FLUENCE EVALUATION FOR OYSTER CREEK REACTOR PRESSURE VESSEL (NON-PROPRIETARY)

TransWare Document Number: **EXL-FLU-001-R-005**
Revision 0

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March 2006

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ACKNOWLEDGMENTS

Mr. Greg Harttraft of AmerGen Energy Company, LLC and Sinan Taner and Michael May of Exelon Generation Company, LLC are acknowledged for assisting in the collection of the operating history, fuels data and the mechanical design data for this evaluation. TransWare Enterprises Inc. personnel who deserve special recognition for their efforts in the performance of this evaluation include Steven Baker for assisting the Oyster Creek Nuclear Generating Station personnel in the collection and formatting of the operating history data; and Charles Schwalbach for his efforts in developing and testing the RAMA geometry model used in this evaluation.

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1
INTRODUCTION

This report presents the results of the reactor pressure vessel (RPV) fluence evaluation performed for the Oyster Creek Nuclear Generating Station (hereinafter referred to as the "Reactor") using the RAMA Fluence Methodology. Projected fluence values are presented for the end of the Reactor's design lifetime of 32 effective full power years (EFPY) and for its extended design lifetime of 50 EFPY. Fluence values are determined for the RPV shell and weld locations specified by AmerGen Energy Co., LLC. This evaluation was performed in accordance with guidelines presented in U. S. Nuclear Regulatory Guide 1.190 [1].

This evaluation also includes the prediction of specific activities for flux wires that were irradiated in the Reactor surveillance capsules from the start of plant operation through the end of cycle 17. Activation measurements were conducted on the flux wires and impact testing was performed on the Charpy specimens extracted from the surveillance capsules [2-4]. Specific activities predicted by the RAMA Fluence Methodology are compared to the activity measurements.

The RAMA Fluence Methodology (hereinafter referred to as the Methodology) has been developed for the Electric Power Research Institute, Inc. (EPRI) and the Boiling Water Reactor Vessel and Internals Project (BWRVIP) for the purpose of calculating neutron fluence in Boiling Water Reactor (BWR) components. The Methodology includes a transport code, model builder codes, a fluence calculator code, an uncertainty methodology, and a nuclear data library. The transport code, fluence calculator, and nuclear data library are the primary software components for calculating the neutron flux and fluence. The transport code uses a deterministic, three-dimensional, multigroup nuclear particle transport theory to perform the neutron flux calculations. The transport code couples the nuclear transport method with a general geometry modeling capability to provide a flexible and accurate tool for calculating fluxes in light water reactors. The fluence calculator uses reactor operating history information with isotopic production and decay data to estimate activation and fluence in the reactor components over the operating life of the reactor. The nuclear data library contains nuclear cross-section data and response functions that are needed in the flux, fluence, and reaction rate calculations. The cross sections and response functions are based on the BUGLE-96 nuclear data library [5]. The Methodology and procedures for its use are described in the following reports: Theory Manual [6] and Procedures Manual [7].

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

2
SUMMARY AND CONCLUSIONS

This section provides a summary of the results of the Oyster Creek reactor pressure vessel fluence evaluation through the projected end of the normal operating life of 32 EFPY and through 50 EFPY. Detailed tables of all RPV fluence results are presented in Section 7 of this report. The primary purpose of this evaluation is to determine the RPV fluence for energy >1.0 MeV. Fluence is also predicted for selected welds located in the beltline region.

Table 2-1 summarizes the peak fluence values generated from this evaluation for energy >1.0 MeV at 32 EFPY and 50 EFPY. One value represents the peak fluence for the weld locations and the other represents the value at the shell locations. The peak fluence for the weld locations is in vertical weld 2-564A at 50 EFPY with a value of 6.16E+18 n/cm^2 for energy >1.0 MeV. The peak fluence for the RPV shells is in the lower intermediate shell at 50 EFPY with a value of 6.97E+18 n/cm^2 for energy >1.0 MeV. [[ ]]

Table 2-1
Peak >1.0 MeV Neutron Fluence for RPV Weld and Shell Locations

Weld/Shell Location	Peak Fluence for 32 EFPY (n/cm^2)	Peak Fluence for 50 EFPY (n/cm^2)
Weld 2-564A	4.12E+18	6.16E+18
Lower Intermediate Shell	4.66E+18	6.97E+18

In addition to the prediction of RPV fluence, specific activities were predicted for the iron and copper flux wires in seven Reactor surveillance capsules. Three tables (Tables 2-2 through 2-4) are provided that summarize the activation comparison results for all the capsules for flux wire measurements, dosimetry measurements and for all measurements combined, respectively. Note that the Methodology provides a direct solution of the reactions, i.e., no multiplicative or other adjustment is made to the results. Additional details of all surveillance capsule activation results, by capsule, are presented in Section 5 of this report.

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In conclusion, the RAMA Fluence Methodology produces accurate results that compare very well with measured data. The Methodology for determining the neutron fluence for the RPV shell and weld locations has been performed in accordance with the guidelines presented in Regulatory Guide 1.190.

3
DESCRIPTION OF THE REACTOR SYSTEM

This section describes the design inputs for the Reactor that were used in the reactor pressure vessel (RPV) fluence evaluation. The basic design inputs include mechanical design drawings, material compositions, and reactor operating history. Mechanical design drawings and structural material data were provided by AmerGen Energy Co., LLC and were used to build the Oyster Creek RAMA geometry model. These data were used in the Oyster Creek surveillance capsule evaluation [11] as well as in this project. Detailed operating history data was provided for this project by AmerGen Energy Co., LLC [12].

3.1 Reactor System Mechanical Design Inputs

The Oyster Creek Nuclear Generating Station (Reactor) employs a boiling water reactor (BWR) nuclear steam supply system. The Reactor is a General Electric BWR/2 class reactor located in Forked River, New Jersey. 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.

Figures 3-1 depicts a planar view of an axial elevation near the core mid-plane. Except for the surveillance capsules that are positioned at the core mid-plane, this geometry is uniform along the full height of the core. All radial regions comprising the fluence model are illustrated. Beginning at the center of the reactor and projecting outwards, the regions include: the core region, including control rod locations and fuel assembly locations (fuel locations are shown only for the northeast quadrant); core reflector region (bypass water); central shroud wall; downcomer water region; reactor pressure vessel (RPV) wall; mirror insulation; biological shield (concrete wall); and cavity regions between the RPV and biological shield.

Also shown in Figure 3-1 are 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 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 (E, F, and I) were removed at the end of cycle 17 and samples analyzed.

Repair tie rods for the core shroud were introduced into the Reactor during reload 15. The tie rods are located in the downcomer region in the close proximity of the outer surface of the shroud wall. The tie rods are not illustrated in Figure 3-1.

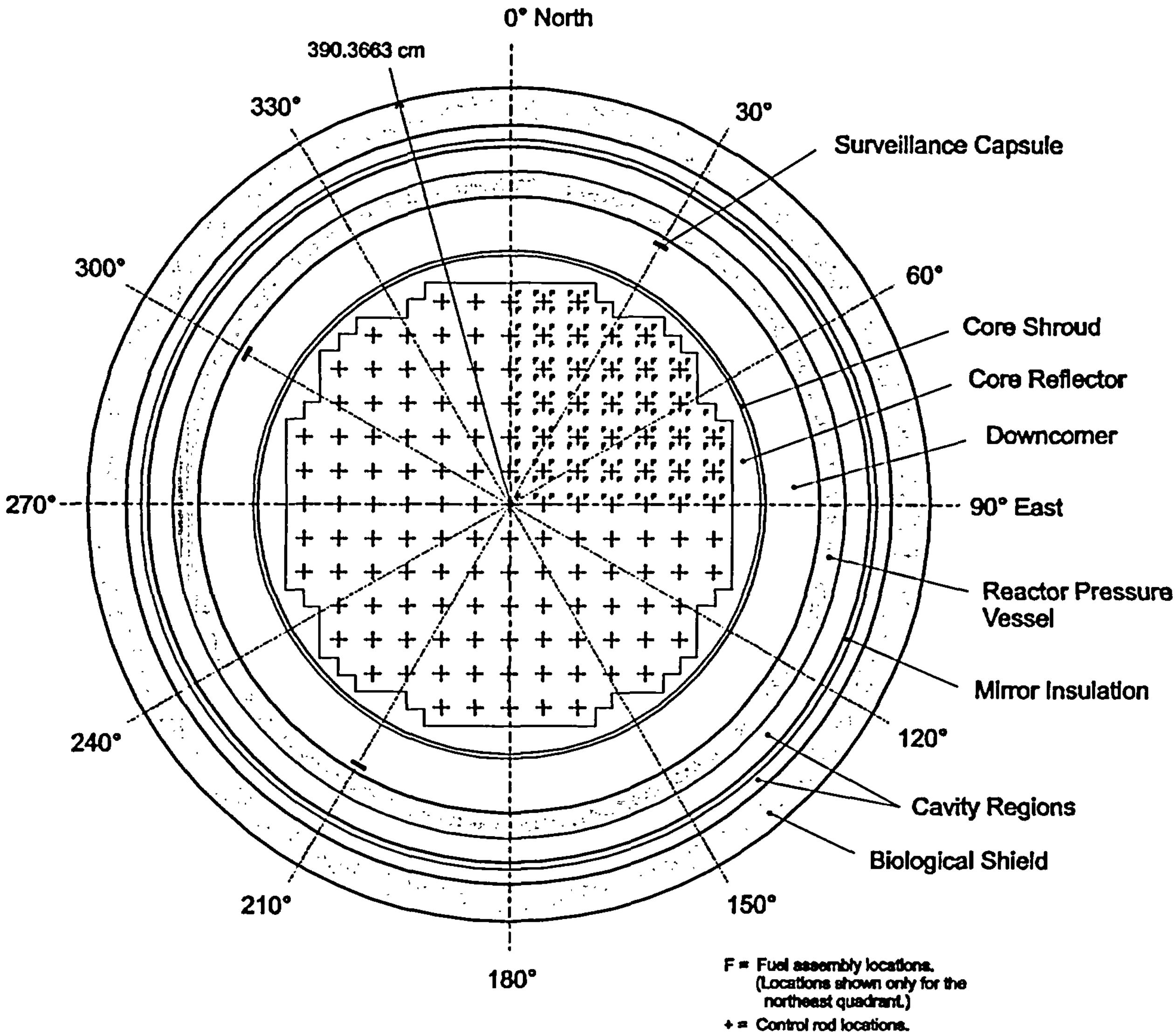

Figure 3-1
Planar View of the Oyster Creek Reactor at the Core Mid-plane Elevation

3.2 Reactor System Material Compositions

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

Table 3-1 provides a summary of the material compositions in the various components and regions of the Reactor. The attributes for the steel, insulation, and air compositions (i.e., material densities and isotopic concentrations) are assumed to remain constant for the operating life of the reactor. The attributes for the ex-core water compositions will vary with the operation of the reactor, but are generally represented at nominal hot operating conditions and are assumed to be constant throughout an operating cycle. The attributes of the fuel compositions in the reactor core region change continuously during an operating cycle due to changes in fuel burnup, power level, 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 used for each operating cycle. The number of data sets used in this analysis is presented in Section 3.3.2.

Table 3-1
Summary of Material Compositions by Region for Oyster Creek

Region	Material Composition
Reactor Core	^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, O_{fuel}, Zr, Water
Core Reflector	Water
Fuel Support Piece	Stainless Steel SS-304
Lower Tie Plate	Stainless Steel SS-304, ZR-2, Inconel-X
Top Guide	Stainless Steel SS-304L
Upper Tie Plate	Stainless Steel SS-304, ZR-2, Inconel-X
Shroud	Stainless Steel SS-304L
Downcomer Region	Water
Surveillance Capsule	Carbon Steel
Reactor Pressure Vessel Clad	Stainless Steel SS-304
Reactor Pressure Vessel Wall	Carbon Steel SA-302B
Cavity Regions	Air (Oxygen)
Insulation	Stainless Steel SS-304
Biological Shield Clad	Carbon Steel
Biological Shield	Reinforced Concrete

3.3 Reactor Operating Data Inputs

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

3.3.1 Power History Data

The reactor power history used in the Reactor fluence evaluation was obtained from daily power history edits provided by AmerGen Energy Co., LLC for the eighteen operating cycles being evaluated [12]. The daily power values represent step changes in power on a daily basis and the power is assumed to be representative of the power over the entire day. The fluence evaluation for the Reactor considered the complete daily operating history of the Reactor from cycles 1 through 18. Also accounted for in the analysis are the shutdown periods. The shutdowns were primarily due to the refueling outages between cycles.

3.3.2 Reactor State Point Data

Reactor operating data for the reactor pressure vessel fluence evaluation was provided as state point data files by AmerGen Energy Co., LLC [12]. The state point files provide a best-available representation of the operating conditions of the unit over the operating lifetime of the Reactor. The data files include three-dimensional data arrays that describe the fuel materials, moderator materials, and relative power distribution in the core.

A total of [[ ]] state point data files are used to represent the first eighteen operating cycles of the Reactor. These data files represent the operating states of the Reactor during these cycles. Table 3-2 shows the number of state point data files for each cycle used in this fluence evaluation. [[

]] The rated thermal power output of the Reactor is specified as 1930 MWt.

A separate neutron transport calculation was performed for each state point. The calculated neutron flux for each state point was combined with the appropriate power history data described in Section 3.3.1 to predict the neutron fluence in the reactor pressure vessel.

Table 3-2
Number of State-point Data Files for Each Cycle in Oyster Creek

Cycle Number	Number of State Point Data Files	Rated Thermal Power (MWt)
1	[[	1930
2		1930
3		1930
4		1930
5		1930
6		1930
7		1930
8		1930
9		1930
10		1930
11		1930
12		1930
13		1930
14		1930
15		1930
16		1930
17		1930
18		1930
19+	]]	1930

3.3.3 Core Loading Pattern

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

The Reactor used four different fuel assembly designs during cycles 1 through 18. Table 3-3 provides a summary of the fuel designs loaded in the Reactor core for these operating cycles. The cycle core loading patterns provided by AmerGen Energy Co., LLC were used to identify the fuel assembly designs in each cycle and their location in the core loading pattern. For each cycle, appropriate fuel assembly models were used to build the reactor core region of the RAMA fluence model for the Reactor.

Table 3-3
Summary of the Oyster Creek Core Loading Pattern

Cycle	General Electric (GE) 7x7 Fuel Assembly Designs	Exxon 7x7 Fuel Assembly Designs	Exxon 8x8 Fuel Assembly Designs	General Electric (GE) 8x8 Fuel Assembly Designs	Dominant Peripheral Fuel Design in the RAMA Model
1	X				GE 7x7
2	X	X			GE 7x7
3	X	X			GE 7x7
4	X	X			GE 7x7
5	X	X	X		Exxon 7x7
6	X	X	X		Exxon 7x7
7	X	X	X		Exxon 7x7
8		X	X		Exxon 7x7
9		X	X		Exxon 8x8
10			X	X	Exxon 8x8
11			X	X	Exxon 8x8
12			X	X	GE 8x8
13			X	X	GE 8x8
14				X	GE 8x8
15				X	GE 8x8
16				X	GE 8x8
17				X	GE 8x8
18				X	GE 8x8
19+				X	GE 8x8

4
CALCULATION METHODOLOGY

The Reactor pressure vessel fluence evaluation was performed using the RAMA Fluence Methodology software package. The Methodology and the application of the Methodology to the Reactor are described in this section.

4.1 Description of the RAMA Fluence Methodology

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

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

The primary outputs from the RAMA Fluence Methodology calculations are neutron flux, neutron fluence, and uncertainty determinations. The RAMA transport code calculates the neutron flux distributions that are used in the determination of neutron fluence. Several transport calculations are typically performed over the operating life of the reactor in order to calculate neutron flux distributions that accurately characterize the operating history of the reactor. The post-processing code (RAFTER) is then used to calculate component fluence and nuclide activations using the neutron flux solutions from the transport calculations and daily operating history data for the plant. The fluence calculated by RAFTER may then be adjusted in accordance with the calculational bias to determine the best estimate fluence and uncertainty in accordance with the intent of U. S. Nuclear Regulatory Guide 1.190.

4.2 The RAMA Geometry Model for Oyster Creek

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

Figure 4-1 illustrates the planar configuration of the Reactor model at an elevation near the core mid-plane elevation. In the radial dimension the model extends from the center of the reactor pressure vessel (RPV) to the outside surface of the biological shield (390.3663 cm). Nine radial regions are defined in the Reactor model: the core region, core reflector, shroud, downcomer, pressure vessel, mirror insulation, biological shield, and inner and outer cavity regions. The pressure vessel wall has cladding on the inner surface. The biological shield wall has cladding on the inner and outer surfaces. The azimuthal dimension assumes octant symmetry which spans from 0 to 45 degrees. Azimuth 0 degrees corresponds to the north compass direction that is specified in the Reactor design drawings. The surveillance capsule is located at azimuth 30 degrees. This also represents the location for the 210 and 300 degree capsules, assuming octant symmetry (see Figure 3-1). The core region shown in Figure 4-1 is meshed into sub-regions that correspond to the fuel assembly locations. [[

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]] It is expected that the peak neutron fluence in the RPV wall will occur in the beltline elevations. For the analysis presented in this report, the RAMA fluence model spans beyond the beltline elevations in order to provide appropriate boundary conditions for performing the neutron transport calculations. [[

]] The elevation parameters are relative to reactor elevation 0 as defined by the reactor vendor. (Elevation 0 corresponds to the inside surface of the pressure vessel wall at the bottom drain plug location.)

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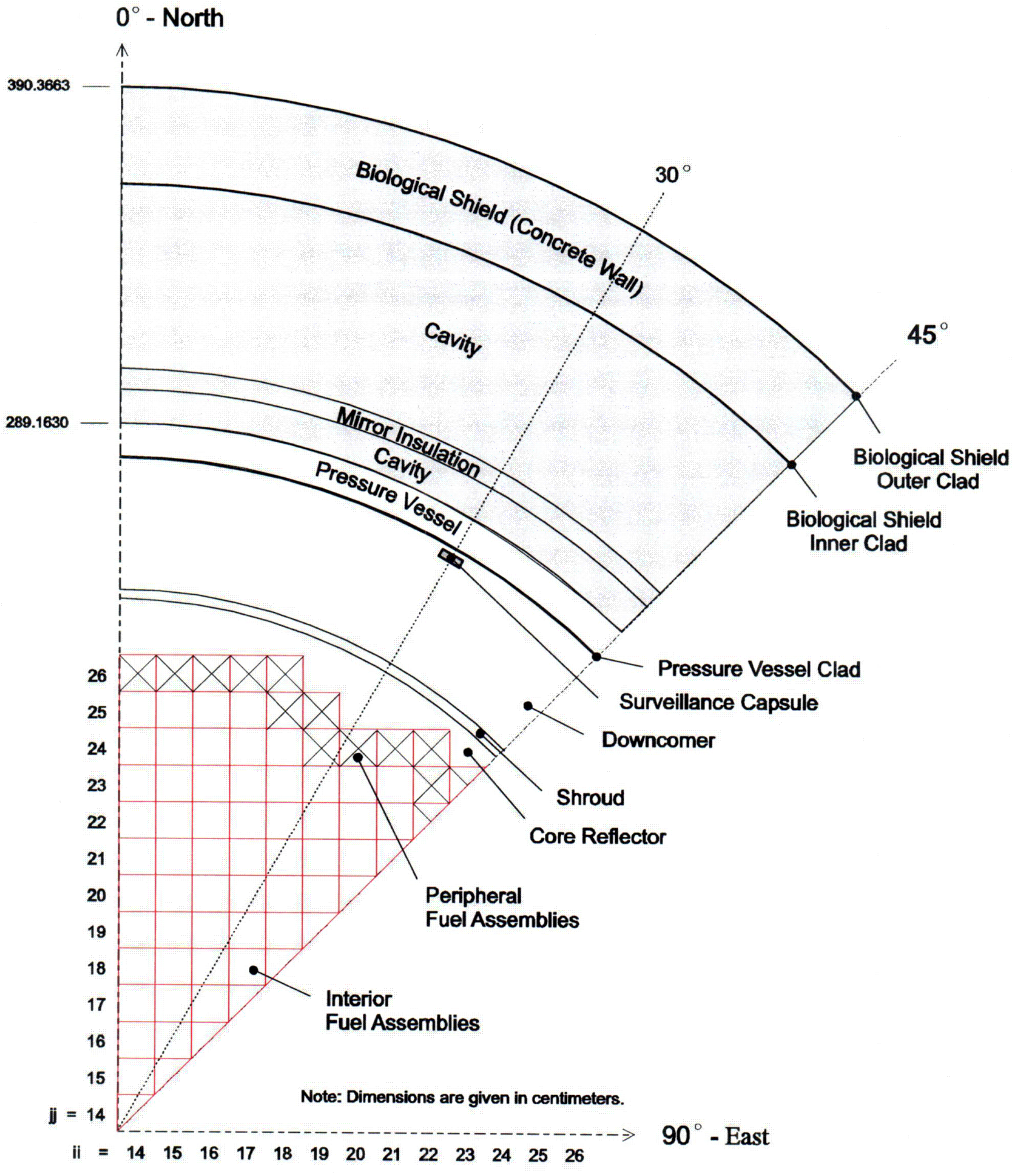

Figure 4-1
Planar View of the Oyster Creek Components and Regions at the Core Mid-plane Elevation

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As the primary interest in this fluence evaluation is the determination of the neutron fluence at specified RPV vertical and horizontal welds, Figure 4-3 identifies these specific weld locations. These weld locations are referenced in the tables in Sections 2 and 7 of this report showing the RPV weld fluence results by their identification numbers shown in Figure 4-3.

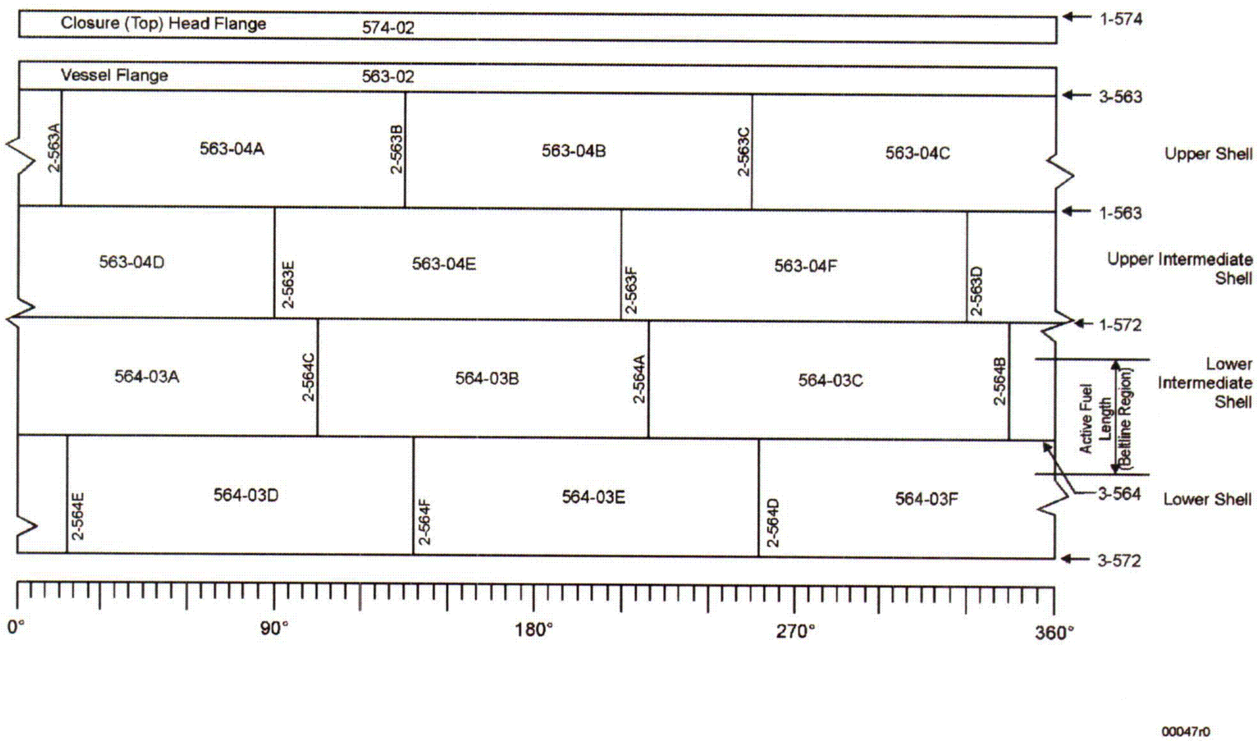

Figure 4-3
Oyster Creek Reactor Pressure Vessel Shell Plates and Weld Location Identifiers

There are several key features of the Reactor design that can be more accurately represented in the RAMA fluence model than in other deterministic methods. Three of these design features are described below.

The reactor core region is modeled to preserve the rectangular shape of the core region. This is illustrated in Figure 4-1. The core region model is characterized in two layers: the interior fuel assemblies and the peripheral fuel assemblies. The peripheral fuel assemblies are the primary contributors to the neutron source in the fluence calculation. The peripheral fuel assemblies are, therefore, modeled to preserve the pin-wise source contribution to the neutron fluence determination.

The surveillance capsule is positioned at azimuth 30 degrees in the model. [[

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[[ ]] Downcomer water surrounds the capsule on all sides.

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4.3 RAMA Calculation Parameters

The RAMA transport code uses a three-dimensional deterministic transport method to calculate neutron flux distributions in reactor problems. [[ ]]

The RAMA transport calculation also uses information from the RAMA nuclear data library to determine the scope of the flux calculation. This information includes the Legendre order of expansion that is used in the treatment of anisotropy of the problem. [[ ]]

The neutron flux is calculated using an iterative technique to obtain a converged solution for the problem. [[ ]]

The impact of these calculation parameter selections on the RAMA fluence evaluation for the Reactor is presented in Section 4.6.

4.4 RAMA Neutron Source Calculation

The neutron source for the RAMA transport calculation is calculated using the input relative power density factors for the different fuel regions and data from the RAMA nuclear data library.

The core neutron source was determined using the cycle-specific three-dimensional burnup distributions. [[

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4.5 RAMA Fission Spectra

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4.6 Parametric Sensitivity Analyses

Several sensitivity analyses were performed to evaluate the stability and accuracy of the RAMA transport calculation for the Reactor model. [[

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5
SURVEILLANCE CAPSULE ACTIVATION RESULTS

This section contains the results from the Reactor surveillance capsule activation analysis. The predicted activations (i.e., specific activities) generated by the RAMA Fluence Methodology were compared to the activation measurements for the capsule flux wires and dosimetry for seven capsules and are presented here. Activation measurement data is available for all the capsule flux wires except the one capsule that was inserted at the Reactor start-up and removed at the end of cycle 1.

One Reactor surveillance capsule was removed at the end of cycle 9 after being irradiated for a total of 8.15 effective full power years (EFPY). BWRVIP Supplemental Surveillance Program (SSP) capsules D, G, and H were irradiated for two cycles for a total of 3.127 effective full power years (EFPY). SSP capsules E, F, and I were irradiated for four cycles for a total of 6.653 effective full power years (EFPY). Details of the dosimetry specimens and analysis are presented in the next subsection.

The calculated-to-measured (C/M) results for the Reactor show a very good agreement between the RAMA calculated values and the measured values. [[

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5.1 Comparison of Predicted Activation to Plant Specific Measurements

There were a total of eight surveillance capsules irradiated in the Reactor and removed from the Reactor after irradiation. Three standard surveillance capsules were inserted at the start of Reactor operation. Flux wires were removed for one of these capsules at the end of cycle 1, however, no measurement data is available. One capsule was removed at the end of cycle 9 and analyzed. Specimens from this capsule included Charpy V-notch, plate and weld, and heat effective zone. Iron, nickel and copper flux wires were also removed (note the nickel flux wires could not be measured due to the substantial decay of the ^{58}Co activity product.). One of the original three capsules is still in the Reactor so is not included in this evaluation. Six of the surveillance capsules were part of the BWRVIP SSP. The SSP capsules were identified as capsules D, E, F, G, H, and I. Capsules D, G, and H were irradiated in the Reactor during cycles 14 and 15. Capsules E, F, and I were irradiated during cycles 14 through 17. The SSP capsules included Charpy V-notch specimens, neutron dosimeters, temperature monitors, and iron and copper flux wires.

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6
REACTOR PRESSURE VESSEL UNCERTAINTY ANALYSIS

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7

CALCULATED NEUTRON FLUENCE FOR REACTOR PRESSURE VESSEL

This section presents the neutron fluence values, as determined by the RAMA Fluence Methodology, for the reactor pressure vessel at the RPV weld locations and RPV shells identified by AmerGen Energy Co, LLC. Values are generated for energy >1.0 MeV at 32 EFPY and 50 EFPY. The results of the fluence evaluation are presented in the tables that follow. Tables 7-1 and 7-2 report the fluence in the RPV weld and RPV shell locations, respectively. The maximum fluence for the RPV weld locations is in vertical weld 2-564A at 50 EFPY with a value of 6.16E+18 n/cm^2 for energy >1.0 MeV. The maximum fluence for the RPV shell locations is in the lower intermediate shell at 50 EFPY with a value of 6.97E+18 n/cm^2 for energy >1.0 MeV.

Table 7-1
Maximum >1.0 MeV Neutron Fluence in RPV Welds

Shell Location	Weld ID#	32 EFPY Fluence (n/cm^2)	50 EFPY Fluence (n/cm^2)
Upper	2-563A	7.94E+10	1.18E+11
	2-563B	6.10E+10	9.03E+10
	2-563C	7.94E+10	1.18E+11
Upper Intermediate	2-563D	3.60E+16	5.23E+16
	2-563E	4.07E+16	6.01E+16
	2-563F	3.60E+16	5.23E+16
Lower Intermediate	2-564A	4.12E+18	6.16E+18
	2-564B	3.02E+18	4.66E+18
	2-564C	2.50E+18	3.85E+18
Circumferential	3-564	2.69E+18	3.76E+18
Lower	2-564D	1.99E+18	2.98E+18
	2-564E	2.20E+18	3.30E+18
	2-564F	2.66E+18	3.71E+18

Table 7-2
Maximum >1.0 MeV Neutron Fluence in RPV Shells

Shell Location	32 EFPY Fluence (n/cm^2)	50 EFPY Fluence (n/cm^2)
Upper	1.92E+11	2.92E+11
Upper Intermediate	7.59E+16	1.07E+17
Lower Intermediate	4.66E+18	6.97E+18
Lower	2.69E+18	3.76E+18

Also provided are the RPV elevations at which the fluence reached a value of 1.0E+17 n/cm^2 for both 32 EFPY and 50 EFPY. Table 7-3 lists the RPV elevations where this occurs for both time projections.

Table 7-3
Elevations at which Fluence Reaches 1.0E+17 n/cm^2 in RPV Shells

Shell Location	Elevation for 32 EFPY Fluence	Elevation for 50 EFPY Fluence
Lower	491.5404 cm (193.52 in)	485.9107 cm (191.30 in)
Upper	934.7868 cm (368.03 in)	945.3969 cm (372.20 in)

8
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