Attachment 2

TransWare Enterprises Report

SEA-FLU-001-R-004, Rev 0

Non-Proprietary Version of
Seabrook Station
Reactor Pressure Vessel Fluence Evaluation at 55 EFPY

NON-PROPRIETARY VERSION OF SEABROOK STATION REACTOR PRESSURE VESSEL FLUENCE EVALUATION AT 55 EFPY

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This document has been prepared in accordance with the requirements of 10CFR50 Appendix B, 10CFR21, and TransWare Enterprises Inc.'s quality assurance program.

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GLOSSARY OF TERMS

AZIMUTHAL QUADRANT SYMMETRY – A type of core and pressure vessel configuration that can be represented by a single quadrant that can be rotated and mirrored to represent the entire 360-degree geometry. The northeast quadrant can be mirrored to represent the northwest and the southeast quadrant can be rotated to represent the southwest quadrant.

BEST-ESTIMATE NEUTRON FLUENCE – See Neutron Fluence.

CALCULATED NEUTRON FLUENCE – See Neutron Fluence.

CONTAINMENT AZIMUTHS – The azimuths in the containment region (including cavity liner plates, primary shield, and reactor support) cited in this report have the 0 degree azimuth corresponding to the RPV 270 degree azimuth (site north).

CONTAINMENT ELEVATIONS – The axial elevations in the containment region (including cavity liner plates, primary shield, and reactor support) cited in this report are relative to sea level.

CORE BELTLINE – The axial elevations corresponding to the active fuel region of the core.

EFFECTIVE FULL POWER YEARS (EFPY) – A unit of measurement representing one full year of reactor operation at the reactor's current rated power level.

FAST NEUTRON FLUENCE – Fluence accumulated by neutrons with energy greater than 1.0 MeV (>1.0 MeV).

NEUTRON FLUENCE – Time-integrated neutron flux reported in units of n/cm^2. The term "best-estimate" fluence refers to the values that are computed in accordance with the requirements of U. S. Nuclear Regulatory Commission Regulatory Guide 1.190 for use in material embrittlement evaluations. The term "calculated" fluence refers to the values that are predicted by the RAMA Fluence Methodology. In this report the best-estimate fluence is synonymous with the calculated fluence since the RAMA Fluence Methodology requires no bias correction to the calculated fluence.

RPV – An abbreviation for reactor pressure vessel. Unless otherwise noted, the reactor pressure vessel refers to the base metal material (i.e., excluding clad/liner).

RPV BELTLINE – The axial elevations corresponding to the regions where the RPV exceeds a fast neutron fluence of 1.0E17 n/cm^2.

RPV ZERO ELEVATION – Axial elevations for the RPV and vessel internals cited in this report assume that RPV zero is at the vessel mating surface of the RPV flange.

1
INTRODUCTION

This report presents the results of the reactor pressure vessel fluence evaluation performed for the Seabrook Station Nuclear Power Plant ("Seabrook Station"). Maximum fast neutron fluence for energy >1.0 MeV is reported for the reactor pressure vessel ("RPV") shells and welds throughout the RPV beltline region assuming a 60-year operating license period for the plant, or the equivalent of 55 effective full power years ("EFPY") of operation.

The fluence evaluation was performed in accordance with guidelines presented in U. S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.190, *"Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence"* [1]. In compliance with these guidelines, comparisons to surveillance capsule flux wire and dosimeter measurements were performed to determine the accuracy of the RPV fluence model. An uncertainty analysis was also performed to determine if a statistical bias exists in the model. It was determined that the Seabrook Station fluence model does not have a statistical bias and that the best-estimate fluence presented in this report is suitable for use in evaluating the effects of embrittlement on RPV material as specified in U. S. Code of Federal Regulations (CFR) 10CFR50 Appendix G, *"Fracture Toughness Requirements"* [2] and U. S. NRC Regulatory Guide 1.99, *"Radiation Embrittlement of Reactor Vessel Materials"* [3].

The fluence values presented in this report were calculated using the RAMA Fluence Methodology [4]. The RAMA Fluence Methodology (hereinafter referred to as RAMA) was developed for the Electric Power Research Institute, Inc. (EPRI) for the purpose of calculating fast neutron fluence in reactor pressure vessels and vessel internal components. As prescribed in Regulatory Guide 1.190, RAMA has been benchmarked against industry standard benchmarks for both pressurized water reactor (PWR) and boiling water reactor (BWR) designs. In addition, RAMA has been compared with several plant-specific dosimetry measurements and reported fluence from several commercial operating reactors. The results of the benchmarks and comparisons to measurements show that RAMA accurately predicts specimen activities, RPV fluence, and vessel component fluence in all light water reactor types. Under funding from EPRI and the Boiling Water Reactor Vessel and Internals Project (BWRVIP), the RAMA methodology has been reviewed by the U. S. NRC and subsequently given generic approval for determining fast neutron fluence in BWR pressure vessels [5] and vessel internal components that include the core shroud and top guide [6]. This prior work has been extended in this report to include comparisons to additional PWR benchmarks and plant-specific dosimetry comparisons which further validates the use of RAMA for all light water reactor designs.

The information and associated evaluations provided in this report have been performed in accordance with the requirements of 10CFR50 Appendix B [7].

2
SUMMARY OF RESULTS

This section provides a summary of the fast neutron fluence evaluation performed for the Seabrook Station pressure vessel. The focus of the RPV evaluation was to determine the maximum fast neutron fluence accumulated in shells, welds, and nozzles in the RPV beltline assuming 55 EFPY of reactor operation.

Table 2-1 summarizes the peak fast neutron fluence at 55 EFPY for the pressure vessel. Figure 7-1 in Section 7 illustrates the location of the RPV welds and shells in the pressure vessel wall.

Table 2-1
Peak Neutron Fluence for Energy >1.0 MeV for Seabrook Station RPV Weld and Shell Locations at the Inner Surface at 55 EFPY

Weld/Shell Location[a]	55 EFPY	
	Elevation [cm (in)]	Peak Fluence at Inner Surface (0T) (n/cm^2)
Peak Weld - Weld 101-171 (22°, 158°, 202°, 338° azimuth)	-592.53 (-233.28)	3.59E+19
Peak Shell – Intermediate Shell (22°, 158°, 202°, 338° azimuth)	-490.87 (-193.26)	3.63E+19

a) The fluence model assumed quadrant rotational symmetry in the azimuthal dimension; thus, results are reported for the four symmetrical locations, as appropriate.

The elevation ranges at which the RPV fluence exceeds 1.0E+17 n/cm^2 are given in Table 2-2. These elevations define the RPV beltline region at 55 EFPY. Based upon the elevation ranges shown in Table 2-2, all RPV circumferential and vertical welds and all shells are in the RPV beltline region. The RPV nozzles lie outside the RPV beltline region with a fluence value of less than 1.0E+17 n/cm^2 at 55 EFPY.

Table 2-2
RPV Fluence Threshold Elevation Range for Seabrook Station

Reactor Lifetime	Lower Elevation [cm (inches)]	Upper Elevation [cm (inches)]
55 EFPY	-815.08 (-320.90)	-289.07 (-113.81)

Section 7 contains tables of results listing the fast neutron fluence for all of the RPV circumferential and vertical welds and shells examined in this evaluation.

In addition to the prediction of RPV fast neutron fluence, comparisons were made between the calculated and measured specific activities for three sets of dosimetry specimens removed from Seabrook Station. [[

]] Note that the results reported for RAMA are a direct solution of the reactions, i.e. no multiplicative or other adjustment factors are applied to the results. A more detailed assessment of the surveillance capsule activation analyses is presented in Section 5 of this report.

Another result from this evaluation is the calculated RPV fluence combined uncertainty value. By combining the measurement uncertainty and analytic uncertainty, the combined RPV fast neutron fluence uncertainty is determined to be [[ ]] with no bias correction to the fluence.

In conclusion, it is determined that RAMA produces results that meet the requirements of U. S. NRC Regulatory Guide 1.190.

3
DESCRIPTION OF THE REACTOR SYSTEM

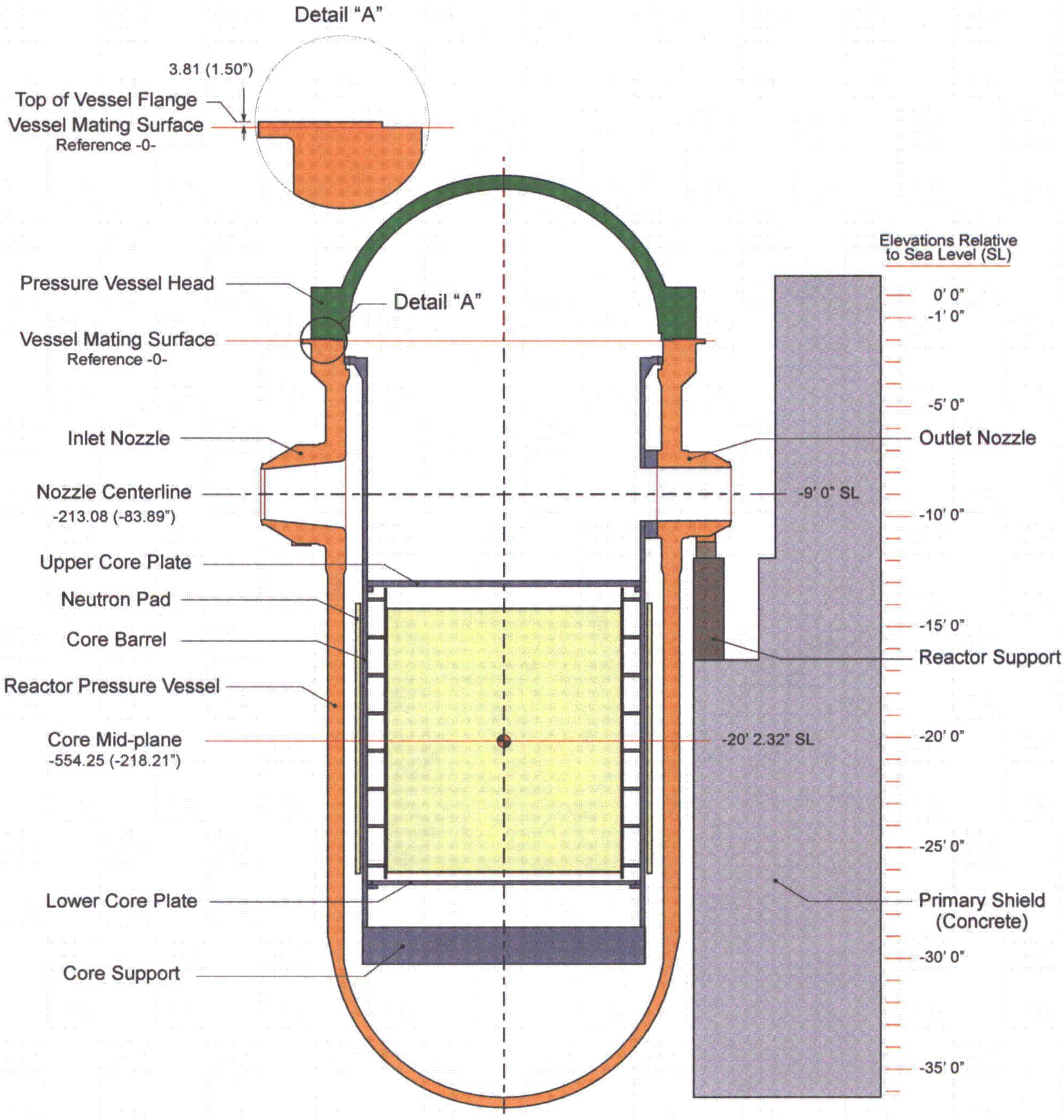
This section provides an overview of the reactor design and operating data inputs that were used to develop the Seabrook Station reactor fluence model. All reactor design and operating data inputs used to develop the model were plant-specific and were provided by NextEra Energy Seabrook, LLC. The inputs for the fluence geometry model were developed from design and as-built drawings for the reactor pressure vessel, vessel internals, fuel assemblies, and containment regions. The reactor operating data inputs were developed from plant records and core simulator data that provided a historical accounting of how the reactor operated for cycles 1 through 12, inclusive. [[
]]

3.1 Overview of the Reactor System Design

Seabrook Station is a Westinghouse 4-loop pressurized water reactor design with a core loading of 193 nuclear fuel assemblies. Seabrook Station began commercial operation in 1990 with a design rated power of 3411 MWt. The reactor implemented two subsequent power uprates. In cycle 11 the rated power was increased to 3587 MWt and in cycle 12 the rated power was increased to 3648 MWt. At the time of this fluence analysis Seabrook Station had completed 12 cycles of operation.

Figure 3-1 illustrates an axial cutaway of the Seabrook Station reactor pressure vessel and containment. This figure shows that the reference elevation, that is, the reactor zero elevation, for the pressure vessel and vessel internals is at the mating surface of the pressure vessel upper flange. This figure also shows that the reference elevation for the structures and primary shield outside the RPV is sea level. The results presented in this report for each component, structure and region are listed with the appropriate reference to elevations in this figure.

Figures 3-2 and 3-3 provide general illustrations of the primary components, structures and regions of the Seabrook Station design at two planar elevations. Figure 3-2 illustrates the basic planar configuration of the Seabrook Station reactor at an axial elevation near the reactor core mid-plane. All of the radial regions of the reactor that are required for fluence projections are shown. Beginning at the center of the reactor and projecting outward, the regions include: the core region; the core baffle plate; the core bypass region; the core barrel; the downcomer region; the surveillance capsules and neutron pads; the reactor pressure vessel; the reactor insulation; the ex-core neutron detectors; the steel liner plates for the primary shield; the primary shield (concrete); and the cavity regions (air).

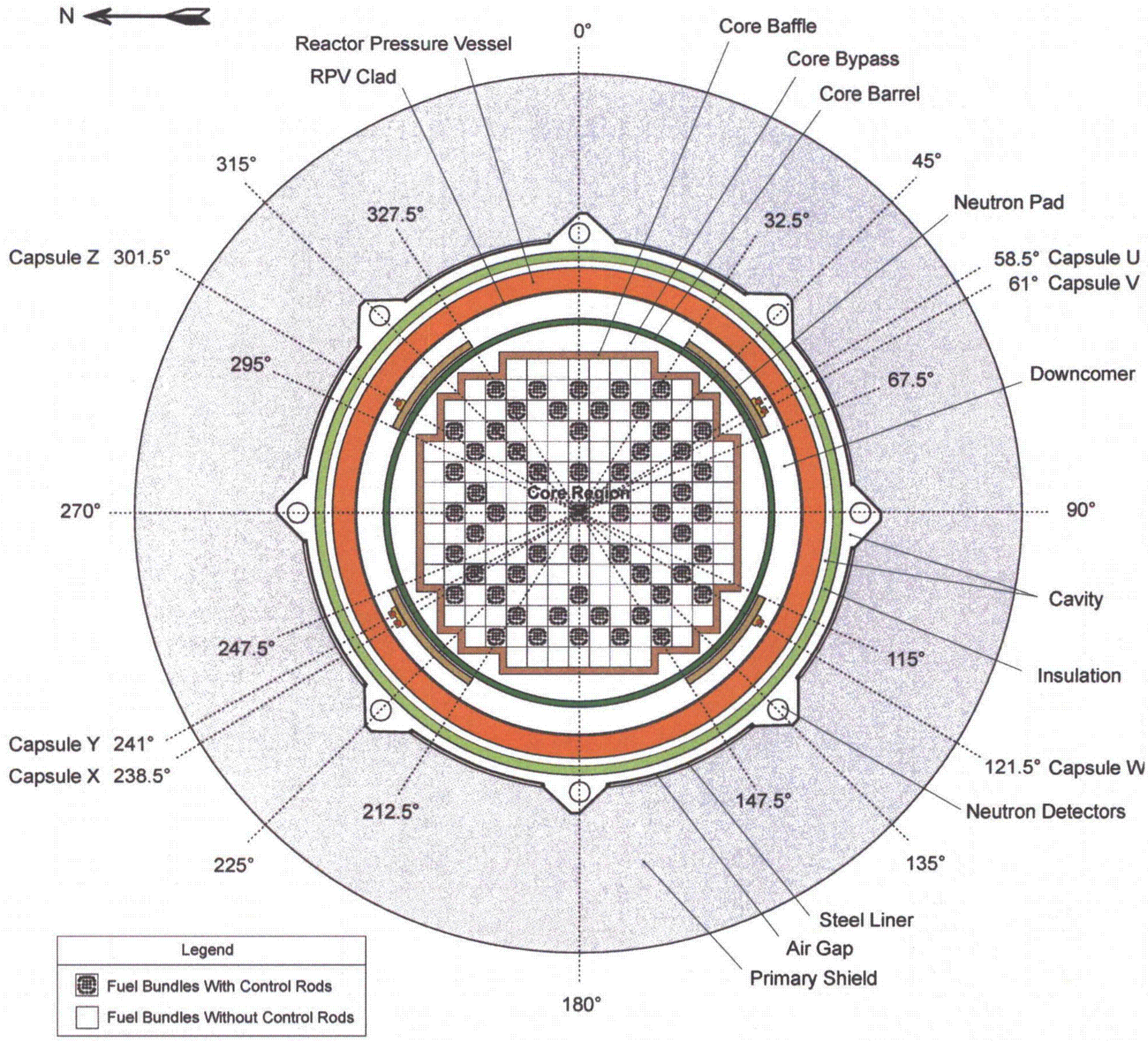


Notes: Drawing is not to scale.

Pressure vessel elevations are given on the left side of the figure in centimeters (inches) relative to the upper flange Vessel Mating Surface.

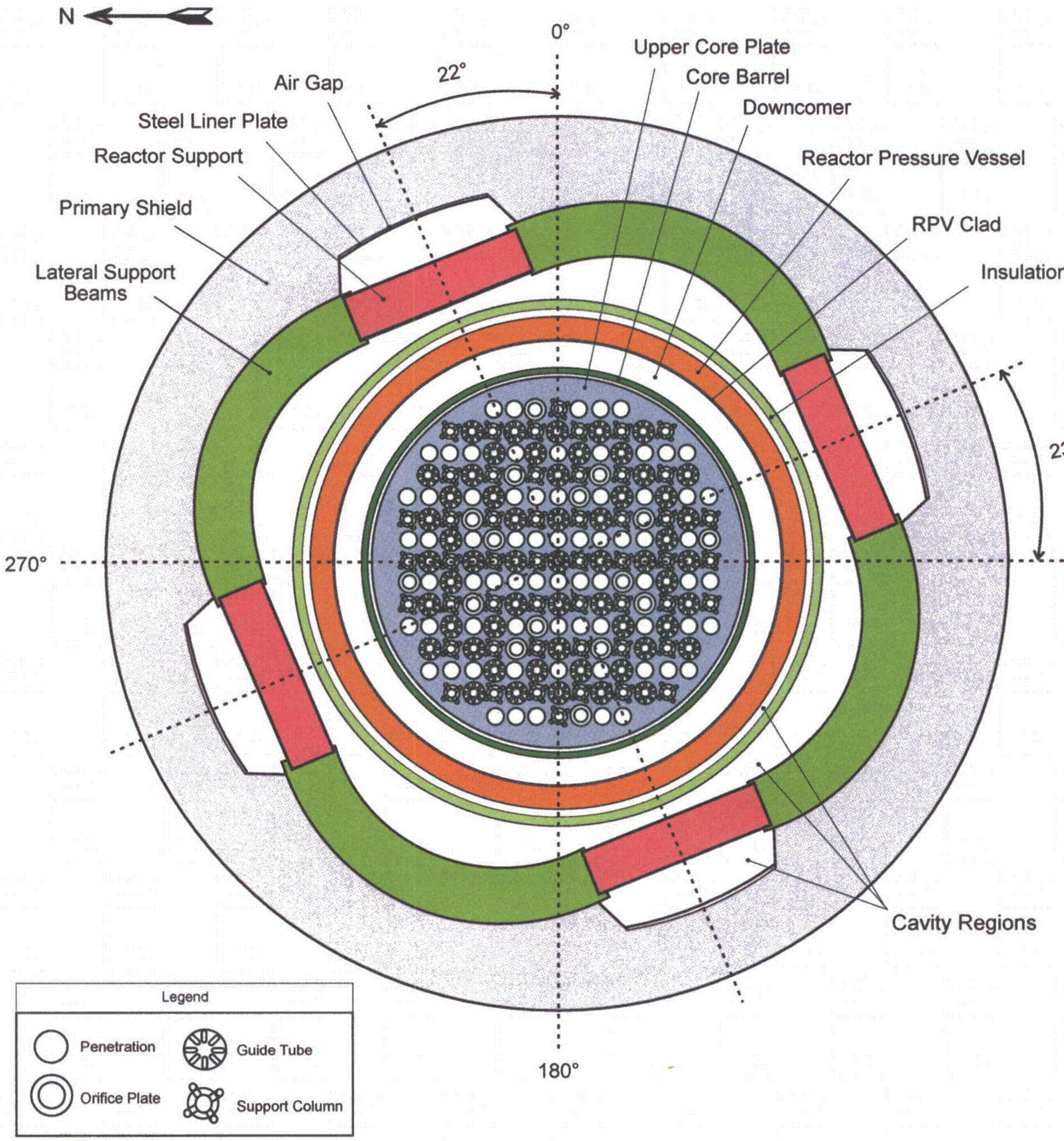
Containment elevations are given on the right side of the figure in feet-inches relative to Sea Level.

Figure 3-1
Illustration of the Seabrook Station Pressure Vessel and Containment Elevations



Notes: Drawing is not to scale.
Azimuths are given relative to the RPV zero azimuth.

Figure 3-2
Planar View of the Seabrook Station at the Core Mid-plane Elevation



Notes: Drawing is not to scale.
Azimuths are given relative to the RPV zero azimuth.

Figure 3-3
Planar View of the Seabrook Station at an Elevation of -13' 0" Relative to Sea Level

Figure 3-3 illustrates the basic planar configuration of the Seabrook Station reactor at an axial elevation of -13' 0" SL. All the radial regions of the reactor that are required for fluence projections are shown. Beginning at the center of the reactor and projecting outward, the regions include: the upper core plate; components occupying upper core plate penetrations (guide tubes, orifice plates, and support columns); the core barrel; the downcomer region; the reactor pressure vessel clad; the reactor pressure vessel; the reactor insulation; the lateral support beams; the reactor supports; the steel liner plates for the primary shield; the primary shield (concrete); and the cavity regions (air).

3.2 Reactor System Mechanical Design Inputs

The mechanical design inputs that were used to construct the Seabrook Station fluence geometry model included as-built and nominal design dimensional data. As-built data for the reactor components and regions of the reactor system is always preferred when constructing plant-specific models; however, as-built data is not always available. In these situations, nominal design information is used.

For the Seabrook Station model, the predominant dimensional information used to construct the fluence model was nominal design data. [[

]]

Another important component of the fluence analysis is the accurate description of the surveillance capsules in the reactor. It is shown in Figure 3-2 that six surveillance capsules were initially installed in the Seabrook Station reactor. The capsules were attached radially to the outside surface of the neutron pads (looking outward from the core region) at the 58.5, 61, 121.5, 238.5, 241 and 301.5-degree azimuths. Surveillance capsules are used to monitor the radiation accumulated in the reactor over a period of time. The importance of surveillance capsules in fluence analyses is that they contain flux wires that are irradiated during reactor operation. When a capsule is removed from the reactor, the irradiated flux wires are evaluated to obtain activity measurements. These measurements are used to validate the fluence model. (See Section 5 which presents a comparison of the calculated-to-measured capsule results.)

Also shown in Figure 3-2 is the radial and azimuthal locations of the ex-core neutron detectors. Eight ex-core neutron detectors were initially installed in the cavity region of the reactor. The detectors are slightly recessed into the primary shield wall at 45-degree increments beginning at 0 degrees.

3.3 Reactor System Material Compositions

Each region of the reactor is comprised of materials that include reactor fuel, steel, borated water, insulation, concrete, and air. Accurate material information is important 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 fast neutron fluence in the reactor components.

Table 3-1 provides a summary of the materials for the various components and regions of the Seabrook Station reactor. The material attributes for the steel, insulation, concrete, and air compositions (i.e. densities and isotopic concentrations) are assumed to remain constant for the operating life of the reactor. 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, changing moderator density levels and boron concentrations. Because of the dynamics of the fuel attributes with reactor operation, several state-point data sets are used to describe the operating states of the reactor for each operating cycle. The number of data sets used in this analysis is presented in Section 3.4.2.1.

3.4 Reactor Operating Data Inputs

An accurate evaluation of reactor vessel and component fluence requires an accurate accounting of the reactor's operating history. The primary reactor operating parameters that affect the determination of fast neutron fluence in light water reactors include reactor power levels, core power distributions, coolant water density distributions, and fuel material (isotopic) distributions. For PWRs, the soluble boron concentration in the coolant is also considered as the absorption properties of boron can effect the determination of the fission source distribution during the neutron transport calculations.

3.4.1 Power History Data

Reactor power history is the measure of reactor power levels on a daily or other periodic basis. For this fluence evaluation, the power history for the Seabrook Station reactor was developed from power history spreadsheets provided by NextEra Energy. The power history data showed that Seabrook Station started commercial operation with a design rated thermal output of 3411 MWt for cycles 1 through 10, and then implemented power uprates of 3587 MWt for cycle 11 and 3648 MWt for cycle 12. It was assumed in this analysis that all future cycles would operate at the 3648 MWt power level.

The power history data for Seabrook Station included daily and monthly average power levels for the reactor over its operating life. This data was used to calculate the vessel fluence and surveillance capsule activities for the reactor. Average reactor powers based on monthly data were used to describe cycles 1 through 5. Daily power histories were used to describe cycles 6

Table 3-1
Summary of Material Compositions by Region for Seabrook Station

Region	Material Composition
Upper Core Plate Support Columns	Stainless Steel
Control Rods and Guide Tubes	Stainless Steel, Silver, Indium, Cadmium
Upper Core Plate	Stainless Steel
Fuel Bundle Top Nozzle	Stainless Steel, Zircaloy, Inconel
Reactor Core	^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, O_{fuel}, Zircaloy, Borated Water
Core Baffle	Stainless Steel
Former Plates	Stainless Steel
Core Bypass Region	Borated Water
Core Barrel	Stainless Steel
Fuel Bundle Lower Nozzle	Stainless Steel, Zircaloy, Inconel
Lower Core Plate	Stainless Steel
Core Support Plate	Stainless Steel
Lower Core Plate Support Columns	Stainless Steel
Downcomer Region	Borated Water
Neutron Pads	Stainless Steel
Surveillance Capsule Specimens	Carbon Steel (SA 336)
Irradiated Specimen Guide Tubes	Stainless Steel
Reactor Pressure Vessel Clad	Stainless Steel
Reactor Pressure Vessel (Base Metal)	Carbon Steel (SA 336)
Cavity Regions	Air
Insulation Clad	Stainless Steel
Insulation	Stainless Steel Foil
Neutron Detector Pipes	Carbon Steel
Reactor Cavity Steel Liner Plates	Carbon Steel
Primary Shield Wall	Reinforced Concrete
Reactor Support	Carbon Steel
Lateral Support Beams	Carbon Steel

through 12. Periods of reactor shutdown due to refueling outages and other events were also accounted for in the model. The power history data was verified by comparing the calculated energy production in effective full power years (EFPY) with power production records provided by NextEra Energy. Table 3-2 lists the accumulated EFPY at the end of each cycle for this fluence evaluation.

3.4.2 Reactor State-Point Data

Core simulator data was provided by NextEra Energy to characterize the historical operating conditions of the reactor for Seabrook Station cycles 1 through 12. The data calculated with core simulator codes represents the best-available information about the reactor cores operating history over the reactor's operating life. In this analysis, the core simulator data was provided in the form of state-point data files. The state-point files included three-dimensional data arrays that described core power distributions, fuel exposure distributions, fuel materials (isotopics), water densities, and soluble boron concentrations in the reactor core.

A separate neutron transport calculation was performed for each of the available state points (see Section 3.4.2.1). The calculated neutron flux for each state point was combined with the appropriate power history data described in Section 3.4.1 in order to provide an accurate accounting of the fast neutron fluence for the reactor pressure vessel. Fluence projections to the end of the reactor's design life were performed using a projected equilibrium cycle. Equilibrium cycles are discussed in Section 3.4.2.2.

3.4.2.1 Beginning of Operation through Cycle 12 State Points

A total of [[ ]] state points were used to represent the operating history of the Seabrook Station reactor for the first 12 operating cycles. These state points were selected from hundreds of exposure points that were calculated with the core simulator code. [[

]]

Table 3-2
State-point Data for Each Cycle in Seabrook Station

Cycle Number	Number of State Points	Rated Thermal Power[a] MWt	Accumulated Effective Full Power Years (EFPY)
1	[[	3411	0.9
2		3411	1.8
3		3411	3.0
4		3411	4.2
5		3411	5.6
6		3411	7.1
7		3411	8.5
8		3411	9.7
9		3411	11.0
10		3411	12.4
11		3587	13.8
12		3648	15.1
13+[[ ]]	]]	3648	55.0

a) The rated thermal power is listed for each cycle. The actual power levels were used for the individual state-point calculations for cycles 1 through 12.

[[ ]]

3.4.2.2 *Projected Reactor Operation*

Projections of plant operations beyond cycle 12 are represented with an "equilibrium" cycle. It is assumed that an equilibrium cycle will incorporate the best-available information on expected cycle length, fuel bundle loading, and operating strategies for future cycles. [[

]]

3.4.2.3 *Limitation of Fluence Projections*

The fluence presented in this report is based on projections of an assumed equilibrium cycle. The significance of the equilibrium cycle is that it defines the flux profiles that are used to project fluence into the future. Providing that the design basis for the equilibrium cycle does not change appreciably, projections based on the equilibrium cycle should remain bounding through 55 EFPY to support licensing, inservice inspection, and flaw evaluation activities.

If the design basis for the equilibrium cycle changes at any point in time that would result in a significant change to the flux profiles for the equilibrium cycle, then an evaluation should be done to determine if an updated fluence evaluation is needed. [[

]]

3.4.3 Core Loading Pattern

It is common in PWRs that more than one fuel assembly design and assembly insert type may be loaded in the reactor core in any given operating cycle. For fluence evaluations, it is important to account for the fuel assembly designs and inserts that are loaded in the core in order to accurately represent the neutron source distribution at the core boundaries (i.e. peripheral fuel locations, the top fuel nodes and the bottom fuel nodes).

3.4.3.1 Fuel Assembly Designs

Seven fuel assembly designs were loaded in the Seabrook Station reactor during cycles 1 through 12[[ ]]. All of the fuel assembly designs consisted of a 17x17 array of pins. Each fuel assembly design had unique material distributions for the top and bottom nozzles and the fuel rod plenum regions. The active fuel height for Seabrook Station is constant at 365.76 cm (12 ft); however, the bottom of active fuel elevation varied with the different fuel assembly designs loaded in the reactor by as much as 2.58 cm (~ 1 in.).

Table 3-3 provides a summary of the fuel designs loaded in the Seabrook Station reactor. The cycle core loading patterns provided by NextEra Energy for each operating cycle were used to identify the fuel assembly designs in each cycle and their location in the reactor core.

3.4.3.2 Fuel Assembly Inserts

Five types of inserts were loaded in the Seabrook reactor during cycles 1 through 12[[ ]]. All inserts consist of a hub and rodlets. The rodlets are connected to a hub and slip into the guide tubes of an assembly. All inserts, except rod cluster control assemblies (RCCA), use the same hub design. The length and composition of the rodlets vary for each insert type.

The common inserts used in Seabrook fuel assemblies are RCCAs and thimble plugs (TP). Other inserts loaded at Seabrook include: burnable poison rod assemblies (BPRA), primary sources (PS), and secondary sources (SS).

Table 3-4 provides a summary of the insert types loaded in the reactor core for these operating cycles. The cycle insert loading patterns provided by NextEra Energy Seabrook, LLC were used to identify the insert types in each cycle and their location in the reactor core.

Table 3-3
Summary of the Seabrook Station Core Loading Pattern

Cycle	STD	STD2	V5H	V5H2	V5HZ2	RFA	Dominant Peripheral Fuel Design
1	193						STD
2	133	60					STD
3	57	136					STD2
4		193					STD2
5	1	112	80				STD2
6	9	24	76	84			V5H
7	1	28		164			STD2
8	1		24	80	88		V5H
9				21	88	84	V5HZ2
10					25	168	V5HZ2
11						193	RFA
12						193	RFA

STD = Standard Fuel Assembly
STD2 = STD with enhanced skeletal structure
V5H = Vantage 5H Design
V5H2 = V5H with Intermediate Flow Mixer (IFM) grids and simplified spring clamps
V5HZ2 = V5H2 with a longer fuel rod that has a larger plenum
RFA = Robust Fuel Assembly

Table 3-4
Summary of the Seabrook Station Insert Loading Pattern

Insert Type	Cycle											
	1	2	3	4	5	6	7	8	9	10	11	12
RCCA	57	57	57	57	57	57	57	57	57	57	57	57
TP	40	134	134	132	132	132	132	132	132	134	132	134
BPRA	92											
PS	2											
SS	2	2	2	4	4	4	4	4	4	2	4	2

RCCA = Rod Cluster Control Assemblies
TP = Thimble Plugs
BPRA = Burnable Poison Rod Assemblies
PS =Primary Sources
SS = Secondary Sources

4
CALCULATION METHODOLOGY

This section provides an overview of the Seabrook Station fluence model that was developed with the RAMA Fluence Methodology software [4]. The RAMA fluence model for Seabrook Station is a plant-specific model that was constructed from the design inputs described in Section 3, *Description of the Reactor System.*

4.1 Description of the RAMA Fluence Methodology

The RAMA Fluence Methodology (RAMA) is a system of computer codes, a data library, and an uncertainty methodology that determines best-estimate fluence in light water reactor pressure vessels and vessel components. The primary codes that comprise the RAMA methodology include model builder codes, a particle transport code, and a fluence calculator code. The data library contains nuclear cross sections and response functions that are needed for each of the codes during execution. The uncertainty methodology is used to determine the uncertainty and bias in the best-estimate fluence calculated by the software.

The primary inputs for RAMA are mechanical design parameters and reactor operating history data. The mechanical design inputs are obtained from plant-specific design drawings, which include as-built measurements when available. The reactor operating history data is obtained from reactor core simulator codes, system heat balance calculations, daily operating logs, and cycle summary reports that describe the operating conditions of the reactor over its operating life time. The primary outputs from RAMA calculations are neutron flux, neutron fluence, dosimetry activation, and an uncertainty evaluation.

The model builder codes consist of geometry and material processor codes that generate input for the particle transport code. The geometry model builder code uses mechanical design inputs and meshing specifications to generate three-dimensional geometry models of the reactor. The material processor code uses reactor operating data inputs to process fuel materials, structural materials and water densities that are consistent with the geometry meshing generated by the geometry model builder code.

The particle transport code performs three-dimensional neutron flux calculations using a deterministic, multigroup, particle transport theory method with anisotropic scattering. The primary inputs prepared by the user for the transport code include the geometry and material data generated by the model builder codes and numerical integration and convergence parameters for the iterative transport calculation. The transport solver is coupled with a general geometry modeling capability based on combinatorial geometry techniques. The coupling of general geometry with a deterministic transport solver provides a flexible, accurate and efficient tool for calculating neutron flux in light water reactor pressure vessels and vessel components. The primary output from the transport code is the neutron flux in multigroup form.

The fluence calculator code determines fluence and activation in the reactor pressure vessel and vessel components over specified periods of reactor operation. The primary inputs to the fluence calculator include the multigroup neutron flux from the transport code, response functions for the various materials in the reactor, reactor power levels for the operating periods of interest, the specification of which components to evaluate, and the energy ranges of interest for evaluating neutron fluence. The fluence calculator includes treatments for isotopic production and decay that are required to calculate specific activities for irradiated materials. The reactor operating history is generally represented with several reactor state points that represent the various power levels and core power shapes generated by the reactor over the life of the plant. These detailed state points are combined with the daily reactor power levels to produce accurate estimates of the fluence and activations accumulated in the plant.

The uncertainty methodology provides an assessment of the overall accuracy of the RAMA Fluence Methodology. Variances in the dimensional data, reactor operating data, dosimetry measurement data, and nuclear data are evaluated to determine if there is a statistically significant bias in the calculated results that might effect the determination of the best-estimate fluence for the reactor. The plant-specific results are also weighted with comparative results from experimental benchmarks and other plant analyses and analytical uncertainties pertaining to the methodology to determine if the plant-specific model under evaluation is statistically acceptable as defined in Regulatory Guide 1.190.

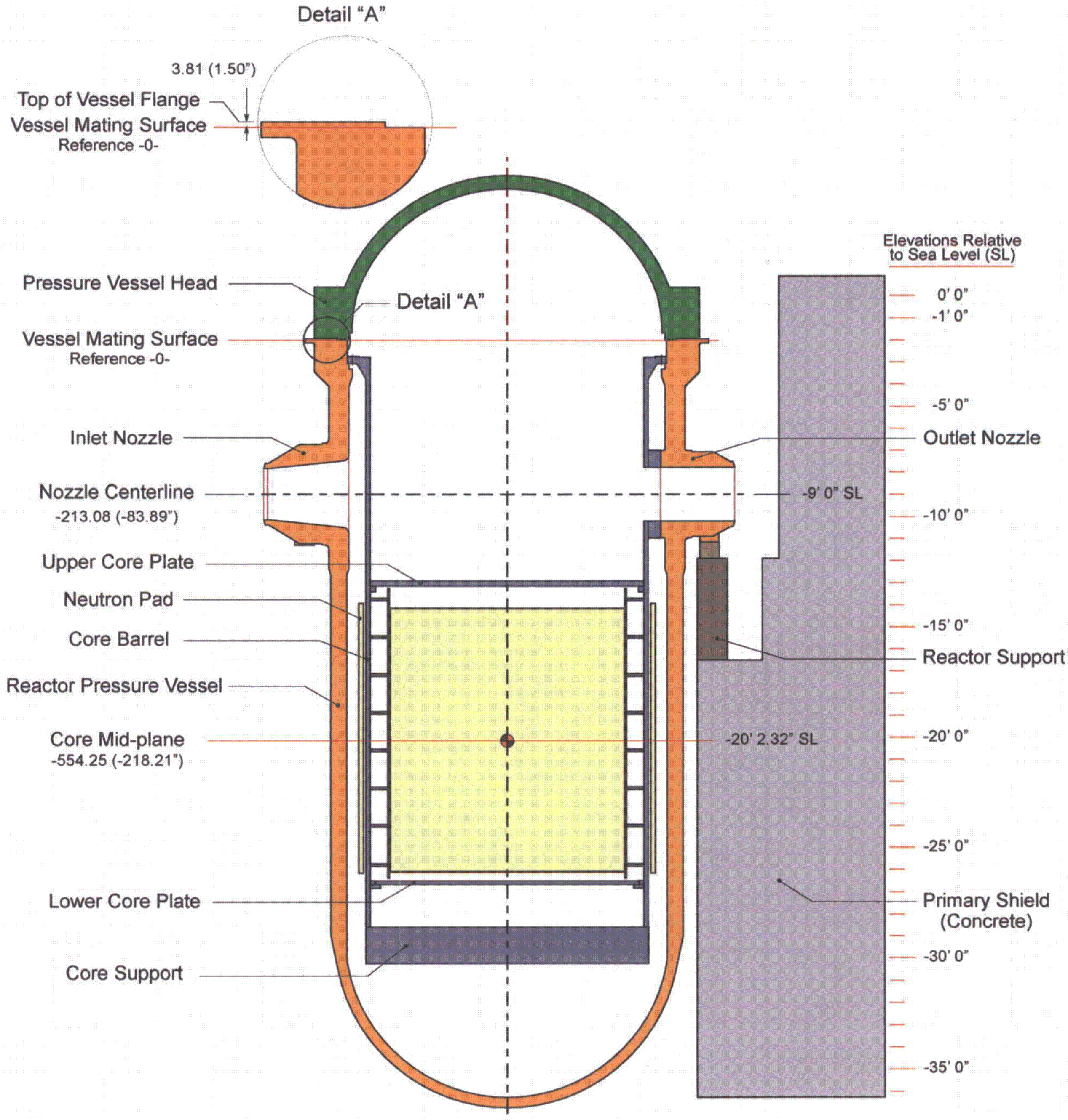
The RAMA nuclear data library contains atomic mass data, nuclear cross-section data and response functions that are needed in the material processing, transport, fluence and reaction rate calculations. The cross-section data and response functions are based on the BUGLE-96 nuclear data library [8].

The RAMA Fluence Methodology is described in the Theory Manual [9] and the general procedures for using the methodology are presented in the Procedures Manual [10].

4.2 The RAMA Geometry Model for the Seabrook Station Reactor

Section 3, *Description of the Reactor System*, describes the design inputs that were provided by NextEra Energy for the Seabrook Station reactor fluence evaluation. These design inputs were used to develop a plant-specific, three-dimensional computer model of the Seabrook Station reactor with the RAMA Fluence Methodology.

Figure 4-1 illustrates an axial cutaway of the Seabrook Station reactor pressure vessel. This figure shows that the reference elevation, that is, the reactor zero elevation, for the pressure vessel and vessel internals is at the mating surface of the pressure vessel upper flange. Although it is not depicted in the figure, the reference elevation for the vessel externals, the primary shield



Notes: Drawing is not to scale.
Pressure vessel elevations are given on the left side of the figure in centimeters (inches) relative to the upper flange Vessel Mating Surface.
Containment elevations are given on the right side of the figure in feet-inches relative to Sea Level.

Figure 4-1
Illustration of the Seabrook Station Pressure Vessel Elevations

and outer cavity structures is sea level. The results presented in this report for each component, structure and region are listed with their appropriate elevations.

Figures 4-2 and 4-3 provide general illustrations of the primary components, structures and regions developed for the Seabrook Station fluence model. Figure 4-2 shows the planar configuration of the reactor model at an elevation corresponding to the reactor core mid-plane elevation. Figure 4-3 shows an axial configuration of the reactor model. Note that the figures are not drawn to scale. They are intended only to provide a perspective for the layout of the model, and specifically how the various components, structures and regions lie relative to the reactor core region (i.e., the neutron source).

Because the figures are intended only to provide a general overview of the model, they do not include illustrations of the geometry meshing developed for the model. To provide such detail is beyond the scope of this document. [[

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The following sub-sections provide an overview of the computer models that were developed for the various components, structures, and coolant flow regions of the Seabrook Station reactor.

4.2.1 The Geometry Model

RAMA uses a generalized three-dimensional geometry modeling system that is based on a combinatorial geometry technique which is mapped to a Cartesian coordinate system. In this analysis, an axial plane of the reactor model is defined by the (x,y) coordinates of the modeling system and the axial elevation at which a plane exists is defined along a perpendicular z-axis of the modeling system. Thus, any point in the reactor model can be addressed by specifying the (x,y,z) coordinates for that point.

Figure 3-2 of Section 3 illustrates a planar cross section view of the Seabrook Station reactor design at an axial elevation corresponding to the reactor core mid-plane elevation. It is shown for this one elevation that the reactor design is a complex geometry composed of various combinations of rectangular, cylindrical and wedge-type bodies. When the reactor is viewed in three dimensions, the varying heights of the different components, structures and regions create additional geometry modeling complexities. An accurate representation of these geometrical complexities in a predictive computer model is essential for calculating accurate, best-estimate fluence in the reactor pressure vessel, the vessel internals, and the surrounding structures.

Figures 4-2 and 4-3 provide general illustrations of the planar and axial geometry complexities that are represented in the Seabrook Station fluence model. For comparison purposes, the planar view illustrated in Figure 4-2 corresponds to the same core mid-plane elevation illustrated in Figure 3-2 for the Seabrook Station reactor, with one variation. The computer model for Seabrook Station assumes quadrant symmetry in the planar dimension.

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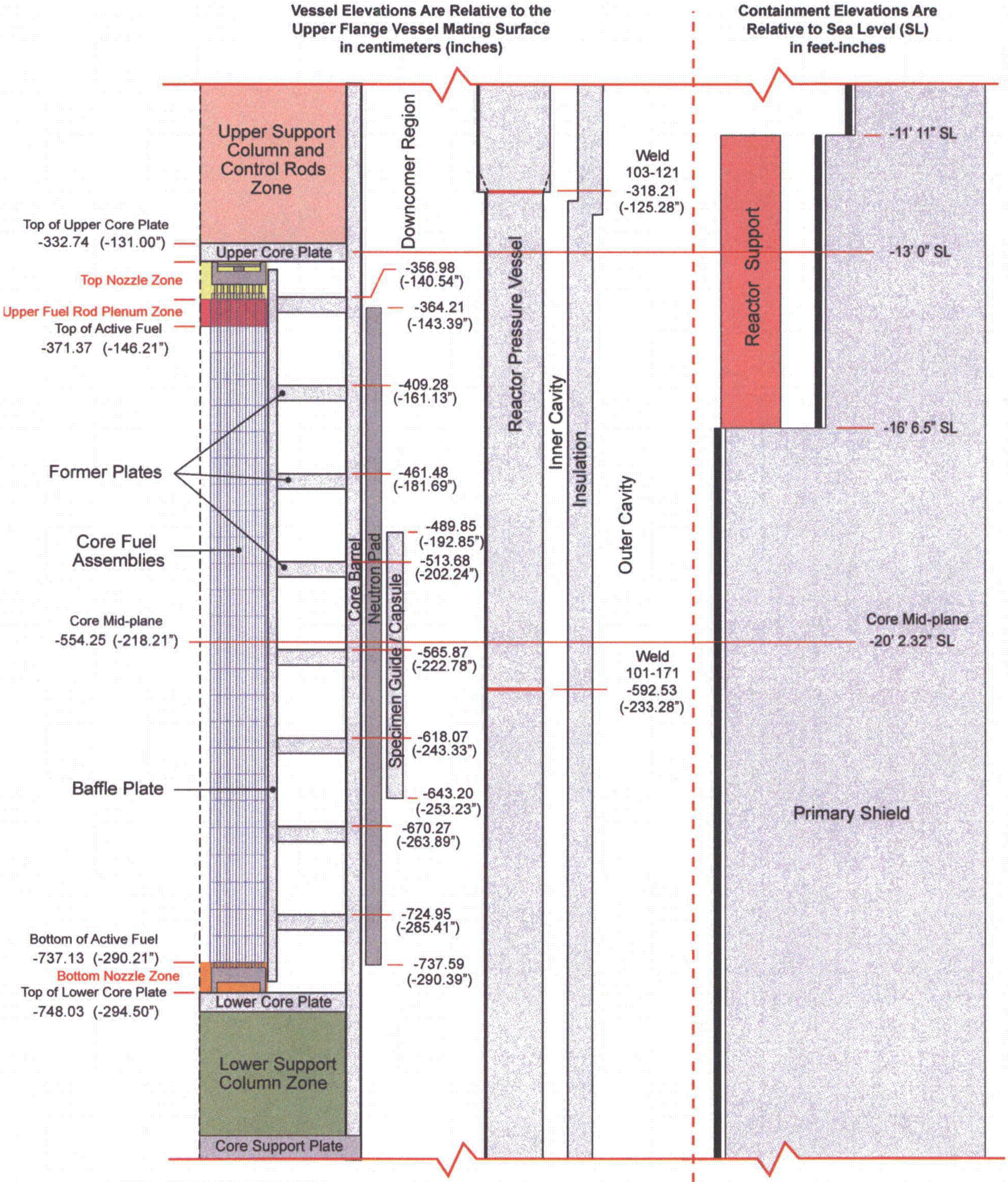
Figure 4-2
Planar View of the Seabrook Station RAMA Model at the Core Mid-plane Elevation



Notes: Drawing is not to scale.
Pressure vessel elevations are given in centimeters (inches) relative to the Vessel Mating Surface.
The fuel elevations are nominal. The actual elevations in the model vary according to fuel assembly designs.
Containment dimensions are given in feet-inches relative to sea level.

Figure 4-3
Axial View of the Seabrook Station RAMA Model

Figure 4-2 illustrates the quadrant geometry that was modeled in this analysis. In terms of the modeling coordinate system, the "northeast" quadrant of the geometry is represented in the model. The 0-degree azimuth, which has a "north" designation, corresponds to the 0-degree azimuth referenced in the plan drawings for the reactor pressure vessel. Degrees are incremented clockwise. Thus, the 90-degree azimuth is designated as the "east" direction. All other components, structures and regions have been appropriately mirror reflected or rotated to this quadrant based upon their relationship to the pressure vessel orientation to ensure that the fluence is correctly calculated relative to the neutron source (i.e., the core region). Although symmetry is a modeling consideration, the results presented in this report for the components and structures are given at their respective azimuths as identified in their respective design drawings.

Figure 4-3 illustrates the axial configuration of the primary components, structures and regions in the fluence model. For discussion purposes, the same components, structures and regions shown in the planar view of Figure 4-2 are also illustrated in Figure 4-3. [[

]] This axial height covers all areas of the reactor pressure vessel that are expected to exceed a fluence threshold of 1.0 x 10^{17} n/cm^2 at 55 EFPY.

Figure 4-3 also reports different units of elevation for the components and structures inside and outside the reactor pressure vessel. The elevations inside the pressure vessel, which include the vessel and vessel insulation, are reported in units of centimeters and inches relative to the RPV upper flange mating surface. The elevations for the containment region, which include the cavity region, ex-core neutron detectors, reactor supports and primary shield are reported in units of feet-inches relative to sea level. These units of elevation are used to report the fluence results for the various components and structures evaluated in this analysis.

As previously noted, the figures are not drawn precisely to scale. They are intended only to provide a perspective of how the various components, structures and regions of the reactor are positioned relative to the reactor core region (i.e., the neutron source) and each other.

4.2.2 The Reactor Core and Core Baffle Models

The reactor core contains the nuclear fuel that is the source of the neutrons that irradiate all components and structures of the reactor. The core is surrounded by a baffle structure that serves to channel the reactor coolant through the core region during reactor operation. The reactor core and core baffle geometries are each rectangular in design and are modeled with rectangular elements to preserve their shapes in the analysis. Models for the reactor core and core baffle structure for the Seabrook Station reactor are illustrated in Figures 4-2 and 4-3.

Figure 4-2 illustrates that the core region is centered in the reactor pressure vessel. The core region is characterized in the analysis with two fundamental fuel zones: interior fuel assemblies and peripheral fuel assemblies. The peripheral fuel assemblies are the primary contributors to the neutron source in the fluence calculation. Because these assemblies are loaded at the core edge where neutron leakage from the core is greatest, there is a sharp power gradient across these assemblies that requires consideration. To account for the power gradient, the peripheral fuel assemblies are sub-meshed[[ ]]. The interior fuel assemblies make a lesser contribution to the reactor fluence and are, therefore, modeled in various homogenized forms in accordance with their contributions to the reactor fluence. For computational efficiency, homogenization treatments are used in the interior core region primarily to reduce the number of mesh regions that must be solved in the transport calculation. The meshing configuration for each fuel assembly location in the core region is determined by parametric studies to ensure an accurate estimate of fluence throughout all regions of the reactor system.

Figure 4-3 illustrates an axial representation for the fuel assemblies in the model. Each fuel assembly design, whether loaded in the interior or peripheral locations in the core, is represented with four material zones: the bottom nozzle zone, the fuel zone, the fuel upper plenum zone and the top nozzle zone. The structural materials in the top and bottom nozzles for each unique assembly design are represented in the model to address the shielding effects that these materials have on the components above and below the core region. The fuel zone contains the nuclear fuel and structural materials for the fuel assemblies. The materials for each fuel assembly are unique during reactor operation and are incorporated into the model using reactor operating data from core simulator codes. The upper plenum region captures fission gases during reactor operation.

It is shown in Figure 4-3 that the Seabrook Station reactor core region has a nominal elevation for the bottom of active fuel at -737.13 cm (-290.21") and that it has an active fuel height of 365.76 cm (12 ft). The true bottom of active fuel elevation was observed to vary by as much as 2.58 cm (~1") depending upon the fuel assembly designs loaded in the reactor. For modeling purposes, a single bottom of active fuel elevation was assumed for each cycle based on the dominant fuel assembly design loaded on the core periphery. The model also accounted for the insert types loaded in the reactor for each cycle. Burnable poison rod assemblies (BPRA) and primary sources (PS) only appeared in the interior fuel assemblies of cycle 1. Two or four secondary sources (SS) appear in each cycle in the interior fuel assemblies. BPRA, PS, and SS rodlets extend downward into the assemblies and nearly reach the bottom of active fuel. Thimble plug (TP) rodlets only extend downward into the fuel plenum region. BPRAs, PSs, and SSs all have the same insert hub that sits in the top nozzle of the assembly as the TPs. Modeling BPRAs, PSs, and SSs as TPs reasonably preserved the materials at the top boundary of the core while simplifying the model. Rod cluster control assemblies (RCCA) were positioned as fully withdrawn in the model, so that the RCCA rodlet end caps reside in the fuel plenum region.

Figures 4-2 and 4-3 illustrate the core baffle that surrounds the core region. The core baffle has a rectangular shape that follows the contour of the core region and is modeled with rectangular elements. [[

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4.2.3 The Core Bypass Model

Figures 4-2 and 4-3 illustrate the core bypass region that lies between the core baffle plates and core barrel. The bypass region is a complex region in which the inner rectangular boundary formed by core region geometry must transition to a cylindrical form to interface with the core barrel that bounds the outer diameter of the bypass region. The RAMA model uses a combination of rectangular and cylindrical bodies to model the bypass region. Figure 4-3 shows that the bypass region has alternating water regions and former plates in the axial dimension. The water regions and former plates are each modeled with the same basic geometry elements, but have different meshing specifications at each elevation based on whether coolant water or steel materials are present.

4.2.4 The Core Barrel Model

Figures 4-2 and 4-3 illustrate the core barrel structure that bounds the core bypass region. The core barrel is cylindrical in design and is modeled with pipe elements.

4.2.5 The Downcomer Region Model

Figures 4-2 and 4-3 illustrate the downcomer region that lies between the core barrel and the reactor pressure vessel. The downcomer region is basically cylindrical in design, but with some geometrical complexities created by the presence of neutron pads and specimen guides in the region. The majority of the downcomer region is modeled with pipe segments. The areas of the downcomer containing the neutron pads and specimen guides are modeled with the appropriate geometry elements to represent their design features and to preserve their radial, azimuthal and axial placement in the downcomer region. These structures are described further in the following sub-section.

4.2.6 The Neutron Pad and Specimen Guide Models

Section 3 describes that there are four neutron pads and six specimen guides / surveillance capsules installed in the Seabrook Station reactor. The four neutron pads are strategically positioned in the downcomer region of the reactor in order to shield the pressure vessel from the high radiation fields generated by the reactor core during reactor operation. There are two neutron pad designs installed in the reactor that differ only by their azimuthal spans. The larger design spans 35 degrees and the smaller design spans 32.5 degrees, for a difference of 2.5 degrees. The six surveillance capsules are installed on the outward sides of the neutron pads such that they are shielded from the neutron source in the same manner as the pressure vessel. There are two surveillance capsules installed on each of the 35-degree neutron pad designs and one surveillance capsule installed on each of the 32.5-degree designs.

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Figures 4-2 and 4-3 illustrate how the neutron pads and surveillance capsules are represented in the Seabrook Station reactor fluence model. The neutron pads are cylindrical in design and, as shown in Figure 4-2, are easily implemented in the cylindrical downcomer region with pipe segments. The specimen guides, which contain the surveillance capsules, are rectangular in design. [[

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The specimen guides also contain the surveillance capsules that are cylindrical in design. [[

]] The specimen guide model also includes a representation for the coolant that surrounds the guides on all sides.

Referring to Figure 4-2, it is shown that the specimen guides are modeled behind the neutron pad at the 58.5° and 61° azimuths. When symmetry is applied to the model, the 58.5° guide represents each of the specimen guides installed at 58.5°, 121.5°, 238.5° and 301.5°, while the 61° guide represents the specimen guides at 61° and 241°.

Referring to Figure 4-3, it is shown that the neutron pads and specimen guides are modeled at their correct axial positions relative to the core region. The neutron pads cover the full height of the core and the surveillance capsules cover about one-third of the core height.

4.2.7 The Reactor Pressure Vessel Model

Figures 4-2 and 4-3 illustrate that the reactor pressure vessel and vessel cladding lie outside the downcomer region and that each are cylindrical in design. Both are modeled with pipe elements. The cladding-pressure vessel interface is a key location for fluence calculations and is preserved in the model. This interface defines the inside surface (0T) for the pressure vessel base metal where the RPV fluence is reported in this fluence evaluation.

The axial view of the pressure vessel model illustrated in Figure 4-3 shows that the vessel diameters and wall thicknesses change at the elevation identified as Weld 103-121. The actual design has a taper (illustrated by the dashed lines) that accomplishes the transition from the thinner to the thicker walls. This model does not include the taper. Instead, the model represents the lower and upper walls, or shells, with a constant diameter and thickness over their full height. This model approximation allows the best-estimate fluence for the weld and the lower shell to be calculated. The approximation will, however, provide a conservative maximum fluence estimate for the upper shell where the inner wall is represented closer to the neutron source than is actual.

4.2.8 The Vessel Insulation and Inner Cavity Models

Figures 4-2 and 4-3 illustrate that the inner cavity region and vessel insulation outside the pressure vessel wall are each cylindrical in design and that they follow the contour of the pressure vessel wall. Both are modeled with pipe elements.

4.2.9 The Outer Cavity Model

Figures 4-2 and 4-3 illustrate that an outer cavity region lies between the vessel insulation and primary shield. The boundaries of the outer cavity region follow the contours of the vessel insulation on the inside surface and the primary shield liner plates on the outside surface. The outer cavity region is essentially cylindrical in design, but with some geometrical complexities. It is shown in the figures that the outer cavity region is perturbed by 1) the presence of ex-core neutron detectors, 2) the presence of reactor support columns, and 3) variations in region diameters created by the contour shapes of the neighboring insulation and liner plate structures. The majority of the outer cavity region is cylindrical in design and is modeled with pipe segments. The areas of the outer cavity containing the ex-core detector tubes and reactor support columns are modeled with the appropriate geometry elements to represent their design features and to preserve their radial, azimuthal and axial placement in the cavity region. These structures are described further in the following sub-sections.

4.2.10 The Ex-core Neutron Detector Model

The ex-core neutron detectors are particularly complex to model as they lie partially in the outer cavity region and partially in the primary shield. Figure 3-2 illustrates how the ex-core neutron detector tubes are recessed, or notched, into the primary shield and that they are positioned at 45° increments around the reactor. The detector tubes are cylindrical in shape and positioned slightly offset from the notch centerline. The detector tube is the primary component in this model and is appropriately represented with a pipe element. [[

]] Figure 4-2 shows how the detector tubes and notches are represented in the model. The notches and detector tubes span an axial height in the model corresponding to the height of the core region (not illustrated).

4.2.11 The Reactor Vessel Support Column Model

Figure 4-3 shows that the support columns rest on a shelf of the primary shield wall at four azimuthal locations around the reactor (not shown). Figure 4-3 further shows that the lower sections of the support columns extend a respectable distance into the core beltline region. For this reason, the support columns and associated hardware are represented in the model in order to approximate the back-scatter contribution of neutrons to the determination of neutron fluence through the pressure vessel wall. [[ ]]

4.2.12 The Primary Shield and Liner Plate Models

Figures 4-2 and 4-3 illustrate the primary shield (concrete) and liner plates that define the outer most regions of the fluence model. Both structures are basically cylindrical in design and are modeled with pipe segments. Their changing diameters, which are created by the ex-core neutron detectors and reactor support columns, are discussed in the previous sections. Note that the air region between the plates and the concrete is preserved in the model for purposes of editing surface fluence for the components.

4.2.13 Above Core and Below Core Component Models

Figure 4-3 includes illustrations of other components and regions that lie above and below the reactor core region. The predominant above-core components represented in the model include the structural components of the fuel assembly upper nozzle (previously discussed), the upper core plate, upper support column and control rods. The predominant below-core components represented in the model include the structural components of the fuel assembly lower nozzle (previously discussed), the lower core plate, the lower support column and the bottom core support plate. With minor exceptions, the components and regions above and below the reactor core region are extensions of the same basic configurations as the core region. Thus, these components and regions are modeled with the same geometrical elements used to model the elevations through the core beltline region as described in the previous paragraphs.

4.2.14 Summary of the Geometry Modeling Approach

To summarize the reactor modeling process, there are several key features of the RAMA code system that allow the reactor design to be accurately represented for RPV fluence evaluations. Following is a summary of some of the key features of the model.

- Rectangular, cylindrical and wedge bodies are mixed in the model in order to provide an accurate geometrical representation of the components and regions in the reactor.
- The reactor core and core baffle geometries are modeled with rectangular bodies to represent their actual shapes in the reactor. The fuel assemblies in the core region are also appropriately sub-meshed with additional rectangular bodies to represent the pin cell regions in the assemblies.
- A combination of rectangular and cylindrical bodies is used to describe the transition parts between the rectangular core/core baffle region and the cylindrical outer core regions (former plates and core bypass).
- The former plates are individually modeled in the core bypass region.
- Cylindrical and wedge bodies are used to model the components and regions that extend outward from the core region (core barrel, downcomer, RPV, etc.).

- The specimen guides and surveillance capsules are modeled at their correct radially, azimuthal and elevational positions behind the neutron pads in the downcomer region.
- The fuel rod plenum, fuel assembly top nozzle, and upper core plate above the core region appropriately include representations of the upper support column and control rod structures.
- The below core region includes appropriate representations for the fuel assembly bottom nozzle, lower core plate and lower support column zones.
- The ex-core neutron detectors are represented as cylinders and are located in the cavity region between the insulation and notched into the primary shield wall.
- The primary shield liner plates are represented as cylindrical bodies. The air gap that exists between the liner plates and primary shield is preserved in the model.
- The reactor support columns and lateral support beams (not shown in the figures) are represented with cylindrical bodies and arc elements in the model, as appropriate.

4.3 RAMA Calculation Parameters

The RAMA transport code uses a three-dimensional deterministic transport method to calculate the neutron flux. The accuracy of the transport method is based on a numerical integration technique that uses ray-tracing to characterize the geometry, anisotropy treatments to determine the directional flow of particles, and convergence parameters to determine the overall accuracy of the flux solution between iterates. The code allows the user to specify values for each of these parameters.

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The RAMA transport calculation employs a treatment for anisotropy that is based on a Legendre expansion of the scattering cross sections. By default, the RAMA transport calculation uses the maximum order of expansion that is available for each nuclide in the RAMA nuclear data library. For the actinide and zirconium nuclides, a P_5 expansion of the scattering cross sections is used. For all other nuclides, a P_7 expansion of the scattering cross sections is used.

The overall accuracy of the neutron flux calculation is determined using an iterative technique to converge the flux iterations. [[

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4.4 RAMA Neutron Source Calculation

RAMA calculates a unique neutron source distribution for each transport calculation using the input relative power density factors for the fuel region and data from the RAMA nuclear data library. The source distribution changes with fuel burnup; thus, the source is determined using core-specific three-dimensional burnup distributions at frequent intervals throughout a cycle. For the fluence model, the peripheral fuel assemblies are modeled to preserve the power gradient at the core edge that is formed from the pin-wise source distributions in these fuel assemblies.

4.5 RAMA Fission Spectra

RAMA calculates a unique weighted fission spectrum for each transport calculation that is based on the relative contributions of the ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu isotopes. The fission spectra for these isotopes is derived from the BUGLE-96 nuclear data library.

5

SURVEILLANCE CAPSULE ACTIVATION AND FLUENCE RESULTS

This section documents the validation and qualification of the application of the RAMA Fluence Methodology to the Seabrook Station reactor in accordance with the requirements of U. S. NRC Regulatory Guide 1.190 (Reg. Guide 1.190). Reg. Guide 1.190 requires fluence calculational methods to be validated by comparison with measurements from operating reactor dosimetry for the specific plant being analyzed or for reactors of similar design. In addition, comparisons to established fluence evaluation benchmarks must be performed to confirm the calculational adequacy of the calculational method.

5.1 Plant-specific Surveillance Capsule Evaluations

This section presents an evaluation of the Seabrook Station surveillance program. In this evaluation, RAMA-calculated activities are compared to activation measurements for various dosimetry irradiated in the Seabrook Station reactor. At the time of this analysis, measurements for three surveillance capsules removed from the reactor had been performed. The capsules are identified in Table 5-1 as U, Y and V. Table 5-1 also provides the azimuthal location where the capsules were irradiated in the reactor and the length of the irradiation period.

Table 5-1
Seabrook Station Surveillance Capsules Analyzed

Capsule ID	Azimuthal Location	Cycles Completed at Capsule Removal	Irradiation in EFPY at Capsule Removal
U	58.5°	1	0.9
Y	241°	5	5.6
V	61°	10	12.4

Figure 3-2 in Section 3 shows the design radial location of the capsules installed in Seabrook Station. Figures 4-2 and 4-3 in Section 4 show the radial, azimuthal and axial positioning of the capsules in the RAMA fluence model. Figure 4-2 shows that Seabrook Station is modeled in azimuthal quadrant symmetry. When symmetry conditions are applied, the capsules shown as 58.5 and 61 degrees provide a representation for all capsules in the reactor. The capsules are positioned at their correct elevations and lengths relative to the reactor core, as illustrated in Figure 4-3. This modeling approach allows the dosimetry that is axially distributed throughout the capsules to be evaluated at their correct elevations in the capsules.

Each of the capsules in Seabrook Station are shielded from the reactor core (the neutron source) by the neutron pads. The neutron pads and surveillance capsules are modeled at their correct azimuthal positions in relation to each other in order to capture the shielding effects of the neutron pads on the activation measurements.

The activation dosimetry in the Seabrook Station surveillance capsules include copper, iron, nickel, and cobalt alloy flux wires located in top, middle, and bottom positions of each capsule. In addition, uranium and neptunium fission powder containers are positioned in the middle of each capsule.

Activation measurements were performed following irradiation for the following reactions [11,12,13]: $^{63}Cu(n,\alpha)^{60}Co$, $^{54}Fe(n,p)^{54}Mn$, $^{58}Ni(n,p)^{58}Co$, $^{59}Co(n,\gamma)^{60}Co$, ^{238}U(n,fission)^{137}Cs, and ^{237}Np(n,fission)^{137}Cs. Activation calculations were performed with RAMA for all dosimetry except the $^{59}Co(n,\gamma)$ reaction, which is primarily a thermal neutron reaction and not applicable to the fast neutron (>1.0 MeV) fluence evaluation documented in this report.

Table 5-2 summarizes the results of the Seabrook Station plant-specific surveillance capsule comparisons. [[

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Table 5-2
Summary of Seabrook Station Plant-specific Surveillance Capsule Measurement Comparisons

Seabrook Station Surveillance Capsule	Number of Measurements	Calculated vs. Measured (C/M)	Standard Deviation
Capsule U (irradiated one cycle)	11	[[	[[
Capsule Y (irradiated five cycles)	10		
Capsule V (irradiated ten cycles)	11		
Total Plant-Specific Comparisons	**32**	**]]**	**]]**

5.1.1 Cycle 1 Surveillance Capsule U Activation Analysis

Activation dosimetry was irradiated in the Seabrook Station Capsule U during the first cycle of operation. The items were removed after being irradiated for a total of 0.9 EFPY.

The predicted ^{137}Cs activity in the fission dosimetry must be corrected to account for photofissions, as well as fissions resulting from ^{235}U impurity and plutonium build-up in the uranium sample. [[

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5.1.2 Cycle 5 Surveillance Capsule Y Activation Analysis

Activation dosimetry was irradiated in the Seabrook Station Capsule Y during the first five cycles of operation. The activation specimens were removed after being irradiated for a total of 5.6 EFPY.

The predicted ^{137}Cs activity in the fission dosimetry must be corrected to account for photofissions, as well as fissions resulting from ^{235}U impurity and plutonium build-up in the uranium sample. [[

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5.1.3 Cycle 10 Surveillance Capsule V Activation Analysis

Activation dosimetry was irradiated in the Seabrook Station Capsule V during the first ten cycles of operation. The items were removed after being irradiated for a total of 12.4 EFPY.

The predicted ^{137}Cs activity in the fission dosimetry must be corrected to account for photofissions, as well as fissions resulting from ^{235}U impurity and plutonium build-up in the uranium sample. [[

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6
REACTOR PRESSURE VESSEL UNCERTAINTY ANALYSIS

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7
CALCULATED REACTOR PRESSURE VESSEL NEUTRON FLUENCE FOR ENERGY >1.0 MeV

This section presents the predicted best-estimate fast neutron fluence with energy >1.0 MeV for the Seabrook Station reactor pressure vessel (RPV) at 55 EFPY. It is reported in Section 6, *Reactor Pressure Vessel Uncertainty Analysis*, that the RAMA-calculated pressure vessel fluence for the Seabrook Station reactor requires no bias adjustment; therefore, the best-estimate fluence is the calculated fluence that was predicted with the RAMA Fluence Methodology.

The reactor pressure vessel fluence reported in this section was determined at the interface of the RPV base metal and cladding, hereafter denoted as the 0T location of the RPV wall. Fluence is presented for the RPV horizontal welds, vertical welds, shells and nozzles residing in the RPV beltline region. The location and identification of the RPV shells and welds are shown in Figure 7-1. Also illustrated in Figure 7-1 are the core beltline region (the elevational range from the bottom of active fuel to the top of active fuel) and the calculated RPV beltline region for the Seabrook Station reactor. While the drawing is not precisely to scale, the drawing does provide a perspective of the relative positioning of these regions to the welds of interest in this evaluation.

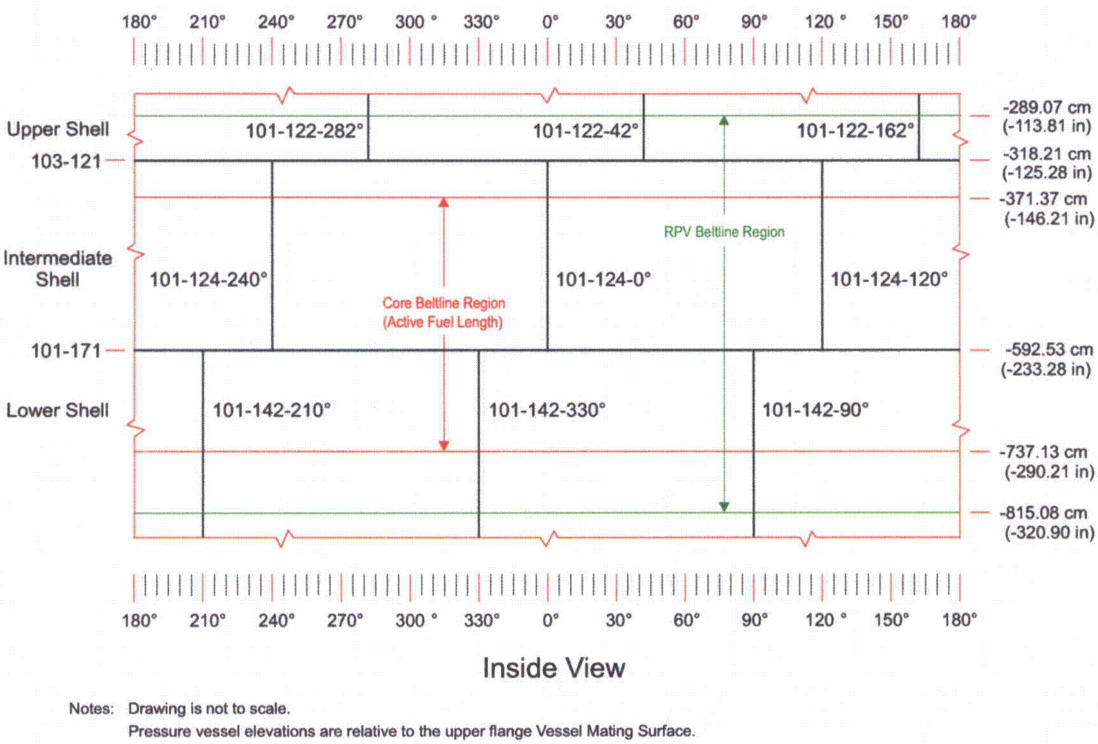

Figure 7-1
Seabrook Station RPV Shell and Weld Location Identifiers

Tables 7-1 and 7-2 report the maximum best-estimate >1.0 MeV neutron fluence at 55 EFPY for the RPV circumferential welds and vertical welds in the RPV beltline region, respectively. RAMA predicted that the maximum fast fluence for the RPV welds at 0T is in weld 101-171 with a value of 3.59E+19 n/cm^2. The calculations further showed that each of the circumferential and vertical welds at 0T in the RPV beltline region exceeded the threshold fluence value of 1.0E+17 n/cm^2 prior to 55 EFPY.

Table 7-1
Maximum >1.0 MeV Neutron Fluence in Seabrook Station RPV Circumferential Welds at 55 EFPY

Weld	Azimuth (Degrees)	Elevation [cm (in)]	Best-Estimate Fluence (n/cm^2) at 0T
101-171	22, 158, 202, 338	-592.53 (-233.28)	3.59E+19
103-121	41, 139, 221, 319	-318.21 (-125.28)	8.22E+17

Table 7-2
Maximum >1.0 MeV Neutron Fluence in Seabrook Station RPV Vertical Welds at 55 EFPY

Weld	Azimuth (Degrees)	Elevation [cm (in)]	Best-Estimate Fluence (n/cm^2) at 0T
101-142-90°	90	-595.07 (-234.28)	2.05E+19
101-142-210°	210	-595.07 (-234.28)	3.46E+19
101-142-330°	330	-595.07 (-234.28)	3.46E+19
101-124-0°	0	-490.93 (-193.28)	2.06E+19
101-124-120°	120	-475.69 (-187.28)	2.40E+19
101-124-240°	240	-475.69 (-187.28)	2.40E+19
101-122-42°	42	-318.21 (-125.28)	8.88E+17
101-122-162°	162	-318.21 (-125.28)	6.09E+17
101-122-282°	282	-318.21 (-125.28)	5.24E+17

Table 7-3 reports the maximum best-estimate >1.0 MeV neutron fluence at 55 EFPY for the RPV shells residing in the RPV beltline region. RAMA predicted that the maximum fluence for the RPV shells at 0T is in the intermediate shell with a value of 3.63E+19 n/cm^2. It is shown in Table 7-3 that the maximum fluence in each of the shells exceeds the threshold fluence value of 1.0E+17 n/cm^2 prior to 55 EFPY. It is noted here that the upper shell of the RPV is thicker than the intermediate shell, and that the actual design has a tapering region that transitions between the thinner and thicker shells. The model does not include this taper. Instead, the model represents each shell with a constant diameter over its full height. This results in a discrepancy between the upper shell maximum fluence and the maximum fluence for the circumferential weld (Weld I.D. 103-121) that exists between the upper and intermediate shells. The maximum fluence for the upper shell and weld occur at the same elevation but, as illustrated in Figure 7-2, the upper shell fluence comes from the inner surface of the upper shell, as modeled, while the weld fluence is taken from the inner surface of the intermediate shell. Use of the maximum fluence reported for the upper shell will provide a conservative estimate for the fluence in the upper shell areas.

Based on a fluence threshold of 1.0E+17 n/cm^2, the RPV beltline is determined to span from a lower elevation of -815.08 cm (-320.90 inches) to an upper elevation of -289.07 cm (-113.81 inches) at 55 EFPY. Based on the RAMA calculations, it was also determined that the RPV nozzle fluence is less than 1.0E+17 n/cm^2 at 55 EFPY.

Table 7-3
Maximum >1.0 MeV Neutron Fluence in Seabrook Station RPV Shells at 55 EFPY

Shell	Azimuth (Degrees)	Elevation [cm (in)]	Best-Estimate Fluence (n/cm^2) at 0T
Lower	22	-594.97 (-234.24)	3.59E+19
Intermediate	22	-490.87 (-193.26)	3.63E+19
Upper	43	-318.21 (-125.28)	8.92E+17

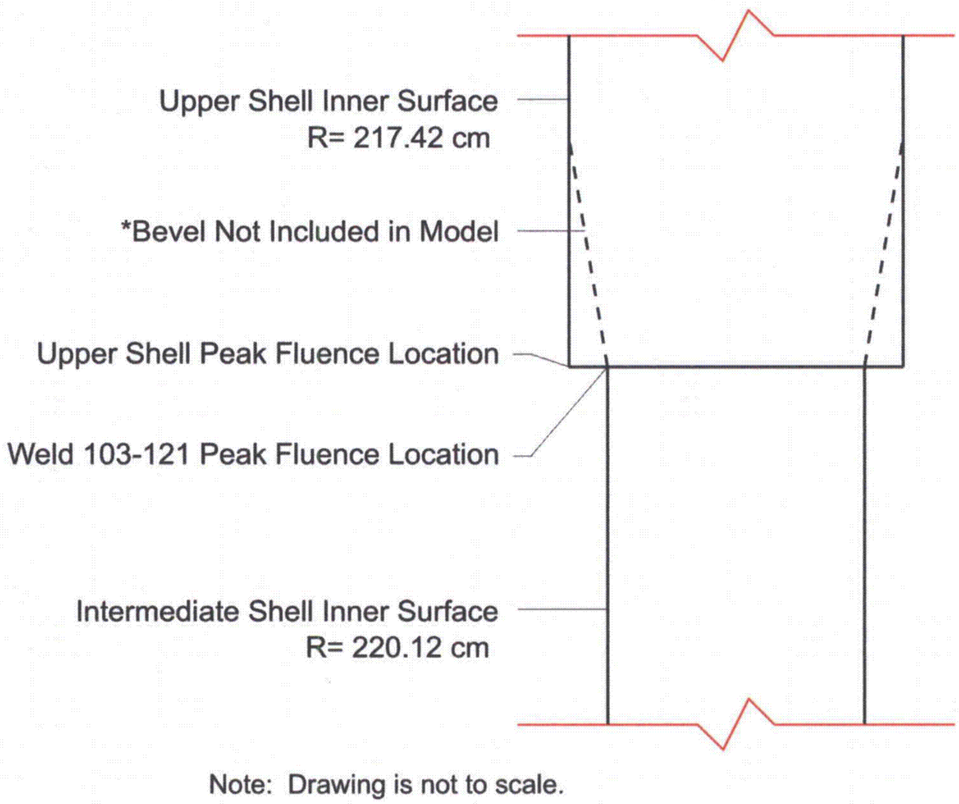

Figure 7-2
Axial View of the Seabrook Station RPV Model Around the Upper Circumferential Weld (103-121)

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