Attachment 4

Westinghouse, WCAP-15353 – Supplement 2 – NP
Revision 0

Palisades Reactor Pressure Vessel Fluence Evaluation

37 pages follow

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Revision 0

Palisades Reactor Pressure Vessel Fluence Evaluation

Westinghouse

WCAP-15353 – Supplement 2 - NP, Revision 0

Palisades Reactor Pressure Vessel Fluence Evaluation

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EXECUTIVE SUMMARY

In early 2000, WCAP-15353, Revision 0[1] describing the methodology used in the fluence evaluations for the Palisades plant was submitted to the NRC staff for review. Subsequent to that review and a further exchange of information documented in Reference 2, the methodology described in WCAP-15353, Revision 0 was approved for application to the Palisades reactor pressure vessel[3]. Subsequent to that approval, additional submittals[4,5] in support of the benchmarking of this fluence methodology were reviewed and approved by the NRC Staff as being in compliance with the requirements of Regulatory Guide 1.190[6].

The fluence analysis described in WCAP-15353, Revision 0[1] included cycle specific evaluations through fuel Cycle 14 (the then current operating cycle). In mid 2010, Supplement 1 to WCAP-15353, Revision 0 was issued to provide an updated neutron fluence assessment for the Palisades pressure vessel that included cycle specific analysis for additional operating cycles for which the design had been finalized (Cycles 15 through 21). This prior supplement included projections for future operation through approximately 44 effective full power years (EFPY). The results of the evaluation documented in Supplement 1 were used as input to vessel materials studies that included updates to surveillance capsule credibility analysis, material chemistry factor determination, Pressurized Thermal Shock (PTS) evaluation, and generation of Pressure-Temperature (PT) limit curves.

The fluence analysis documented in Supplement 1 of WCAP-15353, Revision 0 was limited to an axial range that extended approximately one foot above and below the active fuel stack. This model did not include all of the pressure vessel materials that could potentially exceed the 1.0E+17 n/cm^2 (E > 1.0 MeV) fluence threshold defined in 10 CFR 50 Appendix H[7]. The purpose of this second supplement is to define which materials in the Palisades pressure vessel are projected to exceed the 1.0E+17 n/cm^2 threshold neutron fluence before the end of the license renewal period (EOLE); and, to project the neutron fluence for each of these specific materials. This additional fluence information will be used to perform necessary material evaluations for those materials in the extended beltline region that are projected to exceed the 1.0E+17 n/cm^2 threshold.

The results of the neutron exposure calculations for the extended beltline region of the Palisades pressure vessel are provided in Tables E-1 and E-2. In Table E-1, the axial locations of the maximum neutron exposure location are listed for each of the traditional and extended beltline materials. The axial span of each material is indexed to Z = 0.0 cm at the midplane of the active fuel stack modeled in the neutronic calculations. In Table E-2, the maximum projected neutron fluence for each of the beltline materials is provided. Projected fluence values are listed for EOC21 (23.4 EFPY), EOLE (42.1 EFPY), and EOC36 (44.0 EFPY).

It should be noted that Supplement 2 of this report was generated simply to address the neutron fluence experienced by materials located in the extended beltline regions above and below the reactor core that were not included in either Revision 0 of WCAP-15353 or in Supplement 1 of that report. None of the fluence information that was included in Supplement 1 has been changed in Supplement 2.

Table E-1
Palisades Pressure Vessel Material Locations in the Traditional and Extended Beltline Regions

Material	Axial Location[a,b] [cm]	Notes
Outlet Nozzle to Upper Shell Welds – Lowest Extent 5-114	260.02	Extended Beltline
Inlet Nozzle to Upper Shell Welds – Lowest Extent 5-114	277.72	Extended Beltline
Upper Shell Plates D-3802	204.93	Extended Beltline
Upper Shell Longitudinal Welds 1-112	204.93	Extended Beltline
Upper Shell to Intermediate Shell Circumferential Weld 8-112	204.93	Extended Beltline
Intermediate Shell Plates D-3803	-42.24	Traditional Beltline
Intermediate Shell Longitudinal Welds 2-112	-42.24	Traditional Beltline
Intermediate Shell to Lower Shell Circumferential Weld 9-112	-42.24	Traditional Beltline
Lower Shell Plates D-3804	-42.24	Traditional Beltline
Lower Shell Longitudinal Welds 3-112	-42.24	Traditional Beltline
Lower Shell to Lower Vessel Head Circumferential Weld 10-112	-279.43	Extended Beltline

[a] Axial elevations are indexed to Z = 0.0 at the midplane of the active fuel stack.

[b] Elevations listed represent the location of maximum neutron exposure for the material.

Table E-2
Palisades Maximum Fast Neutron (E > 1.0 MeV) Fluence Experienced by Materials in the Traditional and Extended Beltline

Material	Neutron Fluence [n/cm^2]		
	23.4 EFPY	42.1 EFPY	44 EFPY
Outlet Nozzle to Upper Shell Welds – Lowest Extent 5-114	<1.0E+17	<1.0E+17	<1.0E+17
Inlet Nozzle to Upper Shell Welds – Lowest Extent 5-114	<1.0E+17	<1.0E+17	<1.0E+17
Upper Shell Plates D-3802	9.895E+17	1.529E+18	1.584E+18
Upper Shell Longitudinal Welds 1-112	6.782E+17	9.707E+17	1.001E+18
Upper Shell to Intermediate Shell Circumferential Weld 8-112	9.895E+17	1.529E+18	1.584E+18
Intermediate Shell Plates D-3803	2.157E+19	3.428E+19	3.558E+19
Intermediate Shell Longitudinal Welds 2-112	1.472E+19	2.161E+19	2.232E+19
Intermediate Shell to Lower Shell Circumferential Weld 9-112	2.157E+19	3.428E+19	3.558E+19
Lower Shell Plates D-3804	2.157E+19	3.428E+19	3.558E+19
Lower Shell Longitudinal Welds 3-112	1.472E+19	2.161E+19	2.232E+19
Lower Shell to Lower Vessel Head Circumferential Weld 10-112	<1.0E+17	<1.0E+17	<1.0E+17

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SECTION 1.0

INTRODUCTION

In the assessment of the state of embrittlement of light water reactor (LWR) pressure vessels, an accurate evaluation of the neutron exposure of each of the materials comprising the beltline region of the vessel is required. In Section II F of 10 CFR 50[7] Appendix G, the beltline region is defined as:

> *"the region of the reactor vessel shell material (including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the reactor core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage"*.

In Section II A of 10 CFR 50 Appendix H, the lower limit of neutron exposure for consideration of radiation induced material damage is specified by a neutron fluence (E > 1.0 MeV) threshold of 1.0E+17 n/cm^2. Each of the materials that is anticipated to experience a neutron exposure that exceeds this fluence threshold must be considered in the overall embrittlement assessments for the pressure vessel.

Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [6], describes state-of-the-art calculation and measurement procedures that are acceptable to the NRC staff for determining pressure vessel fluence. Also included in Regulatory Guide 1.190 is a discussion of the steps required to qualify and validate the methodology used to determine the neutron exposure of the pressure vessel wall. One important step in the validation process is the comparison of plant-specific neutron calculations with available measurements.

In early 2000, WCAP-15353, Revision 0[1] describing the methodology used in the fluence evaluations for the Palisades plant was submitted to the NRC staff for review. Subsequent to that review and a further exchange of information documented in Reference 2, the methodology described in WCAP-15353, Revision 0 was approved for application to the Palisades reactor pressure vessel[3]. Subsequent to that approval additional submittals[4,5] in support of the benchmarking of this fluence methodology were reviewed and approved by the NRC Staff as being in compliance with the requirements of Regulatory Guide 1.190[6].

The fluence analysis described in WCAP-15353, Revision 0[1] included cycle specific evaluations through fuel Cycle 14 (the then current operating cycle). In mid 2010, Supplement 1 to WCAP-15353, Revision 0 was issued to provide an updated neutron fluence assessment for the Palisades pressure vessel that included cycle specific analysis for additional operating cycles for which the design had been finalized (Cycles 15 through 21). This supplement included projections for future operation through approximately 44 effective full power years (EFPY). The results of the evaluation documented in Supplement 1 were used as input to vessel materials studies that included updates to surveillance capsule credibility analysis, material chemistry factor determination, Pressurized Thermal Shock (PTS) evaluation, and generation of Pressure-Temperature (PT) limit curves.

Since the PTS screening criterion determination for the Palisades pressure vessel requires the evaluation of all weld heat W5214 surveillance capsule data from Palisades and other PWR's, Supplement 1 also included a compilation of the latest fluence evaluation from withdrawn surveillance capsules containing the W5214 material from sister plants. That compilation of capsule fluence values provided a data set based on the same Regulatory Guide 1.190 compliant methodology described in this report.

The fluence analysis documented in Supplement 1 of WCAP-15353, Revision 0 was limited to an axial range that extended approximately one foot above and below the active fuel stack. This model did not include all the pressure vessel materials that could potentially exceed the 1.0E+17 n/cm^2 (E > 1.0 MeV) fluence threshold defined in 10 CFR 50 Appendix H[7]. The purpose of this second supplement is to define which materials in the Palisades pressure vessel are projected to exceed the 1.0E+17 n/cm^2 threshold neutron fluence before the end of the license renewal period (EOLE); and, to project the neutron fluence for each of these specific materials. The additional fluence information will be used to perform necessary material evaluations for those materials in the extended beltline region that are projected to exceed the 1.0E+17 n/cm^2 threshold.

In subsequent sections of this supplement, the methodologies used to perform neutron transport calculations and dosimetry evaluations are described in some detail and the results of the plant specific transport calculations are given for each of the materials located in the traditional and extended beltline regions of the Palisades pressure vessel. For completeness, comparisons of calculations and measurements demonstrating that the transport calculations meet the requirements of Regulatory Guide 1.190 that were previously discribed in Reference 1 are also included in this supplement. These comparisons demonstrate the adequacy of the methodology for use in the fluence determinations provided in this report. Finally, a listing of updated neutron fluence values, based on the use of an approved Regulatory Guide 1.190 compliant fluence methodology, for several previously withdrawn surveillance capsules that contain some of the Palisades vessel materials is provided for use in data correlation studies.

SECTION 2.0

NEUTRON TRANSPORT CALCULATIONS

As noted in Section 1.0 of this report, the exposure of the Palisades pressure vessel was developed based on a series of fuel cycle-specific neutron transport calculations validated by comparison with plant-specific measurements. Measurement data used in the validation process were obtained from both in-vessel and ex-vessel capsule irradiations. In this section, the neutron transport methodology is discussed in some detail, and the calculated results applicable to the in-vessel surveillance capsules and the pressure vessel beltline materials are presented. A discussion of the Palisades dosimetry evaluations and measurement to calculation comparisons is included in Section 3.0 of this supplement. The data comparisons included in Section 3.0 cover a wide range of both in-vessel and ex-vessel locations. These comparisons along with the benchmarking information described in References 4 and 5 demonstrate that the transport methodology provides results that meet the requirements of Regulatory Guide 1.190 for the pressure vessel and surveillance capsule locations considered in this report.

2.1 – Method of Analysis

In performing the fast neutron exposure evaluations for the Palisades reactor, plant-specific forward transport calculations were carried out using the three-dimensional flux synthesis technique described in Section 1.3.4 of Regulatory Guide 1.190. In particular, the following single channel synthesis approach was employed for all fuel cycles:

$$\Phi(r,\theta,z) = \Phi(r,\theta) * \frac{\Phi(r,z)}{\Phi(r)}$$

where $\phi(r,\theta,z)$ is the synthesized three-dimensional neutron flux distribution, $\phi(r,\theta)$ is the transport solution in r,θ geometry, $\phi(r,z)$ is the two-dimensional solution for a cylindrical reactor model using the actual axial core power distribution, and $\phi(r)$ is the one-dimensional solution for a cylindrical reactor model using the same source-per-unit height as that used in the r,θ two-dimensional calculation.

For the Palisades analysis, all of the transport calculations were carried out using the DORT two-dimensional discrete ordinates code Version 3.2[8] and the BUGLE-96 cross section library[9]. The BUGLE-96 library provides a 67-group coupled neutron-gamma ray cross-section data set produced specifically for light water reactor applications. In these analyses, anisotropic scattering was treated with a P_5 legendre expansion and the angular discretization was modeled with an S_{16} order of angular quadrature. Energy and space dependent core power distributions as well as system operating conditions were treated on a fuel cycle specific basis.

The geometry used for the Palisades transport analysis is discussed in some detail in Reference 3 and the geometric model established for Cycle 15 and beyond was also used for the evaluations documented in Supplement 1 to Reference 1. A plan view of the r,θ model of the reactor geometry at the core midplane is shown in Figure 2.1-1. This model depicts a single quadrant of the reactor. A section view of the r,z model of the Palisades reactor is shown in Figure 2.1-2. The r,z model extended radially from the centerline of the reactor core out to a location interior to the primary biological shield and over an axial span from an elevation approximately four feet below the active fuel to an axial elevation approximately four feet above the active fuel. The one-dimensional radial model used in the synthesis procedure consisted of the same radial mesh intervals included in the r,z model. Thus, radial synthesis factors could be determined on a meshwise basis throughout the entire geometry.

These geometric models formed the basis for the extended beltline evaluation described in this supplement. In completing the current analysis, the two-dimensional r,θ and the one-dimensional r transport calculations from the prior analyses were retained as is and the two-dimensional r,z transport calculations were re-run with expanded models designed to encompass all axial elevations that were anticipated to experience a neutron fluence (E > 1.0 MeV) greater than 1.0E+17 n/cm^2. The expanded r,z calculations were then coupled with the existing r,θ and r calculations to provide the final synthesized three-dimensional neutron fluence solution, $\Phi(r,\theta,z)$, from which the maximum neutron exposure of each of the traditional and extended beltline materials was extracted.

The geometric models used in the fluence analysis were developed from nominal design dimensions for the reactor core and the reactor internals components. However, for the pressure vessel inner radius and thickness, as-built dimensions were available from fabrication records and additional data were subsequently obtained during plant operation. These as-built dimensions were used in the analytical models.

The as-built value of the pressure vessel base metal inner radius was derived from available diameter measurements of the shell of the vessel prior to the addition of cladding. Based on an evaluation of the maximum and minimum vessel measurements, the nominal radius of the pressure vessel base metal was determined to be:

Base Metal IR = 86.35 inches

Based on a nominal cladding thickness of 0.25 inches, the cladding inner radius was determined to be:

Cladding IR = 86.10 inches

The as-built thickness of the pressure vessel base metal was based on a statistical evaluation of 151 thickness measurements taken during the 1995 in-service inspection. The measurement points were taken around the circumference and over the height of the beltline region. Based on the evaluation of these measurement points the total thickness of clad plus base metal was determined to be:

Total Thickness = 9.042 ± 0.088 inches

Based on a nominal cladding thickness of 0.25 inches, the thickness of the pressure vessel base metal shell was determined to be:

Base Metal Thickness = 8.792 inches

These as-built pressure vessel dimensions were submitted to the NRC staff in the form of a reply to a Request for Additional Information (RAI)[2] and were subsequently approved for use in the Palisades fluence analyses.

The core power distributions used in the plant-specific transport analysis for the reactor were provided by Entergy[16]. The data used in the source generation included fuel assembly-specific initial enrichments, beginning-of-cycle burnups and end-of-cycle burnups. Appropriate axial burrnup distributions were also used.

For each fuel cycle of operation, the fuel assembly-specific enrichment and burnup data were used to generate the spatially-dependent neutron source throughout the reactor core. This source description included the spatial variation of isotope dependent (U-235, U-238, Pu-239, Pu-240, Pu-241, and Pu-242) fission spectra, neutron emission rate per fission, and energy release per fission based on the burnup history of individual fuel assemblies. These fuel assembly-specific neutron source strengths derived from the detailed isotopics were then converted from fuel pin cartesian coordinates to the [r,θ], [r,z], and [r] spatial mesh arrays used in the DORT discrete ordinates calculations.

This same qualified methodology was used along with reactor specific input in the determination of the surveillance capsule fluence values discussed in Section 4.0 of this report. It should be noted that these additional surveillance capsule fluence evaluations are not directly related to the extended beltline materials in the Palisades pressure vessel.

Figure 2.1-1

Palisades r,θ Reactor Geometry

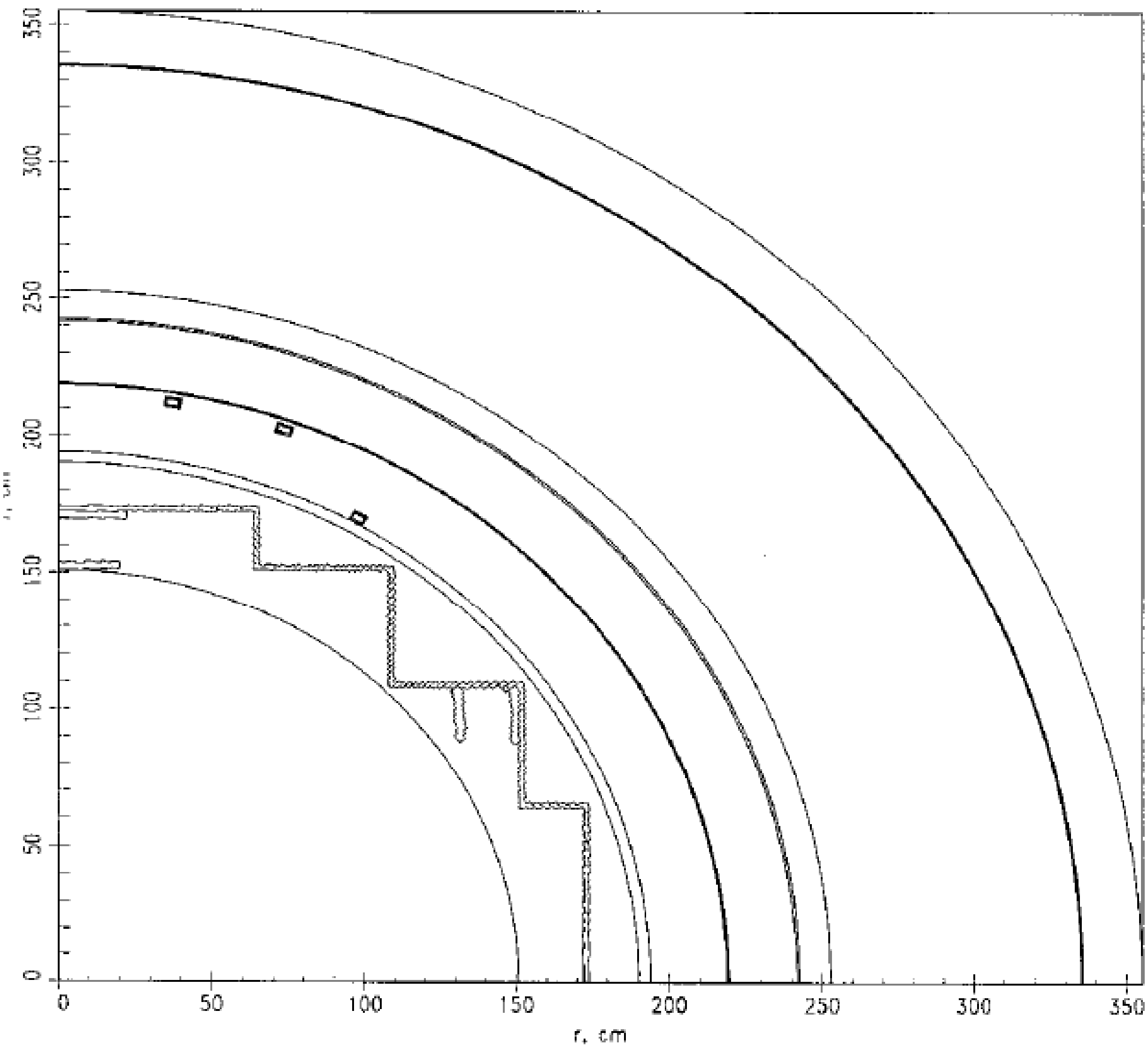
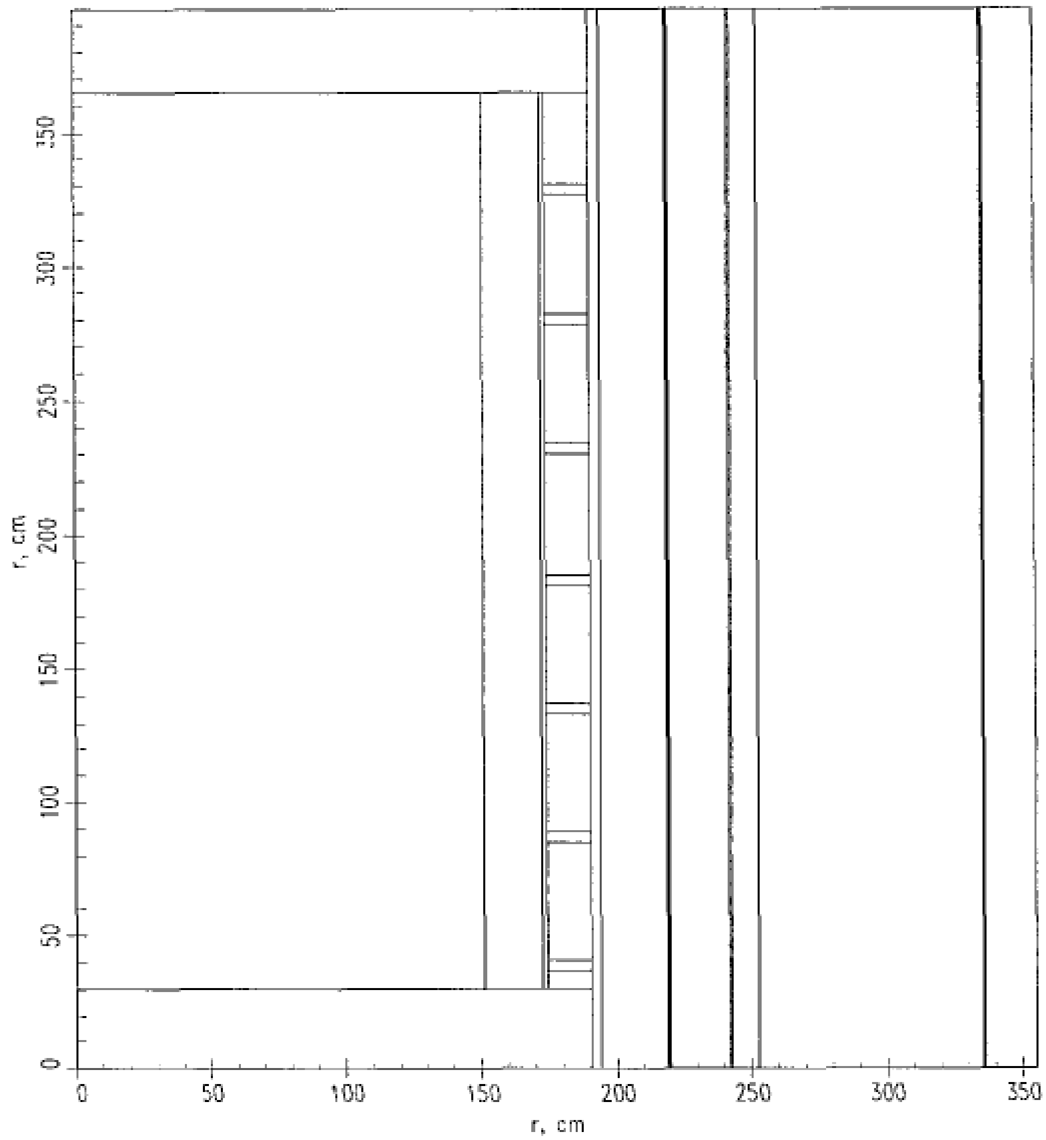


Figure 2.1-2

Palisades r,z Reactor Geometry



2.2 – Calculated Neutron Exposure of Pressure Vessel Beltline Materials

The results of the neutron exposure calculations for both the traditional and extended beltline regions of the Palisades pressure vessel are provided in Tables 2.2-1 and 2.2-2. In Table 2.2-1, the axial location of the maximum neutron exposure of each of the traditional and extended beltline materials is listed. The axial location of each material is indexed to Z = 0.0 cm at the midplane of the active fuel stack modeled in the neutronic calculation. In Table 2.2-2, the maximum projected neutron fluence for each of the beltline materials is provided. Projected fluence values are listed for EOC21 (23.4 EFPY), EOLE (42.1 EFPY), and EOC36 (44.0 EFPY).

From Table 2.2-2, it is noted that, although the upper shell course and the associated upper shell to intermediate shell circumferential weld are projected to exceed the 1.0E+17 n/cm^2 fluence (E > 1.0 MeV) threshold, the nozzles themselves as well as the nozzle to nozzle shell welds remain below 1.0E+17 n/cm^2 through 44 EFPY. Likewise, the lower shell to lower head circumferential weld will remain below the threshold of 1.0E+17 n/cm^2 through 44 EFPY of operation. The current analysis demonstrates that, for 44 EFPY of reactor operation, the axial span of the pressure vessel that is projected to exceed the neutron fluence (E > 1.0 MeV) threshold of 1.0E+17 n/cm^2 is limited to a zone extending from approximately 245 cm below the midplane of the active fuel stack to approximately 245 cm above the midplane of the active fuel stack.

The azimuthal distribution of fuel cycle-specific calculated fast neutron (E > 1.0 MeV) flux and fluence experienced by the materials comprising the beltline region of the Palisades pressure vessel are given in Tables 2.2-3 through 2.2-6, respectively, for plant operation through the conclusion of the twenty-first fuel cycle. Cycle 21 represents the last fuel cycle for which final fuel loading patterns have been designed. As presented, the data in Tables 2.2-3 through 2.2-6 represent the maximum neutron exposures at the pressure vessel clad base metal interface at azimuthal angles of 0°, 15°, 30°, 45°, 60°, 75°, and 90° relative to the core major axes. The limiting weld material for the Palisades pressure vessel occurs along the 60° azimuth (Heat W5214, Weld IDs 2-112A/C and 3-112A/C). All of the data provided in Tables 2.2-3 through 2.2-6 were taken at the axial location of the maximum exposure experienced at each azimuth based on the results of the three-dimensional synthesized neutron fluence evaluations.

In Tables 2.2-7 and 2.2-8, projections of neutron (E > 1.0 MeV) fluence beyond the end of Cycle 21 are provided for the traditional and extended beltline materials, respectively. These projections were based on assumed future operating conditions provided by Entergy. In particular the following assumptions were applied to the analysis:

1 - For Cycle 22, the nominal calculated neutron flux based on the average of the prior uprated fuel cycles (18 through 21) was used. This approach is a realistic representation of the neutron flux that would be expected based on existing preliminary designs for Cycle 22.

2 - For Cycles 23 and beyond, the Cycle 21 neutron flux distribution was applied for all fuel cycles. This is a conservative assumption in that, considering Cycles 15 through 21, the Cycle 21 power distribution results in the highest calculated flux at the location of the critical pressure vessel weld material(60°).

3 - Projected fuel cycle lengths were provided by Entergy as follows:

	Design	**95% Capacity**
Cycle 22	525 EFPD	499 EFPD
Cycle 23	525 EFPD	499 EFPD
Cycle 24	502 EFPD	477 EFPD
Cycles 25+	530 EFPD	504 EFPD

Fuel cycles were assumed to operate with a breaker to breaker capacity factor of 95%.

In completing the projections beyond the end of Cycle 21, operation was assumed to a total of 44 EFPY. Given the assumed operating scenario, this would cover a calendar time period extending to 2033.

In regard to the fluence data provided in Tables 2.2-2, 2.2-5, 2.2-6, 2.2-7 and 2.2-8, it should be noted that the critical longitudinal welds (2-112A, 2-112C, 3-112A, and 3-112C) are exposed to the neutron flux characteristic of the 60° azimuthal location. The beltline circumferential weld 9-112 and all plate materials are exposed to the maximum neutron exposure characteristic of the 75° azimuthal location.

Table 2.2-1
Palisades Pressure Vessel Material Locations in the
Traditional and Extended Beltline Regions

Material	Axial Location[a,b] [cm]	Notes
Outlet Nozzle to Upper Shell Welds – Lowest Extent 5-114	260.02	Extended Beltline
Inlet Nozzle to Upper Shell Welds – Lowest Extent 5-114	277.72	Extended Beltline
Upper Shell Plates D-3802	204.93	Extended Beltline
Upper Shell Longitudinal Welds 1-112	204.93	Extended Beltline
Upper Shell to Intermediate Shell Circumferential Weld 8-112	204.93	Extended Beltline
Intermediate Shell Plates D-3803	-42.24	Traditional Beltline
Intermediate Shell Longitudinal Welds 2-112	-42.24	Traditional Beltline
Intermediate Shell to Lower Shell Circumferential Weld 9-112	-42.24	Traditional Beltline
Lower Shell Plates D-3804	-42.24	Traditional Beltline
Lower Shell Longitudinal Welds 3-112	-42.24	Traditional Beltline
Lower Shell to Lower Vessel Head Circumferential Weld 10-112	-279.43	Extended Beltline

[a] Axial elevations are indexed to Z = 0.0 at the midplane of the active fuel stack.

[b] Elevations listed represent the location of maximum neutron exposure for the material.

Table 2.2-2
Palisades Maximum Fast Neutron (E > 1.0 MeV) Fluence Experienced by Materials in the Tradional and Extended Beltline

Material	Neutron Fluence [n/cm^2]		
	23.4 EFPY	**42.1 EFPY**	**44 EFPY**
Outlet Nozzle to Upper Shell Welds – Lowest Extent 5-114	<1.0E+17	<1.0E+17	<1.0E+17
Inlet Nozzle to Upper Shell Welds – Lowest Extent 5-114	<1.0E+17	<1.0E+17	<1.0E+17
Upper Shell Plates D-3802	9.895E+17	1.529E+18	1.584E+18
Upper Shell Longitudinal Welds 1-112	6.782E+17	9.707E+17	1.001E+18
Upper Shell to Intermediate Shell Circumferential Weld 8-112	9.895E+17	1.529E+17	1.584E+17
Intermediate Shell Plates D-3803	2.157E+19	3.428E+19	3.558E+19
Intermediate Shell Longitudinal Welds 2-112	1.472E+19	2.161E+19	2.232E+19
Intermediate Shell to Lower Shell Circumferential Weld 9-112	2.157E+19	3.428E+19	3.558E+19
Lower Shell Plates D-3804	2.157E+19	3.428E+19	3.558E+19
Lower Shell Longitudinal Welds 3-112	1.472E+19	2.161E+19	2.232E+19
Lower Shell to Lower Vessel Head Circumferential Weld 10-112	<1.0E+17	<1.0E+17	<1.0E+17

Table 2.2-3
Summary of Calculated Maximum Pressure Vessel Neutron Flux (E > 1.0 MeV)
For Cycles 15 through 21 and for Future Projection
Traditional Beltline Materials

Fuel Cycle	Cycle Time (EFPY)	Neutron (E > 1.0 MeV) Flux (n/cm^2-s)			
		0 Deg.	**15 Deg.**	**30 Deg.**	**45 Deg.**
15	1.1	9.671E+09	1.572E+10	1.277E+10	7.924E+09
16	1.2	1.068E+10	1.614E+10	1.330E+10	7.797E+09
17	1.3	1.080E+10	1.875E+10	1.332E+10	7.613E+09
18	1.3	1.292E+10	2.102E+10	1.352E+10	7.337E+09
19	1.3	1.059E+10	1.940E+10	1.445E+10	7.037E+09
20	1.4	1.123E+10	2.021E+10	1.517E+10	8.143E+09
21	1.4	1.138E+10	2.032E+10	1.501E+10	8.506E+09
22	Projected	1.153E+10	2.024E+10	1.454E+10	7.756E+09
23+	Projected	1.138E+10	2.032E+10	1.501E+10	8.506E+09

Fuel Cycle	Cycle Time (EFPY)	Neutron (E > 1.0 MeV) Flux (n/cm^2-s)			
		60 Deg.	**75 Deg.**	**90 Deg.**	
15	1.1	1.105E+10	1.679E+10	1.257E+10	
16	1.2	1.135E+10	1.761E+10	1.401E+10	
17	1.3	9.781E+09	1.968E+10	1.539E+10	
18	1.3	1.088E+10	2.236E+10	1.664E+10	
19	1.3	1.090E+10	2.232E+10	1.743E+10	
20	1.4	1.161E+10	2.197E+10	1.650E+10	
21	1.4	1.172E+10	2.152E+10	1.618E+10	
22	Projected	1.128E+10	2.204E+10	1.669E+10	
23+	Projected	1.172E+10	2.152E+10	1.618E+10	

Data Applicability

Lower Shell Plates D-3804
Lower Shell Longitudinal Welds 3-112
Intermediate Shell Plates D-3803
Intermediate Shell Longitudinal Welds 2-112
Intermediate-Lower Shell Circumferential Weld 2-112

Table 2.2-4
Summary of Calculated Maximum Pressure Vessel Neutron Flux (E > 1.0 MeV)
For Cycles 15 through 21 and for Future Projection
Extended Beltline Materials

Fuel Cycle	Cycle Time (EFPY)	Neutron (E > 1.0 MeV) Flux (n/cm^2-s)			
		0 Deg.	15 Deg.	30 Deg.	45 Deg.
15	1.1	4.105E+08	6.613E+08	5.420E+08	3.363E+08
16	1.2	4.495E+08	6.751E+08	5.597E+08	3.281E+08
17	1.3	4.639E+08	7.990E+08	5.722E+08	3.270E+08
18	1.3	5.553E+08	9.001E+08	5.811E+08	3.154E+08
19	1.3	4.652E+08	8.452E+08	6.347E+08	3.091E+08
20	1.4	4.956E+08	8.844E+08	6.695E+08	3.594E+08
21	1.4	4.830E+08	8.625E+08	6.370E+08	3.610E+08
22	Projected	4.913E+08	8.624E+08	6.200E+08	3.315E+08
23+	Projected	4.830E+08	8.625E+08	6.370E+08	3.610E+08

Fuel Cycle	Cycle Time (EFPY)	Neutron (E > 1.0 MeV) Flux (n/cm^2-s)			
		60 Deg.	75 Deg.	90 Deg.	
15	1.1	4.690E+08	7.135E+08	5.336E+08	
16	1.2	4.777E+08	7.415E+08	5.896E+08	
17	1.3	4.201E+08	8.449E+08	6.611E+08	
18	1.3	4.677E+08	9.607E+08	7.152E+08	
19	1.3	4.788E+08	9.796E+08	7.657E+08	
20	1.4	5.124E+08	9.700E+08	7.282E+08	
21	1.4	4.975E+08	9.136E+08	6.868E+08	
22	Projected	4.811E+08	9.384E+08	7.102E+08	
23+	Projected	4.975E+08	9.136E+08	6.868E+08	

Data Applicability

Upper Shell Plates D-3802

Upper Shell Longitudinal Welds 1-112

Upper Shell-Intermediate Shell Circumferential Weld 8-112

Table 2.2-5
Summary of Calculated Maximum Pressure Vessel Neutron Exposure
Through the Conclusion of Cycle 21
Traditional Beltline Materials

Fuel Cycle	Cycle Time (EFPY)	Cumulative Time (EFPY)	Neutron (E > 1.0 MeV) Fluence (n/cm^2)			
			0 Deg.	**15 Deg.**	**30 Deg.**	**45 Deg.**
1-14	14.4	14.4	1.132E+19	1.576E+19	1.192E+19	7.467E+18
15	1.1	15.5	1.165E+19	1.631E+19	1.237E+19	7.742E+18
16	1.2	16.7	1.206E+19	1.693E+19	1.288E+19	8.041E+18
17	1.3	18.0	1.252E+19	1.773E+19	1.344E+19	8.365E+18
18	1.3	19.3	1.305E+19	1.858E+19	1.400E+19	8.665E+18
19	1.3	20.6	1.347E+19	1.935E+19	1.457E+19	8.944E+18
20	1.4	22.0	1.395E+19	2.023E+19	1.522E+19	9.295E+18
21	1.4	23.4	1.447E+19	2.114E+19	1.590E+19	9.677E+18

Fuel Cycle	Cycle Time (EFPY)	Cumulative Time (EFPY)	Neutron (E > 1.0 MeV) Fluence (n/cm^2)			
			60 Deg.	**75 Deg.**	**90 Deg.**	
1-14	14.4	14.4	1.158E+19	1.576E+19	1.132E+19	
15	1.1	15.5	1.196E+19	1.635E+19	1.175E+19	
16	1.2	16.7	1.240E+19	1.702E+19	1.229E+19	
17	1.3	18.0	1.282E+19	1.786E+19	1.295E+19	
18	1.3	19.3	1.326E+19	1.877E+19	1.363E+19	
19	1.3	20.6	1.369E+19	1.966E+19	1.432E+19	
20	1.4	22.0	1.419E+19	2.060E+19	1.503E+19	
21	1.4	23.4	1.472E+19	2.157E+19	1.576E+19	

Data Applicability

Lower Shell Plates D-3804
Lower Shell Longitudinal Welds 3-112
Intermediate Shell Plates D-3803
Intermediate Shell Longitudinal Welds 2-112
Intermediate-Lower Shell Circumferential Weld 9-112

Table 2.2-6
Summary of Calculated Maximum Pressure Vessel Neutron Exposure
Through the Conclusion of Cycle 21
Extended Beltline Materials

Fuel Cycle	Cycle Time (EFPY)	Cumulative Time (EFPY)	Neutron (E > 1.0 MeV) Fluence (n/cm^2)			
			0 Deg.	**15 Deg.**	**30 Deg.**	**45 Deg.**
1-14	14.4	14.4	5.308E+17	7.394E+17	5.593E+17	3.503E+17
15	1.1	15.5	5.450E+17	7.623E+17	5.780E+17	3.620E+17
16	1.2	16.7	5.623E+17	7.882E+17	5.996E+17	3.746E+17
17	1.3	18.0	5.821E+17	8.223E+17	6.240E+17	3.885E+17
18	1.3	19.3	6.047E+17	8.590E+17	6.477E+17	4.013E+17
19	1.3	20.6	6.232E+17	8.925E+17	6.728E+17	4.136E+17
20	1.4	22.0	6.446E+17	9.307E+17	7.017E+17	4.291E+17
21	1.4	23.4	6.663E+17	9.694E+17	7.303E+17	4.453E+17

Fuel Cycle	Cycle Time (EFPY)	Cumulative Time (EFPY)	Neutron (E > 1.0 MeV) Fluence (n/cm^2)			
			60 Deg.	**75 Deg.**	**90 Deg.**	
1-14	14.4	14.4	5.432E+17	7.394E+17	5.308E+17	
15	1.1	15.5	5.595E+17	7.641E+17	5.493E+17	
16	1.2	16.7	5.778E+17	7.926E+17	5.719E+17	
17	1.3	18.0	5.957E+17	8.286E+17	6.001E+17	
18	1.3	19.3	6.148E+17	8.678E+17	6.293E+17	
19	1.3	20.6	6.338E+17	9.066E+17	6.596E+17	
20	1.4	22.0	6.559E+17	9.485E+17	6.911E+17	
21	1.4	23.4	6.782E+17	9.895E+17	7.219E+17	

Data Applicability

Upper Shell Plates D-3802

Upper Shell Longitudinal Welds 1-112

Upper Shell-Intermediate Shell Circumferential Weld 8-112

Table 2.2-7
Projections of Calculated Maximum Neutron Exposure of Pressure Vessel Beltline Materials
Lower Shell Longitudinal Welds, Lower Shell, Intermediate Shell Longitudinal Welds, Intermediate Shell, Intermediate-Lower Shell Circumferential Weld

End of Fuel Cycle	Cycle Time (EFPY)	Cumulative Time (EFPY)	Neutron (E > 1.0 MeV) Fluence (n/cm^2)			
			0 Deg.	15 Deg.	30 Deg.	45 Deg.
21	1.4	23.4	1.447E+19	2.114E+19	1.590E+19	9.677E+18
22	1.4	24.7	1.496E+19	2.201E+19	1.652E+19	1.001E+19
23	1.4	26.1	1.545E+19	2.289E+19	1.717E+19	1.038E+19
24	1.3	27.4	1.592E+19	2.372E+19	1.779E+19	1.073E+19
25	1.4	28.8	1.642E+19	2.461E+19	1.844E+19	1.110E+19
26	1.4	30.2	1.691E+19	2.549E+19	1.910E+19	1.147E+19
27	1.4	31.6	1.741E+19	2.638E+19	1.975E+19	1.184E+19
28	1.4	32.9	1.790E+19	2.726E+19	2.040E+19	1.221E+19
29	1.4	34.3	1.840E+19	2.815E+19	2.106E+19	1.258E+19
30	1.4	35.7	1.889E+19	2.903E+19	2.171E+19	1.295E+19
31	1.4	37.1	1.939E+19	2.992E+19	2.236E+19	1.332E+19
32	1.4	38.5	1.989E+19	3.080E+19	2.302E+19	1.369E+19
33	1.4	39.8	2.038E+19	3.169E+19	2.367E+19	1.406E+19
34	1.4	41.2	2.088E+19	3.257E+19	2.432E+19	1.443E+19
EOLE		42.1	2.118E+19	3.311E+19	2.472E+19	1.466E+19
35	1.4	42.6	2.137E+19	3.346E+19	2.498E+19	1.480E+19
36	1.4	44.0	2.187E+19	3.434E+19	2.563E+19	1.517E+19

End of Fuel Cycle	Cycle Time (EFPY)	Cumulative Time (EFPY)	Neutron (E > 1.0 MeV) Fluence (n/cm^2)			
			60 Deg.	75 Deg.	90 Deg.	
21	1.4	23.4	1.472E+19	2.157E+19	1.576E+19	
22	1.4	24.7	1.520E+19	2.252E+19	1.647E+19	
23	1.4	26.1	1.571E+19	2.345E+19	1.717E+19	
24	1.3	27.4	1.619E+19	2.433E+19	1.784E+19	
25	1.4	28.8	1.670E+19	2.527E+19	1.854E+19	
26	1.4	30.2	1.721E+19	2.621E+19	1.925E+19	
27	1.4	31.6	1.772E+19	2.715E+19	1.995E+19	
28	1.4	32.9	1.823E+19	2.808E+19	2.066E+19	
29	1.4	34.3	1.874E+19	2.902E+19	2.136E+19	
30	1.4	35.7	1.925E+19	2.996E+19	2.207E+19	
31	1.4	37.1	1.976E+19	3.090E+19	2.277E+19	
32	1.4	38.5	2.028E+19	3.183E+19	2.347E+19	
33	1.4	39.8	2.079E+19	3.277E+19	2.418E+19	
34	1.4	41.2	2.130E+19	3.371E+19	2.488E+19	
EOLE		42.1	2.161E+19	3.428E+19	2.531E+19	
35	1.4	42.6	2.181E+19	3.464E+19	2.559E+19	
36	1.4	44.0	2.232E+19	3.558E+19	2.629E+19	

Table 2.2-8
Projections of Calculated Maximum Neutron Exposure of Pressure Vessel Extended Beltline Materials Upper Shell, Upper Shell Longitudinal Welds, Intermediate-Upper Shell Circumferential Weld

End of Fuel Cycle	Cycle Time (EFPY)	Cumulative Time (EFPY)	Neutron (E > 1.0 MeV) Fluence (n/cm^2)			
			0 Deg.	15 Deg.	30 Deg.	45 Deg.
21	1.4	23.4	6.663E+17	9.694E+17	7.303E+17	4.453E+17
22	1.4	24.7	6.874E+17	1.007E+18	7.570E+17	4.596E+17
23	1.4	26.1	7.082E+17	1.044E+18	7.845E+17	4.752E+17
24	1.3	27.4	7.281E+17	1.079E+18	8.107E+17	4.900E+17
25	1.4	28.8	7.492E+17	1.117E+18	8.385E+17	5.058E+17
26	1.4	30.2	7.702E+17	1.154E+18	8.662E+17	5.215E+17
27	1.4	31.5	7.912E+17	1.192E+18	8.939E+17	5.372E+17
28	1.4	32.9	8.123E+17	1.230E+18	9.217E+17	5.529E+17
29	1.4	34.3	8.333E+17	1.267E+18	9.494E+17	5.686E+17
30	1.4	35.7	8.543E+17	1.305E+18	9.772E+17	5.844E+17
31	1.4	37.1	8.754E+17	1.342E+18	1.005E+18	6.001E+17
32	1.4	38.4	8.964E+17	1.380E+18	1.033E+18	6.158E+17
33	1.4	39.8	9.174E+17	1.417E+18	1.060E+18	6.315E+17
34	1.4	41.2	9.385E+17	1.455E+18	1.088E+18	6.472E+17
EOLE		42.1	9.513E+17	1.478E+18	1.105E+18	6.568E+17
35	1.4	42.6	9.595E+17	1.492E+18	1.116E+18	6.630E+17
36	1.4	44.0	9.805E+17	1.530E+18	1.144E+18	6.787E+17

End of Fuel Cycle	Cycle Time (EFPY)	Cumulative Time (EFPY)	Neutron (E > 1.0 MeV) Fluence (n/cm^2)			
			60 Deg.	75 Deg.	90 Deg.	
21	1.4	23.4	6.782E+17	9.895E+17	7.219E+17	
22	1.4	24.7	6.989E+17	1.030E+18	7.525E+17	
23	1.4	26.1	7.204E+17	1.069E+18	7.821E+17	
24	1.3	27.4	7.409E+17	1.107E+18	8.104E+17	
25	1.4	28.8	7.625E+17	1.147E+18	8.403E+17	
26	1.4	30.2	7.842E+17	1.187E+18	8.702E+17	
27	1.4	31.5	8.059E+17	1.226E+18	9.001E+17	
28	1.4	32.9	8.275E+17	1.266E+18	9.300E+17	
29	1.4	34.3	8.492E+17	1.306E+18	9.599E+17	
30	1.4	35.7	8.709E+17	1.346E+18	9.898E+17	
31	1.4	37.1	8.925E+17	1.385E+18	1.020E+18	
32	1.4	38.4	9.142E+17	1.425E+18	1.050E+18	
33	1.4	39.8	9.359E+17	1.465E+18	1.080E+18	
34	1.4	41.2	9.575E+17	1.505E+18	1.109E+18	
EOLE		42.1	9.707E+17	1.529E+18	1.128E+18	
35	1.4	42.6	9.792E+17	1.545E+18	1.139E+18	
36	1.4	44.0	1.001E+18	1.584E+18	1.169E+18	

SECTION 3.0

NEUTRON DOSIMETRY EVALUATIONS

During the first 14 operating fuel cycles at the Palisades plant, five sets of in-vessel surveillance capsule dosimetry and three sets of ex-vessel dosimetry were irradiated, withdrawn, and analyzed. The results of these dosimetry evaluations provide a measurement data base that can be used to demonstrate that the neutron fluence calculations completed for the Palisades reactor meet the uncertainty requirements described in Regulatory Guide 1.190.[6] That is, the calculations and measurements should agree within 20% at the 1σ level.

These calculation/measurement comparisons were previously completed and documented in Reference 1. However, for completeness, a brief description of the measurement program, dosimetry evaluation procedure, and final results are also included in this supplement to Reference 1.

In addition to the Palisades dosimetry evaluations, this general methodology was also used in the determination of capsule exposures from the other PWR's included in Section 4.0 of this report.

3.1 – Method of Analysis

Evaluations of neutron sensor sets contained in the in-vessel and ex-vessel dosimetry capsules withdrawn to date from the Palisades reactor were completed using current state-of-the art least-squares methodology that meet the requirements of Regulatory Guide 1.190[6].

These least-squares adjustment methods provide the capability of combining the measurement data with the neutron transport calculations resulting in a best estimate neutron energy spectrum with associated uncertainties. Best estimates for key exposure parameters such as $\phi(E > 1.0\ \text{MeV})$ and iron atom displacement rate (dpa/s) along with their uncertainties are then easily obtained from the adjusted spectrum. In general, the least-squares methods, as applied to reactor dosimetry evaluations, act to reconcile the measured sensor reaction rate data, dosimetry reaction cross sections, and the calculated neutron energy spectrum within their respective uncertainties.

For example,

$$R_i \pm \delta_{R_i} = \sum_g (\sigma_{ig} \pm \delta_{\sigma_{ig}})(\phi_g \pm \delta_{\phi_g})$$

relates a set of measured reaction rates, R_i, to a single neutron spectrum, ϕ_g, through the multigroup dosimeter reaction cross section, σ_{ig}, each with an uncertainty δ. The primary objective of the least-squares evaluation is to produce unbiased estimates of the neutron exposure parameters at the location of the measurement.

For the least-squares evaluation of the Palisades dosimetry, the NRC approved methodology based on the use of the FERRET adjustment code[5] was employed to combine the results of the plant-specific neutron transport calculations and sensor set reaction rate measurements to determine best estimate values of exposure parameters along with associated uncertainties at the measurement locations.

The application of the least-squares methodology requires the following input.

1. The calculated neutron energy spectrum and associated uncertainties at the measurement location.

2. The measured reaction rate and associated uncertainty for each sensor contained in the multiple foil set.

3. The energy-dependent dosimetry reaction cross sections and associated uncertainties for each sensor contained in the multiple foil sensor set.

For the Palisades application, the calculated neutron spectrum at each measurement location was obtained from the results of plant-specific neutron transport calculations based on the methodology described in Section 2.0 of this report. The calculated spectrum at each sensor set location was input to the adjustment procedure in an absolute sense (rather than as simply a relative spectral shape). Therefore, within the constraints of the assigned uncertainties, the calculated data were treated equally with the measurements. The sensor reaction rates were derived from the measured specific activities of each sensor set and the operating history of the respective fuel cycles. The dosimetry reaction cross sections were obtained from the SNLRML dosimetry cross-section library [10].

In addition to the magnitude of the calculated neutron spectra, the measured sensor set reaction rates, and the dosimeter set reaction cross sections, the least-squares procedure requires uncertainty estimates for each of these input parameters. The following provides a summary of the uncertainties associated with the least-squares evaluation of the Palisades dosimetry.

Reaction Rate Uncertainties

The overall uncertainty associated with the measured reaction rates includes components due to the basic measurement process, the irradiation history corrections, and the corrections for competing reactions. A high level of accuracy in the reaction rate determinations is assured by utilizing laboratory procedures that conform to the ASTM national consensus standards for reaction rate determinations for each sensor type.

After combining all of these uncertainty components, the sensor reaction rates derived from the counting and data evaluation procedures were assigned the following net uncertainties for input to the least-squares evaluation:

Reaction	Uncertainty
$Cu^{63}(n,\alpha)Co^{60}$	5%
$Ti^{46}(n,p)Sc^{46}$	5%
$Fe^{54}(n,p)Mn^{54}$	5%
$Ni^{58}(n,p)Co^{58}$	5%
$U^{238}(n,f)Cs^{137}$	10%
$Nb^{93}(n,n')Nb^{93m}$	5%
$Np^{237}(n,f)Cs^{137}$	10%
$Co^{59}(n,\gamma)Co^{60}$	5%

These uncertainties are given at the 1σ level.

Dosimetry Cross-Section Uncertainties

As noted above, the reaction rate cross sections used in the least-squares evaluations were taken from the SNLRML library. This data library provides reaction cross sections and associated uncertainties, including covariances, for 66 dosimetry sensors in common use. Both cross sections and uncertainties are provided in a fine multigroup structure for use in least-squares adjustment applications. These cross sections were compiled from the most recent cross-section evaluations and they have been tested with respect to their accuracy and consistency for least-squares evaluations. Further, the library has been empirically tested for use in fission spectra determination as well as in the fluence and energy characterization of 14 MeV neutron sources. Detailed discussions of the contents of the SNLRML library along with the evaluation process for each of the sensors is provided in Reference 10.

For sensors included in the Palisades dosimetry sets, the following uncertainties in the fission spectrum-averaged cross sections are provided in the SNLRML documentation package:

Reaction	Uncertainty
$Cu^{63}(n,\alpha)Co^{60}$	4.08-4.16%
$Ti^{46}(n,p)Sc^{46}$	4.50-4.87%
$Fe^{54}(n,p)Mn^{54}$	3.05-3.11%
$Ni^{58}(n,p)Co^{58}$	4.49-4.56%
$U^{238}(n,f)FP$	0.54-0.64%
$Nb^{93}(n,n')Nb^{93m}$	6.96-7.23%
$Np^{237}(n,f)FP$	10.32-10.97%
$Co^{59}(n,\gamma)Co^{60}$	0.79-3.59%

These tabulated ranges provide an indication of the dosimetry cross-section uncertainties associated with the sensor sets used in LWR irradiations.

Calculated Neutron Spectrum Uncertainties

While the uncertainties associated with the reaction rates were obtained from the measurement procedures and counting benchmarks, and the dosimetry cross-section uncertainties were supplied directly with the SNLRML library, the uncertainty matrix for the calculated spectrum was constructed from the following relationship:

$$M_{gg'} = R_n^2 + R_g * R_{g'} * P_{gg'}$$

where R_n specifies an overall fractional normalization uncertainty, and the fractional uncertainties $R_{g'}$ and R_g specify additional random groupwise uncertainties that are correlated with a correlation matrix given by:

$$P_{gg'} = [1 - \theta]\,\delta_{gg'} + \theta\, e^{-H}$$

where

$$H = \frac{(g - g')^2}{2\gamma^2}$$

The first term in the correlation matrix equation specifies purely random uncertainties, while the second term describes the short-range correlations over a group range γ (θ specifies the strength of the latter term). The value of δ is 1.0 when g = g’ and 0.0 otherwise.

The set of parameters defining the input covariance matrix for the Palisades calculated spectra was as follows:

Flux Normalization Uncertainty (R_n)	15%
Flux Group Uncertainties (R_g, $R_{g'}$)	
(E > 0.0055 MeV)	15%
(0.68 eV < E < 0.0055 MeV)	29%
(E < 0.68 eV)	52%
Short-Range Correlation (θ)	
(E > 0.0055 MeV)	0.9
(0.68 eV < E < 0.0055 MeV)	0.5
(E < 0.68 eV)	0.5

Flux Group Correlation Range (γ)	
(E > 0.0055 MeV)	6
(0.68 eV < E < 0.0055 MeV)	3
(E < 0.68 eV)	2

These uncertainty assignments are consistent with an industry consensus uncertainty of 15-20% (1σ) for the fast neutron portion of the spectrum and provide for a reasonable increase in the uncertainty for neutrons in the intermediate and thermal energy ranges.

3.2 – Dosimetry Evaluations

In this section, comparisons of the measurement results from the Palisades surveillance capsule and reactor cavity dosimetry with corresponding analytical predictions at the measurement locations are presented. These comparisons are provided on two levels. In the first instance, calculations of individual sensor reaction rates are compared directly with the measured reaction rates derived from the counting data obtained from the radiochemical laboratories. In the second case, the calculated values of neutron exposure expressed in terms of ϕ(E > 1.0 MeV), ϕ(E > 0.1 MeV), and iron atom displacements (dpa) are compared with the results of the least squares adjustment procedure described in Section 3.1. It is shown that these two levels of comparison yield consistent and similar results which demonstrate that the transport calculations for Palisades reactor produce neutron exposure results that meet the requirements of Regulatory Guide 1.190.[6]

In Table 3.2-1, measurement/calculation (M/C) ratios for each fast neutron sensor reaction from surveillance capsule and reactor cavity irradiations are listed. This comparison provides a direct comparison, on an absolute basis, of calculation and measurement prior to the application of the least squares adjustment procedure. In Table 3.2-2, comparisons of measured and adjusted neutron exposures are given in terms of adjusted/calculated ratios for the five surveillance capsule dosimetry sets withdrawn to date as well as for the three cycles of reactor cavity midplane dosimetry sets irradiated during Cycles 8, 9, and 10/11.

Table 3.2-1

Comparison of Measured and Calculated Threshold Foil Reaction Rates

Capsule	M/C Ratio					
	^{63}Cu(n,α)	^{46}Ti(n,p)	^{54}Fe(n,p)	^{58}Ni(n,p)	^{238}U(n,f)	^{237}Np(n,f)
A240	1.09	1.21	1.02	0.95		
W290	1.15	1.11	0.99	1.00	0.98	
W290-9	1.12	1.16	0.96	0.98	0.96	0.92
W110	1.17	1.17	1.02	1.01		
SA60-1	1.13	1.19	1.05	1.07	1.15	
84° Cavity						
Cycle 9	1.11	1.10	1.08	1.03	1.13	1.21
Cycle 10/11	1.15	1.11	1.10	1.08	1.32	1.11
74° Cavity						
Cycle 8	1.09	1.14	1.08	1.07	1.06	1.40
Cycle 9	1.03	1.07	1.01	1.01	0.93	1.13
Cycle 10/11	1.08	1.05	1.02	1.03	1.07	1.08
64° Cavity						
Cycle 8	1.09	1.15	1.08	1.06	1.04	1.32
Cycle 9	1.05	1.08	1.01	1.03	1.09	1.24
Cycle 10/11	1.07	1.10	1.05	1.03	1.10	1.12
54° Cavity						
Cycle 10/11	1.09	1.05	1.00		1.06	1.04
39° Cavity						
Cycle 8	1.08	1.21	1.14	1.11	1.06	1.32
Cycle 9	1.06	1.06	0.99	1.00	0.87	0.98
Cycle 10/11	1.03	1.12	1.05	1.05	1.06	1.06
24° Cavity						
Cycle 10/11	1.03	1.08	1.03	1.04	1.19	0.96
Average	**1.09**	**1.12 1.04**	**1.03 1.07 1.14**			
% std dev	**3.9**	**4.7 4.4 3.8**			**10.0**	**12.8**

Reaction	Average M/C	% Standard Deviation
^{63}Cu(n,α)	1.09	3.9
^{46}Ti(n,p)	1.12	4.7
^{54}Fe(n,p)	1.04	4.4
^{58}Ni(n,p)	1.03	3.8
^{238}U(n,f)	1.07	10.0
^{237}Np(n,f)	1.14	12.8
Linear Average	**1.08**	**7.9**

Table 3.2-2

Comparison of Adjusted and Calculated Exposure Parameters

Capsule	Adjusted/Calculated (A/C) Ratio		
	ϕ(E > 1.0 MeV)	**ϕ(E > 0.1 MeV)**	**dpa**
A240	0.983	0.972	0.988
W290	0.988	0.981	0.997
W290-9	0.955	0.937	0.966
W110	1.011	1.001	1.020
SA60-1	1.078	1.067	1.077
84° Cavity			
Cycle 9	1.091	1.083	1.084
Cycle 10/11	1.142	1.133	1.134
74° Cavity			
Cycle 8	1.108	1.120	1.116
Cycle 9	0.999	0.993	0.996
Cycle 10/11	1.044	1.058	1.055
64° Cavity			
Cycle 8	1.086	1.096	1.092
Cycle 9	1.055	1.033	1.038
Cycle 10/11	1.065	1.078	1.075
54° Cavity			
Cycle 10/11	1.026	1.039	1.036
39° Cavity			
Cycle 8	1.116	1.139	1.135
Cycle 9	0.949	0.956	0.957
Cycle 10/11	1.058	1.060	1.060
24° Cavity			
Cycle 10/11	1.062	1.050	1.053
Average	**1.05**	**1.04 1.05**	
% std dev	**5.3**	**5.8**	**5.1**

SECTION 4.0

SURVEILLANCE CAPSULE NEUTRON FLUENCE

In support of embrittlement evaluations for the Palisades reactor pressure vessel, a compilation of calculated neutron fluence (E > 1.0 MeV) values for a series of materials surveillance capsules that contain test samples that apply to the Palisades plant is provided in this section. The compilation, encompassing a total of 18 surveillance capsules irradiated at the Palisades, Indian Point Unit 2, H. B. Robinson Unit 2, and Indian Point Unit 3 reactors is provided in Table 4-1.

For each surveillance capsule listed in Table 4-1, the reported fluence value was calculated using an NRC approved methodology that meets the requirements of Regulatory Guide 1.190[6]. Therefore, this tabulation represents a consistent set of fluence values for use in data correlations. Details of the analysis methodology as applied to each of the four host reactors are given in References 1, 11, 12, 13, 14, and 15.

In providing the data listed in Table 4-1, no new fluence calculations were performed. The data were obtained either from proprietary Palisades specific documents[11, 12] or from non-proprietary public domain documents[1, 13, 14, 15] that have been submitted to the NRC and are available on the ADAMS document system. It should be noted that, relative to the Palisades data listed in Table 4-1, References 1, 11, and 12 did not explicitly report fluence (E > 1.0 MeV) values for the individual capsules. Rather, the irradiation environment was reported in terms of irradiation time and calculated neutron flux (E > 1.0 MeV) averaged over the irradiation period. The fluence values listed in Table 4-1 were computed as the product of the irradiation time and the average neutron flux reported in these documents.

Relative to the data in Table 4-1 and the listed references, it should also be noted that, in addition to the Reg. Guide 1.190 derived fluence values for Indian Point Unit 2, Table 3 of Reference 13 also lists fluence values for H. B. Robinson Unit 2 and Indian Point Unit 3 that were extracted from older references. These older values have been updated and superseded by the fluence values documented in References 14 and 15, respectively. All of these updated fluence values reflect the application of a fluence methodology that meets the requirements of Reg. Guide 1.190.

Table 4-1

Summary of Neutron Fluence (E > 1.0 MeV) Derived from the Application of Methodology Meeting the Requirements of Regulatory Guide 1.190

Reactor	Surveillance Capsule Designation	Fluence (E > 1.0 Mev) [n/cm^2]	Reference
Palisades	A240	4.09e+19	WCAP-15353, R0 (Ref. 1)
Palisades	W290	9.38e+18	WCAP-15353, R0 (Ref. 1)
Palisades	W110	1.64e+19	WCAP-15353, R0 (Ref. 1)
Palisades	SA60-1	1.50e+19	WCAP-15353, R0 (Ref. 1)
Palisades	SA240-1	2.38e+19	CPAL-01-009 (Ref. 11)
Palisades	W100	2.09e+19	CPAL-04-8 (Ref. 12)
Indian Point 2	T	2.53e+18	WCAP-15629, R1 (Table 3) (Ref. 13)
Indian Point 2	Y*	4.55e+18	WCAP-15629, R1 (Table 3) (Ref. 13)
Indian Point 2	Z	1.02e+19	WCAP-15629, R1 (Table 3) (Ref. 13)
Indian Point 2	V*	4.92e+18	WCAP-15629, R1 (Table 3) (Ref. 13)
H. B. Robinson	S	4.79e+18	WCAP-15805, R0 (Table 5-10) (Ref. 14)
H. B. Robinson	V*	5.30e+18	WCAP-15805, R0 (Table 5-10) (Ref. 14)
H. B. Robinson	T*	3.87e+19	WCAP-15805, R0 (Table 5-10) (Ref. 14)
H. B. Robinson	X*	4.49e+19	WCAP-15805, R0 (Table 5-10) (Ref. 14)
Indian Point 3	T*	2.63e+18	WCAP-16251-NP, R0 (Table 5-10) (Ref. 15)
Indian Point 3	Y*	6.92e+18	WCAP-16251-NP, R0 (Table 5-10) (Ref. 15)
Indian Point 3	Z*	1.04e+19	WCAP-16251-NP, R0 (Table 5-10) (Ref. 15)
Indian Point 3	X*	8.74e+18	WCAP-16251-NP, R0 (Table 5-10) (Ref. 15)

Notes:

1 - Relative to the Palisades data, References 1, 11, and 12 did not explicitly report fluence values for the listed capsules. Rather, the irradiation environment was reported in terms of irradiation time and neutron flux averaged over the irradiation period. The fluence values listed in Table 4-1 were computed as the product of the irradiation time and the average neutron flux (E > 1.0 MeV) reported in those documents.

2 - In addition to the Reg. Guide 1.190 derived fluence values for Indian Point Unit 2, Table 3 of Reference 13 also lists fluence values for H. B. Robinson and Indian Point Unit 3 that were taken from older references. These values have been updated and superseded by the fluence values documented in References 14 and 15 that are based on a methodology that meets the requirements of Reg. Guide 1.190.

* Indicates Capsules in other plants that contain W5214 weld material.

SECTION 5.0

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