in the outer four rows of the core;
- The central core shroud, including upper and lower flange pieces;
- The jet pump assemblies, including the jet pump riser tubes, mixer tubes, diffuser tubes, and rams head;
- The reactor pressure vessel, including the clad;
- The dosimetry / surveillance capsules;
- The outer pressure vessel boundary regions that include the mirror insulation, cavity region, and biological shield (concrete wall);
- The top boundary region that includes the top guide, fuel plenum, upper shroud, upper downcomer, and upper pressure vessel; and
- The bottom boundary region that includes the fuel support pieces, core support plate, lower shroud, lower downcomer, and pressure vessel.

Examples of the dimensional information that is used to construct a RAMA BWR fluence model are presented in Tables 5-1 through 5-7. Note that the tables may list design parameters that may not apply to a specific plant or to a specific analyses. The extraneous parameters may be ignored.

Table 5-8 lists typical titles of mechanical design drawings that might contain the design parameters listed in Table 5-1 through 5-7. These titles are provided to assist the data collector with locating the mechanical design drawings of interest.

For nominal designs, it is important to obtain the tolerances for the various dimensional attributes of the components. Tolerance values are used in the uncertainty calculations. Note that any plant-specific tolerances or design information that is omitted will require the use of assumptions in the uncertainty calculation. The assumptions could affect the quality, or uncertainty, of the fluence calculations. A list of the uncertainty parameters that are used in the RAMA fluence calculation is provided in Section 8.1 of this manual.

It is desirable that the as-built values be provided when available because this can improve the overall quality of the calculational model by lessening the effect of assumed variances on the uncertainty calculations.

Table 5-1
Typical Reactor Pressure Vessel, Mirror Insulation, and Biological Shield Design Parameters

Reactor Component	Component Parameter
Reactor Pressure Vessel (RPV)	Clad inner radius
	Clad thickness
	RPV wall inner radius (base metal)
	RPV wall thickness
Mirror Insulation	Inner clad inner radius (if present)
	Inner clad thickness (if present)
	Mirror insulation inner radius
	Mirror insulation thickness
	Outer clad inner radius (if present)
	Outer clad thickness (if present)
Biological Shield (Concrete)	Inner clad inner radius (if present)
	Inner clad thickness (if present)
	Biological shield inner radius
	Biological shield thickness
	Outer clad inner radius (if present)
	Outer clad thickness (if present)

Table 5-2
Typical Shroud Design Parameters

Reactor Component	Component Parameter
Central Shroud	Shroud wall inner radius
	Shroud wall thickness
	Upper flange top elevation
	Upper flange height
	Upper flange inner radius
	Upper flange wall thickness
	Lower flange top elevation
	Lower flange height
	Lower flange inner radius
	Lower flange wall thickness
Upper Shroud	Shroud wall inner radius
	Shroud wall thickness
Shroud Head	Head flange top elevation
	Head flange height
	Head flange inner radius
	Head flange wall thickness
	Shroud head top elevation
	Shroud head outer radius (or spherical radius)
	Shroud head wall thickness
	Shroud head rim bottom elevation
	Shroud head rim height
	Shroud head rim inner radius
	Shroud head rim wall thickness
	Shroud head diagram showing stand pipe penetrations
Steam Separator Stand Pipes	Top elevation of steam separator stand pipes
	Number of stand pipes
	Stand pipe outer radius
	Stand pipe wall thickness

Table 5-3
Typical Jet Pump Design Parameters

Reactor Component	Component Parameter
Jet Pump Assemblies	Number of jet pump assemblies
	Azimuth angles for positioning the jet pump assemblies. Angles are specified at the center of the jet pump assemblies (or riser pipes) beginning at reactor North 0 degrees.
	Delta azimuth angle for positioning the mixer / diffuser tubes relative to the center of the jet pump assembly (or riser tube)
	Radial distance from the center of the reactor to the center of the riser pipes at the core mid-plane elevation
	Radial distance from the center of the reactor to the center of the mixer pipes at the core mid-plane elevation
Jet Pump Riser Pipe	Elevation at the center of the riser inlet
	Riser pipe height from the riser inlet elevation to the bottom of the rams head
	Riser / Rams head connector part height
	Riser pipe inner radius
	Riser pipe wall thickness
	Riser inlet inner radius
	Riser inlet wall thickness
Jet Pump Rams Head	Elevation at the top of the rams head
	Rams head height (u-pipe outer radius)
	Rams head inlet inner radius
	Rams head inlet wall thickness
	Rams head outlet inner radius
	Rams head outlet wall thickness
Jet Pump Nozzle	Nozzle height (including coupler)
	Coupler height
	Coupler inner radius
	Coupler wall thickness
	Nozzle inner radius at the upper elevation
	Nozzle inner radius at the lower elevation
	Nozzle wall thickness

Table 5-3 (Continued)
Typical Jet Pump Design Parameters

Reactor Component	Component Parameter
Jet Pump Mixer Pipes	Elevation at the top of the mixer throat part
	Height of the throat part
	Throat inner radius
	Throat wall thickness
	Height of the mixer pipes
	Mixer inner radius
	Mixer wall thickness
	Elevation at the mixer - diffuser coupling
Jet Pump Diffuser Pipes	Diffuser height from mixer pipe to tail pipe
	Diffuser inner radius at top elevation
	Diffuser inner radius at bottom elevation
	Diffuser wall thickness
Jet Pump Tail Pipes	Tail pipe top elevation
	Tail pipe height (top of baffle plate)
	Tail pipe inner radius
	Tail pipe wall thickness

Table 5-4
Typical Dosimetry Design Parameters

Reactor Component	Component Parameter
Reactor Dosimetry	Number of surveillance capsule assemblies
	Azimuth angles for the capsule assemblies. Angles are specified at the center of the capsule holder beginning at reactor North 0 degrees.
	Dosimetry container bottom elevation
	Dosimetry container height
	Dosimetry container thickness
	Dosimetry container width
	Radial distance from the center of the reactor to the center of the dosimetry container
	Provide drawings showing the capsule holder assembly and dosimetry container designs

Table 5-5
Typical Top Guide Design Parameters

Reactor Component	Component Parameter
Reactor Top Guide	Elevation at the bottom of the top guide
	Top guide height (including top and bottom plates)
	Top guide vertical plate thickness
	Top guide cylinder inner radius
	Top guide cylinder wall thickness
	Top guide lower plate height
	Top guide upper plate height
	Provide drawings showing the top guide design including the top and bottom plate outlines

Table 5-6
Typical Core Support Plate Design Parameters

Reactor Component	Component Parameter
Reactor Core Support Plate	Core support plate top elevation
	Core support plate height
	Core support plate rim inner radius
	Core support plate rim wall thickness
	Number of interior (4-assembly) fuel assembly support pieces
	Equivalent radius of flow area in the interior support pieces
	Number of peripheral (1-assembly) fuel assembly support pieces
	Equivalent radius of flow area in the peripheral support pieces
	Number of control rods
	½ Width of opening in core support plate for control rods
	½ Length of opening in core support plate for control rods
Fuel Support Pieces	Elevation at the top of the fuel support pieces
	Inner radius of interior support piece for one assembly
	Thickness of interior support piece wall
	Inner radius of peripheral support piece
	Thickness of peripheral support piece wall
	Provide drawings of the core support plate and fuel support piece designs

Table 5-7
Typical Fuel Design Parameters

Reactor Component	Component Parameter
Reactor Fuel Assembly	Type of fuel assembly (e.g., standard, water box, water cross)
	Fuel assembly pitch
	Fuel rod array size (e.g., 8x8, 9x9, etc.)
	Number of fuel rods
	Number of water rods
	Fuel rod pitch
	Elevation at the bottom of active fuel
	Active fuel height fuel
Fuel Bundle	Description of the rod layout in the fuel zone
	Fuel rod clad outer radius
	Fuel rod clad wall thickness
	Water rod outer radius
	Water rod wall thickness
Fuel Channel	Channel inside dimension
	Channel wall thickness
	Channel corner radius
Assembly Water Gap	½ Width of the water gap containing the control rod
	½ Width of water gap opposite the control rod
Fuel Assembly Upper Region Parts	Axial distance from the top of active fuel to the bottom of the upper fuel end plugs
	Axial height of the fuel end plugs
	Axial height of the upper tie plate (excluding the handle)
Fuel Assembly Lower Region Parts	Axial distance from the bottom of active fuel to the top of the lower tie plate
	Axial height of the lower tie plate

Table 5-8
Typical Reactor Design Drawing Titles

Reactor Component	Drawing Title
Reactor Pressure Vessel	Reactor Pressure Vessel / Reactor Assembly
	General Arrangement – Elevation
	General Arrangement – Plan
	Nozzle Details
	Pressure Vessel Clad
	Pressure Vessel Insulation / Mirror Insulation
	Reactor Building Biological Shield Wall
	Vessel and Attachments Material Identification
Reactor Core Shroud	Shroud Head
	Shroud Head and Steam Separator Stand Pipes
	Shroud Top Section - Core Structure
	Shroud Bottom Section - Core Structure
	Shroud Support and Jet Pump Interface (Baffle)
	Shroud Support Details and Assembly
	Shroud Support Skirt
Reactor Jet Pump	Jet Pump
	Jet Pump Riser Brace
	Jet Pump Adapter
Reactor Core Support Plate	Reactor Arrangement and Assembly
	Core Support
	Core Support Plate
	Fuel Support
	Orificed Fuel Support
	Peripheral Fuel Support
	Fuel Support Plug
Reactor Top Guide	Top Guide
Reactor Dosimetry	Specimens – Surveillance Program
	Specimen Holder
	Capsule Specimen
	Capsule Bracket
	Capsule Container
Reactor Fuel Assembly	Fuel Bundle Design
	Core Management Plans

5.2 Reactor Operating Data Inputs

Reactor operating data that spans the lifetime of the reactor is needed to perform reactor pressure vessel and reactor component neutron fluence calculations. There are two categories of operating data that are needed for the analysis: daily power factors and reactor state-point data. Daily power factors provide a day-by-day accounting of the reactor power level, including shutdown periods. State-point data provides a snapshot of the reactor operating conditions at selected times in the reactor lifetime. Reactor operating conditions include power distributions, moderator density distributions, and fuel isotopic distributions.

Table 5-9 lists the typical reactor operating data that is used in the RAMA fluence calculation. In concert with the reactor geometry information described in Section 5.1, *Mechanical Design Inputs*, the content and quality of the reactor operating data inputs are also significant contributors to determining accurate neutron fluence.

The following subsections present more information on the operating data used in the RAMA fluence calculation.

Table 5-9
Typical Operating Data Required for Fluence Determination

Data Category	Data Required
Reactor Operating History	Daily core power levels
	Reactor shutdown periods
	Core average axial power shapes
	Control rod densities
Reactor Core Operating State Parameters	Core thermal power (or Core power density)
Reactor Core Nodal Distributions (3D Data)	Core nodal relative power distributions
	Core nodal instantaneous water density distributions (instantaneous voids)
	Core nodal fuel isotopics (^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, $^{16}O_{fuel}$)
	Core nodal fuel exposures
Reactor Fuel Assembly Data	Pin-by-pin local peaking factors (specifically for the fuel assemblies in the outer four rows of the core)
Ex-core Coolant Properties	Water densities
Dosimetry / Surveillance Capsule	Dosimetry / surveillance capsule measurements
	Flux wire measurements

5.2.1 Selecting State Points for the RAMA Fluence Model

The reactor core conditions in a BWR change continuously during reactor operation due to fuel burnup, moderator density variations (void distributions), control rod movements, and power distribution changes. BWRs are particularly interesting, and challenging, to analyze because of the significance that these operating conditions might have on the determination of reactor pressure vessel fluence. Over the course of a single operating cycle, dozens of operating states might occur that could each have some impact on fluence determination. In contrast, there might be only a couple operating states that could provide a reasonable representation of the operating conditions of the reactor over an operating period of interest. Determining the specific state points that are needed for a RAMA fluence calculation can be a challenging exercise for BWRs because of the dynamics in the core. This section describes the steps for selecting reactor state points that can be used in the RAMA fluence calculation.

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5.3 Modeling Conventions for the Reactor Core Region

This section describes the format in which the reactor operating data described in Section 5.2 should be generated for the core region of a RAMA fluence model. The core region model is given special attention because it receives data that is generally processed from another code system (viz., core simulator codes). Thus, the data from these external code systems must be processed into a form that matches the requirements of the RAMA model. The following subsections describe an example reactor core model, naming conventions for the core model, and data formats that are needed by the RAMA software.

5.3.1 Description of the RAMA Reactor Core Model

A primary source of core operating data for a RAMA model is core simulator codes. Core simulator codes employ a three-dimensional nodal geometry model that describes the reactor core region as a set of simple rectangular parallelepiped elements. These models are easy to reproduce with the RAMA model.

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Figures 5-1 and 5-2 also illustrate example naming conventions for the volume identifiers in the nodal and fuel assembly parts of the core region. The naming conventions are important as the PMB and SMB codes use these names to communicate data between the two codes. Additional information on naming conventions is provided in the next subsection.

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Figure 5-1
Reactor Core Nodal Geometry Model in Quadrant Symmetry - Isometric View

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**Figure 5-2
Reactor Core Nodal Geometry Model in Quadrant Symmetry - Top View**

5.3.2 Volume and Material Identifier Formats

Each region, or volume, of the RAMA model is identified by a unique volume identifier. Volume identifiers are alphanumeric strings that allow the regions in the model to be identified by descriptive names. Each region is assigned a material that is identified by a material identifier. Material identifiers are alphanumeric strings that describe the materials in the model.

The fully-qualified form of a volume identifier in the RAMA model is:

PartName:PartNo.VolumeNo

where:

PartName	is a sub-string that identifies a part, or region, in the model;
":"	is a required field separator;
PartNo	is an integer value that identifies a specific instance of the part;
"."	is a required field separator; and
VolumeNo	is an integer value that identifies a sub-volume in the part.

Several parts with the same part name, *PartName*, may exist in a model. The different instances of the part are identified using part numbers, *PartNo*, starting with the integer value "1". A part may be composed of one or more sub-volume elements, *VolumeNo*, which are numbered beginning with the integer value "1".

The fully-qualified form of a material identifier in the RAMA model is:

MatName:MatNo

where:

MatName	is a sub-string that identifies a type of material;
":"	is a required field separator; and
MatNo	is an integer value that identifies a specific type of the material.

Several materials with the same material name, *MatName*, may exist in a model. The different instances of a material are identified using material numbers, *MatNo*, beginning with the integer value "1".

Fuel assembly data may be provided in two forms. The preferred form is a detailed model that assumes that the fuel assemblies are described in three regions: the fuel bundle region, the channel, and the water gap. In this model, the fuel bundle region is further sub-meshed into pin-wise detail to provide explicit representations of the fuel and water rod regions. The other form is a homogenized model that assumes that the fuel assembly information is smeared over the entire nodal volume. The detailed model should be provided when pin-wise powers are available.

Different naming conventions are used for each fuel assembly model. Figure 5-1 illustrates that homogenized fuel assembly nodes have a volume identifier prefix of "CORNOD". (Note that there is a second part to the part name that specifies the location of the part in the model using compass coordinates. Additional information on this is provided in the following subsections.)

Figure 5-2 illustrates that detailed fuel assembly nodes have volume identifier prefixes of "CORFUE" for the fuel bundle region, "CORCHA" for the channel, and "CORBYP" for the water gap. Because the volume identifiers are unique, both forms of fuel assembly data may be mixed in a core model. It is recommended, however, that only the detailed form be used.

The following subsections provide additional information with examples on the recommended naming conventions for the RAMA reactor core model.

5.3.3 Part Names for Detailed Fuel Assembly Models

Detailed fuel assembly data is desired for the RAMA reactor core model. The detailed fuel assembly model includes data for the fuel bundle region, the channel, and the water gap. The fuel bundle region is defined by the pin cell regions. The inner channel water gap which exists between the fuel bundle width and the channel inner wall may be included in the bundle description, but is generally included in the channel definition for modeling convenience.

The fuel bundle region is sub-meshed into equal sub-volumes corresponding to the fuel bundle array size (e.g., 64 volumes in an 8x8 array). In the simplest form, each sub-volume corresponds to a smeared pin cell region that is defined by the pin pitch and which is comprised of a homogenized material. An example of the composition of homogenized material for fuel pins would include fuel pellet, clad, and moderator. For water rods, an example of the composition of homogenized material would include bypass water, water tube, and moderator.

Volume identifiers are generated by the PMB code in a consistent manner. By convention, the volume identifiers for a detailed fuel assembly model have the prefix designations "CORFUE" for the pin cells in the fuel bundle region, "CORCHA" for the channel, and "CORBYP" for the water gap. The fully-qualified forms of the volume identifiers for the detailed fuel assembly model are:

CORFUE_N*jj*E*ii*:*k*.*n*	for the pin cell regions of the fuel bundle,
CORCHA_N*jj*E*ii*:*k*.1	for the fuel channel, and
CORBYP_N*jj*E*ii*:*k*.1	for the water gap.

where:

CORFUE_	is the part name for the pin cells in the fuel bundle region;
CORCHA_	is the part name for the fuel channel region;
CORBYP_	is the part name for the water gap region;
N*jj*E*ii*	Identifies the planar position of the fuel assembly in the core region at coordinates North *jj*, East *ii*;

:*k*	specifies the instance of a fuel assembly part at axial elevation *k*; and
.*n*	specifies the number of a sub-volume in the part.

The suffix part of the part names identifies the planar and axial position indexes, "N*jj*", "E*ii*" and "*k*", and sub-volume numbers, "*n*" in the model. The planar position indexes, "*jj*" and "*ii*" each start with "01" at the lower left corner of the reactor core (i.e., the southwest corner of a full core model). "*jj*" is incremented by "+01" going in the north direction and "*ii*" is incremented by "+01" going in the east direction. The axial elevation "*k*" starts with the value "1" for the nodal region at the bottom of the fuel assembly and increments by "+1" to the top of the fuel assembly.

The sub-volume number "*n*" for the fuel bundle part starts with the value "1" for the pin cell region at the lower left corner of the fuel bundle part and increments by "+1" across the rows and up the columns. The total number of pin cell regions is the product of the array size of the fuel bundle; for example, an 8x8 fuel bundle array has 64 pin cell regions. The insert in Figure 5-2 illustrates volume identifiers for a detailed fuel assembly model located at planar coordinate North 26, East 26, and axial node "*k*". Assuming a fuel bundle array size of 8x8 pins at axial position 12, the volume identifier for the fuel pin in the lower left corner of the fuel bundle is "CORFUE_N26E26:12.1" and the volume identifier for the fuel pin in the upper right corner is "CORFUE_N26E26:12.64". The corresponding volume identifier for the channel region is "CORCHA_N26E26:12.1" and for the water gap is "CORBYP_N26E26:12.1".

Each region of the detailed fuel assembly model must contain a unique material description due to fuel burnup effects. It is necessary, therefore, that unique material identifiers be used for each fuel assembly region. It is required that the material identifiers correspond to the naming conventions for the volume identifiers. Therefore, the fully-qualified forms of the material identifiers for the detailed fuel assembly model are:

CORFUE_N*jj*E*ii*Z*k*:*n*	for the pin cell regions of the fuel bundle,
CORCHA_N*jj*E*ii*Z*k*:1	for the fuel channel, and
CORBYP_N*jj*E*ii*Z*k*:1	for the water gap.

where:

CORFUE_	is the part name for the pin cells in the fuel bundle;
CORCHA_	is the part name for the fuel channel;
CORBYP_	is the part name for the water gap;

`NjjEiiZk` identifies the position of the pin cells in the fuel bundle at planar coordinates North *jj*, East *ii*, and axial node *k*; and

`:n` specifies the number of a sub-volume in the part.

The planar position indexes, "*jj*" and "*ii*", axial index, "*k*", and volume numbers, "*n*", correspond to the same indexes as described for volume identifiers.

As an example, the material identifiers for the lower left pin cell region "CORFUE_N26E26:12.1" is "CORFUE_N26E26Z12:1" and for the upper right pin cell region "CORFUE_N26E26:12.64" is "CORFUE_N26E26Z12:64". The corresponding channel and water gap material identifiers for this fuel assembly part are "CORCHA_N26E26Z12:1" and "CORBYP_N26E26Z12:1".

The material for a pin cell region must account for all material compositions in the region. For fuel rods, the material compositions include the fuel (U, Pu, and O_{fuel}), clad (Zr), and active flow (H and O_{water}). For water rods, the material compositions include the bypass water (H and O_{water}), water tube (Zr), and active flow (H and O_{water}). The materials in the pin cell regions should be smeared, or homogenized, in the pin cell volumes. Note that the pin cells must include a proportionate amount of the active flow from the inner channel water gap if the inner channel water gap is included in the fuel bundle region. This is not an easy task which is why the inner channel water gap is commonly included in the channel region. Also note that two oxygen nuclides are referenced - one that is weighted for fuel and one that is weighted for water. In addition, the effective water density in the fuel bundle region, specifically the void fractions, should be accurately represented in the fuel materials as this can have a significant affect on the transport calculation and, thus, the calculated fluence.

The material descriptions for the fuel channel should include the channel (Zr) and, if appropriate, the inner channel water gap (H and O_{water}). The water gap is modeled explicitly as water (H and O_{water}). Note that control rod materials must not be included in the water gap description.

If pin-wise data is not available, the fuel isotopic data may be based on nodal-average or bundle-average data as described in the next section.

5.3.4 Part Names for Homogenized Fuel Assembly Models

If detailed fuel assembly data is not available in accordance with the previous section, then homogenized fuel assembly data may be used. The homogenized material for this model has the same requirements as the detailed fuel assembly model in that the masses of the fuel rods, water rods, channel, and water gaps must be accounted for.

Figure 5-1 illustrates the naming convention for volume identifiers of homogenized fuel assemblies in a RAMA core model. By convention, the volume identifiers for homogenized fuel nodes have the prefix designation "CORNOD". The suffix part of the name identifies the planar and axial position indexes, "N*jj*", "E*ii*" and "*k*", and sub-volume numbers, "*n*" in the nodal model. The fully-qualified form of a nodal part name is:

CORNOD_N*jj*E*ii*:*k*.1

where:

CORNOD_	is the part name for a fuel assembly represented as a single nodal element;
N*jj*E*ii*	identifies the planar position of the fuel assembly in the core region at coordinates North *jj*, East *ii*;
:*k*	specifies the instance of a fuel assembly part at axial elevation *k*; and
.1	specifies the number of the sub-volume element in the part, which is always "1" for nodal volumes.

The planar position indexes "*jj*" and "*ii*" each start with "01" at the lower left corner of the reactor core (i.e., the southwest corner of a full core model). "*jj*" is incremented by "+01" going in the north direction and "*ii*" is incremented by "+01" going in the east direction. The axial elevation "*k*" starts with the value "1" for the nodal region at the bottom of the fuel assembly and increments by "+1" to the top of the fuel assembly.

Figure 5-1 illustrates part names for a fuel assembly containing 25 axial nodes located at planar coordinate North 16, East 16. The nodal element at the bottom of the fuel assembly has the volume identifier "CORNOD_N16E16:1.1" and the nodal element at the top of the fuel assembly has the volume identifier "CORNOD_N16E16:25.1".

Each homogenized fuel node will have a different fuel material description due to fuel burnup and moderator density distributions. It is necessary, therefore, that unique material identifiers be used to describe the materials for each nodal volume. It is required that the material identifiers correspond to the naming conventions of the volume identifiers. The fully-qualified form of a material identifier in the nodal geometry is:

CORNOD_N*jj*E*ii*Z*k*:1

where:

CORNOD_	is the part name for the nodal volume;
N*jj*E*ii*Z*k*	identifies the position of the fuel node in the core region at planar coordinates North *jj*, East *ii*, and axial node *k*; and

`:1` is an integer value identifying the number of the sub-volume element in the part, which is always "1" for nodal volumes.

The planar position indexes "*jj*" and "*ii*", axial index, "*k*", and volume number, "1", correspond to the same indexes as described for volume identifiers.

As an example, the material identifiers for the fuel material corresponding to the volume identifier in the bottom node "CORNOD_N16E16:1.1" is "CORNOD_N16E16Z1:1" and for the top node "CORNOD_N16E16:25.1" is "CORNOD_N16E16Z25:1".

The material for a fuel assembly node should be smeared over the nodal volume and account for all material compositions in the node, including the fuel (U, Pu, and O_{fuel}), clad (Zr), channel (Zr) and active and bypass water (H and O_{water}). Note that two different oxygen (O) nuclides are used - one which is weighted for fuel materials and one which is weighted for water. Note also that control rod materials must not be included in the material description. If node-wise fuel isotopics are not available, then bundle-average fuel isotopics may be used. When using bundle-average isotopics, the natural ends, if present, should be modeled as natural uranium and the enrichment of the fuel bundle adjusted accordingly. In addition, the effective water density in the core, especially relating to void distributions, should be accurately represented in the fuel materials as this can have a significant affect on the transport calculation and, thus, the calculated fluence.

5.4 Operating Data Format Specification

State-point data for the RAMA core model must be provided in electronic form as ASCII data files. The data should be formatted using the input specifications of the following RAMA transport code input blocks.

Data Block	Description
GAT	Block GAT specifies several attributes for the material regions in the RAMA geometry model. The data provided in this block includes material identifiers, volumes, and relative power factors.
NDE	Block NDE specifies the nuclide number densities of the materials in the RAMA model. The data in this block is linked to Block GAT by the material identifiers.
SPT	Block SPT contains case identifiers that describe the operating cycle and various operating state conditions of the reactor for the state-point data.

Material descriptions in block NDE are provided as nuclide concentrations in units of atoms/barn-cm. The nuclides that are available to describe a material in the problem are listed in the RAMA Fluence Methodology User's Manual [3]. Additional information for each of the above data blocks and the entire code system is also provided in the User's Manual.

6
BUILDING BWR GEOMETRY MODELS

This section describes a basic BWR geometry model that is built using the PMB code. Geometry models for the following reactor components and regions are specifically discussed in this section:

- reactor core region,
- core reflector region beside, above, and below the reactor core,
- core shroud,
- downcomer region,
- jet pumps,
- surveillance capsule,
- pressure vessel,
- mirror insulation,
- cavity regions, and
- biological shield (or concrete wall).

The reactor model that is described in this section is for a typical BWR/4 class plant with jet pumps and a core loading of 764 fuel assemblies. The design inputs that are needed to build the reactor model were described in Section 5, *BWR Design Inputs.*

The geometry models for the parts described in this section include illustrations of example geometry meshing. The example meshing is provided for discussion purposes only. The actual meshing for a model must be determined on a plant-specific basis in accordance with the planar and production model sensitivity studies described in Sections 4.6, *BWR Planar Model Sensitivities*, and 4.7, *BWR Production Model Sensitivities.*

6.1 Reactor Design Description

This section describes the basic design of a BWR for which a geometry model is built. The major reactor components included in the model are the reactor core, the core shroud, the downcomer region with jet pumps, the reactor pressure vessel, surveillance capsules, mirror insulation, and biological shield.

The RAMA geometry modeling system is capable of modeling each reactor component in accurate design detail. While the modeling capability is very flexible, the following restrictions are observed in the model.

- The full geometry model of the reactor system should describe a right circular cylinder element of finite axial height, where the radius is defined by the outer surface of the biological shield wall and the axial height is defined by the elevations of interest in the analysis.
- The geometry model should assume azimuthal symmetry which is defined by the azimuthal range of 0 degrees to 45 degrees.
- The top and bottom planes that define the axial height of the geometry model must be perpendicular to the vertical axis of the right circular cylinder.
- There should be no "holes" in the geometry, i.e., all regions of the solution geometry should be defined by a material.
- There should be no overlapping regions in the geometry.

Figure 6-1 illustrates a planar view of a typical BWR design at an axial elevation near the core mid-plane. The planar view assumes octant symmetry that corresponds to the north-northeast sector, or octant, of the reactor. The north-northeast octant starts at 0 degrees azimuth, which corresponds to the "north" compass direction as specified in vendor drawings, and ends at 45 degrees azimuth.

Figure 6-1 shows that nine radial regions are represented in the model: the core region (comprised of interior, outer, and peripheral fuel assemblies), core reflector, shroud, downcomer with jet pumps and surveillance capsule, pressure vessel, inner and outer cavities, mirror insulation, and biological shield. The reactor pressure vessel figure is shown to have cladding on the inner surface of the pressure vessel wall. The mirror insulation is shown to have cladding on the outer surface of the mirror insulation wall. The biological shield is shown to have cladding on the inner and outer surfaces of the biological shield walls.

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Figure 6-1
Planar View of an Example BWR Configuration

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The peripheral fuel assemblies are the fuel assemblies that contribute most significantly to the ex-core fluence. These assemblies should be meshed in sufficient detail to provide an accurate representation of the pin-wise neutron source distribution at the core edge. The meshing of the fuel assemblies in the core region is described in more detail later in this section.

The downcomer region in Figure 6-1 is shown with one jet pump assembly located at azimuth 30 degrees. The downcomer region also shows one surveillance capsule that is also located at azimuth 30 degrees in the reactor.

The axial span of the model ranges from about 30 cm below the bottom of active fuel to about 45 cm above the top of active fuel. Figure 6-2 shows a side view of the reactor with selected axial dimensions. The bottom elevation of the model extends to below the core support plate and includes the lower fuel tie plate, nose piece, fuel support piece, and core plate. The upper elevation of the model extends to above the top guide and includes the upper fuel plenum, upper tie plate, top guide, and upper shroud plenum.

Most of the core and the ex-core regions extend uniformly from the bottom of active fuel to the top of active fuel. The jet pumps and surveillance capsule are the exceptions. The jet pumps range axially from the bottom elevation to an elevation of about 85% of the core height. The downcomer region above the jet pumps is entirely water. The surveillance capsule is centered axially at the core "mid-plane" elevation. The axial height of the capsule is split evenly around the core mid-plane, where half extends above the core mid-plane and half below the core mid-plane.

6.2 Reactor Core Configuration

The PMB code uses input blocks DEF:CORE and CORNOD to build a reactor core model. Block DEF:CORE describes the basic configuration of the reactor core region. Block CORNOD describes the geometry and mesh types for modeling the fuel assemblies in the core region.

The fuel assembly array provided in the DEF:CORE input block specifies the location of rodded and unrodded fuel assembly cells. These designations are important for ensuring the proper orientation of each fuel assembly geometry in the core layout.

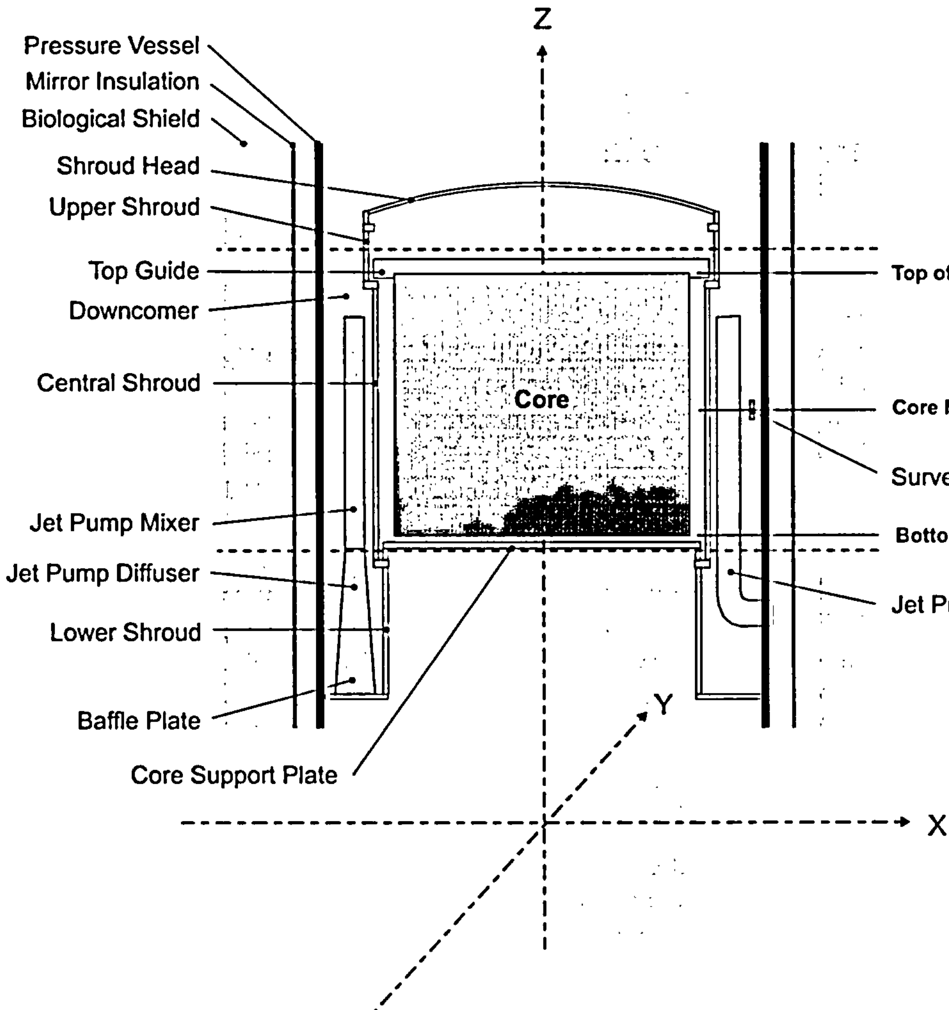

Figure 6-2
Side View of an Example BWR Configuration

6.3 Reactor Core Geometry Description

The reactor core geometry is built using the PMB CORNOD input block. Block CORNOD actually builds three regions of the reactor model: the core region, the core radial reflector region beyond the core periphery, and the central shroud beside the core. The basic core geometry is the same for all operating cycles. The core geometry meshing may change between cycles depending upon the dominant fuel type loaded on the core periphery, e.g., an 8x8 versus a 9x9 fuel assembly design.

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A core reflector region lies between the reactor core region and the central shroud. The reflector region includes transition parts that interface the rectangular shape of the core region with the cylindrical shape of the shroud. Figure 6-4 illustrates a planar view of the core reflector region model with meshing. The planar meshing is uniform along the entire axial height of the core region. Also shown is a list of the volume identifiers that describe the parts and mesh regions in the core reflector model.

The core reflector parts are built as extensions to the columns and rows of fuel assemblies. The reflector parts ("CORRFL_N*jj*E*ii*:*k*.*n*") are constructed from three geometry types: rectangular bodies, basic rectangular bodies with one curved surface, and cylindrical bodies. The reflector parts that interface with the core region are based on rectangular bodies. The size of the rectangular meshing varies with each column and row of parts, as the PMB code employs algorithms to adjust the meshing to avoid generating parts with extremely small and extremely large volume meshes. Rectangular meshing is used until the meshing exceeds the outer edge of the core region. At this point, the part includes a mesh that provides for the transition of a rectangular-based geometry to a cylindrical geometry. The core reflector region is then completed with one or more annuluses that are formed from cylindrical parts. Figure 6-4 shows that two annulus parts ("CORRFL_N*tt*.*k*.*n*") are built between the rectangular-based parts and the shroud wall.

The core shroud is a right circular cylinder pipe that surrounds the reactor core region. The shroud is meshed radially into two annuluses of equal thickness and azimuthally into 30 equal arcs ranging from 0 to 45 degrees. The shroud wall is assumed to maintain the same radius over the height of the core region. Note that the actual shroud radius varies at the top of active fuel where the fuel extends past the upper shroud flange. The assumption will have no effect on the pressure vessel wall or capsule fluence evaluations near the midplane elevation. Figure 6-5 illustrates the core shroud model and the format of the volume identifiers ("CORSHR_N*tt*:*k*.*n*") that describe the model regions.

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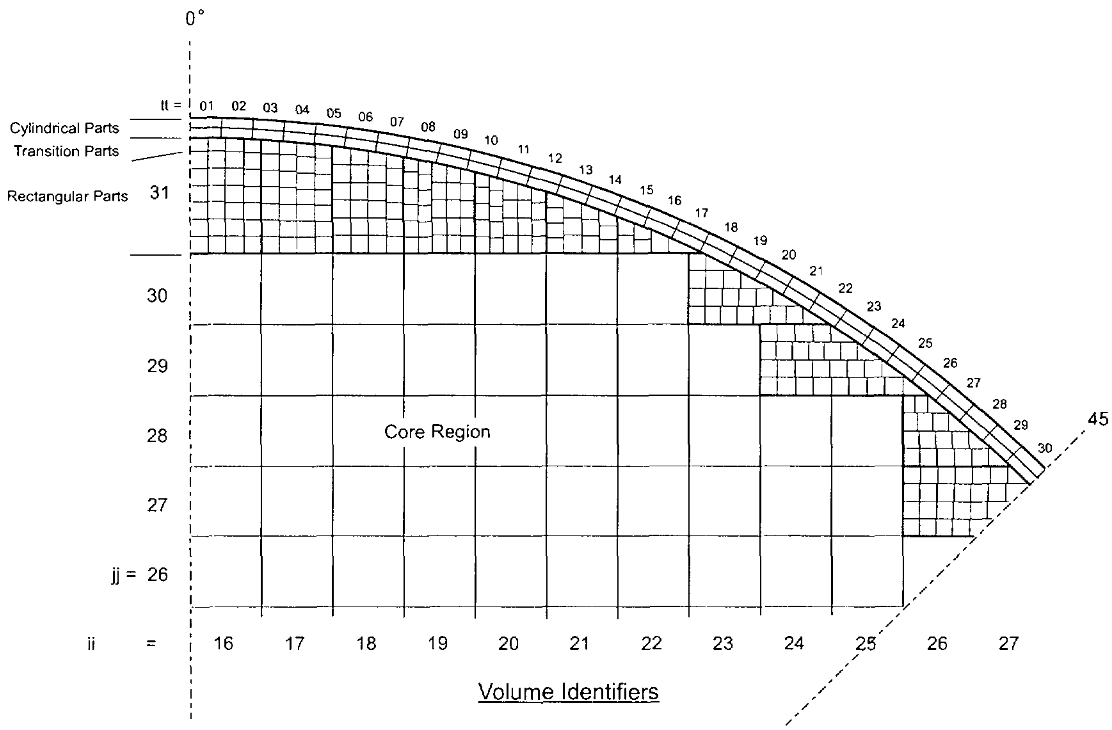
Figure 6-3
Isometric View of an Example Reactor Core Geometry



Rectangular/Transition Parts

CORRFL_N31E{16-17}{01-04}:1-25.1-7	CORRFL_N3004E23:1-25.1-2	CORRFL_N2804E26:1-25.1-3
CORRFL_N31E18{01-04}:1-25.1-6	CORRFL_N3003E23:1-25.1-4	CORRFL_N2803E26:1-25.1-4
CORRFL_N31E19{01-02}:1-25.1-6	CORRFL_N3002E23:1-25.1-6	CORRFL_N2802E26:1-25.1-5
CORRFL_N31E19{03-05}:1-25.1-5	CORRFL_N3001E23:1-25.1-8	CORRFL_N2801E26:1-25.1-6
CORRFL_N31E20{01-02}:1-25.1-5		
CORRFL_N31E20{03-05}:1-25.1-4	CORRFL_N2904E24:1-25.1-5	CORRFL_N2704E26:1-25.1-7
CORRFL_N31E2101:1-25.1-4	CORRFL_N2903E24:1-25.1-7	CORRFL_N2703E26:1-25.1-7
CORRFL_N31E21{02-04}:1-25.1-3	CORRFL_N2902E24:1-25.1-8	CORRFL_N2702E26:1-25.1-6
CORRFL_N31E22{01-02}:1-25.1-2	CORRFL_N2901E24:1-25.1-9	CORRFL_N2701E26:1-25.1-5
CORRFL_N31E22{03-04}:1-25.1		

Cylindrical Parts

CORRFL_N01-30:1-25.1-2

Figure 6-4
Planar View of an Example BWR Core Reflector Geometry

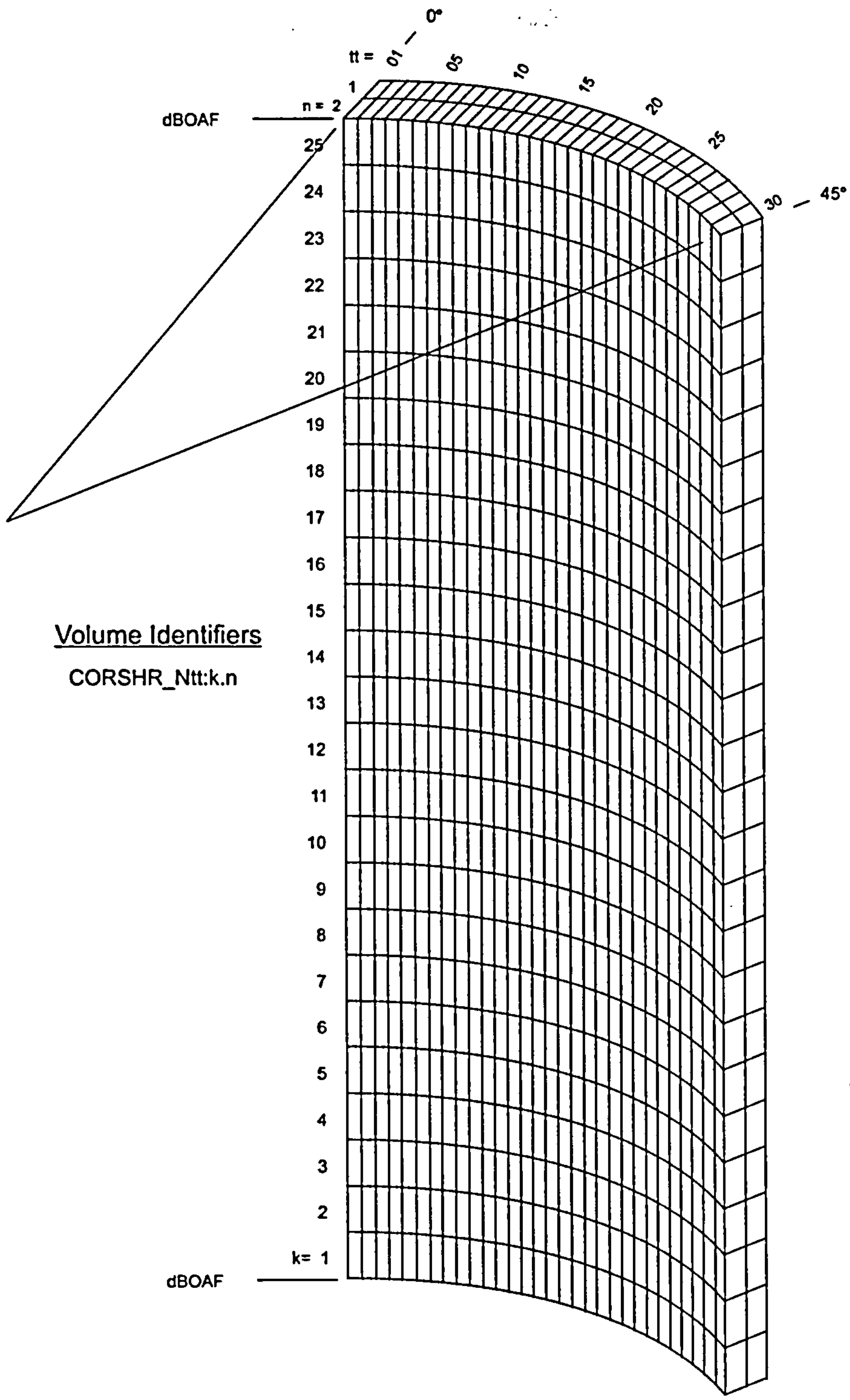


Figure 6-5
Isometric View of an Example Core Shroud Geometry

6.4 Core Elevation External Geometry Description

This section describes the geometry models for the regions that lie beyond the core shroud. These regions include the downcomer, reactor pressure vessel, mirror insulation, cavity regions, and biological shield. The geometry models for the jet pump assembly and surveillance capsule, which are located in the downcomer region, are also described.

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The downcomer region is defined in the PMB DWNCMR input block. Figure 6-6 shows a planar view of an example downcomer region near the core mid-plane elevation in octant symmetry. The downcomer region is divided into two annuluses. The jet pump assembly is located in the inner annulus of the downcomer region and the surveillance capsule is located in the outer annulus of the downcomer region. Both the jet pump assembly and surveillance capsule are positioned at azimuth 30 degrees. Because of the different structural components in the inner and outer annuluses of the downcomer model, each has different meshing requirements. Figures 6-7 and 6-8 illustrate planar views of the downcomer inner annulus with meshing applied.

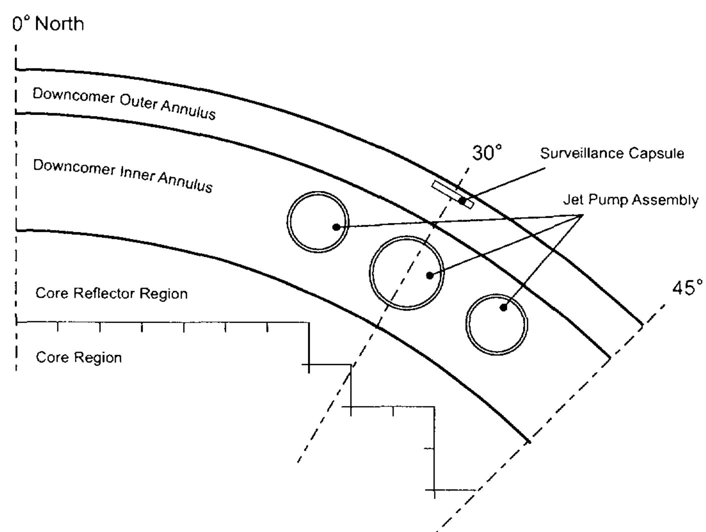

Figure 6-6
Planar View of an Example Downcomer Geometry

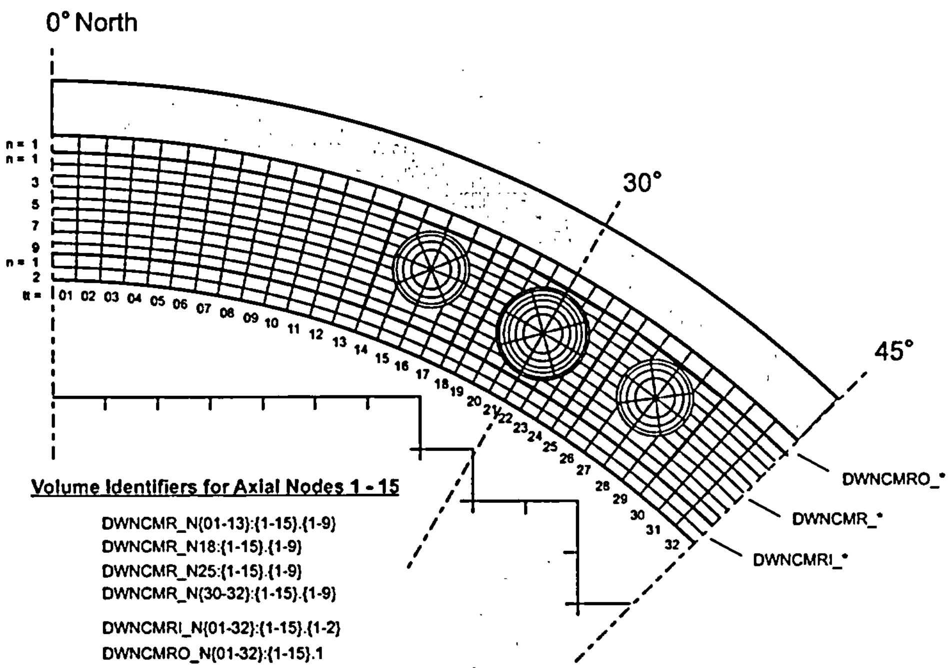

Figure 6-7
Planar View of an Example Downcomer Inner Annulus Geometry

Figure 6-7 shows the downcomer water region with example meshing and volume identifiers that describe the water region parts. The jet pump parts, which are shown as shaded areas, are shown separately in Figure 6-8. Figure 6-8 shows example meshing and volume identifiers for the various steel and water regions of the jet pump model.

The jet pumps in the reactor model range axially from the bottom of active fuel to an elevation of about 85% of the core height. In building the jet pump model, two assumptions are made:

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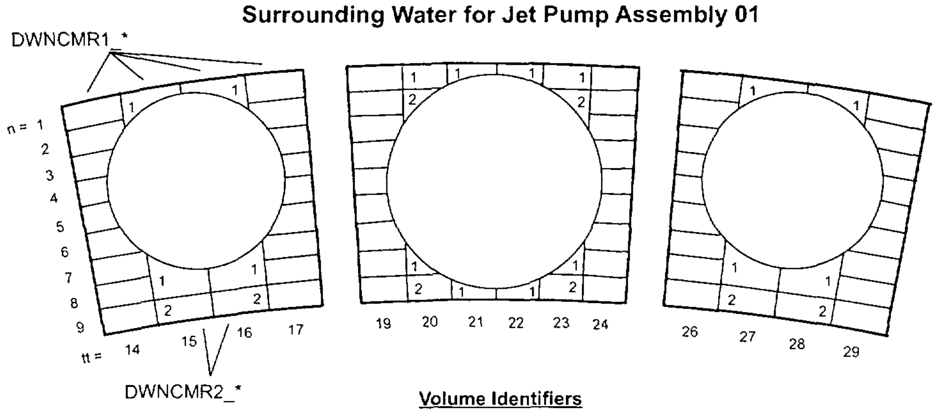
Neither assumption has a significant effect on the pressure vessel fluence calculations. The downcomer inner annulus at the elevations above the jet pumps is comprised entirely of water.

Jet Pump Assembly 01

Surrounding Water for Jet Pump Assembly 01



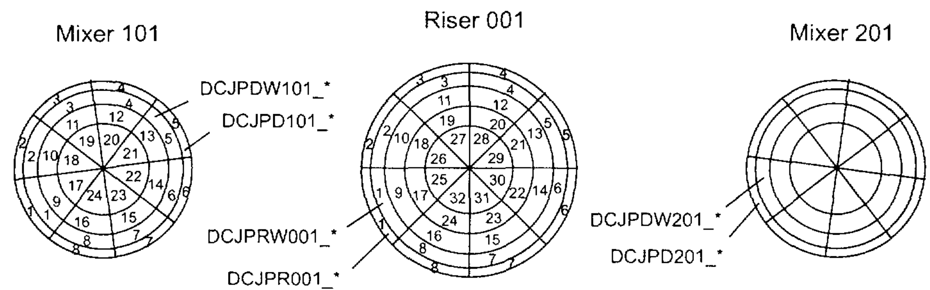
Volume Identifiers

DWNCMR1_N14:{1-15}.{1-9}
DWNCMR1_N{15-16}:{1-15}.1
DWNCMR2_N{15-16}:{1-15}.{1-2}
DWNCMR1_N17:{1-15}.{1-9}

DWNCMR1_N19:{1-15}.{1-9}
DWNCMR1_N20:{1-15}.{1-2}
DWNCMR2_N20:{1-15}.{1-2}
DWNCMR1_N{21-22}:{1-15}.1
DWNCMR2_N{21-22}:{1-15}.1
DWNCMR1_N23:{1-15}.{1-2}
DWNCMR2_N23:{1-15}.{1-2}
DWNCMR1_N24:{1-15}.{1-9}

DWNCMR1_N26:{1-15}.{1-9}
DWNCMR1_N{27-28}:{1-15}.1
DWNCMR2_N{27-28}:{1-15}.{1-2}
DWNCMR1_N29:{1-15}.{1-9}

Riser and Mixer Pipes for Jet Pump Assembly 01



Volume Identifiers

DCJPD101_N01:{1-15}.{1-8}
DCJPDW101_N01:{1-15}.{1-24}

DCJPR001_N01:{1-15}.{1-8}
DCJPRW001_N01:{1-15}.{1-32}

DCJPD201_N01:{1-15}.{1-8}
DCJPDW201_N01:{1-15}.{1-24}

Figure 6-8
Planar View of an Example Jet Pump Geometry

The downcomer outer annulus, which contains a surveillance capsule that is positioned at azimuth 30 degrees, is built from right circular cylinder bodies using the PMB RCPIPE input block.

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The axial height of the surveillance capsule is about 25 cm and is centered between axial planes 12 and 13 of the core geometry model. The height of the surveillance capsule is preserved in the model. The axial meshing above and below the capsule is varied to accommodate the height of the capsule. The downcomer regions that are axially above and below the surveillance capsule elevation use a uniform planar and axial meshing as defined for standard right circular cylinder parts.

The reactor pressure vessel and the regions outside the pressure vessel are modeled as right circular cylinder pipe using the PMB RCPIPE input block.

Figure 6-10 illustrates the reactor pressure vessel geometry and volume identifiers that describe the pressure vessel part. Note that the pressure vessel clad part is omitted from the illustration. The clad is a single annulus part on the inside surface of the pressure vessel wall. It has the same azimuthal and axial meshing as the pressure vessel wall.

Outside the reactor pressure vessel is the cavity region, mirror insulation, and biological shield (concrete wall). These regions provide boundary conditions for the neutron flux calculation and do not require extensive modeling detail. The outer regions are modeled as right circular cylinder pipes similar to the geometries for the central shroud and reactor pressure vessel.

The radial and azimuthal meshing in the outer pressure vessel regions varies as follows:

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Figure 6-9
Planar View of an Example Downcomer Outer Annulus Geometry

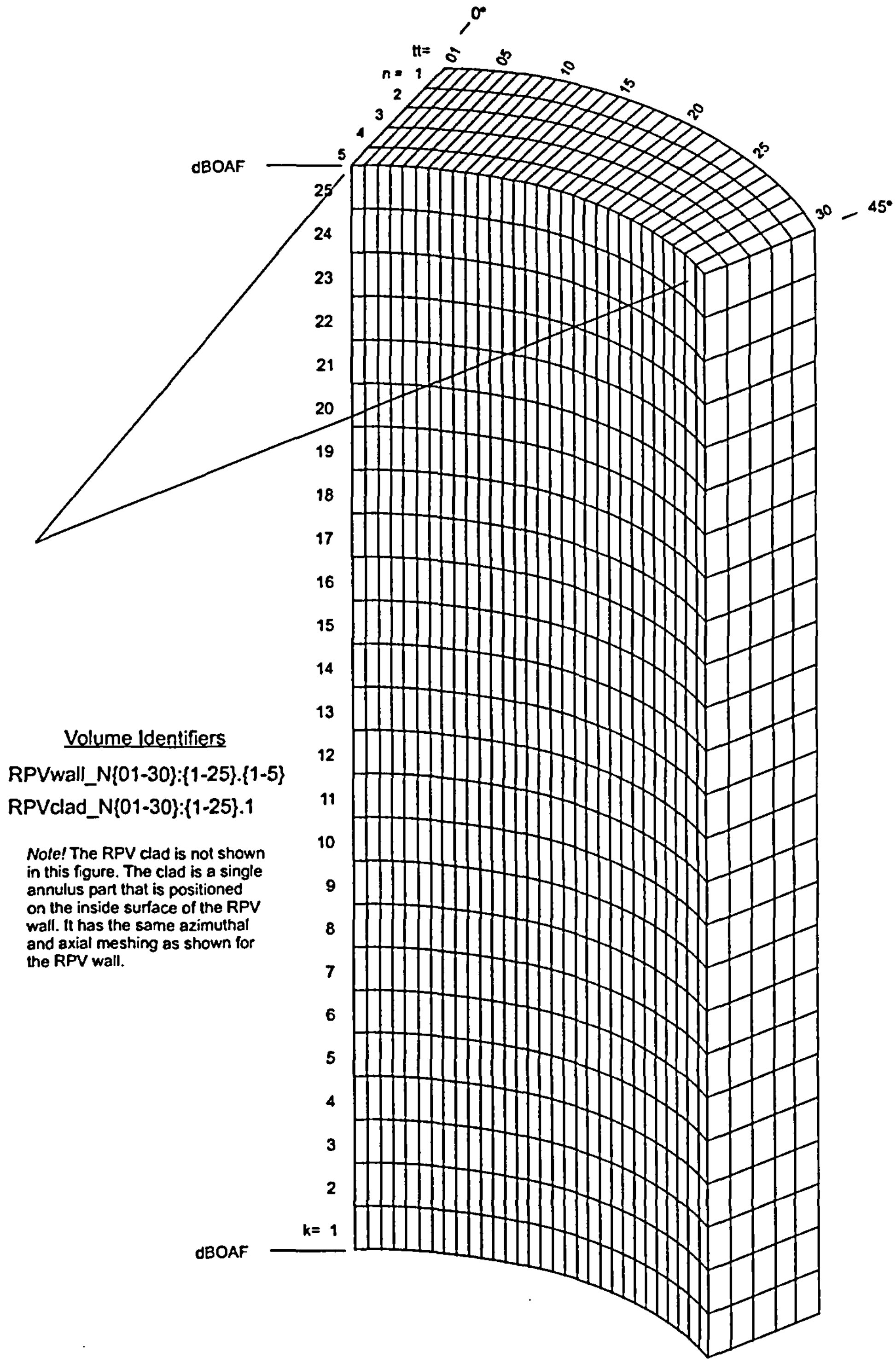

Figure 6-10
Side View of an Example Reactor Pressure Vessel Geometry

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Figure 6-11
Planar View of an Example Outer Reactor Pressure Vessel Geometry

7
RAMA DOSIMETRY CALCULATIONS

An important element in the application of the Methodology is the comparison of predicted activation levels in surveillance capsule dosimetry to measured values. Comparison of activation predictions to measurements is essential for qualifying the applicability of the methodology for use in estimating RPV fluence. The comparisons also form the basis for estimating the bias and uncertainty associated with RPV fluence predictions.

The primary source of activation measurements in BWR reactors is activation samples retrieved from in-vessel surveillance capsules. The activation samples may be in the form of flux wires (typically copper and iron wires) and dosimetry samples consisting of flux wires and/or foil samples.

7.1 Surveillance Capsule Model

The geometry modeling flexibility of the RAMA transport code allows the surveillance capsule to be explicitly represented in the three-dimensional transport calculation. It is possible to model the surveillance capsule in detail, however, that level of detail in the capsule can result in prohibitively long execution times with no significant improvement in accuracy. Therefore, it is generally desirable to represent the capsule with a series of homogenized sub-meshed regions. It is important to include the steel Charpy samples contained within the capsules, since these perturb the flux exposing the dosimeters. A typical surveillance capsule model is described in Section 6.

7.2 RAFTER Activation Prediction

The RAFTER code module of the Methodology is the primary tool for predicting the activation in regions of the reactor system resulting from neutron irradiation. RAFTER utilizes the flux data output punch files (state-point punch files) generated by the RAMA transport code in Section 4.8, *RAMA Transport Calculations*, along with power history information over the period of sample irradiation to represent the spatial- and time-dependent neutron irradiation in the reactor system. In addition to the neutron irradiation information, RAFTER requires the user to identify the locations (i.e., RAMA transport calculation regions) at which activations are to be determined and the activation reactions to be computed.

A list of the activation response reactions that are provided in the RAFTER activation prediction module are presented in the RAMA Fluence Methodology User's Manual [3]. These response reactions originate from the BUGLE-96 nuclear data library. These response reactions are available for two types of neutron collapsing spectra: "flat weighted" collapsing spectra (designated as response set 7001) and "¼ RPV thickness" collapsing spectra (designated as response set 7003). The results calculated for most fast neutron dosimeters are not very sensitive to which set is used (the ^{237}Np dosimeter is slightly sensitive in some cases). In addition, user-specified activation reactions may be supplied to the RAFTER activation prediction module to permit activation predictions for reactions that are not included in the RAFTER computational module.

7.3 Comparison of Predicted Activity to Measurements

The predicted activity from the RAFTER module of the Methodology software system should be compared to the reported specific activity measurements obtained from hotcell evaluations. It is best to use specific activity measurements rather than saturated activity since the effects of irradiation time, power history, and product decay are already incorporated into the RAFTER predicted activity values.

The predicted activity from RAFTER is reported in units of Becquerels/cm^3 (1 Becquerel = 1 disintegration per second or dps). Since specific activity measurements are usually reported in units of either Bq/gm or micro-curies/gm, it is generally necessary to convert the RAFTER predicted values into the corresponding units of the activity measurements. The conversion consists of dividing the RAFTER predicted activity by the density of the specimen, and dividing by the conversion factor of 3.7×10^4 Bq per micro-curie if the measurement units are micro-curies/gm. When making this conversion, it is important to note whether the mass term in the measured specific activity is based upon the mass of the target isotope or the mass of the naturally-occurring element. The appropriate density should be used in the activity conversion.

When comparing RAFTER predicted fission reaction activities, such as ^{238}U(n,fission), to measurements it should be noted that the RAFTER predicted values do not include adjustments for the fission yields of the measured isotopes. As a result, the product isotope fission yield fraction should also be included in the RAFTER conversion for these reactions if the yield correction is not included in the reported measurements. For example, the RAFTER predicted activity for the reaction ^{238}U(n,fission)^{137}Cs should be multiplied by 0.0605 prior to comparison to measured values to account for the 6.05% yield of ^{137}Cs from ^{238}U fissions. In addition, the fission reaction activity prediction from RAFTER does not include any compensation for photofissions (fissions generated by photon, i.e., gamma ray, interactions). Generally a small correction for this effect may be applied to the RAFTER predicted fission reaction activity. It should be noted that although RAMA is capable of predicting the photon distribution in the reactor system this feature has not been qualified for use in RPV and surveillance capsule fluence evaluations.

The comparison of predicted to measured values is usually expressed as the ratio of predicted (or calculated) to measured values, referred to as the "C/M" ratio. The mean values of this ratio for the various measurement samples and the corresponding standard deviation are used as inputs to the uncertainty analysis discussed in Section 8 of this manual.

8
RAMA UNCERTAINTY CALCULATIONS

Calculation of the neutron flux distribution is sensitive to material and geometric representation of the core and reactor internals, the neutron source, the nuclear cross-section data, and the numerical scheme used in the calculation. An uncertainty analysis is incorporated into the Methodology software to estimate the level of uncertainty for the vessel fluence calculation. This section describes the parameters and values that must be determined and input into the Methodology to account for the uncertainty in the fluence calculation. The uncertainty evaluation is composed of three steps: (1) analytic uncertainty analysis, (2) comparison uncertainty analysis, and (3) combined (i.e., overall) uncertainty analysis. This section describes the method of performing each of these analyses. Section 9 describes the application of the uncertainty evaluation to the determination of the best estimate neutron flux and fluence.

8.1 Analytic Uncertainty Evaluation

An analytic uncertainty analysis is required to demonstrate the accuracy of the fluence evaluation methodology [1]. The important sources of uncertainty, i.e. analytical parameters, must be identified and their impact on RPV fluence quantified. The analytic uncertainty is determined by evaluating the sensitivity of the >1.0 MeV fluence at key locations, i.e., the peak fluence location on the RPV inside surface, the RPV T/4 position, and the RPV 3T/4 position. The sensitivity is identified by varying the analytical parameters in the RAMA fluence evaluations and dividing the resulting variation in the predicted (>1.0 MeV) fluence at the key locations by the variation in the analytical parameter. The uncertainty in the fluence at the key locations resulting from each analytical parameter is determined by multiplying the sensitivity by the analytical parameter uncertainty.

In many instances, it is acceptable to use two-dimensional (2D) neutron transport evaluations to determine the analytical parameter sensitivity, however, axial parameter sensitivity evaluations generally require computations that utilize three-dimensional (3D) transport models. A single state point (either an actual state point or a state point based upon a cycle average condition) may be used in the sensitivity evaluations to simplify the computation process since only one RAMA transport evaluation would be needed for each parameter sensitivity evaluation. If a single state point is used in the sensitivity evaluations, it is acceptable to use variations in the predicted (>1.0 MeV) flux instead of the neutron fluence.

In order to determine the impact of the analytical parameter uncertainty on surveillance capsule and RPV fluence it is necessary to establish a 2D and a 3D reference case for use in the parameter sensitivity evaluation. The reference cases should be similar in geometric meshing to the production case used to obtain the predicted surveillance capsule and RPV fluence. The amount of variation in an analytical parameter in a sensitivity case does not need to be directly related to the uncertainty in the parameter. It is only necessary that the variation be large enough to obtain a significant variation in the resulting neutron fluence, but small enough that a first estimate of the flux derivative is accurate (i.e., the deviation in the parameter over the range of the variation is approximately linear).

8.1.1 Geometry Analytical Parameters

Table 8-1 contains a list of the geometry analytical parameters that are important sources of uncertainty in determining RPV fluence. Also shown in the table is the type of model (2D or 3D) that can be used to assess the sensitivity of the parameter. The uncertainty in these parameters is usually determined from the geometric tolerances of the parameters, with the tolerance range representing an uncertainty of $\pm 2\sigma$. In general, there is no bias associated with the geometry analytical parameters since nominal values of the parameters are used in the neutron fluence evaluation.

Table 8-1
Geometry Analytical Parameters

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8.1.2 Material Composition Analytical Parameters

Table 8-2 contains a list of the material composition analytical parameters that are important sources of uncertainty in determining RPV fluence. Also shown in the table is the type of model (2D or 3D) that can be used to assess the sensitivity of the parameter. The uncertainty in the steel composition parameters is usually determined based upon the allowable range of the material's composition in steel. The uncertainty in water density parameters is usually determined based upon system heat balance computations. The uncertainty in stack density is usually based upon fuel vendor estimates. The full range of variation in these parameters represents an uncertainty of 2 . Since nominal values of these parameters are used in the neutron fluence evaluation, there should be no bias associated with the material composition analytical parameters.

Table 8-2
Material Composition Analytical Parameters

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8.1.3 Fission Source Analytical Parameters

Table 8-3 contains a list of the fission source analytical parameters that are important sources of uncertainty in determining RPV fluence. Also shown in the table is the type of model (2D or 3D) that can be used to assess the sensitivity of the parameter.

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Table 8-3
Fission Source Analytical Parameters

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8.1.4 Nuclear Data Analytical Parameters

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Table 8-4
Nuclear Data Analytical Parameters

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8.1.5 Modeling Input Analytical Parameters

There are generally several modeling input parameters that potentially can impact the prediction of RPV neutron fluence. For some of these modeling input parameters, the proper selection for the type of evaluation being performed tends to lessen, if not eliminate, their impact. The uncertainty resulting from the remaining modeling input parameters can be determined using the results obtained from modeling input sensitivity evaluations, as described in Section 4.6, *BWR Planar Model Sensitivities*. In addition, the fluence bias applicable to these parameters is determined based upon the difference between the reference case and the asymptotic case results obtained from the Section 4.6 evaluation.

Table 8-5 contains a list of the modeling input analytical parameters that are important sources of uncertainty in determining RPV fluence. Also shown in the table is the type of model (2D or 3D) that can be used to assess the sensitivity of the parameter. The uncertainty in the individual modeling input parameters is estimated based upon the sensitivity case results. The variation in >1.0 MeV fluence at the key RPV locations resulting from each parameter represents 2 .

Table 8-5
Modeling Input Analytical Parameters

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The variation in RPV >1.0 MeV fluence resulting from variations in each of the modeling input analytical parameters is determined from the modeling sensitivity evaluations performed as described in Section 4.6 of this manual.

8.2 Comparison Uncertainty Evaluation

The comparison uncertainty is determined from the comparison between measured and calculated dosimetry activation results obtained from the fluence benchmark and operating reactor surveillance capsule evaluations. The calculated flux is used to determine calculated activities at the dosimeter locations, as described in Section 7 of this manual.

8.2.1 Operating Plant Measurements

In-vessel surveillance capsule dosimetry measurements obtained from the BWR being evaluated, or from similar BWR designs, are used to validate the methodology for calculation of RPV fluence. Comparisons to measurements for similar BWRs should be used to determine the comparison bias and uncertainty. As additional similar BWR measurement data is accumulated from surveillance capsule evaluations, the data, if acceptably reliable, is added to the measurement database for the type of plant being evaluated.

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8.2.2 Pressure Vessel Simulator Measurements

Measurement comparisons for three pressure vessel simulator benchmark problems (i.e., PCA, VENUS-3, and H. B. Robinson 2 in-vessel and cavity dosimetry benchmarks) are used to validate the applicability of the Methodology to predict RPV fluence. The comparisons of RAMA predicted activation to measured dosimetry from these three established benchmarks contribute to the computation of the overall (or combined) bias and uncertainty.

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8.3 Overall (Combined) Uncertainty Evaluation

As defined in NRC Reg. Guide 1.190, the combined uncertainty is an appropriate combination of the bias and uncertainty obtained from the analytic uncertainty combined with the bias and uncertainty based upon comparisons to operating data and benchmark comparisons.

8.3.1 Overall (Combined) Bias

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8.3.2 Overall (Combined) Standard Deviation

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9 RAMA NEUTRON FLUENCE CALCULATIONS

Determining the neutron fluence in the Reactor Pressure Vessel (RPV) is an essential element in the assessment of RPV damage resulting from neutron irradiation. Generally, the evaluation is restricted to a reduced energy range since low energy neutrons are substantially less likely to inflict irradiation damage than are high-energy neutrons. Material damage assessments are usually restricted to integral neutron fluences with neutron energy >1.0 MeV.

This section describes the procedure for using the Methodology to calculate the accumulated RPV fluence and to determine the best estimate limiting RPV fluence using the overall bias and uncertainty that are described in Section 8. This section also provides guidance for determining an appropriate end-of-life RPV fluence. The following steps form the basis for determining the RPV fluence to be reported in accordance with [1].

9.1 Calculating Accumulated RPV Neutron Fluence

The accumulated RPV neutron fluence with energy >1.0 MeV requires that a RAMA 3D transport calculation be performed for each state point used to define the important operating conditions throughout the operation of the plant (note that state point selection is discussed in Section 5, *BWR Design Inputs)*. The RAMA 3D transport calculations that were discussed in Section 4.8, *RAMA Transport Calculations*, produced flux data punch files during execution. The punch files are used as state-point input data files to the RAFTER module of the Methodology. The individual state points must be appropriately mapped to the reactor power history, as discussed in Section 4.2, *Reactor Design Inputs*. This is required in order for the RAFTER module to accurately account for variations in reactor power over the irradiation period.

It is not uncommon for the locations of interest in the RPV wall to lie between meshed regions. When this is the case, interpolation (or extrapolation for inner surface or outer surface locations) should be used to determine the flux at the desired vessel locations. Depending upon the meshing, interpolation may be required in the radial, azimuthal, and/or axial directions.

9.2 Determining the Best Estimate RPV Neutron Fluence

The best estimate RPV fluence at the maximum fluence location is determined by adjusting the calculated fluence obtained from Section 8.2 using the overall bias and uncertainty from Section 8.3. According to [1], if the overall bias is non-zero and the overall uncertainty does not exceed 20%, then the best estimate flux is calculated by a multiplicative bias correction applied to the calculated RPV fluence values. If the overall uncertainty exceeds 20% but does not exceed 30%, then the best estimate fluence is calculated by a multiplicative bias correction (if the bias is non-zero) with an additional adjustment for the amount that the overall uncertainty exceeds 20%. If the overall uncertainty exceeds 30%, then the neutron fluence methodology exceeds the guidelines presented in [1] and the fluence application must be reviewed by the NRC on an individual basis.

9.3 Determining End-of-Life Neutron Fluence Predictions

Extension of the best estimate neutron fluence to the predicted end-of-life of the RPV requires that representative powers and other state point operating conditions be estimated for the period of irradiation beyond the current operating cycle. These representative conditions should be selected based upon such considerations as the likelihood that the most recent cycle will be representative of future operating cycles and/or the availability of projected plant operating conditions that are likely to be representative of future operating cycles. It is also important to consider the extent to which a conservative versus a best estimate prediction is desired.

Once the conditions have been established, the prediction of RPV fluence to end-of-life consists of using the RAMA transport module to compute the neutron flux distributions for these predictive operating conditions. The resulting flux data punch files are used as state point files that are added to the RAFTER production 3D fluence evaluation case. The power history in the RAFTER input is extended forward to the desired end-of-life Effective Full Power Days (EFPD). The future power history can be represented as a single power step for each future state point with the power specified as the reactor rated power. Any conditions that would affect the current rated core power, such as power uprate, should be included in the RAFTER end-of-life fluence prediction evaluation.

10
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Nuclear Power

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