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U.S. Nuclear Regulatory Commission
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DOCKET 50-255 - LICENSE DPR-20 - PALISADES PLANT
REVISED BEST ESTIMATE FLUENCE EVALUATION USING INDUSTRY DATA

During a meeting on October 19, 1998, the NRC and Consumers Energy agreed: (1) that Palisades reactor vessel fluence values determined by calculations using plant specific measurements were not likely to be approved by the NRC staff at that time or in the near future; and (2) that fluence calculations based on industry average data were more likely to be approved.

At a subsequent meeting on December 7,1998, it was further agreed that Consumers Energy would make three additional submittals modifying the fluence calculation to incorporate industry average data and changes in plant parameters which have defined physical bases.

The first submittal will provide a revised fluence that is determined using industry average data.

The second submittal, planned for the fourth quarter of 1999, will provide new projections based on refinements to fluence calculations that incorporate changes to plant parameters which have defined physical bases.

The third submittal, planned for not later than the third quarter of 2000, will provide new fluence projections based on further fluence reductions being implemented for Cycle 15 and beyond.

This letter is the first submittal. The attachment to this letter provides justification for using industry data to determine best estimate fluence and provides revised best estimate fluence values obtained when industry average data are used in their calculation. When the 0.949 bias factor derived from use of industry data is directly applied on values provided in Consumers Energy's April 4,1996 submittal, the projected date when the limiting material in the Palisades reactor vessel will exceed the 10CFR50.61 screening criteria is mid 2006.

We request that the NRC staff review this information and address it in a Safety Evaluation as soon as practical. This is necessary in view of the presently approved fluence values which result in a projection of exceeding the screening criterion in 2004. This will obviate our preparation of alternate plans by 2001 to comply with 10CFR50.61(b)(4) which requires submittal of analysis three years before RT_{PTS} is projected to exceed the PTS screening criterion.

SUMMARY OF COMMITMENTS

This letter contains no new commitments and no revisions to existing commitments.

for Nathan L. Haskell
Director, Licensing

CC Administrator, Region III, USNRC
Project Manager, NRR, USNRC
NRC Resident Inspector - Palisades

Attachment

ATTACHMENT

CONSUMERS ENERGY COMPANY
PALISADES PLANT
DOCKET 50-255

March 25, 1999

REVISED BEST ESTIMATE FLUENCE EVALUATION USING INDUSTRY DATA

18 Pages

Section 1.0 Introduction

In April of 1996, Consumers Energy submitted an updated Best Estimate neutron fluence evaluation of the Palisades reactor pressure vessel to the NRC Staff for review. The Best Estimate evaluation made use of both state of the art neutron transport analysis techniques and a series of plant specific neutron dosimetry measurements. The fluence evaluation, based on a combination of the plant specific measurements and calculations, resulted in a "Best Estimate" fluence that was lower than the calculated value by a factor of 0.831.

The initial review of this submittal by the NRC Staff indicated that the neutron transport calculations provided by Consumers Energy were acceptable and met all regulatory requirements. However, the Staff questioned the use of measurement in general, and plant specific measurement in particular, to effect a reduction in the calculated pressure vessel fluence.

Subsequent to this initial review by the Staff, additional reviews of the Consumers Energy submittal were conducted by five independent consultants, several rounds of RAI's were addressed, and meetings were held with the Staff, the independent consultants, Consumers Energy and Westinghouse to discuss various issues relative to the use of measured data in the assessment of pressure vessel fluence.

As a result of these efforts, the NRC Staff indicated that, while a collection of plant specific measured data may not be sufficient for use in the assessment of vessel fluence, a larger body of industry wide data may be suitable for this application. During the various interactions with the Staff, data from 21 operating reactors consisting of 158 individual measurement points were presented and discussed in some detail. Comparisons of this data base with calculations indicated that the calculations tended to overpredict measurement by approximately 5%. This observation was also shown to be consistent with benchmarking studies carried out in order to qualify the neutron transport analysis methodology.

The purpose of this submittal is to provide an evaluation of the Palisades reactor vessel fluence based on this larger industry wide data base rather than solely on the Palisades plant specific measured data.

Section 2.0 Calculation of Pressure Vessel Fluence

Calculated values of the fast neutron exposure of the Palisades reactor pressure vessel have been provided to the NRC staff in a series of prior submittals. Review of these submittals and subsequent RAI's led the staff to conclude that the calculational procedures were appropriate and gave excellent agreement with an independent evaluation performed by an NRC contractor. This analytical method is briefly summarized herein. A more detailed discussion of modeling details and input assumptions is available in prior submittals and responses to RAI's (e.g., Consumers Energy June 1997 submittal "Fluence Measurement Process Best Estimate Clarification"). The Staff evaluation of the analytical approach can be found in an SER issued to Consumers Energy on December 20, 1996.

The fluence calculations for Palisades were based on a series of neutron transport calculations performed for each operating cycle of the reactor. The forward transport calculations were carried out in r,θ geometry using the DORT two-dimensional discrete ordinates code and the BUGLE-93 cross-section library. The BUGLE-93 library is a 47 neutron group, ENDFB-VI based, data set produced specifically for light water reactor applications. In these analyses, anisotropic scattering is treated with a P_3 expansion of the scattering cross-sections and the angular discretization is modeled with an S_{16} order of angular quadrature.

The forward calculations were normalized to a core midplane power density and for operation at a thermal power level of 2530 MWt. The spatial core power distributions utilized in the transport calculations were determined using SIMULATE-3. These power distributions were provided as pin-by-pin spatial gradients, initial enrichments, and cycle burnups for each fuel assembly in a quadrant. The neutron source was derived for each fuel pin and for each assembly using burnup dependent values of the fission neutron energy spectrum, neutrons per fission, and energy release per fission evaluated at the mean assembly burnup value. The source spectrum was calculated by determining the fractions of fissions occuring in each of the important uranium and plutonium isotopes for the mid-cycle burnup and calculating the resultant average fission spectrum using the ENDF/B-VI fission spectrum for each isotope. Parameters defining the reactor geometry, materials of construction, and reactor coolant density were based on as-built conditions where available and nominal design conditions elsewhere. Detailed descriptions of these input parameters and their associated uncertainties are provided in prior submittals and responses to subsequent RAI's.

Section 3.0 Comparisons of Calculations with Measurements

The qualification of the methodology used in the evaluation of the reactor vessel fast neutron exposure was carried out in the following three stages:

1 -Comparisons with benchmark measurements from the PCA simulator at ORNL. This phase of the methods qualification addresses the adequacy of basic transport calculation and dosimetry evaluation techniques and cross-sections, but does not test the accuracy of core neutron source calculations nor does it address uncertainties in operational and geometric variables that impact power reactor calculations.

2-Comparisons with the H.B. Robinson power reactor benchmark. This phase of the testing addresses biases and uncertainties that are primarily methods related and would tend to apply generically to all fast neutron exposure evaluations.

3-Comparisons with a series of power reactor measurements from a relatively large group of power reactors. This phase of the testing serves to confirm generic conclusions drawn from the single comparison with the H. B. Robinson power reactor benchmark and provides a sound basis for application of a bias to the analytical on a generic basis.

Information pertinent to each of these phases of qualification testing has previously been supplied to the NRC staff either directly through Consumers Energy submittals or as a result of RAI's provided by the staff. However, for clarity of understanding, the results of each of these sets of calculation/measurement comparisons are also summarized here.

PCA COMPARISONS

The pressure vessel simulator benchmark comparisons used in the qualification of the neutron transport methodology are based on the analysis of the PCA 12/13 experimental configuration. The 12/13 configuration was chosen for the methods evaluation due to the similarity of this particular mockup to the thermal shield - downcomer - pressure vessel designs that are typical of most pressurized water reactors. Of particular note in regard to the areas of similarity are the 12 cm water gap on the core side of the thermal shield, the 13 cm water gap between the thermal shield and the pressure vessel simulator, the 6 cm thick thermal shield, the 22.5 cm thick low alloy steel pressure vessel, and the simulated reactor cavity (void box) positioned behind the pressure vessel mockup. From the viewpoint of fast neutron attenuation, the 12/13 experimental configuration results in a reduction factor for $\phi(E > 1.0 \text{ MeV})$ of approximately 1000 between the reactor core and the inner surface of the pressure vessel; and a corresponding reduction factor of about 30 from the inner surface to the outer surface of the pressure vessel wall. These similarities in the geometry and attenuation properties of the PCA mockup and LWR plant

configurations provide additional confidence that judgments made regarding measurement/calculation comparisons in the simulator environment can be related to the subsequent analyses performed for operating Light Water Reactors.

During the PCA experiments, measurements were obtained at several locations within the mockup to provide traverse data extending from the reactor core outward through the pressure vessel simulator and on into the void box. The specific measurement locations are listed in Table 3-1. All of the measurements were obtained on the lateral centerline of the mockup and were also positioned on the axial midplane of the simulator.

Table 3-1

Summary Of Measurement Locations Within The PCA 12/13 Configuration

Location	ID	Y (cm)
Core Center	A0	-20.75
Thermal Shield Front	A1	11.98
Thermal Shield Back	A2	22.80
Pressure Vessel Front	A3	29.71
Pressure Vessel ¼T	A4	39.51
Pressure Vessel ½T	A5	44.67
Pressure Vessel ¾T	A6	50.13
Void Box	A7	59.13

The measurement locations specified in Table 3-1 provide data sufficient to generate measurement/calculation comparisons throughout the entire 12/13 configuration. Data from locations A3 and A4 provide a verification of the calculated exposure levels near the inner radius of the pressure vessel; whereas, data from locations A4, A5, and A6 establish the means for verification of calculated exposure gradients within the pressure vessel wall itself. Since measurements at operating power reactors can, at best, provide data in the downcomer region internal to the vessel wall or in the cavity external to the vessel wall, these PCA data points located interior to the thick walled vessel establish a key set of comparisons to aid in the accurate determination of exposure gradients within the pressure vessel wall.

The calculated reaction rates and exposure parameters applicable to the PCA 12/13 configuration and comparisons of these analytical predictions with the measurements are provided in Tables 3-2 and 3-3.

An examination of Table 3-2 shows that the calculated slope through the pressure vessel simulator has a divergent trend relative to the measurements. This trend was also observed in prior analyses using ENDF/B-IV cross-sections. In an attempt to determine if the currently observed trend is due to inadequacies in transport cross-sections, or in the

application of the two-dimensional synthesis technique to the analysis of a small reactor system, the PCA analysis was repeated using the TORT three-dimensional discrete ordinates transport code in x,y,z geometry.

The results of the TORT three-dimensional calculations are provided in Table 3-3. Also provided in that table are the results of a least squares adjustment of the data supplied at each measurement location. An examination of Table 3-3 shows a remarkable improvement with the three-dimensional analysis. Also, agreement between the three-dimensional calculation and the best estimate exposure parameters obtained via the least squares adjustment is good with the calculated values tending to overpredict the best estimate at the inner surface of the pressure vessel simulator by approximately 5%. That is, the BE/C ratios of 0.95 at location A3 and 0.96 at location A4 bracket the pressure vessel simulator inner radius that is located between these two measurement points.

Table 3-2

Comparisons Of Measured And Calculated Data From The PCA Benchmark Evaluations

DORT Calculations

Measured Data

Location	Neutron Flux (E>1.0)	^{27}Al (n,α)	^{58}Ni (n,p)	^{115}In (n,n′)	^{238}U (n,f)	^{237}Np (n,f)
A0						
A1		5.48E-09	6.31E-07	1.05E-06		
A2	4.01E-07	7.16E-10	6.72E-08	1.14E-07		7.30E-07
A3		3.13E-10	2.50E-08	3.68E-08	5.91E-08	3.05E-07
A4	4.50E-08	7.15E-11	5.69E-09	1.11E-08	1.79E-08	1.20E-07
A5	2.21E-08	2.92E-11	2.25E-09	5.20E-09	7.88E-09	6.56E-08
A6	9.73E-09	1.12E-11	7.99E-10	2.23E-09	3.26E-09	3.47E-08
A7		4.29E-12		6.43E-10	8.65E-10	9.60E-09

Absolute Calculation

Location	Neutron Flux (E>1.0)	^{27}Al (n,α)	^{58}Ni (n,p)	^{115}In (n,n′)	^{238}U (n,f)	^{237}Np (n,f)
A0	1.60E-04	1.46E-07	2.23E-05	4.00E-05	6.71E-05	3.30E-04
A1	3.76E-06	5.30E-09	6.05E-07	9.77E-07	1.68E-06	8.19E-06
A2	4.18E-07	6.86E-10	6.47E-08	1.06E-07	1.80E-07	9.27E-07
A3	1.38E-07	3.10E-10	2.46E-08	3.60E-08	6.28E-08	2.99E-07
A4	4.54E-08	6.86E-11	5.48E-09	1.07E-08	1.72E-08	1.16E-07
A5	2.13E-08	2.75E-11	2.14E-09	4.82E-09	7.39E-09	6.31E-08
A6	9.15E-09	1.04E-11	7.89E-10	2.03E-09	2.95E-09	3.13E-08
A7	2.16E-09	3.12E-12	1.96E-10	4.82E-10	6.95E-10	7.38E-09

[M/C]

Location	Neutron Flux (E>1.0)	^{27}Al (n,α)	^{58}Ni (n,p)	^{115}In (n,n′)	^{238}U (n,f)	^{237}Np (n,f)
A0						
A1		1.03	1.04	1.08		
A2	0.96	1.04	1.04	1.08		0.79
A3		1.01	1.02	1.02	0.94	1.02
A4	0.99	1.04	1.04	1.04	1.04	1.04
A5	1.04	1.06	1.05	1.08	1.07	1.04
A6	1.06	1.08	1.01	1.10	1.11	1.11
A7		1.38		1.33	1.24	1.30

Table 3-3
Comparisons Of Measured And Calculated Data From The PCA Benchmark Evaluations
TORT Calculations

Neutron Flux (E > 1.0 MeV)

Location	Best Estimate	% Unc.	DORT	TORT	BE/DORT	BE/TORT
A1	3.87e-06	4	3.76e-06	3.83e-06	1.03	1.01
A2	4.24e-07	4	4.18e-07	4.33e-07	1.01	0.98
A3	1.37e-07	4	1.38e-07	1.44e-07	0.99	0.95
A4	4.58e-08	4	4.54e-08	4.78e-08	1.01	0.96
A5	2.23e-08	4	2.13e-08	2.24e-08	1.05	1.00
A6	9.84e-09	5	9.15e-09	9.71e-09	1.08	1.01
A7	2.79e-09	5	2.16e-09	2.62e-09	1.29	1.07

H. B. ROBINSON COMPARISONS

At the onset of fuel Cycle 9, Carolina Power and Light Company entered into a cooperative venture with the NRC sponsored LWR Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP) and the Electric Power Research Institute (EPRI) to perform a series of measurements at the H. B. Robinson Unit 2 reactor. This multi-laboratory cooperative program included measurements both within the reactor cavity and within a replacement internal surveillance capsule attached to the thermal shield. The results of this program have been documented for use as a power reactor benchmark for testing analytical methodologies.

Results of the calculations and dosimetry evaluations are provided in Tables 3-4 through 3-6. In Table 3-4, a comparison of the measured and calculated reaction rates at both the internal surveillance capsule and the reactor cavity locations is provided. The comparisons indicate good agreement at both locations with a slight trend toward overprediction by the calculation. Based on the average of all reaction rates, the calculation exceeds measurement by 3% at the internal surveillance capsule and by 1% in the reactor cavity. The consistency of the agreement at the cavity and internal capsule locations indicates that the attenuation through the vessel wall is being calculated well by the ENDF/B-VI cross-sections. This observation matches that observed with the three-dimensional calculations performed for the PCA.

In Table 3-5, a comparison of the ratio of reaction rates at the surveillance capsule location to that at the cavity location is shown for each measured reaction. Data are provided for both calculation and measurement.

In Table 3-6, calculated exposure parameters in terms of $\phi(E > 1.0\ \text{MeV})$ and dpa are provided along with the results of the least squares adjustment for both the internal capsule and the reactor cavity locations. Again, the trend toward overprediction by the calculation is evident. In this case, for $\phi(E > 1.0\ \text{MeV})$, the observed BE/C ratios are 0.94 and 0.98 for the internal surveillance capsule and reactor cavity capsule, respectively. The results of the least squares evaluations show consistency with the reaction rate comparisons and the calculated and best estimate slopes are likewise in good agreement.

Table 3-4

Comparison of Measured and Calculated Reaction Rates
H.B. Robinson - Cycle 9 Benchmark

	20° Surveillance Capsule			0° Reactor Cavity		
Reaction	Calculated	Measured	M/C	Calculated	Measured	M/C
$^{63}Cu(n,\alpha)^{60}Co$	4.00e-17	3.98e-17	1.00	4.13e-19	4.01e-19	0.97
$^{46}Ti(n,p)^{46}Sc$	6.35e-16	6.49e-16	1.02	5.92e-18	6.17e-18	1.04
$^{54}Fe(n,p)^{54}Mn$	4.01e-15	3.83e-15	0.96	3.78e-17	3.59e-17	0.95
$^{58}Ni(n,p)^{58}Co$	5.43e-15	4.88e-15	0.90	5.63e-17	5.29e-17	0.94
$^{238}U(n,f)FP$	1.79e-14	1.80e-14	1.01	2.54e-16	2.72e-16	1.07
$^{237}Np(n,f)FP$	1.24e-13	1.20e-13	0.97	5.23e-15		
Average			0.97			0.99

Table 3-5

Comparison of Measured and Calculated Surveillance Capsule/Reactor Cavity Reaction Rates
H.B. Robinson - Cycle 9 Benchmark

	Calculation			Measurement		
Reaction	Capsule	Cavity	Ratio	Capsule	Cavity	Ratio
$^{63}Cu(n,\alpha)^{60}Co$	4.00e-17	4.13e-19	96.9	3.98e-17	4.01e-19	99.3
$^{46}Ti(n,p)^{46}Sc$	6.35e-16	5.92e-18	107.	6.49e-16	6.17e-18	105.
$^{54}Fe(n,p)^{54}Mn$	4.01e-15	3.78e-17	106.	3.83e-15	3.59e-17	107.
$^{58}Ni(n,p)^{58}Co$	5.43e-15	5.63e-17	96.6	4.88e-15	5.29e-17	92.2
$^{238}U(n,f)FP$	1.79e-14	2.54e-16	70.5	1.80e-14	2.72e-16	66.2
$^{237}Np(n,f)FP$	1.24e-13	5.23e-15	23.7	1.20e-13		

Table 3-6
Summary of Least Squares Evaluations
H.B. Robinson - Cycle 9 Benchmark

	Calculated	Best Estimate	% Uncertainty	BE/C
		20° Surveillance Capsule		
ϕ(E > 1.0 MeV)	4.89e+10	4.58e+10	7	0.94
dpa/sec	7.75e-11	7.36e-11	9	0.95
Avg. Foil M/C				0.97
		0° Reactor Cavity		
ϕ(E > 1.0 MeV)	9.78e+08	9.54e+08	10	0.98
dpa/sec	4.16e-12	4.11e-12	20	0.99
Avg. Foil M/C				0.99
		Surveillance Capsule/Cavity Ratio		
ϕ(E > 1.0 MeV)	50.0	48.0		0.96
dpa/sec	18.6	17.9		0.96

POWER PLANT DATA BASE COMPARISONS

In prior submittals and responses to RAI's, evaluations of dosimetry sets irradiated at 21 reactors (a total of 158 multiple foil sensor sets) have been evaluated using the same Westinghouse fluence methodology that was employed in the calculation of the Palisades reactor vessel fluence. That is, a methodology based on the use of ENDF/B-VI cross-sections for both neutron transport and dosimetry evaluation. From this data base measurement to calculation comparisons have been made for both in-vessel and ex-vessel locations.

A summary of the comparisons of measurements and calculations for the 21 reactors comprising this data base is provided in Tables 3-7 and 3-8. In Table 3-7, a compilation of the BE/C comparisons based on $\phi(E > 1.0 \text{ MeV})$ as derived from the least squares adjustment of the individual dosimetry sets is given. The data in Table 3-7 indicate that the average [BE]/[C] ratio for the 158 capsule data base is 0.949 with a standard deviation of 7.3%. This ratio from the data base evaluations is fully consistent with those observed for the H. B. Robinson power reactor benchmark (0.94 In-Vessel and 0.98 Ex-Vessel) as well for the PCA benchmark (0.95 - 0.96 at the pressure vessel simulator inner radius). The body of data from the two benchmark evaluations and the power reactor data base clearly indicate that the calculated $\phi(E > 1.0 \text{ MeV})$ based on the analytical methodology used for Palisades is conservative by approximately 5%.

In Table 3-8, similar [M]/[C] comparisons are provided based on the average foil reaction rate for each of the individual dosimetry sets. The data in Table 3-8 indicate that the average [M]/[C] ratio for the 158 capsule data base is 0.971 with a standard deviation of 5.8%. This ratio from the data base evaluations is also consistent with those observed for the H. B. Robinson power reactor benchmark (0.97 In-Vessel and 0.99 Ex-Vessel) as well for the PCA benchmark (0.96 - 0.99 at the pressure vessel simulator inner radius). Again, using this metric, the calculation is seen to overpredict measurement in all cases.

Based on the data base comparisons provided in this section and on the excellent agreement among the data base evaluations and both the simulator and power reactor benchmarks, it is concluded that, in order to provide a "Best Estimate" neutron fluence for the Palisades reactor pressure vessel, the calculated fluence values should be reduced by the application of a bias factor

$$K = 0.949$$

Table 3-7
Summary of [BE]/[C] Ratios from In-Vessel/Ex-Vessel Data Base
Least Squares Adjusted Results

Reactor	Average [BE]/[C]	Standard Deviation	% Standard Deviation	Number of Points
Palisades	0.831	0.044	5.3	17
12	0.843	0.077	9.1	4
15	0.856	0.043	5.1	3
1	0.857	0.104	12.1	20
14	0.872	0.089	10.2	4
5	0.900	0.103	11.5	19
2	0.921	0.070	7.6	20
3	0.929	0.078	8.4	18
11	0.932	0.069	7.4	3
13	0.938	na	na	2
16	0.944	0.095	10.0	4
21	0.981	0.037	3.7	3
18	0.982	0.025	2.6	6
7	0.987	0.099	10.1	4
17	0.991	0.116	11.7	3
19	1.002	0.141	14.1	2
20	1.020	0.026	2.6	2
8	1.022	0.056	5.4	4
9	1.028	0.118	11.4	4
4	1.035	0.098	9.5	12
10	1.058	0.077	7.2	4
Average	0.949	0.069	7.3	158

Table 3-8
Summary of [M]/[C] Ratios from In-Vessel/Ex-Vessel Data Base
Based on Uniformly Weighted Reaction Rates

Reactor	Average [M]/[C]	Standard Deviation	% Standard Deviation	Number of Points
Palisades	0.878	0.045	5.1	17
12	0.891	0.066	7.4	4
3	0.897	0.070	7.8	3
14	0.898	0.075	8.4	20
15	0.914	0.067	7.3	4
5	0.921	0.024	2.6	19
1	0.925	0.074	8.0	20
11	0.947	0.059	6.3	18
2	0.974	0.059	6.0	3
7	0.979	0.083	8.4	2
13	0.980	0.091	9.3	4
18	0.981	0.107	10.9	3
8	0.988	0.088	8.9	6
16	1.003	0.061	6.0	4
19	1.011	0.092	9.1	3
21	1.014	0.082	8.1	2
10	1.022	0.099	9.7	2
20	1.030	0.100	9.7	4
4	1.044	0.054	5.1	4
9	1.045	0.161	15.4	12
17	1.058	0.111	10.5	4
Average	0.971	0.069	5.8	158

Section 4.0 Best Estimate Fluence Determination

The use of "best estimate" values for fast neutron fluence, $\Phi(E > 1.0 \text{ MeV})$, in the assessment of pressure vessel embrittlement is consistent with the requirements of 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."[1] In 10 CFR 50.61, evaluation of the reference temperature [RT_{PTS}] is required to be performed using "best estimate" values of neutron exposure and material properties. Uncertainty in the RT_{PTS} determination (e.g., from uncertainty in the neutron exposure, chemistry factor, or shift correlation) is treated separately by adding an explicit margin term to the calculated value. The use of "best estimate" values for $\Phi(E > 1.0 \text{ MeV})$ is also promulgated in Draft Regulatory Guide DG-1053, "Calculational and Dosimetry methods for Determining Pressure Vessel Neutron Fluence."

For the purposes of the following discussion, the "best estimate" is defined as a neutron exposure in terms of $\Phi(E > 1.0 \text{ MeV})$, resulting from a combination of plant specific neutron transport calculations and available measurement data, to produce an accurate assessment of the pressure vessel exposure, while minimizing the uncertainty associated with the assessment. The general philosophy supporting this definition is that, in order to minimize the uncertainties associated with the reactor vessel exposure projections, plant specific neutron transport calculations must be supported by benchmarking the analytical approach and by combination with measurements.

As noted in Section 3.0 of this submittal, the analytical benchmarking and measurement comparisons are carried out in the following three stages:

1. Comparisons with benchmark measurements from the PCA simulator at ORNL.
2. Comparisons with power reactor benchmarks such as H. B. Robinson 2.
3. Comparisons with industry wide dosimetry data bases.

This benchmarking approach establishes a progression from a purely analytical approach tied to experimental benchmarks to an approach that makes use of industry wide power reactor measurements to remove potential generic biases in the analytical method. Therefore, knowledge regarding the neutron environment applicable to operating reactor pressure vessels is increased and the uncertainty associated with vessel exposure projections is minimized, resulting in an overall "best estimate" neutron exposure evaluation.

For the Palisades application, the best estimate vessel exposure is obtained from the following relationship:

$$\Phi_{Best\ Est.} = K\ \Phi_{Calc.}$$

where: $\Phi_{Best.\ Est.}$ = The best estimate fast neutron exposure at the location of interest; i.e, at the pressure vessel wall.

K = The (BE/C) bias factor derived from the industry wide dosimetry data base discussed in Section 3.0 of this submittal.

$\Phi_{Calc.}$ = The absolute calculated fast neutron exposure at the pressure vessel wall using the methodology described in Section 2.0 of this submittal.

For the case of Palisades, the NRC Staff approved calculated value for the end of cycle 11 peak vessel fluence is given by,

$$\Phi_{Calc.} = 1.60e+19\ n/cm^2$$

and the [BE]/[C] bias factor based on an industry average of 158 In-Vessel and Ex-Vessel dosimetry sensor set evaluations is,

$$K = 0.949$$

This value of K is consistent not only with comparisons based on the power reactor data base, but also with observations from analysis of both the PCA and H. B. Robinson Benchmarks.

Thus, the "Best Estimate" end of cycle 11 peak fast neutron exposure of the Palisades reactor pressure vessel is

$$\Phi_{Best.\ Est.} = [0.949]\ \Phi_{Calc.}$$

or,

$$\Phi_{Best.\ Est.} = 1.52e+19\ n/cm^2$$

The application of the bias factor of 0.949 based on the average fleetwide [BE]/[C] comparison acts to reduce the calculated fluence within the Palisades reactor geometry by approximately 5%. In Table 4-1, the resultant "Best Estimate" values are compared with the least squares evaluations of dosimetry sets removed from four internal surveillance capsules that have been withdrawn over the course of the operating lifetime for Palisades.

Table 4-1

Comparison of Best Estimate Neutron Flux with Results of Dosimetry Evaluations Performed for the Palisades Reactor

Capsule	Best Estimate $\varphi(E > 1.0\ MeV)$ [n/cm^2-s]	Capsule $\varphi(E > 1.0\ MeV)$ [n/cm^2-s]	Capsule Result / Best Estimate
A-240	5.97e+11	5.36e+11	0.90
W-290	6.35e+10	5.63e+10	0.89
W-290-9	3.63e+10	3.12e+10	0.86
W-110	5.81e+10	5.06e+10	0.87

An examination of the comparisons provided in Table 4-1 shows that all of the plant specific dosimetry evaluations for the Palisades internal surveillance capsules fall within 10-15% of the Best Estimate prediction. Furthermore, all four of the Palisades internal capsule dosimetry sets indicate fluence levels below those predicted by application of a best estimate methodology based on comparisons with industry data bases.

Section 5.0 PTS Projections

The peak end of cycle 11 calculated fluence is identified in the December 20, 1996 Safety Evaluation Report as 1.60e+19 n/cm^2. The comparable end of cycle 11 calculated fluence value in the April 4, 1996 submittal for the axial weld locations is 1.18e+19 n/cm^2. The peak cycle 11 calculated flux in the April 4, 1996 submittal is 2.089e+10 n/cm^2-s. The comparable cycle 11 calculated flux value for the axial weld locations is 1.556e+10 n/cm^2-s.

The dates for when the reactor vessel welds are projected to exceed the screening criteria are determined below by application of the 0.949 bias factor to the method employed in the April 4, 1996 submittal, (i.e., fluence rates are based on Cycle 11 fluxes and an 85% capacity factor).

Date = Date(EOC11) + [limiting fluence – accumulated fluence(EOC11)]/fluence rate

Circ. Weld = 1995.65 + [2 .71e+19 - (0.949)(1.60e+19)]/[(0.949)(2.089e+10)(365.25)(24)(3600)(0.85)]
= 2018.1

Axial Weld = 1995.65 + [1.55e+19 - (0.949)(1.18e+19)]/[(0.949)(1.556e+10)(365.25)(24)(3600)(0.85)]
= 2006.5

Section 6.0 Conclusions

Based on the data base comparisons provided in this submittal, and on the excellent agreement among the data base evaluations and NRC endorsed simulator and power reactor benchmarks, it is concluded that, in order to provide a "Best Estimate" neutron fluence for the Palisades reactor pressure vessel, the calculated fluence values should be reduced by the application of a bias factor K = 0.949.

Applying the 0.949 bias factor directly on values provided in Consumers Energy's April 4, 1996 submittal, the projected date that the Palisades reactor vessel would reach the PTS screening criteria on the limiting weld material would be approximately July 2006.