Enclosure 3

WCAP-18124-NP-A Revision 0 Supplement 1-NP, “Fluence Determination with RAPTOR-M3G and FERRET – Supplement for Extended Beltline Materials”

(Non-Proprietary)

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Fluence Determination with RAPTOR-M3G and FERRET – Supplement for Extended Beltline Materials

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ABSTRACT

This supplement is intended to provide the justification necessary to narrow Limitation #1 of WCAP-18124-NP-A and allow licensees to apply the RAPTOR-M3G method for neutron fluence determination in the extended beltline regions of reactor pressure vessels on a generic basis.

This report identifies additional requirements for applying the WCAP-18124-NP-A methodology for neutron fluence determination in the extended beltline region and describes the methodology qualification that has been performed. The qualification process includes an analytic uncertainty analysis, comparisons to alternate calculations performed with a Monte Carlo technique, comparisons of RAPTOR-M3G and FERRET calculations to measurement data collected from the extended beltline region, and estimates of methodology bias and uncertainty. The information provided in this supplement satisfies Regulatory Guide 1.190 for extended beltline region neutron fluence determination with RAPTOR-M3G and FERRET.

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1 INTRODUCTION

Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," describes methods and assumptions acceptable to the NRC staff for determining the RPV neutron fluence with respect to the General Design Criteria (GDC) contained in Appendix A of Title 10 of the Code of Federal Regulations (10 CFR) Part 50. In consideration of the guidance set forth in Regulatory Guide 1.190, GDC 14, 30, and 31 are applicable. GDC 14, "Reactor Coolant Pressure Boundary," requires the design fabrication, erection, and testing of the reactor coolant pressure boundary so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. GDC 30, "Quality of Reactor Coolant Pressure Boundary," requires, among other things, that components comprising the reactor coolant pressure boundary be designed, fabricated, erected, and tested to the highest quality standards practical. GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," pertains to the design of the reactor coolant pressure boundary.

To assess embrittlement of Light Water Reactor (LWR) pressure vessels, an accurate evaluation of the neutron fluence in materials that are predicted to experience significant radiation damage is required. Historically, only those materials directly adjacent to the active core, commonly referred to as "beltline" or "traditional beltline" materials, have been evaluated with respect to reactor pressure vessel (RPV) integrity. However, Regulatory Issue Summary (RIS) 2014-11 (Reference 1) states that any materials predicted to exceed 1.0×10^{17} n/cm^2 (E > 1.0 MeV) at the end of their licensed operating period must be evaluated to determine the changes in fracture toughness. Materials that are not adjacent to the active core yet are predicted to accrue fluence levels greater than 1.0×10^{17} n/cm^2 (E > 1.0 MeV) are now commonly referred to as "extended beltline" materials.

WCAP-18124-NP-A (Reference 2) has been approved by the United States Nuclear Regulatory Commission (NRC) for neutron fluence determination in accordance with Regulatory Guide 1.190 (Reference 3) using the RAPTOR-M3G and FERRET computer codes. The NRC identified two "Limitations and Conditions" associated with the application of RAPTOR-M3G and FERRET. Limitation #1 is repeated below:

> *Applicability of WCAP-18124-NP, Revision 0 is limited to the RPV region near the active height of the core based on the uncertainty analysis performed and the measurement data provided. Additional justification should be provided via additional benchmarking, fluence sensitivity analysis to the response parameters of interest (e.g. pressure-temperature limits, material stress/strain), margin assessment, or a combination thereof, for applications of the method to components including, but not limited to, the RPV upper circumferential weld and the reactor coolant system inlet and outlet nozzles and reactor vessel internal components.*

The objective of this supplement to WCAP-18124-NP-A is to provide the justification necessary to narrow Limitation #1 and allow licensees to apply the RAPTOR-M3G method in the extended beltline regions of RPVs on a generic basis.

Consistent with Section 4 of WCAP-18124-NP-A, this report satisfies Regulatory Position 1.4 of Regulatory Guide 1.190 for the extended beltline region of LWRs, providing methodology qualification and uncertainty estimates for the RAPTOR-M3G and FERRET method. Section 2 articulates the additional requirements necessary to apply the WCAP-18124-NP-A methodology for extended beltline region fluence determination. Section 3 describes an analytic uncertainty analysis performed for the extended beltline region. Section 4 describes extended beltline region comparisons between RAPTOR-M3G calculations and benchmark calculations performed with a Monte Carlo technique. Section 5 describes comparisons between RAPTOR-M3G calculations and benchmark measurements obtained from the extended beltline region. Section 6 provides estimated bias and uncertainty values applicable to the RAPTOR-M3G and FERRET methodology for extended beltline region fluence determination. Section 7 summarizes the compliance of the RAPTOR-M3G and FERRET method with Regulatory Position 1, describes the condition for applying the method in the extended beltline region, and provides a proposal for revised wording of Limitation #1.

2 APPLICATION REQUIREMENTS FOR EXTENDED BELTLINE FLUENCE DETERMINATION

This section describes the additional requirements necessary to apply the WCAP-18124-NP-A methodology for extended beltline region fluence analyses.

2.1 ANGULAR QUADRATURE

RAPTOR-M3G calculations for extended beltline fluence analyses are performed with S_{16} (or higher) level-symmetric angular quadrature sets. This requirement is more restrictive than the angular quadrature requirement in WCAP-18124-NP-A.

This application requirement is consistent with guidance in Regulatory Positions 1.3.1 and 1.3.5 of Regulatory Guide 1.190, which states that a higher order quadrature may be needed in reactor cavity fluence calculations. Extended beltline region benchmarking described in Section 3 through Section 5 of this supplement employ S_{16} level-symmetric angular quadrature sets.

2.2 SPATIAL DIFFERENCING

RAPTOR-M3G calculations for extended beltline fluence analyses are performed with the theta-weighted (TW) spatial differencing scheme described in WCAP-18124-NP-A. The directional theta-weighted (DTW) spatial differencing scheme that was also approved in WCAP-18124-NP-A will not be used for extended beltline region fluence determination as described in this Supplement. This requirement is more restrictive than the spatial differencing requirement in WCAP-18124-NP-A.

Extended beltline region benchmarking described in Section 3 through Section 5 of this supplement employ the TW spatial differencing scheme. This application requirement is consistent with the prior approval of WCAP-18124-NP-A.

2.3 MEASUREMENT UNCERTAINTY FOR NIOBIUM SENSORS

In Section 3.4 of WCAP-18124-NP-A, the uncertainty applicable to typical niobium sensor measurements is increased to from 5% to 10%. This change is based on the increased uncertainties associated with the liquid scintillation measurement process used for niobium sensors as compared to the gamma spectroscopy measurement process applied for other non-fission sensors. This change has no impact on the previous results but is noted here for completeness.

3 ANALYTIC UNCERTAINTY ANALYSIS

This section describes an analytic uncertainty analysis that was performed to assess the effects of input and methodological uncertainties on calculated neutron fluence (E > 1.0 MeV) results in the extended beltline region. These comparisons address Regulatory Position 1.4.1 of Regulatory Guide 1.190.

This analytic uncertainty analysis is comprised of parameter variations that were previously considered in WCAP-18124-NP-A combined with several new parameters (discussed below) that are significant for extended beltline region fluence analyses.

The parameter variations from WCAP-18124-NP-A include the Core Neutron Source Uncertainties, (comprised of Peripheral Assembly Source Strength, Pin Power Distribution, Peripheral Assembly Burnup, and Axial Power Distribution components) and the Geometric and Temperature Uncertainties (comprised of Internals Dimensions, Vessel Inner Radius, Vessel Thickness, Coolant Temperature, and Core Periphery Modeling components). The magnitudes of these parameter variations are unchanged from WCAP-18124-NP-A and are fully described therein.

The new parameters that have been considered in this analytic uncertainty analysis for the extended beltline region are as follows:

[

]^{a,c}.

- [

]a,c.

Results of the analytic uncertainty analysis are shown in Table 3-1 through Table 3-4. Results applicable to the inner radius of the RPV are shown in Table 3-1 and Table 3-2, while results applicable to the outer radius of the RPV are shown in Table 3-3 and Table 3-4. [

]a,c.

The results shown in Table 3-1 through Table 3-4 represent the maximum difference observed between the base case and the permutation cases for each elevation. Each elevation examined a span of azimuthal angles at increments of 5° or less. Amongst the azimuthal angles that were considered for each elevation, the median differences that were observed were lower than the reported maximum difference. Therefore, the analytic uncertainty estimates provided in Table 3-1 through Table 3-4 include a degree of conservatism.

The results in Table 3-1 through Table 3-4 show net analytic uncertainty values that span from [

]a,c

[
]a,c.

It is important to emphasize that the results presented in Table 3-1 through Table 3-4 are *estimates* of the analytic uncertainty associated with each parameter. The results exhibit a degree of sensitivity to the specific details of the models that were considered. [

]a,c.

The results of this analytic uncertainty analysis are combined with other uncertainty components to arrive at an estimate of the RAPTOR-M3G methodology uncertainty in the extended beltline region in Section 6. This analytic uncertainty analysis addresses Regulatory Position 1.4.1 of Regulatory Guide 1.190 for the extended beltline region.

Table 3-1: Uncertainty Analysis Results - [ ]a,c

a,c

Table 3-2: Uncertainty Analysis Results - [ ]a,c

a,c

Table 3-3: Uncertainty Analysis Results - [ ]a,c

a,c

Table 3-4: Uncertainty Analysis Results - [ ]a,c

a,c

4 MONTE CARLO CALCULATIONAL BENCHMARK COMPARISONS

Regulatory Position 1.4.2.3 of Regulatory Guide 1.190 states that the calculational benchmarks in NUREG/CR-6115 should be used for methods qualification. However, the benchmarks in NUREG/CR-6115 do not include materials in the RPV extended beltline and, as such, are not useful for qualifying the WCAP-18124-NP-A methodology for extended beltline applications described in this supplement. [

]^{a,c}.

[

]a,c. These comparisons are consistent with the guidance contained in Regulatory Position 1.4.2.3 of Regulatory Guide 1.190.

a,c

Figure 4-1: [ ]a,c

a,c

Figure 4-2: [ ]a,c

a,c

Figure 4-3: [ ]a,c

a,c

Figure 4-4: [ ]a,c

a,c

Figure 4-5: [

]a,c

a,c

Figure 4-6: [ ]a,c

Note: [ ]a,c.

5 MEASUREMENT DATA BENCHMARK COMPARISONS

Regulatory Guide 1.190 recommends that methods qualification should be based on comparisons with measurement data from operating power reactors. This section provides the comparisons of power reactor measurements from the extended beltline region to calculations performed with RAPTOR-M3G and FERRET. These comparisons address the portions of Regulatory Position 1.4.2.1 of Regulatory Guide 1.190 that relate to operating reactor measurements.

5.1 IN-VESSEL MEASUREMENTS

This subsection describes a retrospective dosimetry analysis of stainless steel samples harvested from scrap surveillance capsule material from two different pressurized water reactors (PWRs).

The material samples were extracted from the top support plugs, which are structural components of surveillance capsules that extend between 1 and 2 feet above the top of the core and are attached to the thermal shield outside the core barrel. Thus, the top support plugs are ideally located between the core and the materials of interest in the RPV extended beltline region. Figure 5-1 shows the relative locations of the active fuel, the top support plugs, and the reactor vessel nozzles.

Four material samples from each capsule, each weighing approximately 100 mg, were produced using a milling machine in the high-level hot cell. Due to the length of time that elapsed between the end of irradiation and the laboratory measurement of the samples, amongst the reactions that are primarily sensitive to fast neutrons, only the Mn-54 reaction product from the Fe-54 (n,p) reaction was recoverable.

Calculated reaction rates were obtained with the methodology described in WCAP-18124-NP-A employing the TW spatial differencing scheme. In these calculations, the angular discretization was modeled with an S_{16} level-symmetric angular quadrature set.

The RAPTOR-M3G calculations produced a detailed neutron spectrum at the measurement locations for each fuel cycle in the BUGLE-96 group structure. Response functions applicable to the Fe-54 (n,p) reaction were obtained from the BUGLE-96 library and applied to the transport calculation results, and time-averaged reaction rates were calculated. These time-averaged calculated reaction rates were used as the basis for comparisons to the measured reaction rates.

Results of the comparisons are shown in Table 5-1 and depicted in Figure 5-2 and Figure 5-3.

Note that there are larger uncertainties associated with retrospective dosimetry comparisons than with dedicated reactor dosimetry measurements. In this case, the exact irradiation location of the material samples is not well-known, the geometric modeling of the surrounding support structure is limited, and the measurement collection process imparted an additional degree of uncertainty. Nonetheless, most of the measurements agree with the calculations within the 20% range defined in Regulatory Guide 1.190 as being acceptable for RT_{PTS} and RT_{NDT} determination – even those measurements collected from locations that are distant from the top of the core.

Table 5-1: Top Support Plug Retrospective Dosimetry Measured-to-Calculated (M/C) Reaction Rate Ratios

Plant / Sample Number	Location Relative to Core Top (cm)	Reaction Rate M/C
Plant #1 / Sample #1	56	0.89
Plant #1 / Sample #2	49	0.83
Plant #1 / Sample #3	43	0.76
Plant #1 / Sample #4	36	0.84
Plant #2 / Sample #1	27	1.05
Plant #2 / Sample #2	22	1.00
Plant #2 / Sample #3	16	0.89
Plant #2 / Sample #4	10	0.67
	Average	**0.86**
	Std. Dev. %	**14%**

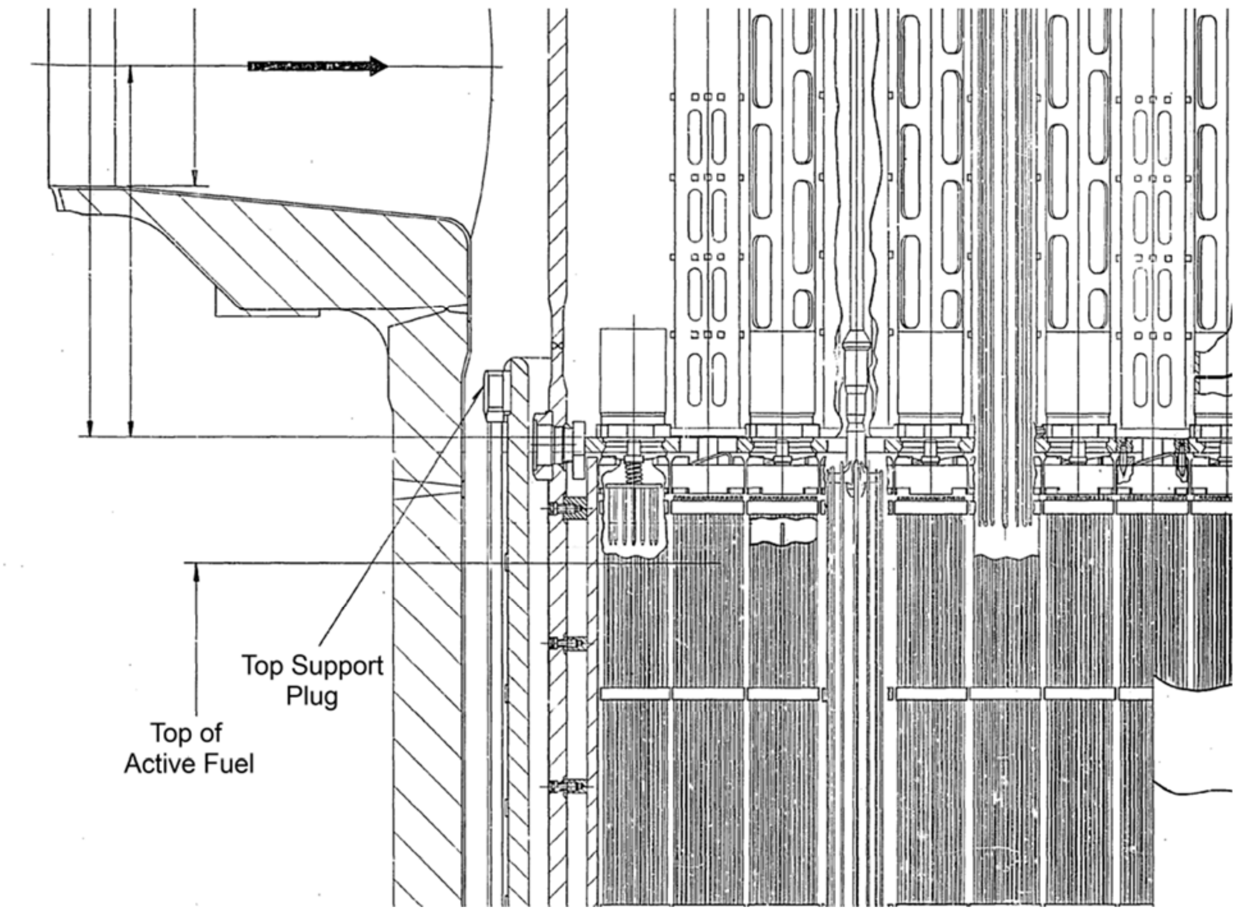

Figure 5-1: Top Support Plug Location Relative to the Top of the Active Fuel and the Reactor Vessel Nozzles

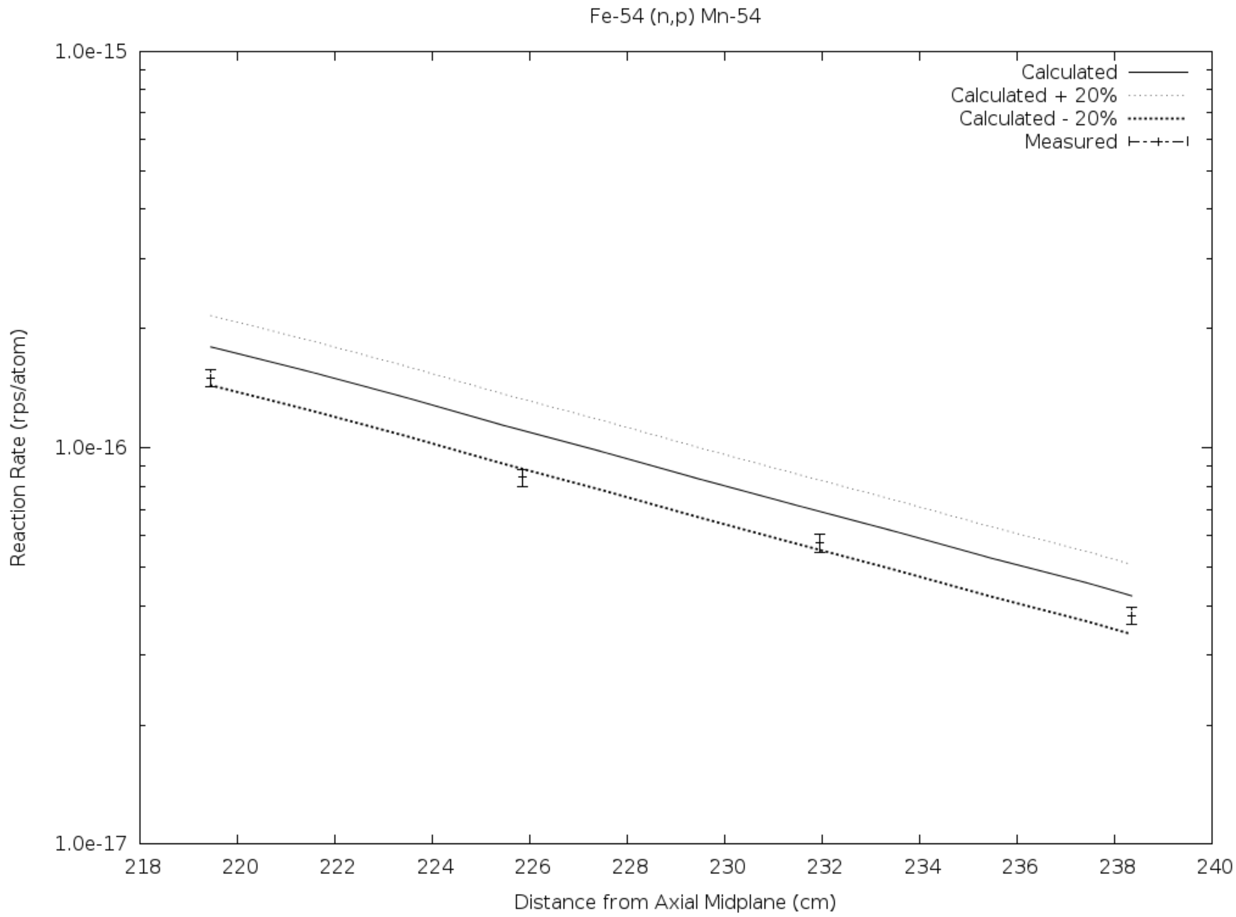

Figure 5-2: Comparison of Top Support Plug Measurements with Calculations from Plant #1

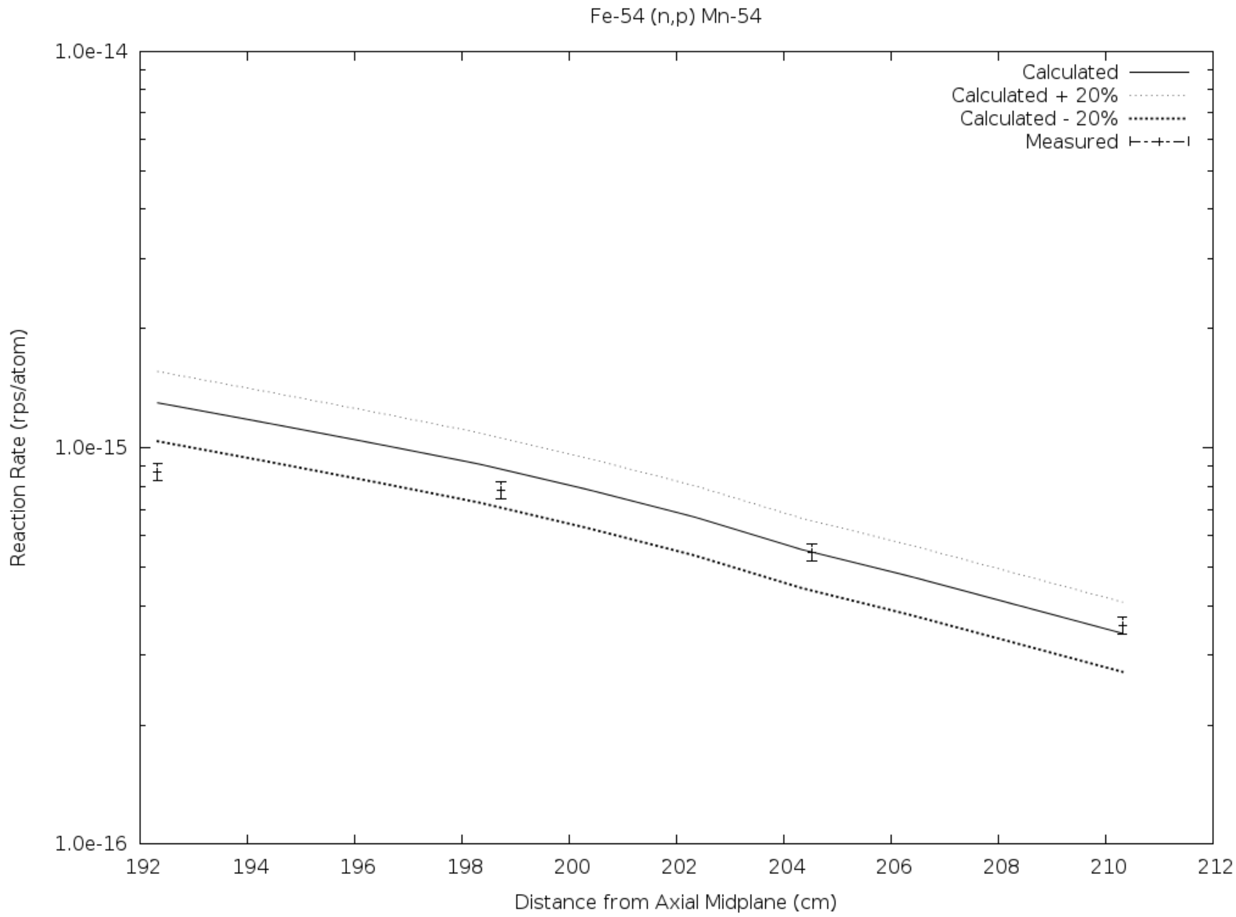

Figure 5-3: Comparison of Top Support Plug Measurements with Calculations from Plant #2

5.2 EX-VESSEL MEASUREMENTS

This subsection describes benchmarking of the RAPTOR-M3G methodology against ex-vessel neutron dosimetry measurements collected from the extended beltline region of a 4-Loop PWR. The presentation of results is limited to the dosimetry data that is applicable to the extended beltline region.

5.2.1 Description of Ex-Vessel Neutron Dosimetry

Comprehensive multiple foil sensor sets were installed at several locations in the reactor cavity of a 4-Loop PWR to characterize the neutron environment within the beltline and extended beltline regions of the pressure vessel.

The placement of the individual multiple foil sensor sets within the reactor cavity is illustrated in Figure 5-4 and Figure 5-5. Information on the measured neutron reactions including neutron energy response and product half-life, as well as the positions of the dosimetry materials in the capsules, are provided in Table 5-2. Locations of the individual capsules comprising each of the three sets of dosimetry withdrawn to date are summarized in Table 5-3.

In Figure 5-4, a plan view of the azimuthal locations of the four strings is depicted. In Figure 5-5, the axial extent of each of the sensor set strings is illustrated along with the locations of the multiple foil holders. At the 135-degree and 180-degree locations, multiple foil sets were positioned opposite the core midplane, opposite the top and bottom of the active fuel stack, in the vicinity of the RPV supports, as well as near the RPV lower shell to bottom head girth weld. The RPV support capsules are located at an elevation approximately 75 cm above the top of the core. The lower shell to bottom head girth weld capsules are located at an elevation approximately 125 cm below the bottom of the core.

5.2.2 Method of Analysis

The transport calculations were carried out using the RAPTOR-M3G three-dimensional discrete ordinates code and the BUGLE-96 cross-section library. The BUGLE-96 library provides a 67-group coupled neutron-gamma ray group cross-section data set produced specifically for LWR applications. In these analyses, anisotropic scattering was treated with a P_3 Legendre expansion and the angular discretization was modeled with an S_{16} level-symmetric angular quadrature set.

A plan view of the reactor model is shown in Figure 5-6. In addition to the core, reactor internals, RPV, and concrete bioshield, the model also included explicit representations of the surveillance capsules, RPV clad as well as the RPV nozzles and supports. Section views of the reactor model are shown in Figure 5-7 and Figure 5-8.

In developing the model of the reactor geometry, nominal design dimensions were used for the various structural components. Water temperatures (and densities) in the core, bypass, and downcomer regions of the reactor were taken to be representative of full-power operating conditions. These coolant temperatures were varied on a cycle-specific basis. The reactor core itself was treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel assembly grids, guide tubes, etc.

The *r,θ,z* geometric mesh description of the reactor model consisted of 233 radial by 186 azimuthal by 469 axial mesh intervals. Mesh sizes were chosen to ensure that proper convergence of the inner iterations was achieved on a pointwise basis. The pointwise inner iteration flux convergence criterion used in the calculations was 0.001.

The core power distributions used in the plant-specific transport analysis included fuel-assembly-specific initial enrichments, burnups, and axial power distributions. This information was used to develop spatial- and energy-dependent core source distributions averaged over the fuel cycle. Therefore, the results from the neutron transport calculations provided data in terms of the fuel cycle-averaged neutron exposure rate, which, when multiplied by the appropriate fuel cycle length, provide the incremental fast neutron exposure for the fuel cycle. The energy distribution of the source was based on an appropriate fission split for uranium and plutonium isotopes based on the initial ^{235}U enrichment and burnup history of the individual fuel assemblies. From the assembly-dependent fission splits, composite values of energy release per fission, neutron yield per fission, and fission spectrum were determined. These fuel-assembly-specific neutron source strengths derived from the detailed isotopics were then converted from fuel pin Cartesian coordinates to the *r,θ,z* spatial mesh arrays used in the RAPTOR-M3G discrete ordinates calculations.

5.2.3 Dosimetry Analysis Results

In Table 5-4, comparisons of measured and calculated reaction rates are listed for the sensors contained in five sensor sets withdrawn from measurement locations in the vicinity of the RPV supports. For the individual threshold foils, the average measured-to-calculated (M/C) ratio ranges from 0.60 to 1.01 with an overall average of 0.75 and a standard deviation of 28%.

In Table 5-5, similar comparisons are provided for the two sensor sets withdrawn from the measurement locations near the lower shell to bottom held girth weld. For the individual threshold foils, the average M/C ratio ranges from 0.94 to 1.16 with an overall average of 1.02 and a standard deviation of 14%.

In Table 5-4 and Table 5-5, the overall average was based on an equal weighting of the sensor types with no account taken of the spectral coverage of the individual sensors.

In Table 5-6 and Table 5-7, best-estimate-to-calculated (BE/C) ratios for neutron (E > 1.0 MeV) fluence rate and iron atom displacement rate resulting from the least-squares evaluation of each sensor set are provided for the extended beltline measurement locations. For the sensor set locations in the vicinity of the RPV supports, the average BE/C ratio is 0.79 with a standard deviation of 19% for neutron (E > 1.0 MeV) fluence rate, and 0.88 with a standard deviation of 20% for iron atom displacement rate. The corresponding average BE/C ratios for the sensor set locations near the lower shell to bottom head girth weld are 0.95 with a standard deviation of 12% for neutron (E > 1.0 MeV) fluence rate, and 0.98 with a standard deviation of 14% for iron atom displacement rate.

The results provided in Table 5-4, Table 5-5, Table 5-6, and Table 5-7 indicate that the calculations are over-predicting the neutron exposure in the extended beltline region, in particular at the high end of the energy spectrum. For instance, the bottom of the 90% neutron response for the copper, titanium, iron, and nickel dosimeters is 4.53 MeV, 3.70 MeV, 2.27 MeV, and 1.98 MeV, respectively. Neutrons with this much energy constitute a small fraction of the neutron (E > 1.0 MeV) fluence rate in the extended beltline region. The BE/C values in Table 5-6 and Table 5-7 account for the spectral coverage of the different sensors, and provide an estimate of the key damage parameters, fast neutron ($E > 1.0$ MeV) fluence rate and iron atom displacement rate, that results from a spectrum- and uncertainty-weighted reconciliation of all of the measurements and calculations. The BE/C values in Table 5-6 and Table 5-7 suggest that the calculated damage parameters are moderately conservative relative to the best-estimate values.

Table 5-2: Measured Dosimetry Reactions

Material	Reaction of Interest	Neutron Energy Response(1)	Product Half-Life(2)	Dosimeter Capsule Position(3)
Copper	$^{63}Cu\ (n,\alpha)\ ^{60}Co$	4.53-11.0 MeV	5.271 y	2-Cd
Titanium	$^{46}Ti\ (n,p)\ ^{46}Sc$	3.70-9.43 MeV	83.788 d	2-Cd
Iron	$^{54}Fe\ (n,p)\ ^{54}Mn$	2.27-7.54 MeV	312.13 d	1-B & 2-Cd
Nickel	$^{58}Ni\ (n,p)\ ^{58}Co$	1.98-7.51 MeV	70.86 d	2-Cd
^{238}U	$^{238}U\ (n,f)\ ^{137}Cs$	1.44-6.69 MeV	30.07 y	3-Cd
Niobium	$^{93}Nb\ (n,n')\ ^{93m}Nb$	0.95-5.79 MeV	16.13 y	3-Cd
^{237}Np	$^{237}Np\ (n,f)\ ^{137}Cs$	0.68-5.61 MeV	30.07 y	3-Cd
Cobalt - Al	$^{59}Co\ (n,g)\ ^{60}Co$	Thermal	5.271 y	1-B & 2-Cd

Note(s):
(1) Energies between which 90% of activity is produced (^{235}U fission spectrum) per ASTM E844-18.
(2) Half-life data is from ASTM E1005-16.
(3) "B" denotes bare and "Cd" denotes cadmium-shielded.

Table 5-3: Multiple Foil Sensor Set Locations within the 4-Loop Reactor Cavity

Relative Azimuth (Degrees)	EVND Set 1		EVND Set 2		EVND Set 3	
	RPV Supports	Bottom Head Girth Weld	RPV Supports	Bottom Head Girth Weld	RPV Supports	Bottom Head Girth Weld
0	E	-	K	-	M	Q
45	A	-	-	-	R	V

Table 5-4: Measured-to-Calculated (M/C) Reaction Rates – 4-Loop Reactor Ex-Vessel Capsules Located in the Vicinity of the RPV Supports

Reaction	Capsule					Average	% Std. Dev.
	E	A	K	M	R		
^{63}Cu (n,α) ^{60}Co	0.68	-	0.60	0.60	0.53	0.60	10.2
^{46}Ti (n,p) ^{46}Sc	0.76	0.65	0.69	0.73	0.59	0.68	9.8
^{54}Fe (n,p) ^{54}Mn	0.76	0.67	0.66	0.63	0.49	0.64	15.2
^{58}Ni (n,p) ^{58}Co	0.78	0.68	0.68	0.74	0.54	0.68	13.3
^{238}U(Cd) (n,f) ^{137}Cs	1.09	0.92	-	-	-	1.01	12.0
^{93}Nb (n,n′) ^{93m}Nb	-	-	1.33	1.02	0.58	0.98	38.6
^{237}Np(Cd) (n,f) ^{137}Cs	1.18	0.82	-	-	-	1.00	25.5
Average of M/C Results						0.75	28

Table 5-5: Measured-to-Calculated (M/C) Reaction Rates – 4-Loop Reactor Ex-Vessel Capsules Located Opposite the Lower Shell to Bottom Head Girth Weld

Reaction	Capsule		Average	% Std. Dev.
	Q	V		
^{63}Cu (n,α) ^{60}Co	1.28	1.03	1.16	15.3
^{46}Ti (n,p) ^{46}Sc	1.11	0.94	1.03	11.7
^{54}Fe (n,p) ^{54}Mn	1.00	0.87	0.94	9.8
^{58}Ni (n,p) ^{58}Co	1.03	0.90	0.97	9.5
^{93}Nb (n,n′) ^{93m}Nb	1.20	0.88	1.04	21.8
Average of M/C Results			1.02	14

Table 5-6: Best-Estimate-to-Calculated (BE/C) Exposure Rates – 4-Loop Reactor Ex-Vessel Capsules Located in the Vicinity of the RPV Supports

Capsule	Neutron (E > 1.0 MeV) Fluence Rate BE/C	Iron Atom Displacement Rate BE/C
E	0.95	1.03
A	0.78	0.84
K	0.87	1.03
M	0.82	0.91
R	0.55	0.61
Average	0.79	0.88
% Std. Dev.	19	20

Table 5-7: Best-Estimate-to-Calculated (BE/C) Exposure Rates – 4-Loop Reactor Ex-Vessel Capsules Located Opposite the Lower Shell to Bottom Head Girth Weld

Capsule	Neutron (E > 1.0 MeV) Fluence Rate BE/C	Iron Atom Displacement Rate BE/C
Q	1.04	1.07
V	0.87	0.88
Average	0.95	0.98
% Std. Dev.	12	14

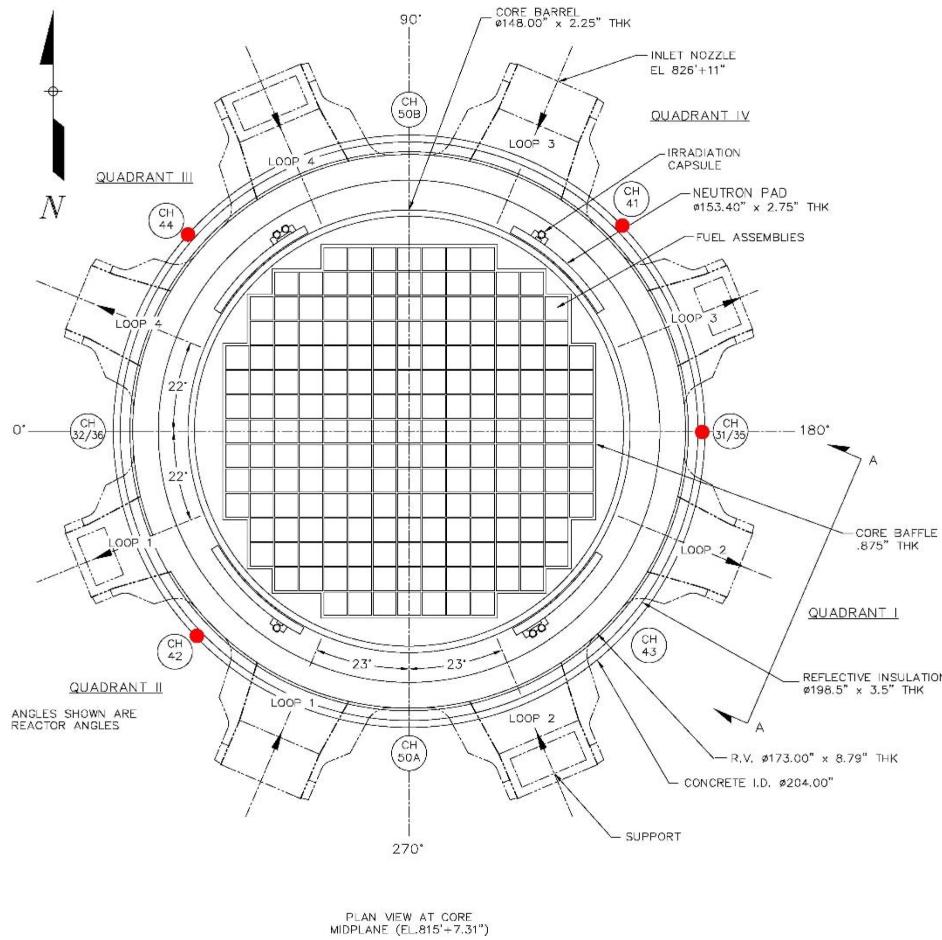

Figure 5-4: 4-Loop Reactor Geometry – Plan View at Core Midplane

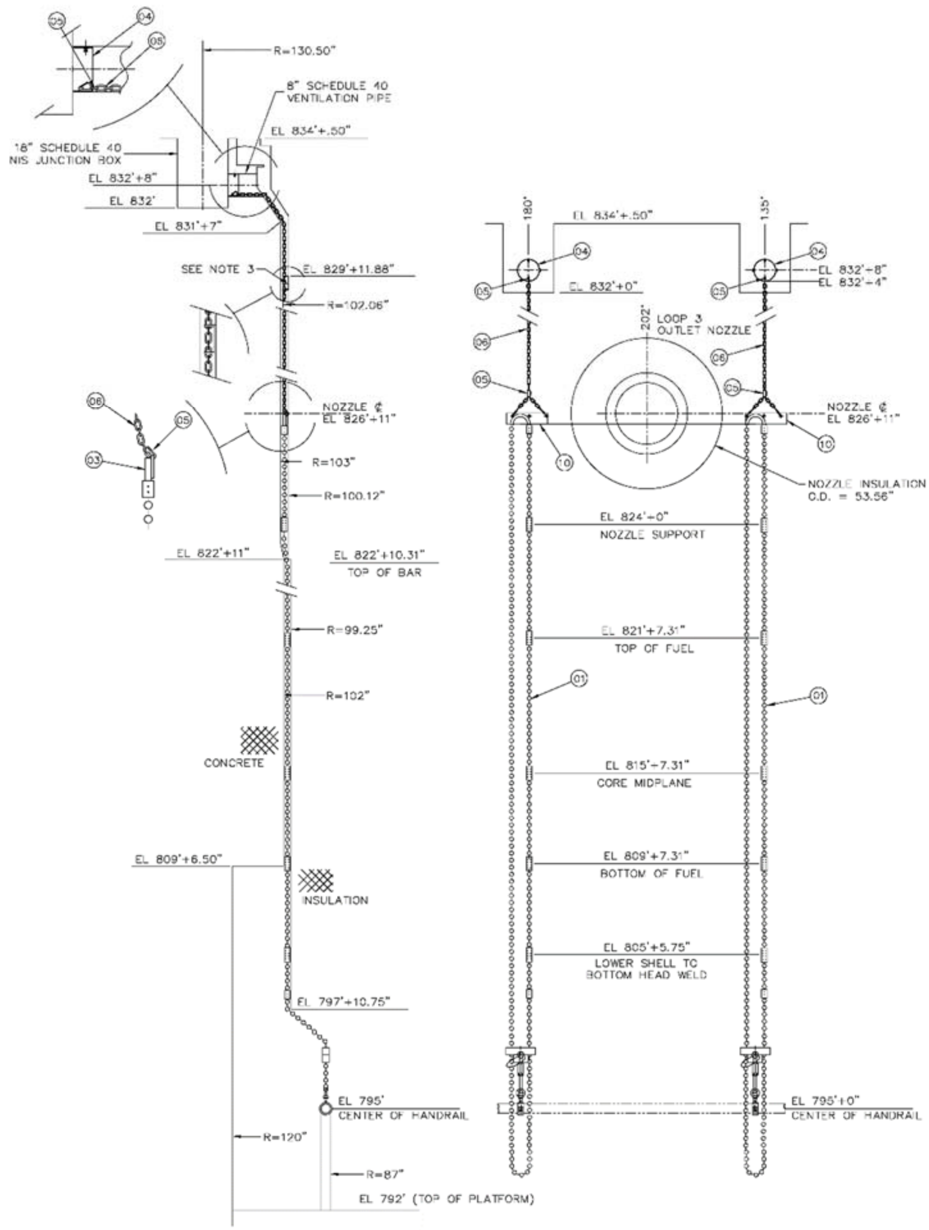

Figure 5-5: Axial Location of EVND Sensor Sets in the 4-Loop Reactor

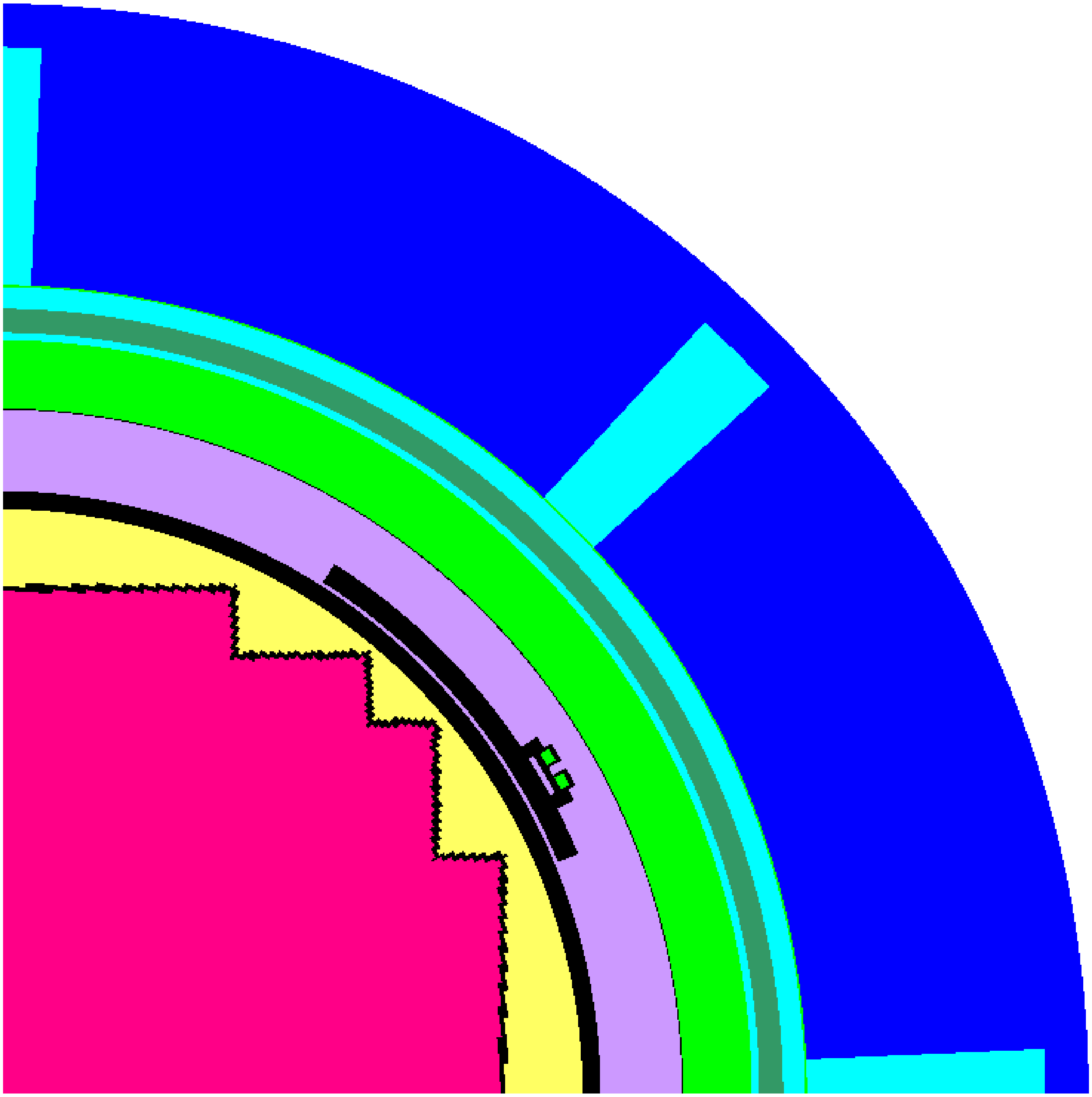

Figure 5-6: 4-Loop Reactor RAPTOR-M3G Model Geometry - Plan View at Core Midplane

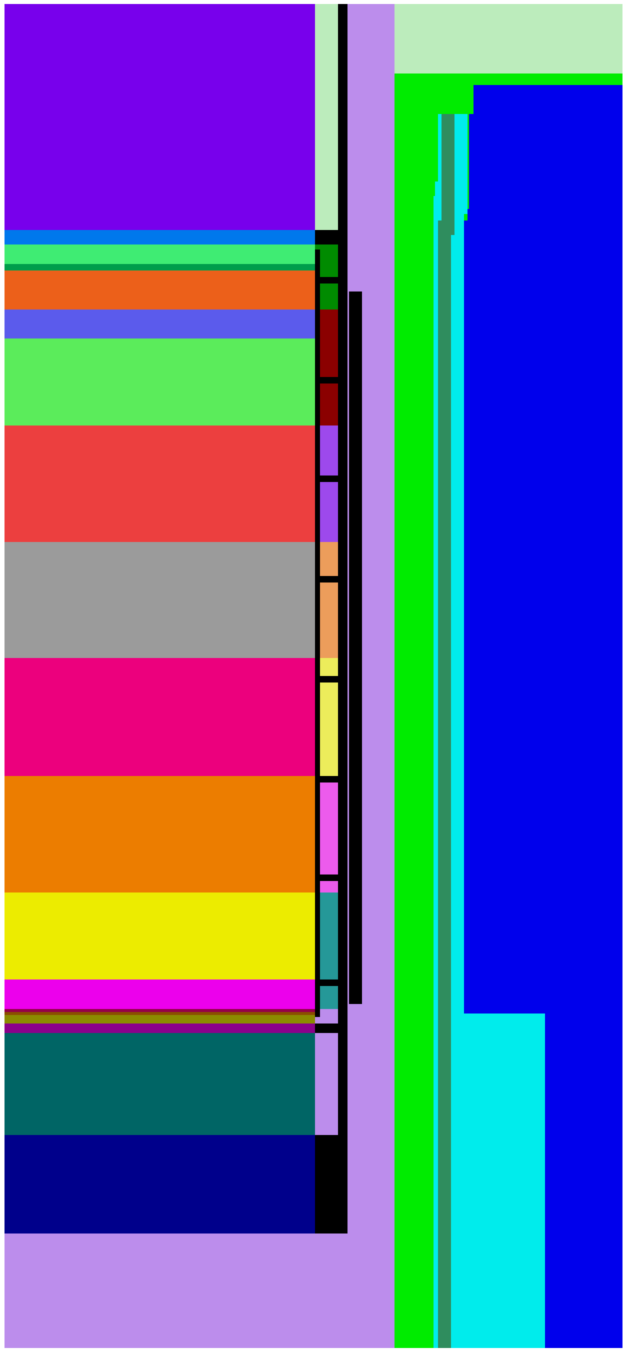

Figure 5-7: 4-Loop Reactor RAPTOR-M3G Model Geometry - Section View at Outlet Nozzle Centerline

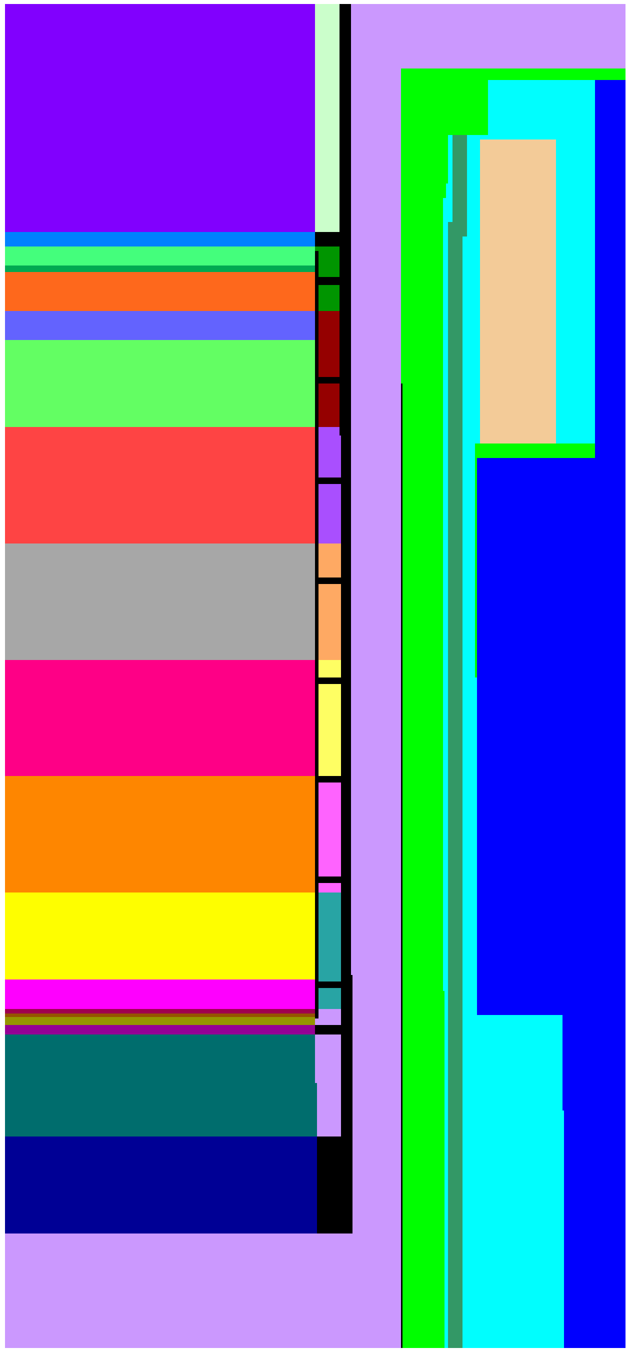

Figure 5-8: 4-Loop Reactor RAPTOR-M3G Model Geometry - Section View at Inlet Nozzle Centerline

6 METHODOLOGY BIAS AND UNCERTAINTY ESTIMATE FOR THE EXTENDED BELTLINE REGION

Regulatory Guide 1.190 requires the determination of overall bias and uncertainty associated with the fluence determination methodology. This section provides the final estimates of bias and uncertainty for the RAPTOR-M3G and FERRET methodology in the extended beltline region, thereby addressing Regulatory Position 1.4.3 of Regulatory Guide 1.190.

To arrive at an estimate of the methodology uncertainty for the beltline region, WCAP-18124-NP-A combines the results of the analytic uncertainty analysis with additional uncertainty components from the simulator benchmarks, the operating reactor benchmark, and “other factors” not explicitly considered. The magnitudes of these uncertainty components were 3%, 5%, and 5%, respectively. The methodology uncertainty for the beltline region was estimated at 13%[1] and the methodology was determined to be unbiased.

For the extended beltline region, [

]^{a,c}:

[ ]^{a,c}

[

]^{a,c}.

[1] Note that the total analytic uncertainty in Table 4-31 of WCAP 18124 NP-A pertains to assessed uncertainty values associated with ex-vessel neutron dosimetry. It contains a “Dosimetry Position” uncertainty component that is associated with a ± 5 cm uncertainty in the radial, azimuthal, and axial directions. This uncertainty component does not apply when assessing RPV exposure at specific locations.

[

]a,c:

- [

]a,c.

7 SUMMARY, CONCLUSIONS, AND CONDITIONS

In this report, the requirements for the use of the WCAP-18124-NP-A methodology for extended beltline region neutron fluence determination have been described. The WCAP-18124-NP-A methodology has been qualified for the extended beltline region according to the requirements set forth in Regulatory Guide 1.190.

Table 7-1 provides the detailed correspondence between the components of Regulatory Position 1 of Regulatory Guide 1.190 and the material provided in WCAP-18124-NP-A and supplemented by this report for the beltline and extended beltline regions. This information satisfies Regulatory Guide 1.190 for the beltline and extended beltline regions.

The following condition must be met in order to apply the WCAP-18124-NP-A methodology for neutron fluence determination in the extended beltline region:

- [

  ]a,c.

With this condition established, revised wording for Limitation #1 of WCAP-18124-NP-A is proposed as follows:

> *Applicability of WCAP-18124-NP-A, Revision 0, Supplement 1, is limited to RPV materials [ ]a,c. Additional justification should be provided via supplemental benchmarking, fluence sensitivity analysis to the response parameters of interest, margin assessment, or a combination thereof, for applications of the method to components including, but not limited to, the reactor vessel internal components.*

With Regulatory Position 1 satisfied as described above and the condition satisfied for each application, the WCAP-18124-NP-A methodology, as supplemented by this report, meets the requirements for neutron fluence determination in Regulatory Guide 1.190 for the extended beltline region.

Table 7-1: Compliance of the RAPTOR-M3G and FERRET Methodology with Regulatory Position 1 of Regulatory Guide 1.190 for the Beltline and Extended Beltline Regions

RG 1.190 Reg. Pos.	Title	WCAP-18124-NP-A (Beltline Region)	Supplement 1 (This Report, Extended Beltline Region)	Comment
1.1.1	Materials and Geometry	Section 2.2 describes the approach to geometric modeling for fluence determination with RAPTOR-M3G.	No Changes.	[ ]^{a,c}.
1.1.2	Cross-Sections	Section 2.3 describes the use of the BUGLE-96 library and testing that has been performed with the BUGLE-B7 library that shows only minor differences between the BUGLE 96 and BUGLE-B7 libraries for neutron fluence applications.	No Changes.	No Comment.
1.2	Core Neutron Source	Section 2.4 describes the process applied to construct accurate representations of the core source term.	No Changes.	No Comment.
1.3.1	Discrete Ordinates Transport Calculation	Section 2.5 describes the application of the discrete ordinates methodology in RAPTOR-M3G.	Section 2.1 and 2.2 of this report require the use of S16 or higher level-symmetric angular quadrature sets and TW spatial differencing.	[ ]^{a,c}.
1.3.2	Monte Carlo Transport Calculation	Not Applicable.	Not Applicable.	RAPTOR-M3G is a discrete ordinates radiation transport code.
1.3.3	Fluence Determination	Section 2.2 describes mesh grid representations of surveillance capsules.	No Changes.	No Comment.
1.3.4	Synthesis of the 3-D Fluence	Not Applicable.	Not Applicable.	RAPTOR-M3G directly solves for the 3-dimensional neutron exposure.

Table 7-1 (continued): Compliance of the RAPTOR-M3G and FERRET Methodology with Regulatory Position 1 of Regulatory Guide 1.190 for the Beltline and Extended Beltline Regions

RG 1.190 Reg. Pos.	Title	WCAP-18124-NP-A (Beltline Region)	Supplement 1 (This Report, Extended Beltline Region)	Comment
1.3.5	Cavity Fluence Calculations	Section 4.3 provides comparisons of RAPTOR-M3G calculations to measurements from the beltline region of a PWR reactor cavity.	The adequacy of RAPTOR-M3G calculations for extended beltline reactor cavity locations is addressed in the Sections 3, 4, and 5.	[ ]a,c.
1.4.1	Analytic Uncertainty Analysis	Section 4.4 provides the analytic uncertainty analysis for the beltline region.	Section 3 provides a comprehensive analytic uncertainty analysis for the extended beltline region.	No Comment.
1.4.2.1	Operating Reactor Measurements	Section 5.4 provides comparisons of RAPTOR-M3G calculation results to in-vessel and ex-vessel measurement data retrieved from beltline locations.	Section 5 provides comparisons of RAPTOR-M3G calculation results to in-vessel and ex-vessel measurement data retrieved from extended beltline locations.	No Comment.

Table 7-1 (continued): Compliance of the RAPTOR-M3G and FERRET Methodology with Regulatory Position 1 of Regulatory Guide 1.190 for the Beltline and Extended Beltline Regions

RG 1.190 Reg. Pos.	Title	WCAP-18124-NP-A (Beltline Region)	Supplement 1 (This Report, Extended Beltline Region)	Comment
1.4.2.2	Pressure Vessel Simulator Measurements	Sections 4.2 and 4.3.1 compare calculation results from RAPTOR-M3G to measurements from the PCA, VENUS-1, and H. B. Robinson-2 experiments.	No Changes.	No Comment.
1.4.2.3	Calculational Benchmark	Section 4.3.2 provides comparisons of RAPTOR-M3G calculation results to the calculational benchmarks from NUREG/CR-6115.	Section 4 provides comparisons of RAPTOR-M3G calculation results to alternate calculation results obtained with a Monte Carlo technique for the beltline and extended beltline regions.	No Comment.
1.4.3	Estimate of Fluence Calculational Bias and Uncertainty	Section 4.5 provides an estimate of the bias and uncertainty for the beltline region.	Section 6 provides an estimate of the bias and uncertainty for the extended beltline region.	No Comment.

8 REFERENCES

1. U.S. Nuclear Regulatory Commission Regulatory Issue Summary 2014-11, “Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components,” U.S. Nuclear Regulatory Commission, October 2014. [*ADAMS Accession Number ML14149A165*]

2. Westinghouse Report WCAP-18124-NP-A, Revision 0, “Fluence Determination with RAPTOR-M3G and FERRET,” July 2018. [*ADAMS Accession Number ML18204A010*]

3. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.190, Revision 0, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” March 2001. [*ADAMS Accession Number ML010890301*]

4. RSICC Data Library Collection DLC-185, “BUGLE-96, Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications,” Radiation Shielding Information Computational Center, Oak Ridge National Laboratory (ORNL), July 1999.