ATTACHMENT 3

WCAP-15353-NP, REVISION 0, SUPPLEMENT 1

PALISADES REACTOR PRESSURE VESSEL FLUENCE EVALUATION

29 pages follow

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WCAP-15353 – Supplement 1-NP
Revision 0

May 2010

Palisades Reactor Pressure Vessel Fluence Evaluation

Westinghouse

WCAP-15353 – Supplement 1-NP, Revision 0

Palisades Reactor Pressure Vessel Fluence Evaluation

S. L. Anderson*, Fellow Engineer
Radiation Engineering & Analysis

May 2010

Reviewed:
P. M. Song*, Principal Engineer
Radiation Engineering & Analysis

Approved:
A. R. Dulloo*, Manager
Radiation Engineering & Analysis

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P.O. Box 355
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EXECUTIVE SUMMARY

Calculations of the neutron exposure of the Palisades reactor pressure vessel were previously completed and documented in WCAP-15353, Revision 0, "Palisades Reactor Vessel Neutron Fluence Evaluation," January 2000.[3] This evaluation was submitted for review by the NRC Staff and, after consideration of RAI's addressed in Reference 4, the fluence methodology as well as the final results were approved by the Staff.

The fluence analysis described in WCAP-15353, Revision 0[3] included cycle specific evaluations through Cycle 14 (the then current operating cycle). This supplement to WCAP-15353 provides an updated neutron fluence assessment for the Palisades pressure vessel beltline region that includes cycle specific analysis for additional operating cycles for which the design has been finalized (Cycles 15 through 21) and includes projections for future operation through approximately 44 effective full power years (EFPY). Updated evaluations of surveillance capsule credibility analysis and determination of material chemistry factors are being completed in parallel with this fluence calculation and will be documented elsewhere.

Based on the cycle specific analysis through Cycle 21 (approximately 23.4 EFPY) and the projection scenario for future operation provided by Entergy, the maximum neutron exposure of the pressure vessel beltline materials through 44 EFPY is summarized as follows.

End of Fuel Cycle	Estimated Calendar Date	Cumulative Time (EFPY)	Neutron (E > 1.0 MeV) Fluence (n/cm^2)			
			0 Deg.	15 Deg.	30 Deg.	45 Deg.
21	10/2010	23.4	1.447E+19	2.114E+19	1.590E+19	9.677E+18
22	4/2012	24.7	1.496E+19	2.201E+19	1.652E+19	1.001E+19
23	10/2013	26.1	1.545E+19	2.288E+19	1.717E+19	1.038E+19
24	4/5/2015	27.4	1.592E+19	2.372E+19	1.779E+19	1.073E+19
25	10/2016	28.8	1.642E+19	2.461E+19	1.844E+19	1.110E+19
26	4/2018	30.2	1.691E+19	2.549E+19	1.909E+19	1.147E+19
27	10/2019	31.5	1.741E+19	2.637E+19	1.975E+19	1.184E+19
28	4/2021	32.9	1.790E+19	2.726E+19	2.040E+19	1.221E+19
29	10/2022	34.3	1.840E+19	2.814E+19	2.105E+19	1.258E+19
30	4/2024	35.7	1.889E+19	2.903E+19	2.170E+19	1.295E+19
31	10/2025	37.1	1.939E+19	2.991E+19	2.236E+19	1.332E+19
32	4/2027	38.4	1.988E+19	3.079E+19	2.301E+19	1.369E+19
33	10/2028	39.8	2.038E+19	3.168E+19	2.366E+19	1.406E+19
34	4/2030	41.2	2.087E+19	3.256E+19	2.432E+19	1.443E+19
35	10/2031	42.6	2.137E+19	3.344E+19	2.497E+19	1.480E+19
36	4/2033	44.0	2.186E+19	3.433E+19	2.562E+19	1.517E+19

End of Fuel Cycle	Estimated Calendar Date	Cumulative Time (EFPY)	Neutron (E > 1.0 MeV) Fluence (n/cm^2)			
			60 Deg.	75 Deg.	90 Deg.	
21	10/2010	23.4	1.472E+19	2.157E+19	1.575E+19	
22	4/2012	24.7	1.520E+19	2.252E+19	1.647E+19	
23	10/2013	26.1	1.571E+19	2.345E+19	1.717E+19	
24	4/5/2015	27.4	1.619E+19	2.433E+19	1.784E+19	
25	10/2016	28.8	1.670E+19	2.527E+19	1.854E+19	
26	4/2018	30.2	1.721E+19	2.621E+19	1.925E+19	
27	10/2019	31.5	1.772E+19	2.714E+19	1.995E+19	
28	4/2021	32.9	1.823E+19	2.808E+19	2.065E+19	
29	10/2022	34.3	1.874E+19	2.902E+19	2.136E+19	
30	4/2024	35.7	1.925E+19	2.995E+19	2.206E+19	
31	10/2025	37.1	1.976E+19	3.089E+19	2.277E+19	
32	4/2027	38.4	2.027E+19	3.182E+19	2.347E+19	
33	10/2028	39.8	2.078E+19	3.276E+19	2.417E+19	
34	4/2030	41.2	2.129E+19	3.370E+19	2.488E+19	
35	10/2031	42.6	2.180E+19	3.463E+19	2.558E+19	
36	4/2033	44.0	2.231E+19	3.557E+19	2.628E+19	

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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 Appendix G to 10 CFR 50[1], 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".* Each of the materials comprising the beltline region must be considered in the overall embrittlement assessments for the pressure vessel. Therefore, plant-specific exposure assessments must include evaluations as a function of position over the beltline region.

Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [2], 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[3] 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 4, the methodology described in WCAP-15353, Revision 0 was approved for application to the Palisades reactor pressure vessel. Subsequent to that approval additional submittals[7,8] in support of the benchmarking of this fluence methodology were reviewed and approved by the NRC Staff.

The fluence analysis described in WCAP-15353, Revision 0[3] included cycle specific evaluations through Cycle 14 (the then current operating cycle). This supplement to WCAP-15353 provides an updated neutron fluence assessment for the Palisades pressure vessel beltline region that includes cycle specific analysis for additional operating cycles for which the design has been finalized (Cycles 15 through 21) and includes projections for future operation through approximately 44 effective full power years (EFPY). The results of this evaluation are intended for use as input to vessel materials analyses (to be documented elsewhere) that include updates to surveillance capsule credibility analysis and material chemistry factor determination.

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, this report also includes the latest fluence evaluation from capsules containing the W5214 material. This compilation of capsule fluence values is based on the same fluence methodology described in this report.

In subsequent sections of this supplement, the methodologies used to perform neutron transport calculations and dosimetry evaluations are described in some detail, the updated results of the plant specific transport calculations are given for the beltline region of the Palisades pressure vessel. Comparisons of calculations and measurements demonstrating that the transport calculations meet the requirements of Regulatory Guide 1.190 that were previously included in Reference 3 are also included in this supplement for completeness. 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 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.

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[5] and the BUGLE-96 cross section library[6]. 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.

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 current evaluations. A plan views 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 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 one foot below the active fuel to an axial elevation one foot 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.

The core power distributions used in the plant-specific transport analysis for the reactor were provided by Entergy.[15] The data used in the source generation included fuel assembly-specific initial enrichments, beginning-of-cycle burnups and end-of-cycle burnups. Appropriate axial distributions were also obtained.

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.

Figure 2.1-1

Palisades r,θ Reactor Geometry

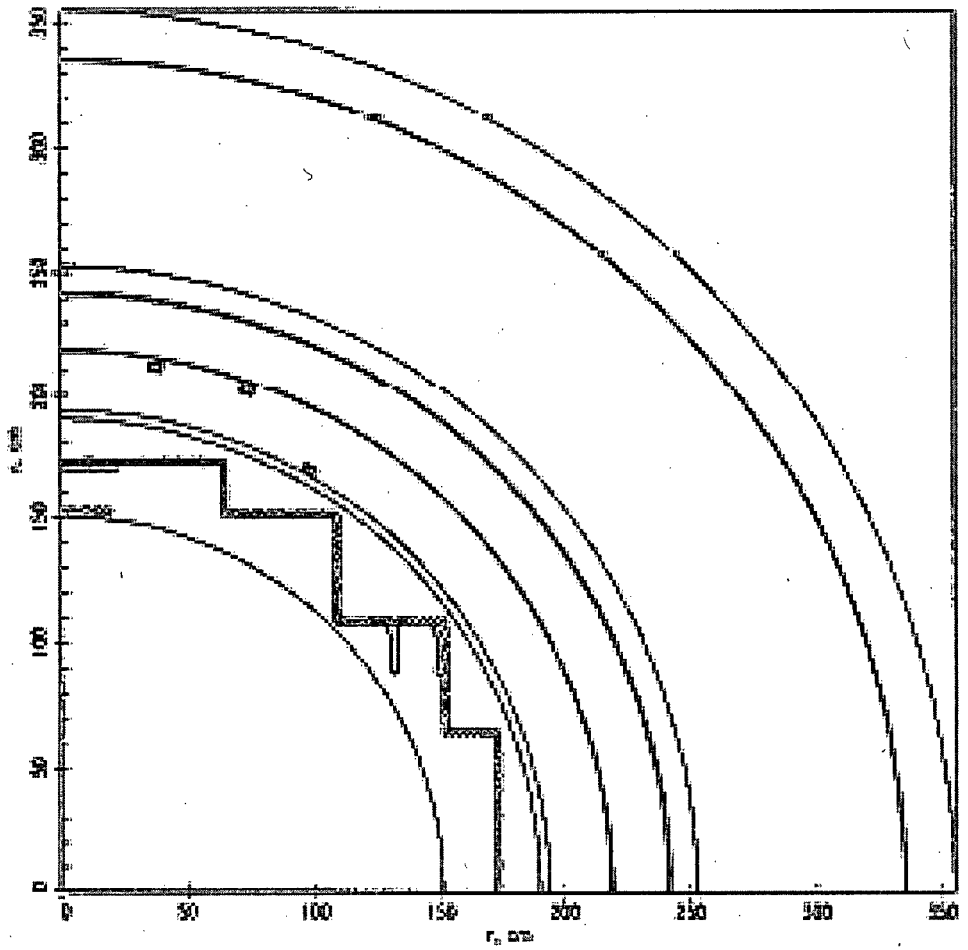
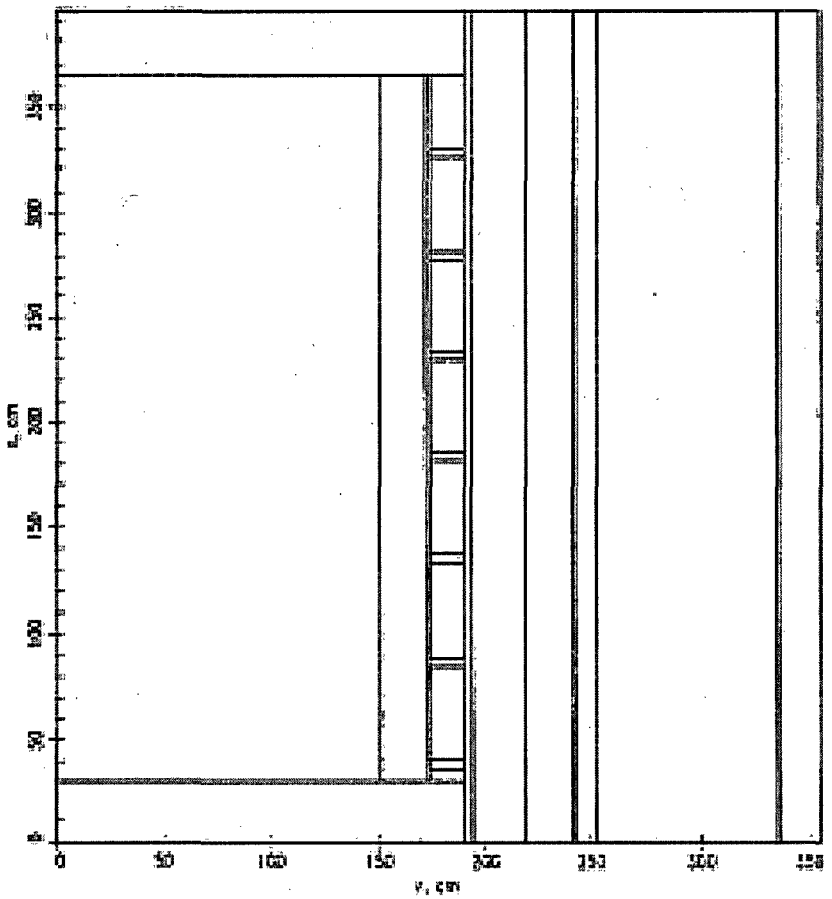


Figure 2.1-2

Palisades r,z Reactor Geometry



2.2 – Calculated Neutron Exposure of Pressure Vessel Beltline Materials

The plant- and 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 is given in Tables 2.2-1 and 2.2-2, 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 2.2-1 and 2.2-2 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 2.2-1 and 2.2-2 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 Table 2.2-3, projections of neutron (E > 1.0 MeV) fluence beyond the end of Cycle 21 are provided. 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 (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 2.2-1, 2.2-2, and 2.2-3, 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 is exposed to the maximum neutron exposure characteristic of the 75° azimuthal location.

Table 2.2-1

Summary of Calculated Maximum Pressure Vessel Neutron Flux (E > 1.0 MeV) For Cycles 15 Through 21 and for Future Projection

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	1.558E+10	1.277E+10	7.924E+09
16	1.2	1.068E+10	1.604E+10	1.330E+10	7.797E+09
17	1.3	1.080E+10	1.860E+10	1.332E+10	7.613E+09
18	1.3	1.292E+10	2.094E+10	1.352E+10	7.337E+09
19	1.3	1.059E+10	1.924E+10	1.445E+10	7.037E+09
20	1.4	1.123E+10	2.004E+10	1.517E+10	8.143E+09
21	1.4	1.138E+10	2.016E+10	1.501E+10	8.506E+09
22	Proj.	1.153E+10	2.024E+10	1.454E+10	7.756E+09
23+	Proj.	1.138E+10	2.016E+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	1.681E+10	1.257E+10	
16	1.2	1.135E+10	1.762E+10	1.401E+10	
17	1.3	9.781E+09	1.967E+10	1.539E+10	
18	1.3	1.088E+10	2.235E+10	1.664E+10	
19	1.3	1.090E+10	2.230E+10	1.743E+10	
20	1.4	1.161E+10	2.198E+10	1.650E+10	
21	1.4	1.172E+10	2.151E+10	1.618E+10	
22	Proj.	1.128E+10	2.204E+10	1.669E+10	
23+	Proj.	1.172E+10	2.151E+10	1.618E+10	

Table 2.2-2

Summary of Calculated Maximum Pressure Vessel Neutron Exposure
Through the Conclusion of Cycle 21

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	8.366E+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	9.296E+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	1.575E+19	

Table 2.2-3

Projections of Calculated Maximum Pressure Vessel Neutron Exposure

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	2.288E+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	1.909E+19	1.147E+19
27	1.4	31.5	1.741E+19	2.637E+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	2.814E+19	2.105E+19	1.258E+19
30	1.4	35.7	1.889E+19	2.903E+19	2.170E+19	1.295E+19
31	1.4	37.1	1.939E+19	2.991E+19	2.236E+19	1.332E+19
32	1.4	38.4	1.988E+19	3.079E+19	2.301E+19	1.369E+19
33	1.4	39.8	2.038E+19	3.168E+19	2.366E+19	1.406E+19
34	1.4	41.2	2.087E+19	3.256E+19	2.432E+19	1.443E+19
35	1.4	42.6	2.137E+19	3.344E+19	2.497E+19	1.480E+19
36	1.4	44.0	2.186E+19	3.433E+19	2.562E+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	1.575E+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	31.5	1.772E+19	2.714E+19	1.995E+19	
28	1.4	32.9	1.823E+19	2.808E+19	2.065E+19	
29	1.4	34.3	1.874E+19	2.902E+19	2.136E+19	
30	1.4	35.7	1.925E+19	2.995E+19	2.206E+19	
31	1.4	37.1	1.976E+19	3.089E+19	2.277E+19	
32	1.4	38.4	2.027E+19	3.182E+19	2.347E+19	
33	1.4	39.8	2.078E+19	3.276E+19	2.417E+19	
34	1.4	41.2	2.129E+19	3.370E+19	2.488E+19	
35	1.4	42.6	2.180E+19	3.463E+19	2.558E+19	
36	1.4	44.0	2.231E+19	3.557E+19	2.628E+19	

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.[2] 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 3. However, for completeness, a brief description of the measurement program, dosimetry evaluation procedure, and final results are also included in this supplement to Reference 3.

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[8].

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[8] 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. [9]

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 9.

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.[2]

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[2]. 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 3, 12, 13, and 14.

In providing the data listed in Table 4-1, no new fluence calculations were performed. The data were obtained either from Palisades specific documents[10, 11] or from public domain documents[3, 12, 13, 14] 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 3, 10, and 11 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 12 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 13 and 14, 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. 3)
Palisades	W290	9.38e+18	WCAP-15353, R0 (Ref. 3)
Palisades	W100-1	1.64e+19	WCAP-15353, R0 (Ref. 3)
Palisades	SA60-1	1.50e+19	WCAP-15353, R0 (Ref. 3)
Palisades	SA240-1	2.38e+19	CPAL-01-009 (Ref. 10)
Palisades	W100-2	2.09e+19	CPAL-04-8 (Ref. 11)
Indian Point 2	T	2.53e+18	WCAP-15629, R1 (Table 3) (Ref. 12)
Indian Point 2	Y*	4.55e+18	WCAP-15629, R1 (Table 3) (Ref. 12)
Indian Point 2	Z	1.02e+19	WCAP-15629, R1 (Table 3) (Ref. 12)
Indian Point 2	V*	4.92e+18	WCAP-15629, R1 (Table 3) (Ref. 12)
H. B. Robinson	S	4.79e+18	WCAP-15805, R0 (Table 5-10) (Ref. 13)
H. B. Robinson	V*	5.30e+18	WCAP-15805, R0 (Table 5-10) (Ref. 13)
H. B. Robinson	T*	3.87e+19	WCAP-15805, R0 (Table 5-10) (Ref. 13)
H. B. Robinson	X*	4.49e+19	WCAP-15805, R0 (Table 5-10) (Ref. 13)
Indian Point 3	T*	2.63e+18	WCAP-16251-NP, R0 (Table 5-10) (Ref. 14)
Indian Point 3	Y*	6.92e+18	WCAP-16251-NP, R0 (Table 5-10) (Ref. 14)
Indian Point 3	Z*	1.04e+19	WCAP-16251-NP, R0 (Table 5-10) (Ref. 14)
Indian Point 3	X*	8.74e+18	WCAP-16251-NP, R0 (Table 5-10) (Ref. 14)

Notes:

1 - Relative to the Palisades data, References 1, 10, and 11 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 12 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 13 and 14 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

REFERENCES

1. Code of Federal Regulations Title 10 Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix G, "Fracture Toughness Requirements" and Appendix H, "Reactor Vessel Materials Surveillance Requirements," January 1992.

2. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, March 2001.

3. WCAP-15353, Revision 0, "Palisades Reactor Pressure Vessel Neutron Fluence Evaluation," G. K. Roberts et al., January 2000.

4. LTR-REA-00-630, "Transmittal of Responses to Requests for Additional Information on WCAP-15353 in Support of the Palisades Pressure Vessel Fluence Evaluation," G. K. Roberts, July 13, 2000.

5. CCC-650, "DOORS 3.2, One-, Two-, and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," April 1998. Available from the Radiation Safety Information Computational Center, Oak Ridge National Laboratory.

6. 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," March 1996. Available from the Radiation Safety Information Computational Center, Oak Ridge National Laboratory.

7. WCAP-14040-NP-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004..

8. WCAP-16083-NP-A, Revision 0, "Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry," S. L. Anderson, May 2006.

9. DLC-178, "SNLRML Recommended Dosimetry Cross-Section Compendium," July 1994. Available from the Radiation Safety Information Computational Center, Oak Ridge National Laboratory.

10. Westinghouse Project Letter CPAL-01-009, "Fluence Analysis of Palisades Surveillance Capsule SA-240-1," W. R. Rice, April 30, 2001.

11. Westinghouse Project Letter CPAL-04-8, "Fluence Analysis for Reactor Vessel Surveillance Capsule W100," S. P. Swigart, February 11 2004.

12. WCAP-15629, Revision 1, "Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation," T. J. Laubham, December 2001.

13. WCAP-15805, Revision 0, "Analysis of Capsule X from the Carolina Power & Light Company H. B. Robinson Unit 2 Reactor Vessel Radiation Surveillance Program," T. J. Laubham, et al., March 2002.

14. WCAP-16251-NP, Revision 0, "Analysis of Capsule X from Entergy's Indian Point Unit 3 Reactor Vessel Radiation Surveillance Program," T. J. Laubham, et al, July 2004.

15. Palisades Calculation EA-DOR-09-01, "Reactor Pressure Vessel Fluence Calculations for Cycle 20 and an Estimate of Cycle 21," Thomas W. Allen, June 10, 2009.

ATTACHMENT 4

BACKGROUND INFORMATION, DESCRIPTION OF PROPOSED CHANGES, AND EVALUATION DISCUSSION

10 pages follow

BACKGROUND INFORMATION

This submittal provides in Attachments 1 and 2 updated pressurized thermal shock (PTS) evaluations for the Palisades Nuclear Plant (PNP) reactor pressure vessel beltline materials. Attachment 3 provides a revised PNP reactor vessel (RV) fluence evaluation in support of the PTS evaluations.

These evaluations were generated in response to a license renewal requirement under 10 CFR 54.21(c)(1)(iii) (Reference 1) to adequately manage the effects of aging on the intended functions described in 10 CFR 54.4 for the period of extended operation through the license renewal period. Section 4.2.2 of the PNP license renewal application (Reference 2) states "At the appropriate time, prior to exceeding the PTS screening criteria, Palisades will select the optimum alternative to manage PTS in accordance with NRC regulations, and will make the applicable submittals to obtain NRC review and approval."

The PNP license renewal application indicates that the limiting RV welds are projected to reach the PTS screening criteria in 2014, prior to the end of the license renewal period. The limiting locations were the beltline axial welds fabricated with weld wire heat no. W5214.

10 CFR 50.61 (Reference 3) requires under specific circumstances the following actions:

1. Implement a flux reduction program pursuant to 10 CFR 50.61(b)(3) that is reasonably practicable to avoid exceeding the screening criteria,
2. Submit a safety analysis pursuant to 10 CFR 50.61(b)(4) to determine what, if any, modifications to equipment, systems and plant operation are necessary to prevent failure of the RV from a postulated PTS event, or
3. Perform a thermal-annealing treatment of the RV pursuant to 10 CFR 50.61(b)(7) to recover fracture toughness.

In addition, 10 CFR 50.61 states that the PTS assessment must be updated whenever there is a significant change in projected values of RT_{PTS}, or upon request for a change in the expiration date for operation of the facility.

Subsequent to license renewal application approval, the NRC revised the Code of Federal Regulations to provide an additional option for management of PTS under 10 CFR 50.61a, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events." This new option involves inspection of the RV beltline region and determination of limiting RT_{MAX-X} values for each axial weld, plate, forging, and circumferential weld. The 10 CFR 50.61a regulation requires that an application for implementation of

10 CFR 50.61a be submitted for review and approval at least three years before the limiting RT_{PTS} value calculated under 10 CFR 50.61 is projected to exceed the PTS screening criteria.

Entergy Nuclear Operations, Inc. (ENO) employed Structural Integrity Associates, Inc. to review and update the RV PTS evaluation to ensure that required actions, including inspection of the RV beltline region welds, would be completed in accordance with regulatory requirements.

On June 22, 2010, ENO representatives met with NRC staff to share the preliminary results of the recently completed PTS evaluation for the limiting axial welds fabricated with weld wire heat no. W5214. During the meeting, the use of surveillance capsule data was presented to demonstrate that the axial welds fabricated with weld wire heat no. W5214 will not reach the PTS screening criterion limit until April 2017. Based on the preliminary results and other factors, ENO had changed the schedule for inspection of the RV beltline region welds from the fall 2010 refueling outage to the spring 2012 refueling outage.

Following the planned inspection of the RV beltline region in the spring of 2012, ENO intends to calculate RT_{MAX-X} values for each RV beltline material and to submit an application requesting approval for implementation of 10 CFR 50.61a. Submittal of this application to use 10 CFR 50.61a is planned to occur no less than three years before the limiting axial welds fabricated with weld wire heat no. W5214 are projected to reach PTS screening criteria cited in 10 CFR 50.61.

During the June 22, 2010, meeting, ENO agreed to submit revised PTS evaluations for the RV beltline materials. The revised PTS evaluations would communicate and document that compliance with 10 CFR 50.61 requirements will continue to be satisfied while the RV inspection is completed, the RT_{MAX-X} values for each RV beltline materials are determined, and the NRC reviews the application to implement 10 CFR 50.61a.

DESCRIPTION OF PROPOSED CHANGES

The current PNP PTS evaluation was submitted to the NRC in 2000 (Reference 5) and was also referenced later in the PNP renewed operating license application. This PTS evaluation was projected to remain valid until 2014. In preparation for entering the period of extended plant operation, ENO has updated the PTS evaluation to reflect the material chemistry factors based upon available surveillance capsule data and fluence projections for future power operation.

In the updated PTS evaluation, the revised values of RT_{PTS} in Attachments 1 and 2 have been determined in accordance with 10 CFR 50.61. The equations for determining RT_{PTS} are as follows:

$RT_{PTS} = RT_{NDT(U)} + M + \Delta RT_{PTS}$

$\Delta RT_{PTS} = CF \times FF$

where,

$RT_{NDT(U)}$ = the reference temperature of nil ductility transition for the unirradiated material.

M = margin term to cover for uncertainties in the value of initial RT_{NDT} and the scatter in the shift.

$$M = 2 * \sqrt{\sigma_I^2 + \sigma_\Delta^2}$$

σ_I = the standard deviation for the initial RT_{NDT} (°F). For non-Linde 80 type welds, if a generic initial RT_{NDT} value is used, σ_I = 17°F. If a measured value is used for the initial RT_{NDT}, σ_I = 0°F.

σ_Δ = the standard deviation for ΔRT_{NDT} (°F). The values for σ_Δ are 28°F for welds and 17° for base metal (plates or forgings).

ΔRT_{PTS} = the mean value of the transition temperature shift due to irradiation.

CF = chemistry factor in °F, which is a function of the copper and nickel content, obtained from either the tables or a fitted CF value from surveillance data.

FF = fluence factor = $f^{(0.28 - 0.10 \log(f))}$

where,

f = neutron fluence, in units of 10^{19} n/cm2 (E > 1 MeV), at the clad/base metal interface.

10 CFR 50.61 states that σ_Δ need not exceed 0.5 times the mean reference temperature shift (0.5* ΔRT_{NDT}), or the standard value of σ_Δ of 28°F for welds and 17°F for base metal (plates or forgings), whichever is lower. Note that the margin term, M, may be reduced by half if credit is obtained for credible surveillance data.

The current RT_{PTS} values are documented in Section 4.2.2 of the license renewal application in Table 4.2.2-1, "Estimated RT_{PTS} on March 24, 2031" (Reference 2).

Inspection of the information in Table 4.2.2-1 of the license renewal application reveals that the current projected RT_{PTS} values are based on the method in the PTS rule given in 10 CFR 50.61(c)(1) using the best estimate chemistry for the various beltline region welds and plates, the corresponding chemistry factor and the fluence values from the vessel fluence evaluation in WCAP-15353 (Reference 6). With the exception of plate heat no. C-1279, no credit was given for surveillance data to improve the current RT_{PTS} projections since the data available at that time were limited for the materials other than this plate and were determined to be non-credible.

For plate heat no. C-1279, the current projected RT_{PTS} values are based on the method in the PTS rule given in 10 CFR 50.61(c)(2) and (c)(3), for use of surveillance data and the fluence values from the vessel fluence evaluation in WCAP-15353, Revision 0 (Reference 6). Credit was given for credible surveillance data for both the fitted chemistry factor and a reduced margin term for plate heat no. C-1279.

The current fluence projection used to generate the current RT_{PTS} values is described in WCAP-15353, Revision 0. That evaluation, along with the benchmarking method, was submitted for review by the NRC and the methodology and the final results were approved as part of the PTS evaluation in 2000 (Reference 7). The fluence projection at the time determined that the peak fluence at the clad-to-base-metal interface at the 60° limiting axial weld was 1.158×10^{19} n/cm^2 (E > 1 MeV) at the end of cycle 14 (i.e., October 1999). The calculated exposure rate at the pressure vessel for fuel cycle 15 was used for extrapolating the fast neutron fluence into the future with an assumed capacity factor of 89%.

The revised RT_{PTS} values are included in Attachments 1 and 2. The revised RT_{PTS} values are based upon two methods provided in 10 CFR 50.61. The first method is described in 10 CFR 50.61(c)(1) and uses the copper and nickel chemistry to determine a chemistry factor. The second method is described in 10 CFR 50.61(c)(2) and (c)(3) and uses surveillance data. The methods used to produce revised RT_{PTS} values for the specific beltline materials are shown in Table 1.

Table 1 Methods for Calculating Revised RT_{PTS} Values for Beltline Materials 10 CFR 50.61	
Material	**Method**
Intermediate shell, axial welds 2-112 A/B/C, material heat no. W5214	Paragraphs (c)(2) and (c)(3)
Lower shell, axial welds 3-112 A/B/C, material heat no. W5214 and 34B009	Paragraphs (c)(2) and (c)(3) and Paragraph (c)(1)
Intermediate to lower shell, circumferential weld 9-112, material heat no. 27204	Paragraphs (c)(2) and (c)(3)
Intermediate shell, plate D-3803-1, material heat no. C-1279	Paragraph (c)(1)
Intermediate shell, plate D-3803-2, material heat no. A-0313	Paragraph (c)(1)
Intermediate shell, plate D-3803-3, material heat no. C-1279	Paragraph (c)(1)
Lower shell, plate D-3804-1, material heat no. C-1308A	Paragraph (c)(1)
Lower shell, plate D-3804-2, material heat no. C-1308B	Paragraph (c)(1)
Lower shell, plate D-3804-3, material heat no. B-5294	Paragraph (c)(1)

The revised RT_{PTS} values are based on the initial properties referenced in Section 4.2.2 of the license renewal application in Table 4.2.2-1, "Estimated RT_{PTS} on March 24, 2031."

As noted in Table 1, the revised RT_{PTS} values for the weld heat 34B009, plate D-3803-2 (heat A-0313), plate D-3804-1 (heat C-1308A), plate D-3804-2 (heat C-1308B), and plate D-3804-3 (heat B-5294) are based upon the chemistry factors referenced Section 4.2.2 of the license renewal application in Table 4.2.2-1, "Estimated RT_{PTS} on March 24, 2031."

Also noted in Table 1, the revised RT_{PTS} values for weld heat number W5214, and weld heat number 27204, are based upon chemistry factors derived from surveillance capsule data.

Evaluation of the surveillance capsule data for weld heat no. W5214, weld heat no. 27204, and plate heat no. C-1279 has been completed. The evaluation was performed in accordance with 10 CFR 50.61 and published NRC guidance given in Reference 9. The new information was gathered by performing a survey of all relevant surveillance data for weld heat no. W5214, weld heat no. 27204, and

plate heat no. C-1279. The Charpy data for these heats were compiled and refitted consistently using the CVGRAPH hyperbolic tangent curve-fitting methodology (Reference 8).

For weld heat W5214, the evaluation indicates that the data from two (2) supplemental surveillance capsules from PNP is credible. However, when all sources of the data are combined, weld heat number W5214 is not fully credible based on scatter in the data, but it meets the other tests of credibility. For conservatism, ENO has elected to use a chemistry factor based upon surveillance capsule data from all sources of data versus only the two (2) supplemental surveillance capsules from PNP. Thus, the revised RT_{PTS} value for weld heat W5214 uses a fitted chemistry factor with the full margin term.

Evaluation of the surveillance capsule data for weld heat number 27204 reveals that the surveillance capsule data is fully credible. 10 CFR 50.61 states that σ_Δ need not exceed 0.5 times the mean reference temperature shift ($0.5^* \Delta RT_{NDT}$), or the standard value of σ_Δ of 28°F for welds and 17°F for base metal (plates or forgings), whichever is lower. Note that the margin term, M, may be reduced by half if credit is obtained for credible surveillance data. Thus, the revised RT_{PTS} value for weld heat 27204 uses a reduced margin term.

Evaluation of the surveillance data for plate heat number C-1279 indicates that not all of the surveillance capsule data is within the two sigma scatter band. Thus, for conservatism, the surveillance data for plate heat number C-1279 is treated as non-credible and no credit was taken for the surveillance data and a full margin term was used. It is noted that this is a change from the information provided in the PNP license renewal application which used a fitted CF value and reduced margin term for plate heat number C-1279.

The revised RT_{PTS} values for weld heat no. W5214, weld heat no. 34B009, and the beltline plate materials are based upon the full margins referenced in Section 4.2.2 of the license renewal application in Table 4.2.2-1, "Estimated RT_{PTS} on March 24, 2031."

The revised RT_{PTS} values were generated using updated radiation exposure calculations from WCAP-15353-NP, Revision 0, Supplement 1, entitled "Palisades Reactor Pressure Vessel Fluence Evaluation," provided in Attachment 3. The current RT_{PTS} values were generated with a detailed fluence evaluation that contained input data for plant operation through the end of cycle 14 (i.e., October 1999) documented in WCAP-15353, Revision 0, entitled "Palisades Reactor Pressure Vessel Neutron Fluence Evaluation" (Reference 6). The methodology for determining radiation exposure has not been changed or altered by WCAP-15353-NP, Revision 0, Supplement 1. The revised fluence evaluation provides an updated fluence assessment for the RV beltline region that includes cycle specific analyses for known core configurations for additional

operating cycles 15 through 21, and projections for future operation based on core design of cycle 21 with a load factor of 95% for future plant operation.

EVALUATION DISCUSSION

Table 2 below summarizes the revised RT_{PTS} values projected for the PNP RV beltline materials through the license renewal period, ending March 24, 2031. The axial welds made from heat nos. W5214 and 34B009 are projected to exceed the PTS screening criterion of 270°F prior to the end of the license renewal period. The other RV beltline materials are projected to remain below the PTS screening criteria of 270°F for the plate materials and 300°F for the circumferential weld during the license renewal period.

Based on the new evaluation, the axial welds fabricated with weld wire heat no. W5214 are projected to remain below the PTS screening criterion of 270°F until April 2017. ENO plans to inspect the beltline region of the RV during the spring 2012 refueling outage. Following completion of this inspection, ENO intends to calculate the RT_{MAX-X} values for each RV beltline material and to submit a report to NRC to justify continued operation using the PTS screening criteria in Table 1 of 10 CFR 50.61a. Submittal of an application for implementation of 10 CFR 50.61a is anticipated to occur no less than three years before the limiting axial welds fabricated with weld wire heat nos. W5214 and 34B009 are projected to reach the 10 CFR 50.61 PTS screening criteria limit.

The revised RT_{PTS} values have been determined using methods described in the PTS rule given in 10 CFR 50.61, paragraphs (c)(1), (c)(2), and (c)(3). Paragraph (c)(1) uses copper and nickel chemistry to determine a chemistry factor. Paragraph (c)(1) was used to determine the revised RT_{PTS} values for axial weld 3-112A/B/C (heat no. 34B009), plate D-3803-2 (heat no. A-0313), plate D-3804-1 (heat no. C-1308A), plate D-3804-2 (heat no. C-1308B), plate D-3803-1 (heat no. C-1279), plate D-3804-3 (heat no. B-5294), and plate D-3803-3 (heat no. C-1279). The revised RT_{PTS} values for these RV materials are based on copper and nickel values (and therefore the chemistry factors), the initial $RT_{NDT(U)}$ values, and the margins specified in the license renewal submittal, except for plate heat no. C-1279 which now uses the full margin term.

10 CFR 50.61 paragraphs (c)(2) and (c)(3) use surveillance capsule data to establish the chemistry factor. The surveillance capsule data was evaluated to the credibility criteria specified in 10 CFR 50.61 and to the NRC guidance published on November 19, 1997, "Meeting Summary for November 12, 1997 Meeting with Owners Group Representative and NEI Regarding Review of Response to Generic Letter 92-01, Revision 1, Supplement 1 Responses" (Reference 9). Revised RT_{PTS} values based upon surveillance capsule data use either a full or reduced margin term, depending on the scatter in the data.

The revised RT_{PTS} value for circumferential weld 9-112 (heat no. 27204) uses a reduced margin term of 44°F. The initial $RT_{NDT(U)}$ value for this RV material is the same as specified in the license renewal submittal.

The revised RT_{PTS} values for axial weld 2-112A/B/C (heat no. W5214), axial weld 3-112A/B/C (heat no. W5214), plate D-3803-1 (heat no. C-1279) and plate D-3803-3 (heat no. C-1279) use a full margin term. The initial $RT_{NDT(U)}$ values for these RV materials are the same as those specified in the license renewal application.

The fluence methodology used to produce the revised RT_{PTS} values is the same as that specified in the license renewal application. The fluence projections for the revised RT_{PTS} values have been updated to reflect actual plant operating data for known core configurations through cycle 21. Future fluence projections are based upon core design for cycle 21 and a projected plant operating capacity factor of 95%.

Table 2

Palisades Reactor Vessel Beltline Metal Properties on Extended Operating License Expiration Date (3/24/31)

RPV Material	Heat No.	Cu (wt%)	Ni (wt%)	CF (°F)	Surface Fluence (E19 n/cm^2)	FF	$RT_{NDT}(U)$ (°F)	RT_{NDT} Shift (°F)	Margin (°F)	RT_{PTS} (°F)
Axial Weld 2-112A/B/C	W5214	0.213	1.007	227.74*	2.161	1.209	-56	275.4	65.5	284.9
Axial Weld 3-112A/B/C	W5214	0.213	1.007	227.74*	2.161	1.209	-56	275.4	65.5	284.9
Axial Weld 3-112A/B/C	34B009	0.192	0.98	217.7	2.161	1.209	-56	263.2	65.5	272.7
Circ Weld 9-112	27204	0.203	1.018	216.13*	3.429	1.322	-56	285.7	44	273.7
Plate D-3803-1	C-1279	0.24	0.50	157.5	3.429	1.322	-5	209.5	34	238.5
Plate D-3803-2	A-0313	0.24	0.52	160.4	3.429	1.322	-30	212.0	34	216.0
Plate D-3803-3	C-1279	0.24	0.50	157.5	3.429	1.322	-5	209.5	34	238.5
Plate D-3804-1	C-1308A	0.19	0.48	128.8	3.429	1.322	0	170.3	34	204.3
Plate D-3804-2	C-1308B	0.19	0.50	131	3.429	1.322	-30	173.2	34	177.2
Plate D-3804-3	B-5294	0.12	0.55	82	3.429	1.322	-25	108.4	34	117.4

* Fitted CF values based on use of plant-specific surveillance data from all available sources

REFERENCES

1. 10 CFR 54.21, "Contents of application – technical information."

2. Palisades Nuclear Plant Application for Renewed Operating License, dated March 22, 2005 (Accession No. ML050940429).

3. 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events."

4. "Summary of June 22, 2010, Pre-application Meeting with Entergy Nuclear Operations, Inc., to Discuss Pressurized Thermal Shock Analysis (TAC No. ME4021)," Mahesh Chawla (NRC), dated July 13, 2010 (Accession No. ML101890904).

5. Consumers Energy letter to the NRC, dated February 21, 2000, "Palisades Reactor Vessel Neutron Fluence Reevaluation" (Accession No. ML003686516).

6. WCAP-15353, Revision 0, "Palisades Reactor Pressure Vessel Neutron Fluence Evaluation," dated January 2000 (Accession No. ML003686582).

7. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Reactor Pressure Vessel Fluence Reevaluation, Consumer Energy Company, Palisades Plant, Docket No. 50-255, dated November 14, 2000 (Accession No. ML003768802).

8. CVGRAPH Version 5.0.2, Hyperbolic Tangent Curve-Fitting Program, Developed by ATI Consulting, 2000.

9. Memorandum from Keith R. Wichman (NRC), "Meeting Summary for November 12, 1997 Meeting with Owners Group Representative and NEI Regarding Review of Response to Generic Letter 92-01, Revision 1, Supplement 1 Responses," dated November 19, 1997.

ATTACHMENT 5

DESCRIPTION OF COMMITMENTS

1. Entergy Nuclear Operations, Inc. (ENO) will perform a volumetric inspection of the reactor vessel beltline region welds during the 2012 refueling outage.

2. ENO will transmit a revised pressurized thermal shock evaluation under 10 CFR 50.61a, including an evaluation of near-surface flaws from the volumetric examination, for review and approval no less than three (3) years before the reactor vessel limiting axial welds fabricated with weld wire heat no. W5214 are projected to reach the PTS screening criterion cited under 10 CFR 50.61.