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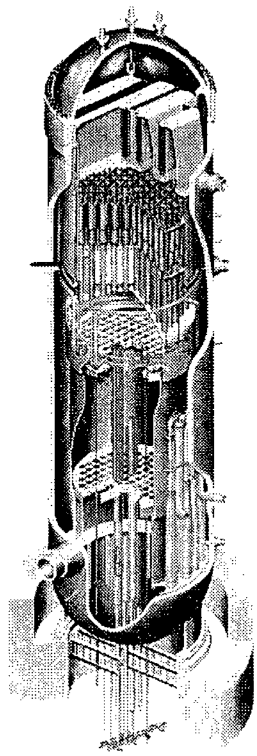
Evaluation of RAMA Fluence Methodology Calculational Uncertainty



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Evaluation of RAMA Fluence Methodology Calculational Uncertainty

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REPORT SUMMARY

This report documents the overall calculational uncertainty associated with the application of the Radiation Analysis Modeling Application (RAMA) Fluence Methodology to BWR reactor pressure vessel fluence evaluations.

Background

The RAMA Fluence Methodology calculates activation and neutron fluence in BWR components. RAMA includes a transport code, model builder codes, a fluence calculator code, an uncertainty methodology, and a nuclear data library. The transport code, fluence calculator, and nuclear data library are the primary software components for calculating the neutron flux and fluence.

The U. S. Nuclear Regulatory Commission approved RAMA for application in accordance with U. S. Regulatory Guide 1.190. Compliance with the provisions of this Regulatory Guide requires that RAMA be qualified using comparisons to plant-specific measurement data and industry benchmark problems. This project performed data comparisons from several plant-specific surveillance capsule and simulator benchmark problems in order to qualify RAMA for use in reactor pressure vessel (RPV) fluence evaluations for BWRs. This report presents plant and benchmark data that have been evaluated with RAMA.

Objectives

To conduct an uncertainty and bias assessment required by Regulatory Guide 1.190, using the RAMA Fluence Methodology, by comparing calculated-to-measured activations of plant surveillance capsules, BWR internal components, and benchmark simulation experiments.

Approach

The project team compiled a total of 416 measurement samples from 22 BWR surveillance capsules. The team obtained the 413 measurements from the Pool Critical Assembly (PCA) Pressure Vessel Facility and the VENUS-3 vessel simulation benchmarks. They performed statistical analysis to determine the overall uncertainty and bias. Additionally, the team obtained measurements from samples removed from a BWR core shroud, top guide, and jet pump riser brace pads.

Results

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EPRI Perspective

Accurate neutron fluence determinations are required for a number of reasons: 1) to determine neutron fluence in the 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 overall calculational uncertainty is well within the uncertainty guidelines provided in Regulatory Guide 1.190.

Keywords

Fluence
Embrittlement
Boiling water reactor
Vessel and internals
Reactor pressure vessel

ABSTRACT

This document reports the overall calculational uncertainty associated with the application of the RAMA Fluence Methodology to BWR reactor pressure vessel fluence evaluations. The individual uncertainty components are described. Comparisons to measurements are presented for surveillance capsule activity specimens, along with comparisons to measurements obtained from other irradiated components including top guide, shroud, and jet pump riser brace pad samples.

ACKNOWLEDGMENTS

EPRI and the Boiling Water Reactor Vessel and Internals Project (BWRVIP) are acknowledged for their support of the RAMA Fluence Methodology project. Special thanks are extended to Robert Carter of EPRI for his guidance, comments, and overall support in completing this document.

The following TransWare personnel are acknowledged for their contributions to the programming and populating of the uncertainty database for this project: Eric Jones and Kevin Retzke.

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1 INTRODUCTION

The RAMA Fluence Methodology [1] (hereinafter referred to as RAMA) calculates activation and neutron fluence in boiling water reactor (BWR) components. RAMA includes a transport code, model builder codes, a fluence calculator code, an uncertainty methodology, and a nuclear data library. The transport code, fluence calculator, and nuclear data library are the primary software components for calculating the neutron flux and fluence.

RAMA has been approved by the U. S. Nuclear Regulatory Commission [2] for application in accordance with U. S. Regulatory Guide 1.190 [3]. Compliance with the provisions of the Regulatory Guide requires that RAMA be qualified using comparisons to plant-specific measurement data and industry benchmark problems. Data comparisons from several plant-specific surveillance capsule and simulator benchmark problems have been performed in order to qualify RAMA for use in RPV fluence evaluations for BWRs. Plant and benchmark data that have been evaluated with RAMA are presented in Table 1-1.

Table 1-1
Description of Reactors Used in Comparisons to Measurement Data

Reactor Type	Fuel Assembly Configuration	Jet Pumps	Reactor Name
BWR/2	560	0	Oyster Creek Nuclear Generating Station
BWR/4	548	20	Cooper Nuclear Station
BWR/4	560	20	James A. FitzPatrick Nuclear Power Station; & Edwin I. Hatch Nuclear Plant Units 1 and 2
BWR/4	764	20	Hope Creek Nuclear Generating Station; Peach Bottom Atomic Power Station Units 2 and 3; & Susquehanna Steam Electric Station Units 1 and 2
BWR/6	624	20	Clinton Power Station
Experimental			Pool Critical Assembly Pressure Vessel Facility Benchmark
Experimental			VENUS-3 Benchmark

This report documents the results of this qualification effort for the application of RAMA in BWR fluence evaluations. Included in this report is an assessment of the overall uncertainty of RAMA.

1.1 Implementation Requirements

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

2 SUMMARY AND CONCLUSIONS

RAMA has been used to evaluate eleven different BWRs with plant classes ranging from BWR/2s through BWR/6s, with the exception that no BWR/3 plant evaluation is included in the current BWR uncertainty assessment. A total of 416 measurement samples are included in the plant-specific capsule comparison evaluation obtained from 22 capsules. Table 2-1 summarizes the calculated-to-measured (C/M) ratio and standard deviation for various BWR class reactors.

Table 2-1
Measurement Comparisons by Reactor Class

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It is shown that there is no significant variation in the predictive capability of RAMA for the various BWR classes.

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3
DESCRIPTION OF THE REACTOR SYSTEMS

This section describes the reactor core and component configuration for the reactor systems used in the RAMA uncertainty evaluation.

3.1 Reactor System Mechanical Design Inputs

The reactor systems are modeled with RAMA. RAMA employs a three-dimensional modeling technique to describe the reactor geometry for the neutron transport calculations. Detailed mechanical design information is used in order to build an accurate three-dimensional computer model representation of each reactor system. Pertinent details of each reactor design used in this uncertainty evaluation are described in the following subsections.

3.1.1 BWR/2 with 560 Fuel Assembly Configuration

The Oyster Creek Nuclear Generating Station is a General Electric BWR/2 class reactor. The reactor core consists of 560 fuel assemblies with a rated thermal power of 1930 MWt. Note that BWR/2 class plants are pre-jet pump designs.

Figure 3-1 illustrates the planar view of the axial elevation at the core mid-plane. The figure shows the azimuthal positions of the surveillance capsules in the downcomer region at 30, 210, and 300 degrees. The surveillance capsules are positioned radially near the inner surface of the RPV wall. Three capsules were inserted at the beginning of reactor operation. One of these capsules was removed at the end of cycle 1. One was removed at the end of cycle 9 and analyzed. One of these original capsules is still in the reactor. Six special surveillance capsules were loaded at the beginning of cycle 14 at azimuth 210 degrees as part of the BWRVIP Supplemental Surveillance Program. Three of these capsules (capsules D, G, and H) were removed at the end of cycle 15 and samples were analyzed. The remaining three capsules (E, F, and I) were removed at the end of cycle 17 and samples were analyzed.

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Figure 3-1
Planar View of the BWR/2 - 560 Fuel Assembly Oyster Creek Reactor at the Core Mid-Plane Elevation

3.1.2 BWR/4 with 560 Fuel Assembly Configuration

The James A. FitzPatrick Nuclear Power Station and Edwin I. Hatch Nuclear Power Plant Units 1 and 2 are General Electric BWR/4 class reactors with core loadings of 560 fuel assemblies. The initial rated thermal power output was 2436 MWt for all units.

The FitzPatrick reactor was subsequently uprated to a rated power of 2536 MWt in cycle 13. Both Hatch reactors have been uprated three times to rated powers of 2558 MWt, 2763 MWt, and 2804 MWt. Hatch Unit 1 was uprated to these power levels in cycle 17, cycle 19, and cycle 22, respectively. Hatch Unit 2 was uprated to these power levels in cycle 13, cycle 15, and cycle 18, respectively.

Figure 3-2 illustrates the basic planar geometry configuration of the 560 fuel assembly BWR/4 reactors at the axial elevation corresponding to the core mid-plane. The figure shows the azimuthal positions of the surveillance capsules in the downcomer region at 30, 120, and 300 degrees and the jet pump assemblies at 30, 60, 90, 120, 150, 210, 240, 270, 300, and 330 degrees.

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**Figure 3-2
Planar View of Typical BWR/4 - 560 Fuel Assembly Reactors at the Core Mid-Plane Elevation**

3.1.3 BWR/4 with 548 Fuel Assembly Configuration

Cooper Nuclear Station is a General Electric BWR/4 class reactor with a core loading of 548 fuel assemblies. The core configuration is similar to the 560 fuel assembly BWR/4 class reactors except that 12 peripheral assemblies are replaced with dummy assemblies. The rated thermal power output of the reactor is 2381 MWt.

Figure 3-3 illustrates the basic planar geometry configuration of the 548 fuel assembly BWR/4 reactor at the axial elevation corresponding to the core mid-plane. The figure shows the azimuthal positions of the surveillance capsules in the downcomer region at 30, 120, and 300 degrees; the jet pump assemblies at 30, 60, 90, 120, 150, 210, 240, 270, 300, and 330 degrees; and the location of the dummy assemblies.

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Figure 3-3
Planar View of the BWR/4 - 548 Fuel Assembly Cooper Reactor at the Core Mid-Plane Elevation

3.1.4 BWR/4 with 764 Fuel Assembly Configuration

Hope Creek Nuclear Generating Station, Peach Bottom Atomic Power Station Units 2 & 3, and Susquehanna Steam Electric Station Units 1 & 2 are General Electric BWR/4 class reactors with core loadings of 764 fuel assemblies. The initial rated thermal power output was 3293 MWt for all units.

Both Peach Bottom reactors were subsequently uprated to a rated power of 3458 MWt in cycle 15 of Unit 2 and cycle 11 of Unit 3. Both Susquehanna reactors have also been uprated to a rated power of 3441 MWt in cycle 9 of Unit 1 and cycle 7 of Unit 2, followed by an additional uprate to 3489 MWt in cycle 13 of Unit 1 and cycle 11 of Unit 2.

Figure 3-4 illustrates the basic planar geometry configuration of the 764 fuel assembly BWR/4 reactors at the axial elevation corresponding to the core mid-plane. The figure shows the azimuthal positions of the surveillance capsules in the downcomer region at 30, 120, and 300 degrees and the jet pump assemblies at 30, 60, 90, 120, 150, 210, 240, 270, 300, and 330 degrees.

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Figure 3-4
Planar View of Typical BWR/4 - 764 Fuel Assembly Reactors at the Core Mid-Plane Elevation

3.1.5 BWR/6 with 624 Fuel Assembly Configuration

Clinton Power Station is a General Electric BWR/6 class reactor with a core loading of 624 fuel assemblies. The initial rated thermal power output of the reactor was 2894 MWt. A power up-rate was achieved in cycle 9, raising the rated thermal power to 3473 MWt.

Figure 3-5 illustrates the basic planar geometry configuration of the reactor at an axial elevation corresponding to the reactor core mid-plane. This figure shows the positioning of the surveillance capsules relative to the inside surface of the reactor pressure vessel wall. The azimuthal positions of the surveillance capsules in the downcomer region are at 3, 177, and 183 degrees and the jet pump assemblies at 30, 60, 90, 120, 150, 210, 240, 270, 300, and 330 degrees.

This reactor design differs from earlier BWR class reactors in that the surveillance capsules are positioned at the flats of the core edge, i.e., they are not shielded by the jet pumps.

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Figure 3-5
lanar View of the BWR/6 - 624 Fuel Assembly Clinton Reactor at the Core Mid-Plane Elevation

4 CALCULATION OF UNCERTAINTY AND BIAS

U. S. Nuclear Regulatory Guide 1.190 [3] provides the basis for determining the best-estimate neutron fluence to be used in estimating the impact of irradiation of the reactor pressure vessel. The fluence is determined by a calculational methodology that must be qualified by comparison of predicted activation to measured values obtained from plant samples and vessel simulation benchmarks.

As a part of the qualification, calculation-to-measurement (C/M) comparisons are used to identify biases (i.e., systematic prediction errors) in the calculations. Any statistically significant calculational biases, if present, are used to modify the calculated fluence by applying a correction to account for the biases.

The C/M comparisons are also used to obtain an estimate of the calculational uncertainty associated with the methodology. This estimate of the calculational uncertainty is referred to in this report as the "comparison uncertainty". In addition, an independent estimate of the calculational uncertainty is determined by evaluating the sensitivity of the calculated fluence to the uncertainty in the modeling input parameters, which is referred to in this report as the "analytic uncertainty". These independent estimates of fluence uncertainty are combined using appropriate weighting factors, as proposed in Regulatory Guide 1.190, to obtain an overall calculational uncertainty. The overall calculational uncertainty is used to assure that the methodology meets the uncertainty requirements of Regulatory Guide 1.190, which requires that acceptable methodologies must have an overall uncertainty (1σ) of 20% or less. The overall uncertainty also provides an explicit uncertainty on the fluence for those applications that require conservative rather than best-estimate fluence values, such as in probable risk assessment (PRA) evaluations.

This section documents the various components that comprise the calculational bias and uncertainty for RAMA RPV fluence evaluations.

4.1 RAMA Calculational Bias

The RAMA calculational bias is determined by comparing plant-specific predicted activity to measured values obtained from various BWRs. Section 5 describes the extent of RAMA comparisons to measurements that have been performed to date. A total of 11 BWRs are included in the comparison database. The comparisons are obtained from 416 samples obtained from 22 surveillance capsules.

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4.2 RAMA Calculational Uncertainty

Independent estimates of the RAMA calculational uncertainty are obtained from the comparison uncertainty (i.e., uncertainty in the comparisons to measurements) and the analytic uncertainty (i.e., the sensitivity of the fluence to uncertainties in modeling parameters and inputs). The overall calculational uncertainty is determined by combining the independent uncertainty estimates using appropriate weighting factors that reflect the applicability of the uncertainty estimate to the RAMA fluence evaluation. The following sections describe the determination of the RAMA calculational uncertainty.

4.2.1 Comparison Uncertainty

There are two components to the RAMA comparison uncertainty: the BWR plant-specific measurement comparisons and the simulation benchmark comparisons. Each of these components is treated separately in the determination of the RAMA comparison uncertainty.

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4.2.2 Analytic Uncertainty

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4.2.3 Overall Calculational Uncertainty

The overall calculational reactor pressure vessel uncertainty is the weighted sum of the plant-specific comparison uncertainty, the benchmark comparison uncertainty, and the analytic uncertainty. Table 4-1 shows the standard weighting factors used to determine the combined uncertainty from a RAMA uncertainty evaluation. Note that the combined uncertainty is presented as a range that reflects the typical variation in analytic uncertainty for the various BWRs.

Table 4-1
RAMA Overall Calculational Uncertainty

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5 SURVEILLANCE CAPSULE ACTIVATION MEASUREMENT COMPARISONS

This section documents the results of the comparison of RAMA predicted activation values to surveillance capsule activation measurements. RAMA evaluations have been performed for each of the plants included in the current uncertainty evaluation. As a part of the fluence evaluations, the predicted activations (i.e., specific activities) generated by RAMA were compared to the activation measurements for the capsule flux wires and dosimetry of each plant. The comparisons for each plant are combined to determine the overall comparison bias and uncertainty for BWR applications of RAMA.

5.1 Plant-Specific Surveillance Capsule Comparisons

RAMA has been used to evaluate eleven different BWRs with plant types ranging from BWR/2s through BWR/6s, with the exception that no BWR/3 plant evaluation is included in the current BWR uncertainty assessment. BWR/5s are implicitly included in the BWR/4 reactor class because the later BWR/4s included features of the BWR/5s. The only notable distinction between the BWR/4s and BWR/5s is the BWR/5s have a higher-rated core flow.

A total of 416 measurement samples are included in the plant-specific capsule comparison evaluation obtained from 22 capsules. Table 5-1 summarizes the RAMA comparison statistics by BWR reactor class. Table 5-1 confirms that there is no significant variation in the RAMA prediction capability for the various BWR reactor classes. Sections 5.1.1 through 5.1.5 provide detailed plant comparisons of the RAMA predictions to the capsule activation measurements for each reactor class.

Table 5-1
Measurement Comparison by Reactor Class

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5.1.1 BWR/2 560 Fuel Assembly Configuration Surveillance Capsule Comparisons

The RAMA fluence evaluation for the Oyster Creek RPV [4] provides comparison data for the BWR/2 class of reactors. One reactor surveillance capsule was removed at the end of cycle 9 after being irradiated for a total of 8.1 effective full power years (EFPY). BWRVIP Supplemental Surveillance Program (SSP) capsules D, G, and H were irradiated for two cycles for a total of 3.1 EFPY. SSP capsules E, F, and I were irradiated for four cycles for a total of 6.6 EFPY. Table 5-2 summarizes the comparison of RAMA predicted activation to the Oyster Creek capsule measurements.

Table 5-2
Measurement Comparison for Oyster Creek Surveillance Capsules

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5.1.2 BWR/4 560 Fuel Assembly Configuration Surveillance Capsule Comparisons

The RAMA fluence evaluations for the James A. FitzPatrick RPV [5], the Edwin I. Hatch Unit 1 RPV [6], and the Edwin I. Hatch Unit 2 RPV [7] provide comparison data for the BWR/4 class of reactors with core loadings of 560 fuel assemblies. Comparison results from the fluence evaluations of each of these reactor systems are provided in this section.

Two surveillance capsule activation analyses were performed for the FitzPatrick reactor. Surveillance capsule flux wires were removed and analyzed at the end of cycle 6 and at the end of cycle 12. The cycle 6 flux wires were irradiated for 6.0 EFPY. The cycle 12 flux wires were irradiated for 13.4 EFPY. Table 5-3 summarizes the comparison of RAMA predicted activation to the FitzPatrick capsule measurements.

Table 5-3
Measurement Comparison for FitzPatrick Surveillance Capsules

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Table 5-4
Measurement Comparison for Hatch Unit 1 Surveillance Capsules

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One surveillance capsule activation analysis has been performed for the Hatch Unit 2 reactor. Surveillance capsule flux wires were removed and analyzed at the end of cycle 8. These flux wires were irradiated for 6.6 EFPY. Table 5-5 summarizes the comparison of RAMA predicted activation to the Hatch Unit 2 capsule measurements.

Table 5-5
Measurement Comparison for Hatch Unit 2 Surveillance Capsule

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5.1.3 BWR/4 548 Fuel Assembly Configuration Surveillance Capsule Comparisons

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Table 5-6
Measurement Comparison for Cooper Surveillance Capsules

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5.1.4 BWR/4 764 Fuel Assembly Configuration Surveillance Capsule Comparisons

The RAMA capsule and fluence evaluations for Hope Creek Nuclear Generating Station [10], Peach Bottom Atomic Power Station Unit 2 [11], Peach Bottom Atomic Power Station Unit 3 [12], Susquehanna Steam Electric Station Unit 1 [13], and Susquehanna Steam Electric Station Unit 2 [14] provide comparison data for the BWR/4 class of reactors with core loadings of 764 fuel assemblies. Comparison results from the fluence evaluations of each of these reactor systems are provided in this section.

The Hope Creek flux wire dosimeter was removed at the end of cycle 1 with an accumulated irradiation of 1.0 EFPY. Table 5-7 summarizes the comparison of RAMA predicted activation to the Hope Creek flux wire measurements.

Table 5-7
Measurement Comparison for Hope Creek Flux Wire Dosimeter

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One surveillance capsule activation analysis has been performed for the Peach Bottom Unit 2 reactor. Surveillance capsule flux wires were removed and analyzed at the end of cycle 7. These flux wires were irradiated for 7.5 EFPY. Table 5-8 summarizes the comparison of RAMA predicted activation to the Peach Bottom Unit 2 capsule measurements.

Table 5-8
Measurement Comparison for Peach Bottom Unit 2 Surveillance Capsule

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One surveillance capsule activation analysis has been performed for the Peach Bottom Unit 3 reactor. Surveillance capsule flux wires were removed and analyzed at the end of cycle 7. These flux wires were irradiated for 7.6 EFPY. Table 5-9 summarizes the comparison of RAMA predicted activation to the Peach Bottom Unit 3 capsule measurements.

Table 5-9
Measurement Comparison for Peach Bottom Unit 3 Surveillance Capsule

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Two surveillance capsule activation analyses were performed for the Susquehanna Unit 1 reactor. Surveillance capsule flux wires were removed and analyzed at the end of cycles 1 and 6. These flux wires were irradiated for 1.4 EFPY and 6.7 EFPY, respectively. Table 5-10 summarizes the comparison of RAMA predicted activation to the Susquehanna Unit 1 capsule measurements.

Table 5-10
Measurement Comparison for Susquehanna Unit 1 Surveillance Capsules

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One surveillance capsule activation analysis has been performed for the Susquehanna Unit 2 reactor. Surveillance capsule flux wires were removed and analyzed at the end of cycle 5. These flux wires were irradiated for 6.2 EFPY. Table 5-11 summarizes the comparison of RAMA predicted activation to the Susquehanna Unit 2 capsule measurements.

Table 5-11
Measurement Comparison for Susquehanna Unit 2 Surveillance Capsule

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5.1.5 BWR/6 624 Fuel Assembly Configuration Surveillance Capsule Comparisons

The RAMA fluence evaluation for the Clinton Power Station RPV [15] provides comparison data for the BWR/6 class of reactors. One surveillance capsule activation analysis has been performed for the Clinton reactor. Surveillance capsule flux wires were removed and analyzed at the end of cycle 1. These flux wires were irradiated for 0.99 EFPY. Table 5-12 summarizes the comparison of RAMA predicted activation to the Clinton capsule measurements.

Table 5-12
Measurement Comparison for Clinton Surveillance Capsule

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6 COMPARISON TO OTHER MEASUREMENTS

This section documents the results of the comparison of RAMA predicted activation values to activation measurements from vessel simulation benchmarks, a BWR numerical vessel benchmark, a PWR vessel benchmark, shroud and top guide samples from a BWR/4 reactor with a core loading of 764 fuel assemblies, and jet pump riser brace pad samples from a BWR/4 reactor with a core loading of 560 fuel assemblies.

6.1 Comparison of Predicted Activation to Vessel Simulation Benchmark Measurements

In accordance with the guidelines provided in Regulatory Guide 1.190 [3], and as specified in the RAMA theory and procedures manuals [16] and [17], it is appropriate to include comparisons of vessel simulation benchmark measurements in the overall fluence uncertainty evaluation whenever a statistically significant set of plant-specific comparison data is not available. The Pool Critical Assembly (PCA) Pressure Vessel Facility and the VENUS-3 experimental benchmarks have been evaluated using RAMA [18]. The PCA experimental benchmark includes 27 activation measurements at the mid-plane elevation in various simulated reactor components. The VENUS-3 experimental benchmark includes 386 activation measurements at a range of elevations in various simulated reactor components. Table 6-1 summarizes the results obtained from the application of RAMA to the vessel simulation benchmarks.

Table 6-1
Summary of Comparisons to Vessel Simulation Benchmark Measurements

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6.2 Comparison to other Vessel Benchmark Measurements

In addition to the vessel simulation benchmark evaluations, RAMA has been used in the performance of two other fluence evaluations. These evaluations include: the BWR Pressure Vessel Numerical Benchmark (documented in [18]), and the H. B. Robinson Unit 2 Pressure Vessel Benchmark (documented in [18]). While the results of these other benchmarks do not contribute to the uncertainty evaluation, they do provide further confirmation that RAMA accurately predicts surveillance capsule and vessel neutron flux distributions. A summary of the results of these other benchmarks is provided in the following paragraphs.

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6.3 Comparison to Core Shroud and Top Guide Measurements

Core shroud and top guide samples were removed from the Susquehanna Unit 2 reactor, a BWR/4 reactor with a core loading of 764 fuel assemblies, after eleven cycles of operation (15.3 EFPY). Details of the sample locations and the RAMA activation and fluence evaluation on these samples are provided in [19]. Three shroud samples and three top guide samples were evaluated with RAMA. A summary of the comparisons to specific activity measurements for these samples is provided in Table 6-2.

Table 6-2
Average Activation Results for Susquehanna Unit 2 Core Shroud and Top Guide Samples

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6.4 Comparison to Jet Pump Riser Brace Pad Measurements

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A
RAMA COMPARISON TO MEASUREMENT DATA

This appendix contains all of the comparisons to measured data for every surveillance capsule, core shroud, top guide, and jet pump riser brace pad measurement uncertainty that has been evaluated using RAMA. The comparison data is presented in subsections that correspond to the summary presentations provided in the body of the report. Thus, the surveillance capsule data is presented by BWR reactor class, followed by vessel simulator benchmarks, other vessel benchmarks, core shroud, top guide and jet pump riser brace pad data.

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