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Three-dimensional RAMA Fluence Methodology Benchmarking

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Benchmarking of the RAMA Fluence Methodology software, that has been performed in accordance with U. S. Nuclear Regulatory Commission Regulatory Guide 1.190, is described. The RAMA Fluence Methodology has been developed by TransWare Enterprises Inc. through funding provided by the Electric Power Research Institute, Inc. (EPRI) and the Boiling Water Reactor Vessel and Internals Project (BWRVIP). The purpose of the software is to provide an accurate method for calculating neutron fluence in BWR pressure vessels and internal components. The Methodology incorporates a three-dimensional deterministic transport solution with flexible arbitrary geometry representation of reactor system components, previously available only with Monte Carlo solution techniques. Benchmarking was performed on measurements obtained from three standard benchmark problems which include the Pool Criticality Assembly (PCA), VENUS-3, and H. B. Robinson Unit 2 benchmarks, and on flux wire measurements obtained from two BWR nuclear plants. The calculated to measured (C/M) ratios range from 0.93 to 1.04 demonstrating the accuracy of the RAMA Fluence Methodology in predicting neutron flux, fluence, and dosimetry activation.

1. Introduction

Accurate fluence calculations are becoming more important as nuclear plants age. These calculations are required for the following reasons:
1) to determine neutron fluence in the reactor pressure vessel 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.

Because some of the components that need to be evaluated are located above and below the active fuel zone, a three-dimensional analysis methodology is required to obtain an accurate flux solution. The RAMA Fluence Methodology [1] has been developed by TransWare Enterprises Inc. through funding provided by the Electric Power Research Institute, Inc. (EPRI) and the Boiling Water Reactor Vessel and Internals Project (BWRVIP). The Methodology is intended to facilitate the building of complex reactor geometry models, perform accurate three-dimensional flux calculations,

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calculate neutron activation, predict fluence, and apply uncertainty analysis to the fluence evaluation. The benchmarking that has been performed as part of the RAMA Fluence Methodology qualification in accordance with U. S. Nuclear Regulatory Guide 1.190 [2] is described in this paper.

2. Methodology Description

The RAMA Fluence Methodology includes a parts model builder code for use in generating reactor geometry models, a state-point model builder code for processing reactor operating data, a neutron transport code for calculating neutron flux, a code module for calculating activations and fluences, a method for calculating uncertainties, and a nuclear data library. The Methodology uses a deterministic, three-dimensional, multigroup nuclear particle transport theory code, RAMA, to perform neutron flux calculations. RAMA couples the nuclear transport method with a general geometry modeling capability to provide a flexible and accurate tool for determining fluxes for any light water reactor design. The code uses the method of characteristics solution technique, a three-dimensional ray-tracing method based on combinatorial geometry, a fixed source iterative solution with anisotropic scattering, a thermal-group upscattering treatment, and a nuclear cross-section data library based upon the ENDF/B-VI data file.

The Methodology has been developed to utilize the dimensional, material and operational history data that is available from the utilities. The two model builder codes are provided to assist the user in building the complex geometries of the components found in BWR reactors. The first code is the Parts Model Builder code that generates the reactor geometry model. The inputs have been structured to accept the dimensions typically found on the reactor vessel and internal design drawings found at the utilities. The second code is the State-point Model Builder code that processes materials for the geometry model. The inputs for the core, such as fuel isotopics, power distribution and density distribution, are obtained directly from the 3-D simulator codes used by the utilities and vendors for reload design and licensing activities. The model builder codes work together to efficiently build the geometry and material inputs for the RAMA transport calculation.

The RAMA Fluence Methodology geometry modeling system is modular and is based on combinatorial geometry techniques. This allows the user to build each part or region (e.g., core, reflector, 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 approach allows the user to easily replace a region of one meshing description with another region using a different meshing description. Rebuilding the entire geometry model every time a part description changes, such as when fuel designs change between operating cycles, or when performing sensitivity calculations, is avoided with this feature. The combinatorial geometry technique also allows the user to build the exact geometry. The fuel is modeled with rectangular elements while the shroud, downcomer, jet pumps, and vessel are modeled with cylindrical elements in the solution geometry.

After the inputs have been defined, the flux calculation is performed using the three-dimensional RAMA transport code, which is based on the deterministic Method of Characteristics transport theory. The RAMA transport code includes special treatments for flexible geometry, geometry ray-tracing, variable angular quadrature sets, anisotropic scattering, and vacuum and reflective boundary conditions. The neutron source is calculated internally within the RAMA transport code from the three-dimensional power distribution and the fuel isotopic concentrations obtained from 3-D core simulator codes. The RAMA nuclear data library is based on the 47 neutron group BUGLE-96 cross section library [3]. It includes P_7 Legendre scattering cross sections for the lighter nuclides such as hydrogen and oxygen, and P_5 scattering for the heavy nuclides.

The calculated fluxes from the RAMA transport code and the detailed operating history data for the reactor are used in the RAFTER post-processing code to compute component fluences. RAFTER also calculates the dosimetry activities for the various activation reactions. These can then be compared to the measurement data.

3. Benchmarking

Benchmarks have been performed as part of the RAMA Fluence Methodology qualification in accordance with Regulatory Guide 1.190. Three standard benchmarks have been performed: the Pool Criticality Assembly (PCA) [4], VENUS-3 [5], and H. B. Robinson Unit 2 [6] benchmarks. In addition, two surveillance capsule evaluations were performed for two BWR nuclear plants, Susquehanna Unit 2 and Hope Creek.

The input parameters for each benchmark problem were determined by conducting extensive sensitivity analyses. These analyses are described below in the discussions of each benchmark problem. Because the benchmarking solved a wide range of problems with different geometry and material specifications, the final input parameters were not the same for all the benchmark problems and were dependent on the geometry being modeled. As an example, there was a range in the maximum planar width between adjacent parallel rays in the ray tracing calculation of RAMA of 0.5 cm to 1.0 cm. The maximum axial height between adjacent parallel rays ranged from 1.0 cm to 10.0 cm for all the benchmark problems. Some of the input parameters were established because an input that defined a more detailed model would increase the run time or computer resource utilization to an unacceptable value. For example, the final benchmark calculations utilized an S8 angular quadrature set because a higher angular quadrature set produced a prohibitively long run time for production calculations.

Table 1 summarizes the comparison of the RAMA Fluence Methodology calculated to measured (C/M) results for the various benchmarks.

Table 1 Summary of RAMA Fluence Methodology comparison of calculated to measured (C/M) results

Benchmark	Average C/M	Standard Deviation
PCA	0.99	0.05
VENUS-3	1.03	0.05
H. B. Robinson 2 - Surveillance Capsule	0.95	0.04
H. B. Robinson 2 - Cavity	1.04	0.04
Susquehanna Unit 2	0.98	0.08
Hope Creek	0.93	0.04

3.1 Pool Criticality Assembly Pressure Vessel Facility Benchmark

The Pool Criticality Assembly (PCA) Pressure Vessel Facility Benchmark consists of the PCA reactor loaded with plate fuel elements and the ex-core components that are used to simulate pressure vessel surveillance configurations in light water reactors. The ex-core components include a thermal shield, a simulated pressure vessel, and a void box that simulates the reactor cavity. A planar view of the PCA facility is shown in Figure 1 and an axial view is shown in Figure 2. The RAMA model for the PCA benchmark takes advantage of half-core symmetry so that only one half of the core is explicitly modeled. The total height of the PCA model is 88.900 cm. The active core region is 60.008 cm high and is axially centered in the model. The active core region contains 19 axial nodes of varying height. The regions above and below the height of the active core are 14.446 cm high and each contains eight axial nodes. The planar power distribution and the axial cosine profile, as reported in [4], are combined to form a three-dimensional power distribution for the modeled PCA core regions. The resulting core neutron source is normalized to a value of one core fission neutron per second in the PCA core in order to coincide with the normalization of the experimental results.

Measured reaction rates are reported for six dosimeter reactions: ^{237}Np(n,fission)^{137}Cs, ^{238}U(n,fission)^{137}Cs, ^{115}In(n,n′)^{115m}In, ^{58}Ni(n,p)^{58}Co, ^{27}Al(n,α)^{24}Na, and ^{103}Rh(n,n′)^{103m}Rh. The detectors are distributed throughout the PCA geometry at the core mid-plane in seven locations to provide spatial and spectral variations. As shown in Figure 1, the detectors identified by circles reside directly in front of the thermal shield, directly behind the thermal shield, directly in front of the simulated pressure vessel, approximately 1/4T, 1/2T and 3/4T in the pressure vessel, and directly behind the pressure vessel in the void box region. All reported dosimeter measurements except the ^{103}Rh(n,n′)^{103m}Rh are included in the evaluation of the PCA benchmark. These ^{103}Rh(n,n′)^{103m}Rh measurements are excluded from the evaluation due to suspected erroneous BUGLE-96 response cross section data for the ^{103}Rh(n,n′)^{103m}Rh reaction [7]. The measured dosimeter reaction rates are expressed as equivalent fission neutron flux. The equivalent fission neutron flux utilizes an equivalent ^{235}U fission spectrum dosimeter cross section defined as:

$$\sigma_{eq} = \sum_{g=1}^{\#energygroups} \sigma_{R_g} \chi_g \qquad (1)$$

where σ_{R_g} is the energy group dependent activation response cross section and χ_g is the energy group dependent ^{235}U fission production spectrum. The equivalent ^{235}U fission spectrum flux is then defined as:

$$\phi_{eq} = \frac{\sum_{g=1}^{\#energygroups} \sigma_{R_g} \phi_g}{\sigma_{eq}} \qquad (2)$$

where ϕ_g is the energy group dependent neutron flux.

Sensitivity analyses were performed to evaluate the stability and accuracy of RAMA for the PCA benchmark reference case with respect to mesh size and solution parameters. The parameter variations that were studied included mesh size in the X direction parallel to the core, mesh size in the Y direction perpendicular to the core, planar distance between parallel rays, axial distance between parallel rays, and maximum Legendre scattering order. The variation in the mesh in the Y direction perpendicular to the core face is the most sensitive parameter in the transport calculation. The transport calculation is observed in Figures 3 through 6 to systematically approach an asymptotic result as the mesh size is decreased. In addition, the ^{237}Np predicted detector response is especially sensitive to meshing in the water regions, which illustrates the significance of its lower energy threshold relative to the other dosimeter reactions.

The average C/M comparison for the 27 dosimeters positioned at the core mid-plane in the ex-core components is 0.99 with a standard deviation of $\pm$ 0.05. The average C/M comparison for each of the dosimeter types ^{237}Np, ^{238}U, ^{115}In, ^{58}Ni, and ^{27}Al is 1.00 $\pm$ 0.06, 0.95 $\pm$ 0.02, 1.02 $\pm$ 0.05, 1.02 $\pm$ 0.04, 0.97 $\pm$ 0.03 and 0.99 $\pm$ 0.05 respectively.

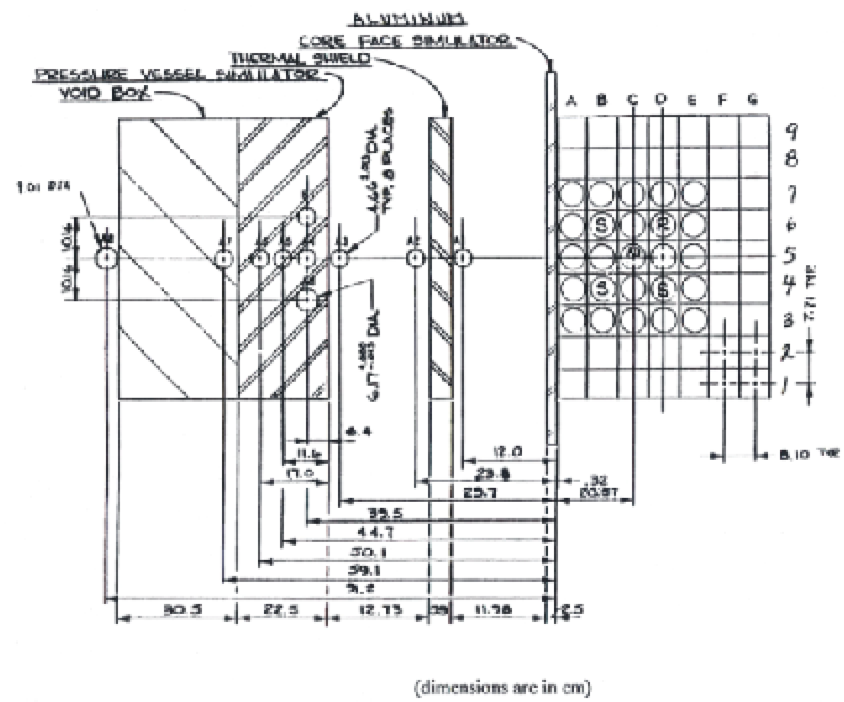

Fig. 1 Planar view of pool criticality assembly (PCA) pressure vessel facility

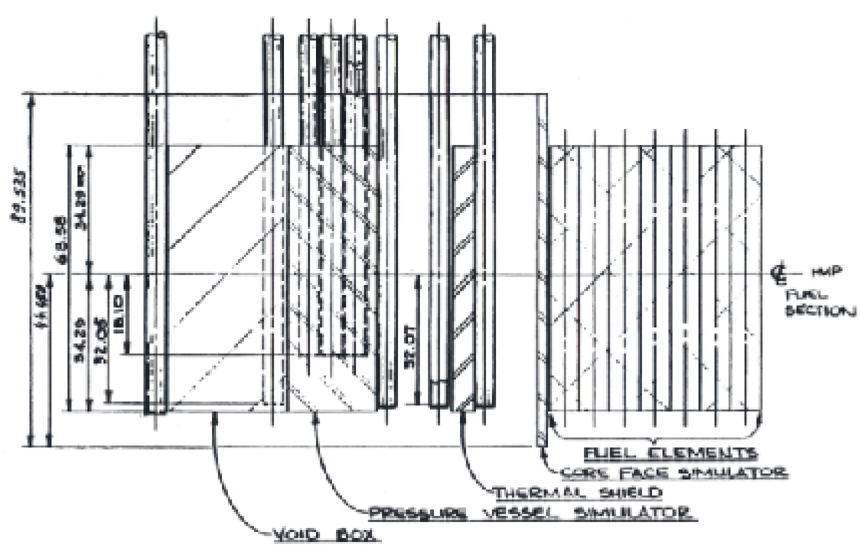

Fig. 2 Elevation view of pool criticality assembly (PCA) pressure vessel facility

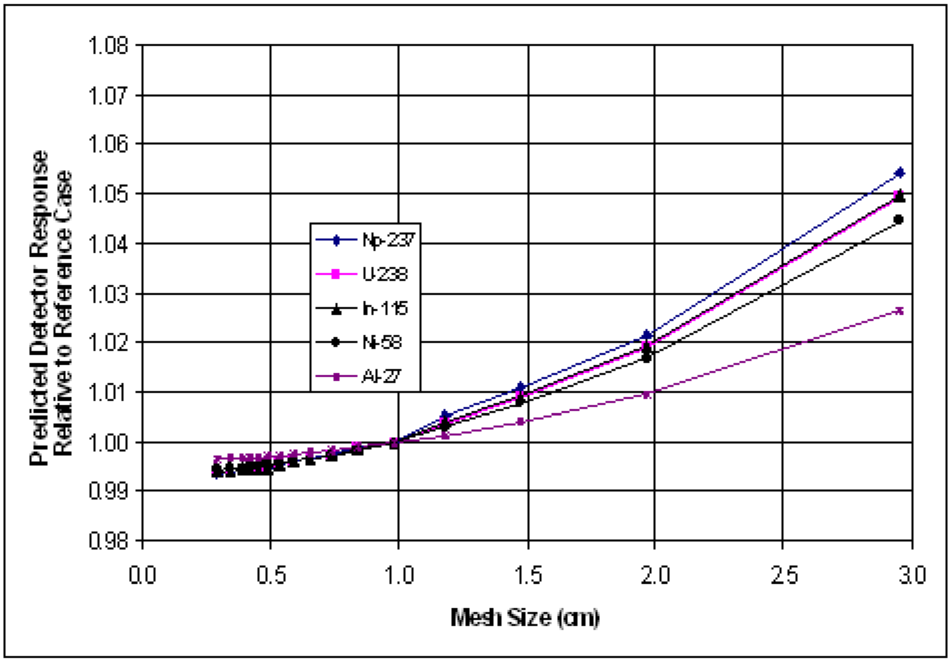

Fig. 3 Thermal shield mesh sensitivity in direction perpendicular to core face

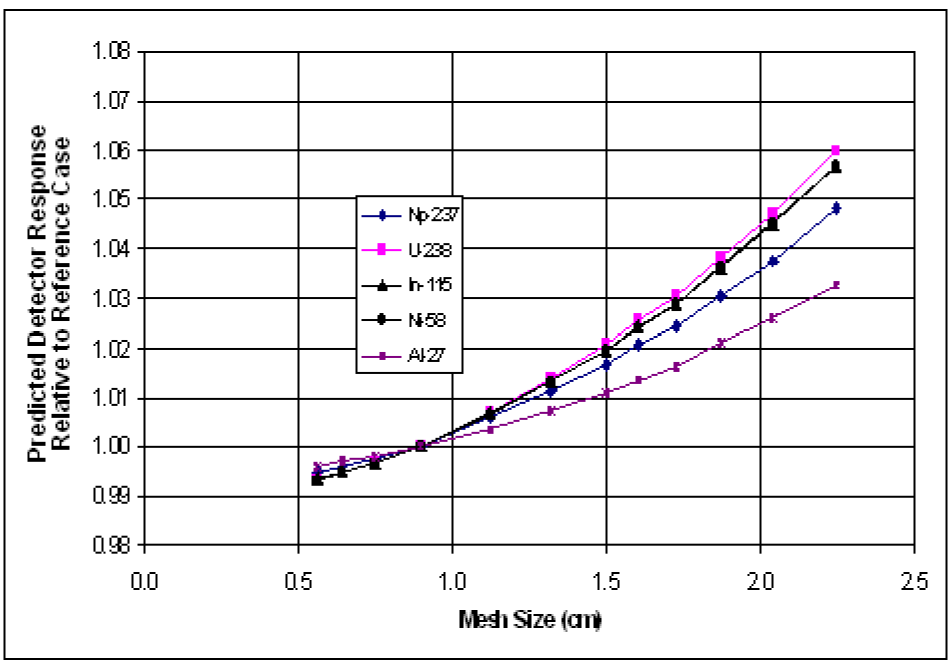

Fig. 4 Pressure vessel mesh sensitivity in direction perpendicular to core face

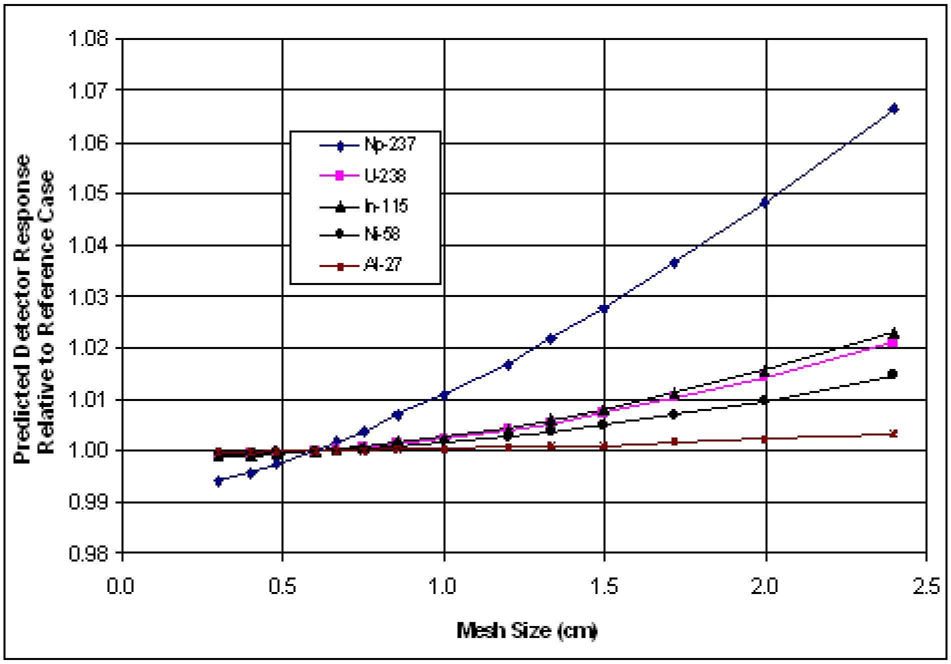

Fig. 5 Water gap between aluminum window and thermal shield mesh sensitivity in direction perpendicular to core face

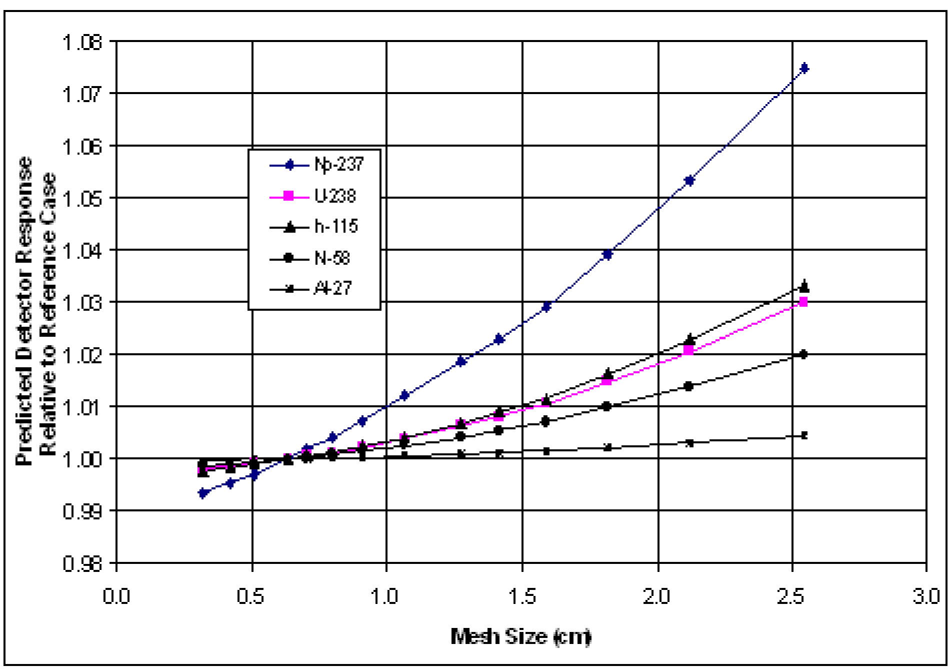

Fig. 6 Water gap between thermal shield and pressure vessel mesh sensitivity in direction perpendicular to core face

3.2 VENUS-3 Benchmark

The VENUS-3 benchmark is prescribed by the U. S. Nuclear Regulatory Commission for use in benchmarking pressure vessel neutron fluence prediction methodologies. The dimensions and material compositions required to perform the VENUS-3 benchmark were obtained from [5]. The VENUS-3 benchmark consists of the VENUS reactor and the ex-core components that are used to simulate pressure vessel surveillance configurations in light water reactors. A planar view of the VENUS facility is shown in Figure 7, and an axial view is shown in Figure 8. The VENUS-3 core is composed of 12 simulated 15x15 PWR fuel assemblies with a pin pitch of 1.26 cm. There are a total of 2,548 fuel pins, 52 non-fuel pins, and a 10x10 pin area at the center of the core that is occupied by a core water hole and the inner baffle. Four types of pins are loaded in the core: fuel 3/0 (3.3 wt. % ^{235}U), fuel 4/0 (4.0 wt. % ^{235}U), pyrex pins, and partial length shielded assembly (PLSA) pins. The pyrex pins simulate PWR control rod clusters. The PLSA pins consist of fuel 3/0 fuel pins above core mid-plane and stainless steel rods below core mid-plane that simulate a PWR partially shielded peripheral assembly. The active core height is 50 cm. Below the active core are a lower reflector region, a bottom grid, the bottom support region, and a lower filling. Above the active core are an intermediate grid, an upper reflector, an upper grid, and an upper filling. Between the core hole and the core is an inner stainless steel baffle. Outside the core is an outer stainless steel baffle that is typical of a PWR core baffle. A water reflector region occupies the space between the outer baffle and the barrel. Beyond the barrel is a stainless steel neutron pad, the air-filled jacket, and the stainless steel pressure vessel.

The VENUS-3 benchmark provides measured reaction rates at thirty locations and fourteen elevations distributed inside the core, in the inner and outer baffles, in the reflector region between the outer baffle and barrel, and in the barrel. Measured reaction rates are reported for three dosimeter reactions: ^{58}Ni(n,p)^{58}Co, ^{115}In(n,n′)^{115m}In, and ^{27}Al(n,α)^{24}Na. The detectors are distributed throughout the VENUS-3 geometry to provide spatial and spectral variations.

The RAMA model for the VENUS-3 benchmark consists of the quadrant of the core corresponding to the detector locations. Axially, VENUS-3 is modeled from 88.0 cm to 168.0 cm for a total height of 80.0 cm. The active core region is 50.0 cm high, and was modeled with 14 axial nodes of varying height ranging from 3.0 to 4.0 cm. The regions above the height of the active core are 13 cm high and contain 5 axial nodes. The regions below the height of the active core are 17 cm high and contain 5 axial nodes. Radially, the VENUS-3 model extends from the center of the core to the outside surface of the pressure vessel (84.455 cm). The components from the barrel to the vessel are modeled as cylindrical regions with centers that are coincident with the center of the core. The fuel lattice is represented in the X-Y plane of the model as square homogenized pin cell regions. The neutron fission source is normalized as specified in [5] in order to coincide with the normalization of the measurement results.

Sensitivity analyses were performed to evaluate the stability and accuracy of RAMA for the VENUS-3 benchmark reference case with respect to mesh size and solution parameters. The parameter variations that were studied included planar mesh size, planar distance between parallel rays, axial distance between parallel rays, convergence criterion, angular quadrature set, and maximum Legendre scattering order. The maximum absolute deviation in the predicted detector response relative to a reference case was less than or equal to 1% for all of the sensitivity cases, except for the variation in angular quadrature set. An increase in the angular quadrature from S8 to S16 produced a deviation in results of approximately 5%. The transport calculation was observed to systematically approach an asymptotic result for each parameter evaluated in the sensitivity analyses as seen with the PCA benchmark (see figures 3-6 as examples).

The average C/M ratio for all measured reaction rates is 1.03 with a standard deviation of ±0.05. The average C/M ratio for each of the dosimeter types ^{58}Ni, ^{115}In, and ^{27}Al is 1.04 $\pm$ 0.05, 1.03 $\pm$ 0.03, and 0.99 $\pm$ 0.04 respectively.

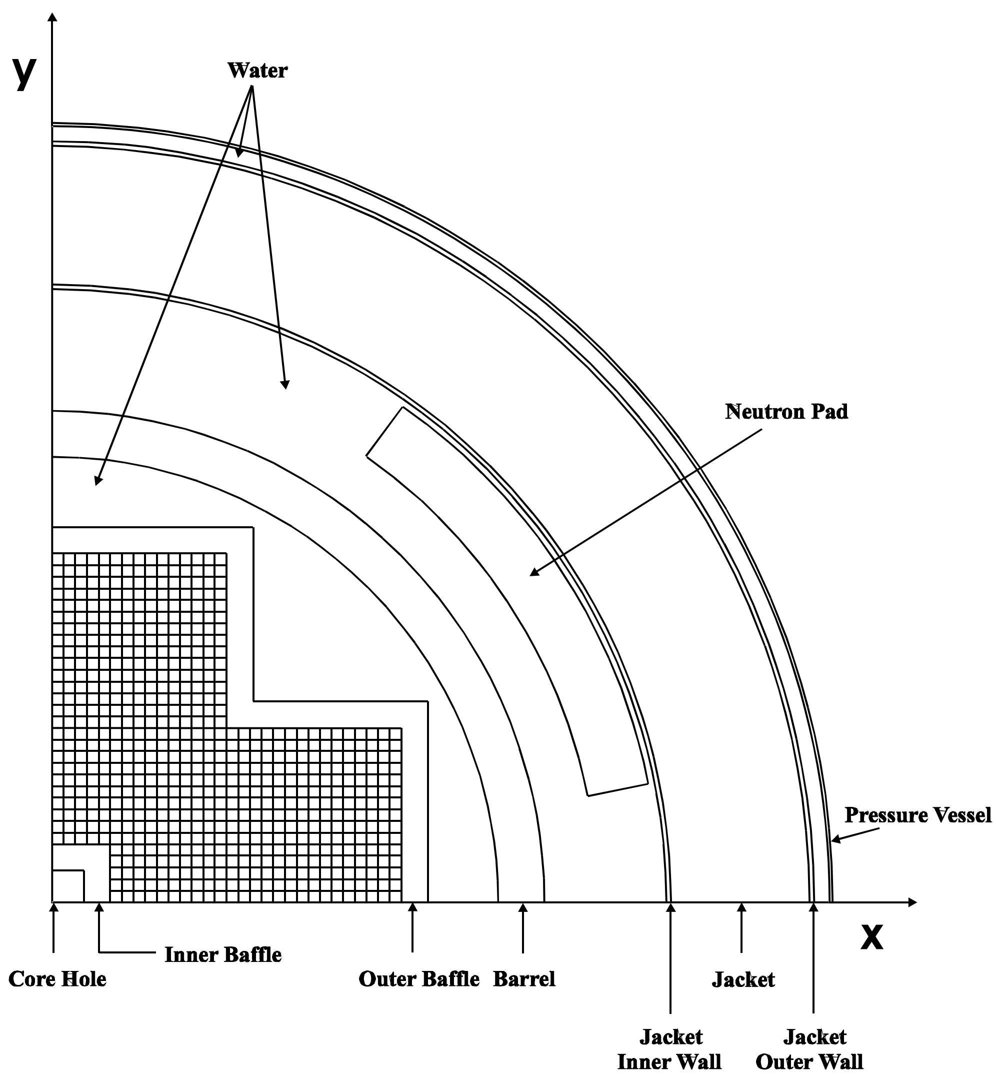

Fig. 7 Planar view of the VENUS-3 benchmark facility

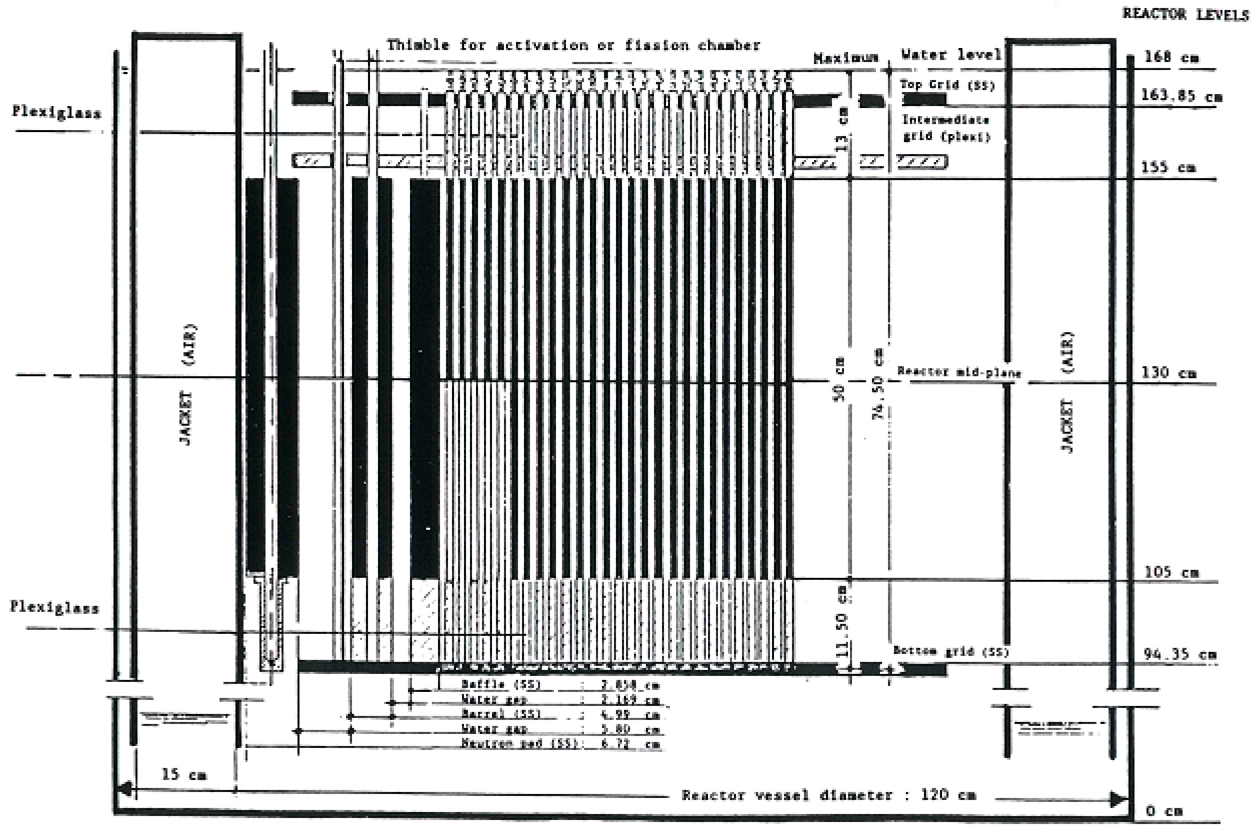

Fig. 8 Elevation view of the VENUS-3 benchmark facility

3.3 H. B. Robinson Unit 2 Benchmark

The H. B. Robinson Unit 2 (HBR-2) benchmark was performed at the 2300 MWt PWR on dosimeters in the surveillance capsule location inside the pressure vessel and in the cavity outside the pressure vessel. The dimensions and material compositions required to perform the HBR-2 benchmark were obtained from [6]. Measured reaction rates are reported for six dosimeter reactions: ^{237}Np(n,f)^{137}Cs, ^{238}U(n,f)^{137}Cs, ^{58}Ni(n,p)^{58}Co, ^{54}Fe(n,p)^{54}Mn, ^{46}Ti(n,p)^{46}Sc, and ^{63}Cu(n,α)^{60}Co. The surveillance capsule is at the azimuthal angle of 20˚ and the cavity dosimeters are located at the azimuthal angle of 0˚ in the model. The dosimeters were irradiated during operating cycle 9 only.

The RAMA model for the HBR-2 plant assumes octant symmetry and is described over azimuths 0˚ to 45˚as shown in Figure 9. The active core region contains 12 axial nodes each having a height of 30.48 cm. Only the four center planes (i.e., planes 5, 6, 7 and 8) from -60.96 cm to 60.96 cm for a total height of 121.92 cm are modeled. Radially, the HBR-2 model extends from the center of the core to the outside surface of the biological shield (340.0 cm). The components from the core barrel to the biological shield are modeled as cylindrical regions with centers that are coincident with the center of the core. The fuel peripheral assemblies are represented in the X-Y plane of the model as a 15 by 15 array of homogenized pin cells. Fuel assemblies one row inward from the periphery are sub-meshed into a 5 by 5 array at equal-volume regions. The center assemblies are fully homogenized (see Figure 9).

The power distribution provided in [6] and uranium and plutonium number densities are used to calculate the source in the reactor core. The uranium and plutonium number densities for the fuel assemblies are generated using the initial average fuel enrichment and the fuel exposure in [6]. Both material and power data are provided for eight state points for operating cycle 9 as well as a cycle 9 average data set.

Sensitivity analyses were performed to evaluate the stability and accuracy of RAMA for the HBR-2 benchmark case with respect to mesh size and solution parameters. The parameter variations that were studied included planar mesh size, planar distance between parallel rays, axial distance between parallel rays, convergence criterion, angular quadrature set, maximum Legendre scattering order, the depth to which the rays penetrate the reflective boundary region, and the height of the model. The maximum absolute deviation in the predicted detector response relative to a reference case was less than or equal to 1% for all of the sensitivity cases, except for the variation in angular quadrature set. An increase in the angular quadrature from S8 to S24 produced a deviation in results of approximately 5% in the capsule region and 8% in the cavity region. The transport calculation was observed to systematically approach an asymptotic result for each parameter evaluated in the sensitivity analyses as seen with the PCA benchmark (see figures 3-6 as examples).

The dosimeters were irradiated during cycle 9 operation only. Calculations were performed using the eight state-point operating data sets provided for cycle 9 as well as the cycle 9 average operating data set. Using the eight state-point operating data, the average C/M results for all the dosimeters in the surveillance capsule and for the cavity are 0.95 and 1.04 respectively, with a standard deviation of ±0.04 for both regions. Using the cycle 9 average operating data set, the average C/M result for all the dosimeters in the surveillance capsule is 0.98 with a standard deviation of ±0.06 and for the cavity dosimeters is 1.05 with a standard deviation of ±0.07.

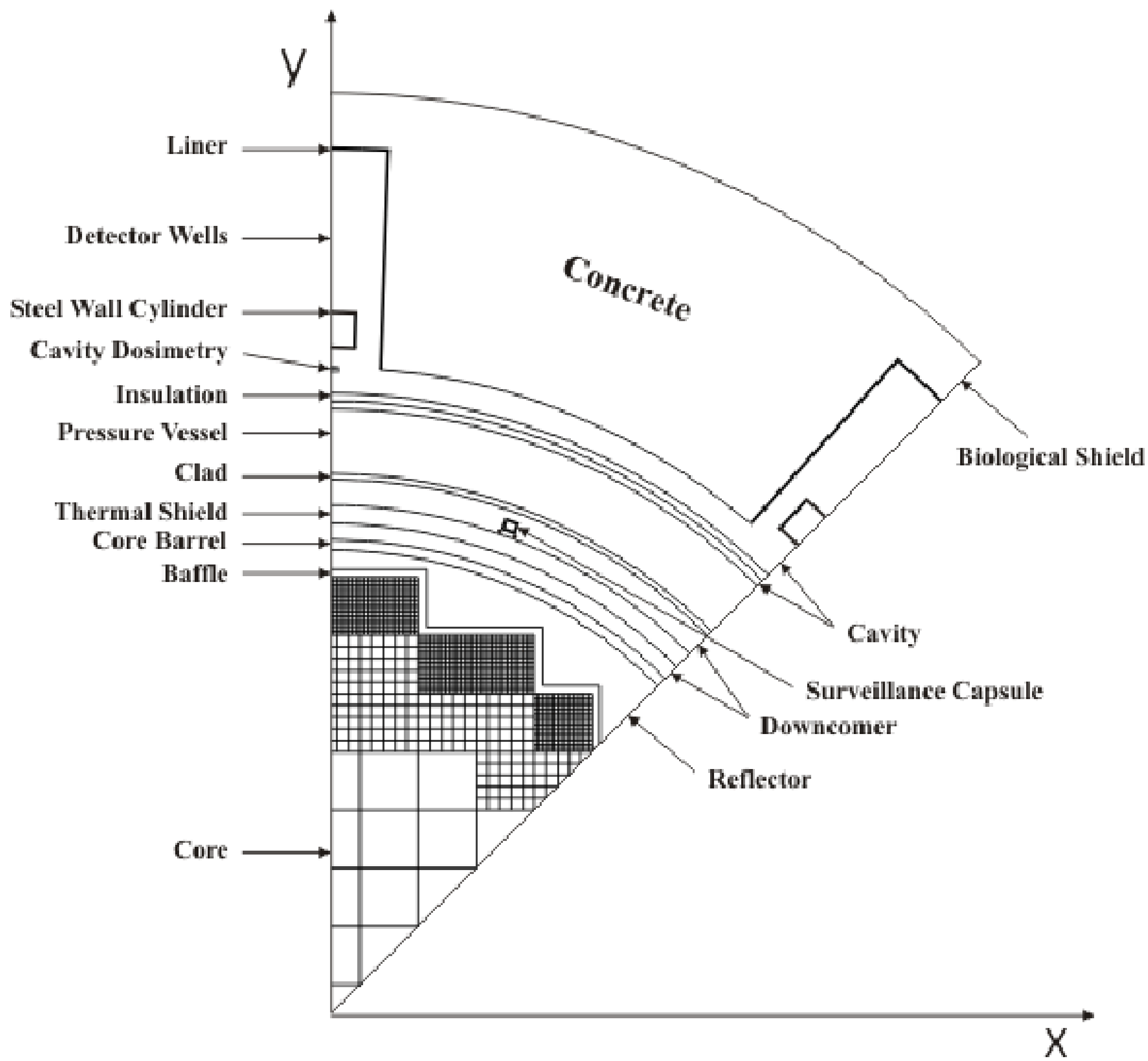

Fig. 9 Planar view of the H. B. Robinson unit 2 reactor

3.4 Susquehanna Unit 2 and Hope Creek Capsule Benchmark

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

The RAMA model for the Susquehanna Unit 2 and Hope Creek plants assumes octant symmetry and is described over azimuths 0˚ to 45˚. There were nine radial regions defined in the model: the fuel region, core reflector, shroud, downcomer with jet pumps, pressure vessel, inner and outer cavities, mirror insulation, and biological shield. The reactor core region is modeled with rectangular elements to preserve the rectangular shape of the core region. The cylindrical components and regions outside the core region are modeled with cylindrical elements to preserve their correct geometric shape. The jet pump assembly design in the downcomer region is properly modeled using cylindrical pipe elements for the jet pump riser and mixer pipes.

Each region of the reactor is comprised of materials that include reactor fuel, steel, water, insulation, and air. Accurate material information is essential for the fluence evaluation as the material compositions determine the scattering and absorption of neutrons throughout the reactor system and, thus, affect the determination of neutron fluence in the reactor components. The attributes for the steel, insulation, and air compositions (i.e., material densities and isotopic concentrations) are assumed to remain constant for the operating life of the reactor. The attributes for the water compositions in the ex-core regions will vary with the operation of the reactor, but are generally represented at nominal hot operating conditions and assumed to be constant for an operating cycle. The attributes of the fuel compositions in the reactor core region change continuously during an operating cycle due to changes in power level, fuel burnup, control rod movements, and changing moderator density levels (voids). Because of the dynamics of the fuel attributes with reactor operation, one to several data sets describing the operating state of the reactor core are required for each operating cycle.

Numerous sensitivity analyses were performed to evaluate the stability and accuracy of RAMA for the BWR/4 benchmark with respect to mesh size and solution parameters. The parameter variations that were studied included planar mesh size, planar distance between parallel rays, axial distance between parallel rays, convergence criterion, angular quadrature set, maximum Legendre scattering order, the depth to which the rays penetrate the reflective boundary, and the height of the model. The maximum absolute deviation in the >1.0 MeV capsule flux relative to a reference model was less than or equal to 1% for all of the sensitivity cases, except for the variation in the planar mesh size and the variation in angular quadrature set. The mesh size was reduced to one half the reference model mesh size and resulted in a change in the >1.0 MeV flux at the capsule location of approximately 3.5%. An increase in the angular quadrature from S8 to S24 produced a deviation in results of approximately 7%. The transport calculation was observed to systematically approach an asymptotic result for each parameter evaluated in the sensitivity analyses as seen with the PCA benchmark (see figures 3-6 as examples).

The Susquehanna Unit 2 surveillance capsule was removed at the end of cycle 5 after being irradiated for a total of 6.22 effective full power years (EFPY). The Hope Creek flux wire holder was removed at the end of cycle 1 after being irradiated for a total of 1.03 EFPY. Each cycle for both reactors was modeled with multiple state-points in RAMA and detailed power histories in RAFTER.

The average C/M ratio for the Susquehanna Unit 2 capsule flux wire evaluation is 0.98 with a standard deviation of ±0.08.The average C/M ratio for the Hope Creek flux wire evaluation is 0.93 with a standard deviation of ±0.04.

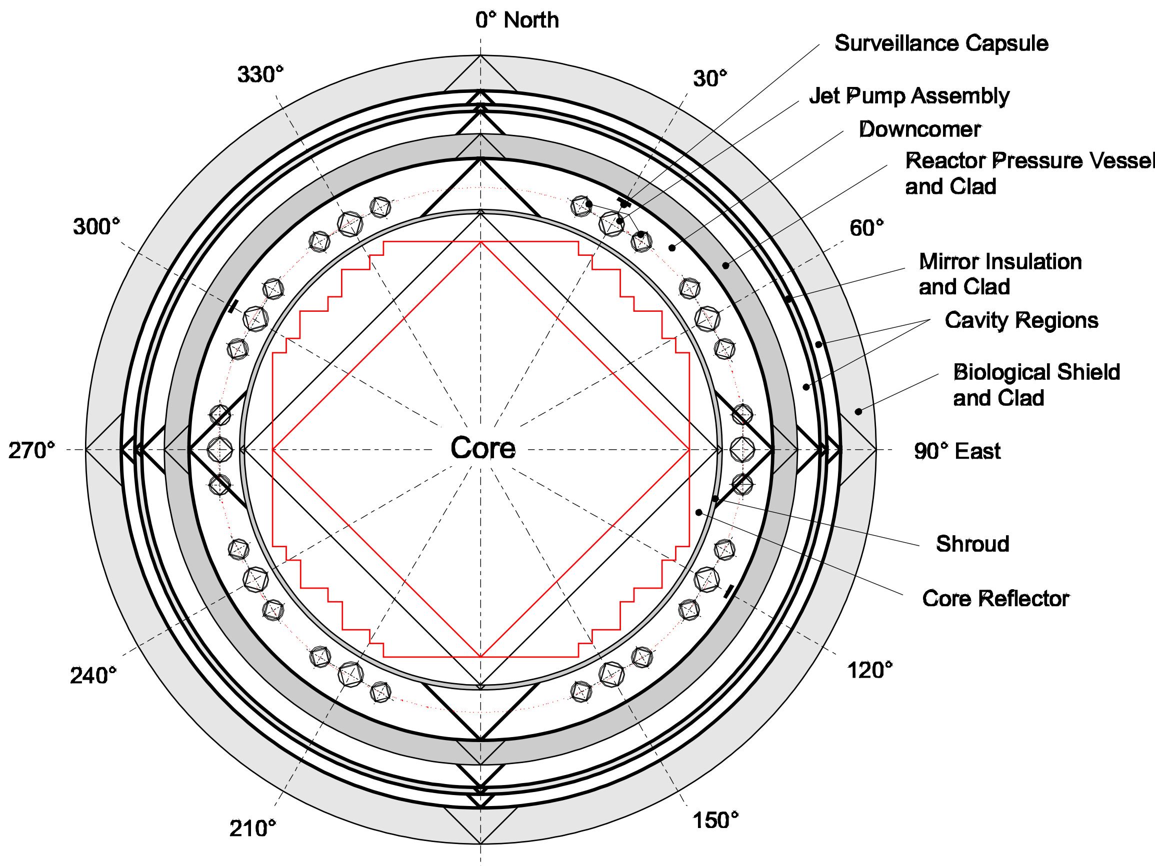

Fig. 10 Planar view of the BWR/4 reactor vessel and internals

4. Conclusions

Table 1 shows a summary of the C/M comparison for the benchmarks reported in this paper. The C/M ratios are in very good agreement indicating the RAMA Fluence Methodology is accurately predicting neutron flux, dosimetry measurements, and component fluence. The sensitivity analyses performed for each benchmark demonstrate that the RAMA Fluence Methodology results are stable with respect to the important mesh and solution parameters used in the neutron transport calculation.

The majority of the benchmarking presented in this paper has been performed against measured data obtained from core mid-plane elevations. Additional benchmarking of the RAMA Fluence Methodology is planned for measurements obtained from the top guide and shroud components of a commercial BWR. The measurements in these components are away from the core mid-plane and are in closer proximity to the reactor core. These studies will further qualify the RAMA Fluence Methodology for fluence evaluations.

Acknowledgments

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