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PALISADES

CONSUMERS ENERGY RESPONSE TO TECHNICAL MEETING W/NRC ON FEBRUARY 26, 1997

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ATTACHMENT 1

CONSUMERS ENERGY COMPANY
PALISADES PLANT
DOCKET 50-255

Consumers Energy Response to Technical meeting with NRC on February 26, 1997

37 Pages

Consumers Energy Response
to Technical Meeting with the NRC on February 26, 1997

EXECUTIVE SUMMARY

On April 4, 1996, Consumers Energy Company submitted a reevaluation of the Palisades reactor pressure vessel fluence data. Since that submittal, two Requests for Additional Information (RAI), dated May 31, 1996, and November 5, 1996, were received and answered by Consumers Energy. Technical meetings were also held with the NRC staff on May 5, 1996, and August 14, 1996, in which additional questions were raised by the NRC and answered by Consumers Energy.

The NRC Safety Evaluation dated December 20, 1996, concluded that Consumers Energy had determined the Palisades reactor vessel fluence in accordance with Draft Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Fluence". This draft guide requires that plants qualify their calculational methods using available measurements and allows a plant to adjust the calculations using a correction factor determined from a statistically significant measurement data base.

A large fraction of the RAIs address the NRC staff concerns relative to the measurement process, measurement accuracy, and the ultimate use of the measurements to determine the Best Estimate pressure vessel fluence with its associated uncertainty. At the February 26, 1997, technical meeting between Consumers Energy and the NRC staff, additional issues relative to the use of measurements were raised by the NRC Staff. Attachment 1 of this submittal answers these new issues.

At present, the NRC staff has accepted the calculated fluence submitted by Consumers Energy, but has rejected the use of measurements pending further review. It is the view of Consumers Energy that our April 4, 1996, submittal, including the use of a plant specific [BE]/[C] correction, meets all current regulatory criteria and provides the Best Estimate fluence required by the regulations for all vessel integrity evaluations. The following is a summary of the four issues raised by the staff at the February 26, 1997, meeting. They are discussed in detail later in the General Discussion Section.

Consumers Energy Response to Technical Meeting with the NRC on February 26, 1997

<u>NRC Issue 1:</u> *Why are the plant specific bias factors so different?*

<u>Consumers Energy Summary Response:</u>

The Palisades specific bias factor is within the expected variational range of the average plant specific bias factors for the industry data. To date, dosimetry sets irradiated at 21 reactors (a total of 158 multiple foil sensor sets) have been evaluated using current fluence methodology. This data base includes both in-vessel and ex-vessel irradiations. Based on the entire data base, the average Best Estimate to Calculation [BE]/[C] bias is 0.95 with an associated standard deviation of 7%. Individual plant specific bias factors range from 0.83 to 1.06. Thus, all of the average plant specific bias factors fall within ±12% of the data base average.

Given that the standard deviation associated with calculations alone is typically 15% (Refer to NRC Safety Evaluation of December 20, 1996), an average [BE]/[C] bias of 0.95 with a standard deviation of 7% lies well within the range that is expected. The Palisades plant specific bias of 0.83 is also within the range that is expected with a calculational uncertainty of 15%.

A comparison of Measurement to Calculation ratios from a 20-plant Siemens-KWU data base with similar data from the 21-plant Westinghouse data base shows agreement. It is evident from the independent evaluations of these two data bases (41 operating reactors) that the use of calculation alone tends to produce conservative estimates of the neutron fluence.

<u>NRC Issue 2:</u> *The bias from each dosimeter is different, therefore, a different set of dosimeters would lead to a different answer? Related: Where does the observed spectrum bias come from?*

<u>Consumers Energy Summary Response:</u>

An examination of the Measurement/Calculation ratios [M]/[C], for each of the foil reactions comprising the multiple foil sensor sets used by Westinghouse shows that the observed ratios for individual reactions differ from the average [M]/[C] ratio within a range of ±8%. This variation is consistent with the standard deviation associated with the overall data base, but it also indicates that the ratio of [M]/[C] varies somewhat with energy. Thus, the measured data indicates that, in addition to a bias in the overall magnitude of the calculations, a small variation in the calculated vs. measured energy distribution also exists. Spectral variations in [M]/[C] ratios similar to those observed in the Westinghouse data base have also been reported in various technical papers including I. Remec's documentation of the H. B. Robinson benchmark cited in DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Fluence".

The most probable causes of the mismatch between calculated and best estimate spectra are due to uncertainties associated with basic nuclear data. These include, but are not limited to, the fission spectra of uranium and plutonium at neutron energies above 4 MeV, effects of approximations used in processing cross-sections into the multi-group structures used in the transport analysis, and uncertainties in the transport cross-sections themselves.

<u>NRC Issue 3:</u> *Why is the spread in Palisades [M]/[C] comparisons tighter than at other plants?*

<u>Consumers Energy Summary Response:</u>

The spread in Palisades [M]/[C] comparison are comparable to other plants. The standard deviations associated with the Palisades [M]/[C] comparisons are no better or worse than those observed at other operating reactors. Comparisons among the 21 domestic reactor evaluations completed by Westinghouse and the 20 Siemens-KWU reactors comprising a large German data base show that the Palisades data are consistent with the average data from other plants and with the uncertainties associated with the measurement process itself.

<u>NRC Issue 4:</u> *What are the causes of the [BE]/[C] bias exhibited at Palisades?*

<u>Consumers Energy Summary Response:</u>

The individual affects causing the overall bias observed at Palisades and other plants are to a large extent unknown. The plant has made extensive efforts to quantify and remove the known biases from the calculations. One additional known source of bias is the fouling of the flow venturi in the steam generator feed system. This has caused the plant to run at approximately 98% of its rated thermal power. This directly effects the [BE]/[C] bias. Other possible sources of bias include, the calculation of pin powers in low powered peripheral assemblies, as-built core barrel and shroud thicknesses compared to nominal values, coolant temperatures in the bypass region and peripheral fuel assemblies, and undiscovered errors in the ENDF/B-VI cross-sections used in the transport analyses. Errors in the ENDF/B-VI cross-section data base may be exaggerated at Palisades due to the lack of a thermal shield and the location of the in-vessel dosimetry.

Since the identification and quantification of these individual effects is extremely difficult, the use of plant specific measurements represent a practical means to quantify the net overall bias in the calculation. Bench marking the calculational methodology to test reactor benchmark problems does not identify or quantify variations or errors in the input to power reactor plant specific calculations that cannot be easily determined from existing documentation.

General Discussion

The Best Estimate fast neutron fluence evaluation for the Palisades reactor pressure vessel was performed using a methodology that conforms to the guidance provided in draft regulatory guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Fluence". From the DISCUSSION Section of DG-1053, the following comments relative to the accepted fluence methodology are noted:

1. The methodology presented is intended as a "Best Estimate", rather than a bounding or conservative fluence determination. The effect of fluence uncertainty in the embrittlement prediction has been included separately in an explicit margin term applied to the projected RT_{NDT} or RT_{PTS}.

2. In the accepted methodology, the fluence prediction is made with a calculation and measurements are used to qualify the methodology.

    a. The methods qualification by comparison to measurements must be made to ensure a reliable and accurate vessel fluence determination.

    b. In this qualification, calculation to measurement comparisons are used to identify biases in the calculations and to provide reliable estimates of the fluence uncertainties.

    c. When the measurement data are of sufficient quality and quantity that they allow a reliable estimate of the calculational bias (i.e., they represent a statistically significant measurement data base), the comparisons to measurement may be used to:

        1) Determine the effect of various modeling approximations and any calculational bias.

2) Modify the calculations by applying a correction to account for bias or by model adjustment or both.

d. The sensitivity of the calculation to the important input and modeling parameters must be determined and combined with the uncertainties of the input and modeling parameters to provide an estimate of the overall calculational uncertainty.

e. The prediction of the vessel fluence must be made by an absolute calculation in which the transport of neutrons from the core is calculated out to the vessel and cavity, rather than by a simple spatial extrapolation of the fluence measurements.

The Palisades fluence evaluation is in conformance with all of these requirements established by DG-1053, including the application of a bias factor (see Section 2.c.2 above) to the calculated results in order to arrive at a "Best Estimate" fluence with a reduced uncertainty relative to that associated with the calculation alone.

Previous discussions with the NRC staff have indicated that, with respect to the calculational aspects of the fluence methodology, the staff is in essential agreement with the approach used for Palisades. However, considerable disagreement on the use of plant specific measurement data to bias analytical results (see Section 2.c.2 above) was stated by the NRC staff. Concerns of the NRC staff were general in nature and centered around the reliability of the measurement process. This submittal is provided to address several of the specific issues raised by the NRC staff at the NRC-Consumers Energy technical meeting held February 26, 1997.

Relative to the discussion provided in this submittal, a distinction should be made between Best Estimate/Calculation, [BE]/[C] ratios and Measurement/Calculation, [M]/[C] ratios. In this case, Best Estimate values refer to the combination of calculation and measurement via a least squares adjustment procedure to arrive at the best estimate of the neutron flux (E > 1.0 MeV) with an associated uncertainty. The least squares procedure provides a weighting of calculated and measured input based on the energy response and uncertainty associated with each input parameter. The [BE]/[C] ratios represent a comparison of the results of the least squares adjustment with the analytical prediction of the neutron flux (E > 1.0 MeV). The [M]/[C] ratios, on the other hand, provide a direct comparison of actual calculated and measured individual foil reaction rates. Using the [M]/[C] data, a direct comparison of calculated and measured neutron flux (E > 1.0 MeV) can not be made without a suitable weighting of the individual foil results.

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Concern was expressed by the NRC staff relative to the use of the least squares adjustment approach to determine the Best Estimate fluence. Requests for Additional Information (RAI) dated May 31, 1996, and November 5, 1996, were provided by the NRC staff relative to the least squares approach, the data used in the analysis, the magnitude of the adjustments, and the sensitivity of the adjustment to various input parameters. Consumers Energy has made no attempt to readdress the least squares adjustment approach in this document. In the NRC staff evaluation of the Palisades submittal, a detailed assessment of the transport calculations was provided, however, the details of the NRC staff review of the RAI relative to the least squares procedure were not provided. Therefore, this RAI and the Consumers Energy responses are included with this submittal for further review.

<u>NRC Issue 1:</u> *Why are the plant specific bias factors so different?*

<u>Consumers Energy Response:</u>

The Palisades specific bias factor is within the expected variational range of the average plant specific bias factors for the industry data. Pressure vessel fast neutron fluence estimates based solely on the results of benchmarked plant specific neutron transport calculations typically have an associated uncertainty of ±15% at the 1σ level. The observed standard deviations in plant specific [BE]/[C] bias factors as well as plant specific average foil reaction rate [M]/[C] ratios are 7% and 6%, respectively. Thus, on a statistical basis, both the [BE]/[C] and [M]/[C] plant specific comparisons are consistent with an uncertainty of 15% in the analytical predictions.

To date, dosimetry sets irradiated at 21 reactors (a total of 158 multiple foil sensor sets) have been evaluated using the current Westinghouse fluence methodology based on ENDF/B-VI transport and dosimetry cross-sections. The evaluations include data from both in-vessel and ex-vessel irradiations.

The evaluation of these dosimetry sets and subsequent comparison with calculation indicates that, on average, the analytically predicted fluence exceeds the best estimate value by approximately 5%. The data base average [BE]/[C] bias factor is 0.95 with an associated standard deviation of 7%. The plant specific average [BE]/[C] bias factors range from 0.83 to 1.06, thus, all of the observed biases fall within ±12% of the average value. Considering the entire set of 158 individual measurement points, all of the data fall within ± 25% of the mean value of 0.95. It is expected that an individual plant will have errors in its calculational inputs that are isolated to that plant. An average [BE]/[C] of 0.95 implies that in general the calculations are biased high. However this average value does not have any meaning for a specific plant since each plant will have its own bias that is a combination of both the generic methodological bias and the plant specific bias and these two biases cannot reasonably be separated. A summary of the [BE]/[C] bias factors for the 21 reactors currently comprising the data base is provided in Table 1-1. These data are shown graphically in Figure 1-1.

A similar comparison of the [M]/[C] ratios for the average reaction rates based on an equal weighting of all individual foil [M]/[C] ratios results in a similar observation. That is, the calculations tend to exceed the average measured foil reaction rates by approximately 3%. The data base average [M]/[C] ratio is 0.97 with an associated standard deviation of 6%. The plant specific average [M]/[C] ratios range from 0.88 to 1.06, thus, all of the comparisons fall within ± 11% of the average value. A summary of the uniformly weighted [M]/[C] ratios for the 21 plant data base is shown in Table 1-2. These comparisons are illustrated in Figure 1-2.

Given that uncertainties associated with calculation alone are typically 15% at the 1σ level, an average [BE]/[C] bias of 0.95 with a standard deviation of 7% lies well within the range of expected values.

A Siemens-KWU data base consisting of measurement to calculation comparisons from 20 German reactors has been cited as an example of a good set of [M]/[C] comparisons based on a reliable measurement data base. Like the Westinghouse data, the Siemens-KWU evaluations include both in-vessel and ex-vessel measurement locations. The source of the German data was the following publication:

> Polke, E., "Siemens-KWU Experience in Evaluating Fluence Detectors Inside and Outside the RPV in German Light Water Reactor Plants," Proceedings of the Ninth ASTM-Euratom Symposium on Reactor Dosimetry, Prague, Czechoslovakia, September 1996.

In the publication by E. Polke, the following conclusions are drawn:

> "There are a large amount of fluence data and theoretical to experimental ratios available. From these data it was found that theoretical fluences are about 4% to 8% higher than the experimental values."
>
> Using an Fe detector, "on average the calculated fluence is 8% higher than the measured data. The standard deviation of the C/E (C/M) ratio is 6%."
>
> Using a Nb detector, "on average the calculated fluence is 4% higher than the measured data. The standard deviation of the C/E (C/M) ratio is 6%."
>
> For both the Fe and Nb data, individual measurement point comparisons show that, "the deviation between the theoretical fluence and the experimental fluence E > 1.0 MeV is, with one exception, less than ±20%."

From these quotations, it is evident that average experimental to calculational bias, average [M]/[C] ratio, in the 20-plant Siemens-KWU data base lies in the range of 0.92 to 0.96 with an associated standard deviation of 6%. These values are in agreement with the average [BE]/[C] bias of 0.95 with a standard deviation of 7% and the average [M]/[C] ratio of 0.97 with a standard deviation of 6% determined from the 21-plant Westinghouse data base.

It is evident from the independent evaluations of these two data bases (41 operating reactors) that the use of calculation alone tends to produce conservative estimates of the neutron fluence. Because the damage trend curve in 10 CFR 50.61 is based mainly on measurements the use of fluence calculations alone would be conservative compared to the rule.

The 17 point [BE]/[C] data base from the Palisades Plant also demonstrates conservatism in the analytical determination of vessel fluence. The average [BE]/[C] bias of 0.83 has an associated standard deviation of 5.3% with all of the data points falling within a range of 0.73 to 0.92. The standard deviation of the Palisades Plant specific [BE]/[C] is smaller than the data base standard deviation. This is expected since the Palisades [BE]/[C] is a subset of the overall data base. Previous responses to NRC staff RAI have demonstrated that, on a statistical basis, all of the Palisades data are consistent and represent a reliable measurement data base as required by DG-1053.

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Table 1-1

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

Reactor	Average [BE]/[C] Ratio	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	----	---	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

Note: The standard deviations listed in the table are based solely on the variation of the individual [BE]/[C] data points comprising each data set. The uncertainty associated with the least squares evaluation (typically about 8%) is not included.

Table 1-2

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

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

Figure 1-1

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

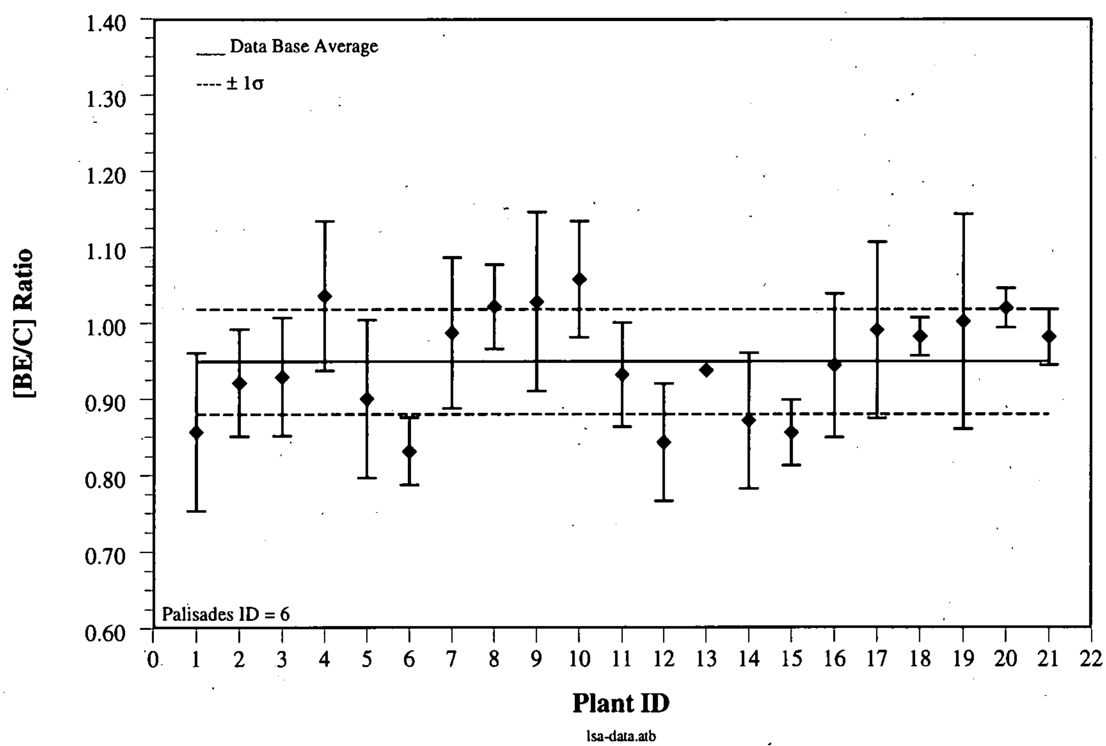
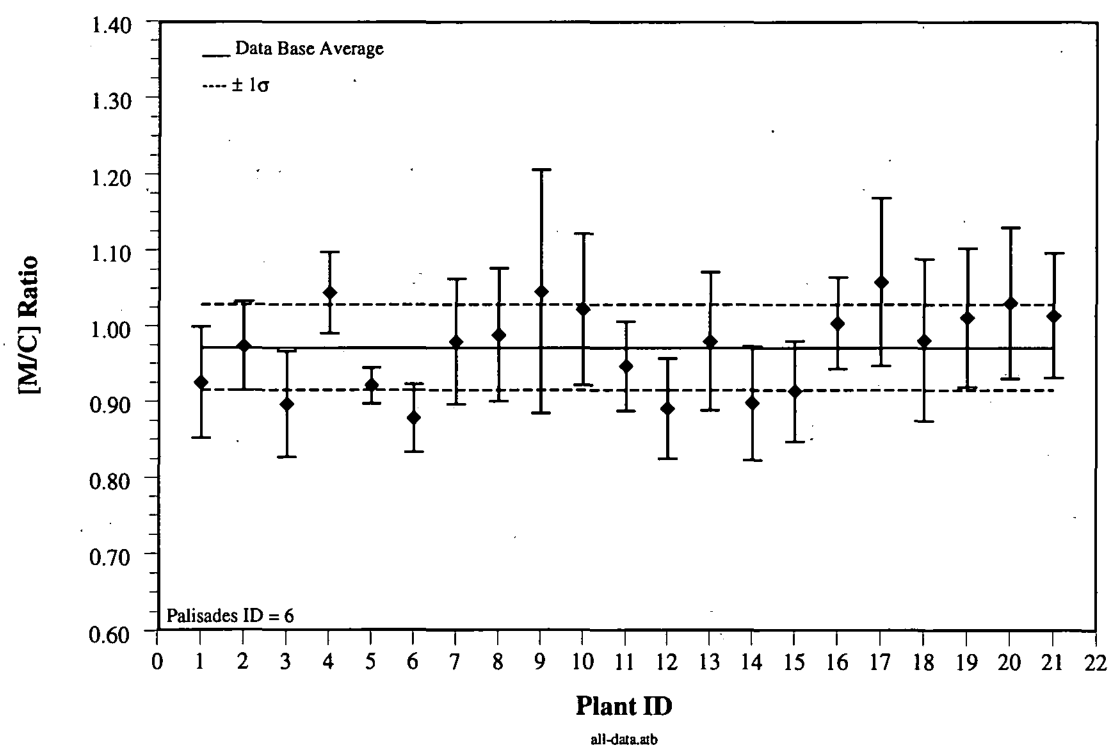


Figure 1-2

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



<u>NRC Issue 2:</u> *Where Does the Observed Spectrum Bias Come From?*

<u>Consumers Energy Response:</u>

An examination of [M]/[C] ratios for each of the foil reactions comprising the multiple foil sensor sets used by Westinghouse shows that the observed ratios for individual reactions differ from the overall average [M]/[C] ratio for all foil reactions by approximately ±8%. This observation is illustrated by the data comparisons shown in Table 2-1. For ease of comparison, the data presented in Table 2-1 have been normalized to an unweighted average of 1.00 for the six foil reactions. The normalized data allows for comparisons among data sets to observe similar spectral variations. The Palisades data is presented in Table 2-2.

This variation in the Palisades normalized [M]/[C] ratios for individual reaction rates are consistent with the 8% standard deviation associated with the overall data base average, but, nevertheless, indicates that the ratio of measurement to calculation varies somewhat with energy. That is, the measured data indicate that, in addition to an observed bias in the overall magnitude of the calculations, small variations in the calculated vs. measured energy distribution also exist, i.e., spectral variations.

Spectral variations in [M]/[C] ratios similar to those observed in the Westinghouse data base have also been reported elsewhere. The following references contain data that indicate that spectral mismatches between calculation and experiment have been observed in a fairly wide variety of applications:

1. Williams, M. L., et. al., "Transport Calculations of Neutron Transmission Through Steel Using ENDF/B-V, Revised ENDF/B-V, and ENDF/B-VI Iron Evaluations," Ann. Nucl. Energy, Vol. 18, No. 10, pp. 549-565, (1991).

2. Sajo, E., et. al., "Comparison of Measured and Calculated Neutron Transmission Through Steel for a Cf-252 Source," Ann. Nucl. Energy, Vol. 20, No. 9, pp. 585-604, (1993).

3. Sajo, E., et. al., "Pressure Vessel Neutron Spectrum Analysis of the Czech LRO/VVER-440 Benchmark Experiment," Proceedings of the ANS 1996 Topical Meeting on Radiation Protection and Shielding, pp. 181-188, N. Falmouth, Massachusetts (April 1996).

4. Bevilaqua, A., et. al., "Special Dosimetry at Saint Laurent B1 MOX-Loaded Unit," Proceedings of the Eighth ASTM-Euratom Symposium on Reactor Dosimetry, pp. 132-139, Vail, Colorado, September 1993.

5. Remec, I. and Kam, F. B. K., "H. B. Robinson-2 Pressure Vessel Benchmark," NUREG/CR-6453, to be published.

In References 1, 2, and 3 listed above, detailed comparisons of calculated and measured energy spectra are provided for several evaluations involving deep penetrations in steel. In all cases, the differences between calculation and measurement varied with neutron energy. In Reference 2, the following summary conclusion was noted:

> "The results obtained in this study appear to indicate that the ENDF/B-VI cross-sections will not entirely resolve the spectrum discrepancies observed in the energy interval above 1 MeV."
>
> "The discrepancies could indicate that further refinement is needed in the iron cross-section and/or the Cf-252 fission spectrum."

The data comparisons provided in References 4 and 5 listed above are characteristic of pressurized water reactor systems similar to those comprising the Westinghouse dosimetry data base and are provided in Tables 2-3 and 2-4, respectively.

The comparisons in Reference 4 are based on a collaborative set of measurements and calculations performed by Commissariat a l ' Energie Atomique and Electricite' de France for the French reactor at Saint Laurent and include data from both an in-vessel surveillance capsule and ex-vessel dosimetry sets.

The data comparisons from Reference 5 are based on measurements and calculations at both in-vessel and ex-vessel locations performed in support of the NRC sponsored Light Water Reactor Surveillance Dosimetry Improvement Program (LWR-PVSDIP). The data from Reference 5 is cited as a suitable benchmark in DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Fluence."

A comparison of normalized [M]/[C] comparisons from References 4 and 5 with those observed in the Westinghouse dosimetry data base is provided in the Table 2-5.

From Table 2-5, it is evident that the spectral variations observed in the Westinghouse dosimetry data base are also evident in both the St. Laurent and H. B. Robinson comparisons. It may also be noted that the St. Laurent and H. B. Robinson comparisons fall within 1 standard deviation of the data base average values.

A comparison of the overall Westinghouse data base with the Palisades plant specific normalized [M]/[C] data is also illustrative. This comparison is provided in Table 2-6.

The comparisons shown in Table 2-6 show that, like St. Laurent and H. B. Robinson, the Palisades plant specific dosimetry data base exhibits similar trends (i.e., Fe and Ni lower than other dosimetry) and is statistically compatible with the overall Westinghouse dosimetry data base.

In attempting to determine the cause of the observed discrepancies in the calculated and measured relative neutron energy distributions, it is important to recall that the calculations involve assumptions regarding the energy spectrum of the neutron source as well as assumptions related to processing basic cross-section data from the ENDF/B-VI data files to the multi-group structures characteristic of the transport calculations.

In terms of the source spectrum, two factors can combine to produce uncertainties that would vary with neutron energy. The first of these is the relative lack of knowledge of the high energy end of the fission spectrum for the individual isotopes that are fissioning in a light water reactor core; the second is the uncertainty in the mix of uranium and plutonium isotopes that make up the total power production for a given fuel cycle. The combination of these two factors along with the fact that the mix of fissioning isotopes varies both with position and time leads to a net effect on the calculation that may produce energy dependent biases.

In addition to these energy dependent uncertainties in the neutron source term, remaining deficiencies in the ENDF/B-VI cross-sections themselves produce effects that could cause energy dependent biases that are a function of penetration.

The most probable cause of the observed mismatch in calculated and best estimate spectra is a combination of uncertainties in these source term and cross-section processing assumptions that enter into the transport calculation along with uncertainties in the transport cross-sections. These observed spectral effects can be duplicated analytically by performing sensitivity studies to demonstrate that the [M]/[C] observations are consistent with the uncertainties in each of these parameters. However, it is not possible to separate out each of the individual effects that act in concert to produce a net effect, the observed spectral bias, manifested in the observations in the normalized [M]/[C] ratios. Nevertheless, the comparisons of measurement with calculation by themselves provide an excellent indication of the net effect of uncertainties in all of these variables.

In regard to these comparisons, it should be noted that the observed spectral differences are not large and are easily accounted for in the uncertainty estimates associated with the spectrally weighted best estimate evaluations. The least squares adjustment approach accounts for these spectral differences by combining the individual measurements and their uncertainties with the transport calculated spectrum and its uncertainties to arrive at a Best Estimate of the true spectrum at the measurement locations. Thus, the observed spectral mismatch is accounted for in the overall uncertainty derived for the Best Estimate fluence.

Table 2-1

[M]/[C] Comparisons from the Westinghouse Dosimetry Data Base

Foil Reaction	Absolute [M]/[C] Ratio	Normalized [M]/[C] Ratio
^{63}Cu (n,α) ^{60}Co	1.023 ± 0.068	1.054 ± 0.070
^{46}Ti (n,p) ^{46}Sc	0.976 ± 0.058	1.005 ± 0.060
^{54}Fe (n,p) ^{54}Mn	0.916 ± 0.061	0.943 ± 0.063
^{58}Ni (n,p) ^{58}Co	0.903 ± 0.062	0.930 ± 0.063
^{238}U (n,f) FP	0.982 ± 0.088	1.011 ± 0.091
^{237}Np (n,f) FP	1.021 ± 0.116	1.051 ± 0.119
Average	0.971 ± 0.056	1.000 ± 0.080

Table 2-2

[M]/[C] Comparisons from the Palisades Data Base

Foil Reaction	Absolute [M]/[C] Ratio	Normalized [M]/[C] Ratio
^{63}Cu (n,α) ^{60}Co	0.922 ± 0.046	1.049 ± 0.052
^{46}Ti (n,p) ^{46}Sc	0.942 ± 0.049	1.073 ± 0.056
^{54}Fe (n,p) ^{54}Mn	0.836 ± 0.033	0.952 ± 0.038
^{58}Ni (n,p) ^{58}Co	0.843 ± 0.026	0.959 ± 0.030
^{238}U (n,f) FP	0.847 ± 0.079	0.964 ± 0.090
^{237}Np (n,f) FP	0.880 ± 0.101	1.002 ± 0.115
Average	0.879 ± 0.072	1.000 ± 0.070

Table 2-3

[M]/[C] Comparisons from the St. Laurent Data Base

Foil Reaction	Absolute [M]/[C] Ratio	Normalized [M]/[C] Ratio
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	1.000	1.024
$^{46}Ti\ (n,p)\ ^{46}Sc$		
$^{54}Fe\ (n,p)\ ^{54}Mn$	0.936	0.958
$^{58}Ni\ (n,p)\ ^{58}Co$	0.907	0.929
^{238}U (n,f) FP	1.010	1.034
^{237}Np (n,f) FP	1.031	1.055
Average	0.977	1.000

Table 2-4

[M]/[C] Comparisons from the H.B. Robinson 2 Data Base

Foil Reaction	Absolute [M]/[C] Ratio	Normalized [M]/[C] Ratio
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	1.076	0.993
$^{46}Ti\ (n,p)\ ^{46}Sc$	1.124	1.037
$^{54}Fe\ (n,p)\ ^{54}Mn$	1.042	0.961
$^{58}Ni\ (n,p)\ ^{58}Co$	1.005	0.928
^{238}U (n,f) FP	1.157	1.068
^{237}Np (n,f) FP	1.099	1.014
Average	1.084	1.000

Table 2-5

Comparison of Normalized [M]/[C] Ratios from the Westinghouse Dosimetry Data Base with Independent Evaluations from other Pressurized Water Reactors

Reaction	[M]/[C] Data Base	[M]/[C] St Laurent	[M]/[C] H. B. Robinson
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	1.054 ± 0.070	1.024	0.993
$^{46}Ti\ (n,p)\ ^{46}Sc$	1.005 ± 0.060		1.037
$^{54}Fe\ (n,p)\ ^{54}Mn$	0.943 ± 0.063	0.958	0.961
$^{58}Ni\ (n,p)\ ^{58}Co$	0.930 ± 0.063	0.929	0.928
$^{238}U\ (n,f)$ FP	1.011 ± 0.091	1.034	1.068
$^{237}Np\ (n,f)$ FP	1.051 ± 0.119	1.055	1.014
Average	1.000 ± 0.080	1.000	1.000

Table 2-6

Comparison of Normalized [M]/[C] Ratios from the Westinghouse Dosimetry Data Base with Plant Specific Evaluations Performed for Palisades

Reaction	[M]/[C] Data Base	[M]/[C] Palisades
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	1.054 ± 0.070	1.049 ± 0.052
$^{46}Ti\ (n,p)\ ^{46}Sc$	1.005 ± 0.060	1.073 ± 0.056
$^{54}Fe\ (n,p)\ ^{54}Mn$	0.943 ± 0.063	0.952 ± 0.038
$^{58}Ni\ (n,p)\ ^{58}Co$	0.930 ± 0.063	0.959 ± 0.030
$^{238}U\ (n,f)$ FP	1.011 ± 0.091	0.964 ± 0.090
$^{237}Np\ (n,f)$ FP	1.051 ± 0.119	1.002 ± 0.115
Average	1.000 ± 0.080	1.000 ± 0.070

<u>NRC Issue 3:</u> *Why is the spread on Palisades M/C comparisons tighter than at other plants?*

<u>Consumers Energy Response:</u>

The standard deviations associated with the Palisades [M/C] comparisons are consistent with other plants, they are no better or worse than those observed at other operating reactors.

The standard deviations associated with the [M/C] ratios developed from the Palisades Plant specific dosimetry data base are both reasonable and consistent with those observed in industry wide comparisons. In Table 3-1 absolute [M/C] ratios obtained from the Palisades irradiations are compared with an industry wide data base compiled by Westinghouse from irradiations at 21 domestic reactors and also with a Siemens-KWU data base reported to consist of [M/C] ratios from 20 German light water reactors. The Palisades, Westinghouse, and Siemens-KWU data bases all contain comparisons at both in-vessel and ex-vessel locations.

An examination of Table 3-1 indicates that the standard deviations in non-fission reaction rates range from 4% - 7% and for fission reaction rates from 9% - 12%. Variations of this magnitude are consistent with the associated uncertainties in the measurement process and are not unexpected. Also from Table 3-1, it can be seen that the corresponding standard deviations in the Palisades data base range from 4% - 5% and 9% - 12% for non-fission and fission reaction rates respectively. In some cases the Palisades data have a smaller standard deviation than the data base as a whole. Since the data base spans many plants and the Palisades data are from a single plant, this trend should be expected.

More detailed summaries of the absolute [M]/[C] ratios for the individual sensors that comprise the multiple foil sets used in both in-vessel and ex-vessel irradiations are provided in Tables 3-2 through 3-7 for the ^{63}Cu (n,α), ^{46}Ti (n,p), ^{54}Fe (n,p), ^{58}Ni (n,p), ^{238}U (n,f), and ^{237}Np (n,f), reactions respectively. These data are illustrated graphically in Figures 3-1 through 3-5, except for ^{46}Ti (n,p) due to the limited data.

From the data listed in Tables 3-2 through 3-7, the percent standard deviation associated with the Palisades foil measurements can be compared with the range of percent standard deviations observed in the 21 plant data base. These comparisons are summarized in Table 3-8.

From the comparisons listed in Table 3-8, the statistical behavior of the Palisades Plant specific [M]/[C] ratios is consistent with observations from the 21 plant data base for all foil reactions. Again, as noted earlier the standard deviations of the Palisades Plant specific foil data fall close to the data base average for all reactions.

Table 3-1[a]

Comparison of Palisades Absolute [M]/[C] Ratios with Corresponding Ratios from Westinghouse and Siemens-KWU Industry Wide Data Bases

Reaction	[M]/[C][b] Westinghouse	[M]/[C][c] Siemens-KWU	[M]/[C][d] Palisades
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	1.02 ± 7%		0.92 ± 5%
$^{46}Ti\ (n,p)\ ^{46}Sc$	0.98 ± 6%		0.94 ± 5%
$^{54}Fe\ (n,p)\ ^{54}Mn$	0.92 ± 7%	0.92 ± 6%	0.84 ± 4%
$^{58}Ni\ (n,p)\ ^{58}Co$	0.90 ± 7%		0.84 ± 4%
$^{238}U\ (n,f)$ FP	0.98 ± 9%		0.85 ± 9%
$^{93}Nb\ (n,n')\ ^{93m}Nb$		0.96 ± 6%	
$^{237}Np\ (n,f)$ FP	1.02 ± 11%		0.88 ± 12%
Average	0.97 ± 6%	0.94 ± 6%	0.88 ± 5%

NOTES:

[a] Siemens-KWU data were taken from:
Polke, E., "Siemens-KWU Experience in Evaluating Fluence Detectors Inside and Outside the RPV in German Light Water Reactor Plants," Proceedings of the Ninth ASTM-Euratom Symposium on Reactor Dosimetry, Prague, Czechoslovakia, September 1996.

[b] The Westinghouse data base consists of M/C comparisons from 158 multiple foil sensor sets irradiated at 21 operating reactors. The data base represents both in-vessel and ex-vessel comparisons.

[c] The Siemens-KWU data base consists of M/C comparisons from in-vessel and ex-vessel irradiations at 20 operating reactors. The total number of data points were not reported.

[d] The Palisades data base consists of M/C comparisons from 17 multiple foil sensor sets irradiated at both in-vessel and ex-vessel locations.

Table 3-2

Summary of [M]/[C] Ratios from In-Vessel/Ex-Vessel Data Base
$^{63}Cu\ (n,\alpha)\ ^{60}Co$

Reactor	Average [M]/[C] Ratio	Standard Deviation	% Standard Deviation	Number of Points
3	0.890	0.076	8.5	18
Palisades	0.922	0.046	5.0	17
5	0.943	0.082	8.7	19
8	0.962	0.008	0.8	4
11	0.984	0.041	4.1	3
10	0.986	0.039	4.0	4
7	0.987	0.048	4.8	4
14	0.989	0.044	4.4	4
12	0.998	0.057	5.8	4
1	1.019	0.107	10.5	20
15	1.024	0.032	3.1	3
19	1.035	0.036	3.5	4
2	1.035	0.056	5.4	20
18	1.051	0.012	1.2	6
4	1.053	0.088	8.4	12
16	1.053	0.022	2.1	4
19	1.074	0.004	0.4	2
13	1.096	0.037	3.4	2
20	1.100	0.026	2.4	2
21	1.117	0.010	0.9	3
17	1.177	0.080	6.8	3
Average	1.023	0.068	6.7	158

Table 3-3

Summary of [M]/[C] Ratios from In-Vessel/Ex-Vessel Data Base
$^{46}Ti\ (n,p)\ ^{46}Sc$

Reactor	Average [M]/[C] Ratio	Standard Deviation	% Standard Deviation	Number of Points
3	0.899	0.057	6.3	16
Palisades	0.942	0.049	5.2	16
5	0.946	0.039	4.1	12
1	0.992	0.092	9.3	8
4	1.023	0.084	8.2	16
2	1.056	0.040	3.8	17
Average	0.976	0.058	5.9	85

Table 3-4

Summary of [M]/[C] Ratios from In-Vessel/Ex-Vessel Data Base
$^{54}Fe\ (n,p)\ ^{54}Mn$

Reactor	Average [M]/[C] Ratio	Standard Deviation	% Standard Deviation	Number of Points
3	0.804	0.058	7.3	18
Palisades	0.836	0.033	3.9	17
14	0.845	0.069	8.1	4
18	0.869	0.037	4.2	6
15	0.874	0.027	3.1	3
1	0.877	0.085	9.7	20
7	0.882	0.058	6.5	4
11	0.894	0.041	4.5	3
5	0.899	0.063	7.0	19
12	0.908	0.133	14.6	4
8	0.915	0.028	3.1	4
19	0.926	0.078	8.5	2
2	0.927	0.035	3.8	19
13	0.929	0.006	0.6	2
16	0.938	0.042	4.5	3
21	0.938	0.054	5.7	3
20	0.947	0.032	3.4	2
10	0.968	0.013	1.3	4
9	0.984	0.045	4.5	4
4	1.004	0.110	11.0	12
17	1.075	0.101	9.4	3
Average	0.916	0.061	6.6	156

Table 3-5

Summary of [M]/[C] Ratios from In-Vessel/Ex-Vessel Data Base
^{58}Ni (n,p) ^{58}Co

Reactor	Average [M]/[C] Ratio	Standard Deviation	% Standard Deviation	Number of Points
14	0.821	0.037	4.5	4
3	0.838	0.079	9.5	12
Palisades	0.843	0.026	3.1	16
1	0.846	0.096	11.3	19
18	0.866	0.018	2.1	6
15	0.867	0.006	0.7	3
12	0.869	0.093	10.8	4
11	0.883	0.046	5.2	3
5	0.887	0.059	6.6	19
9	0.891	0.083	9.4	4
19	0.897	0.040	4.5	2
20	0.899	0.029	3.2	2
13	0.906	0.006	0.6	2
10	0.907	0.054	5.9	4
2	0.920	0.031	3.4	19
7	0.922	0.080	8.7	3
8	0.922	0.045	4.8	4
16	0.938	0.032	3.4	3
21	0.939	0.049	5.2	3
4	1.022	0.174	17.0	12
17	1.092	0.079	7.2	3
Average	0.903	0.062	6.8	147

Table 3-6

Summary of [M]/[C] Ratios from In-Vessel/Ex-Vessel Data Base
^{238}U (n,f) FP

Reactor	Average [M]/[C] Ratio	Standard Deviation	% Standard Deviation	Number of Points
Palisades	0.847	0.079	9.3	15
12	0.850	0.073	8.6	4
1	0.859	0.101	11.8	20
15	0.874	0.115	13.1	2
17	0.876	0.128	14.6	3
5	0.924	0.105	11.4	19
2	0.934	0.088	9.4	20
11	0.949	0.025	2.7	2
3	0.957	0.079	8.3	18
14	0.967	0.143	14.8	4
9	1.000	0.057	5.7	4
7	1.001	0.118	11.7	4
4	1.013	0.121	11.9	12
8	1.013	0.080	7.9	4
16	1.032	0.086	8.4	2
13	1.062	0.045	4.2	2
19	1.082	0.165	15.2	2
21	1.082	0.039	3.6	3
20	1.098	0.003	0.3	2
18	1.098	0.030	2.7	6
10	1.102	0.085	7.7	4
Average	0.982	0.088	9.0	152

Table 3-7

Summary of [M]/[C] Ratios from In-Vessel/Ex-Vessel Data Base
^{237}Np (n,f) FP

Reactor	Average [M]/[C] Ratio	Standard Deviation	% Standard Deviation	Number of Points
12	0.831	0.024	2.9	3
14	0.870	0.111	12.8	4
Palisades	0.880	0.101	11.5	14
13	0.907	0.053	5.9	2
15	0.929	0.036	3.9	2
5	0.929	0.170	18.3	14
1	0.956	0.139	14.5	18
2	0.971	0.094	9.6	20
3	0.990	0.116	11.7	16
21	0.992	0.023	2.3	3
18	1.021	0.049	4.8	6
11	1.023	0.112	10.9	3
16	1.057	0.227	21.5	2
17	1.070	0.125	11.7	3
19	1.076	0.177	16.5	2
7	1.096	0.123	11.2	4
20	1.107	0.052	4.7	2
8	1.130	0.104	9.2	4
10	1.147	0.200	17.5	3
4	1.148	0.157	13.7	10
9	1.316	0.044	3.3	3
Average	1.021	0.116	11.3	138

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Table 3-8

Comparison of Standard Deviations in the Palisades Plant Specific Measurements with the Range of Standard Deviations from Plants Comprising the Data Base

Reaction	Data Base Average % Standard Deviation	Data Base Range % Standard Deviation	Palisades % Standard Deviation
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	7%	0.4% - 10.5%	5%
$^{46}Ti\ (n,p)\ ^{46}Sc$	6%	3.8% - 9.3%	5%
$^{54}Fe\ (n,p)\ ^{54}Mn$	7%	0.6% - 14.6%	4%
$^{58}Ni\ (n,p)\ ^{58}Co$	7%	0.6% - 17.0%	4%
^{238}U (n,f) FP	9%	0.3% - 15.2%	9%
^{237}Np (n,f) FP	11%	2.9% - 21.5%	12%
Average	6%		5%

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Figure 3-1

[M]/[C] Ratios from In-Vessel/Ex-Vessel Data Base
$^{63}Cu\ (n,\alpha)\ ^{60}Co$

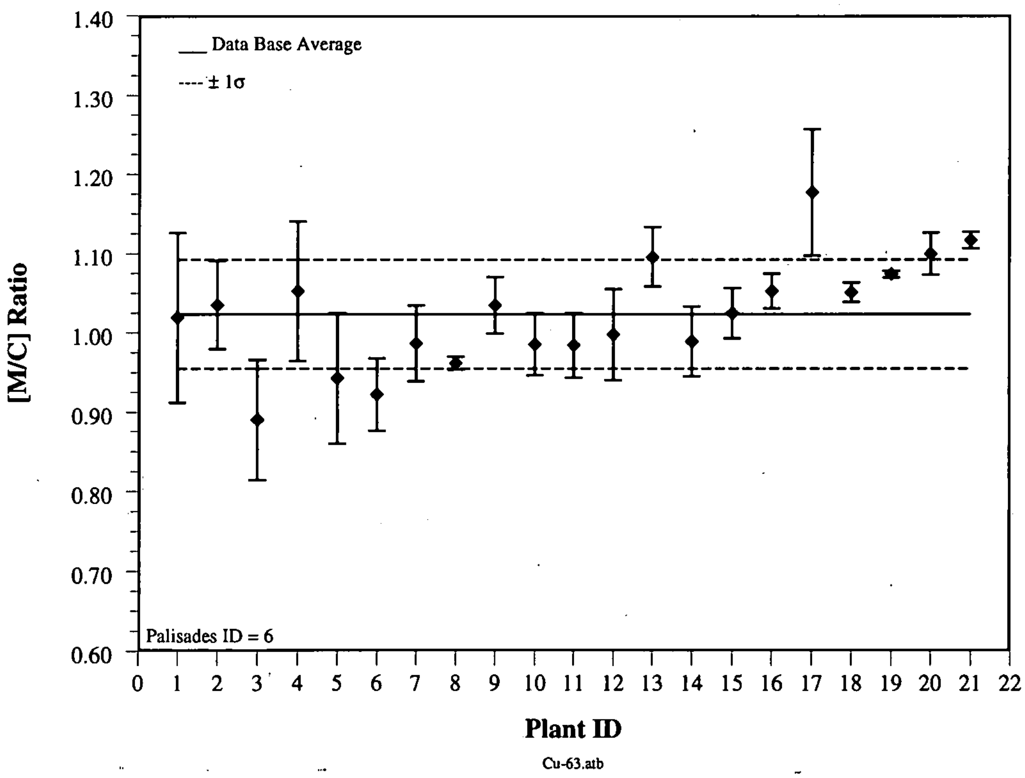
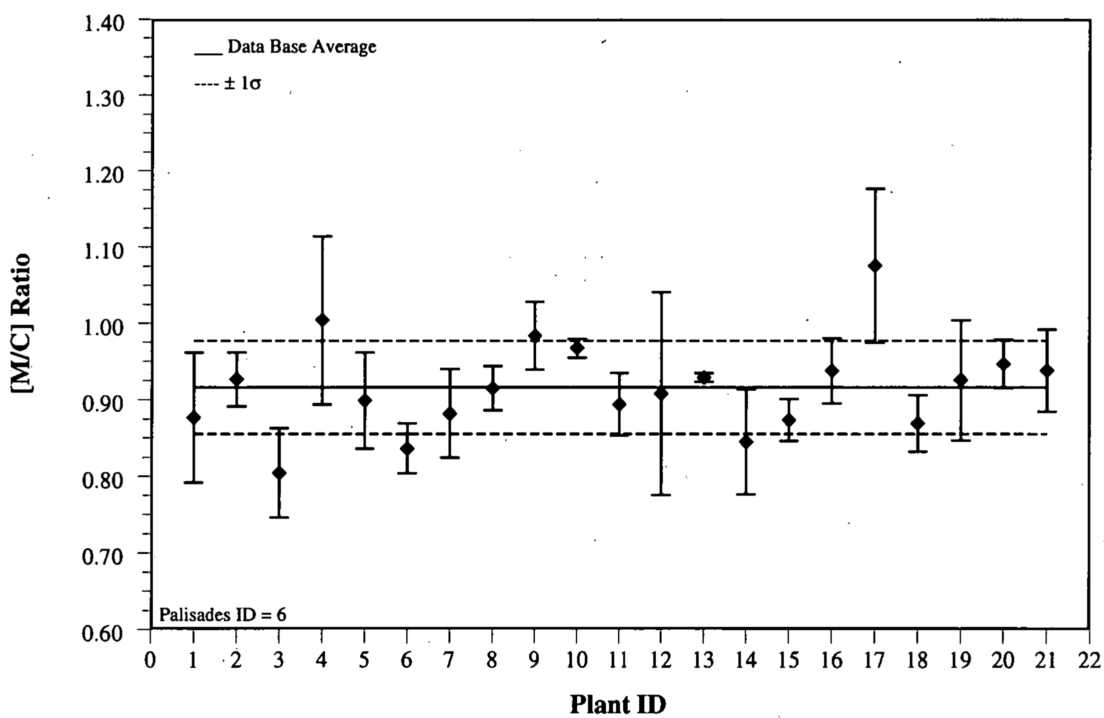


Figure 3-2

[M]/[C] Ratios from In-Vessel/Ex-Vessel Data Base
$^{54}Fe\ (n,p)\ ^{54}Mn$



Fe-54.atb

Figure 3-3

[M]/[C] Ratios from In-Vessel/Ex-Vessel Data Base
$^{58}Ni\ (n,p)\ ^{58}Co$

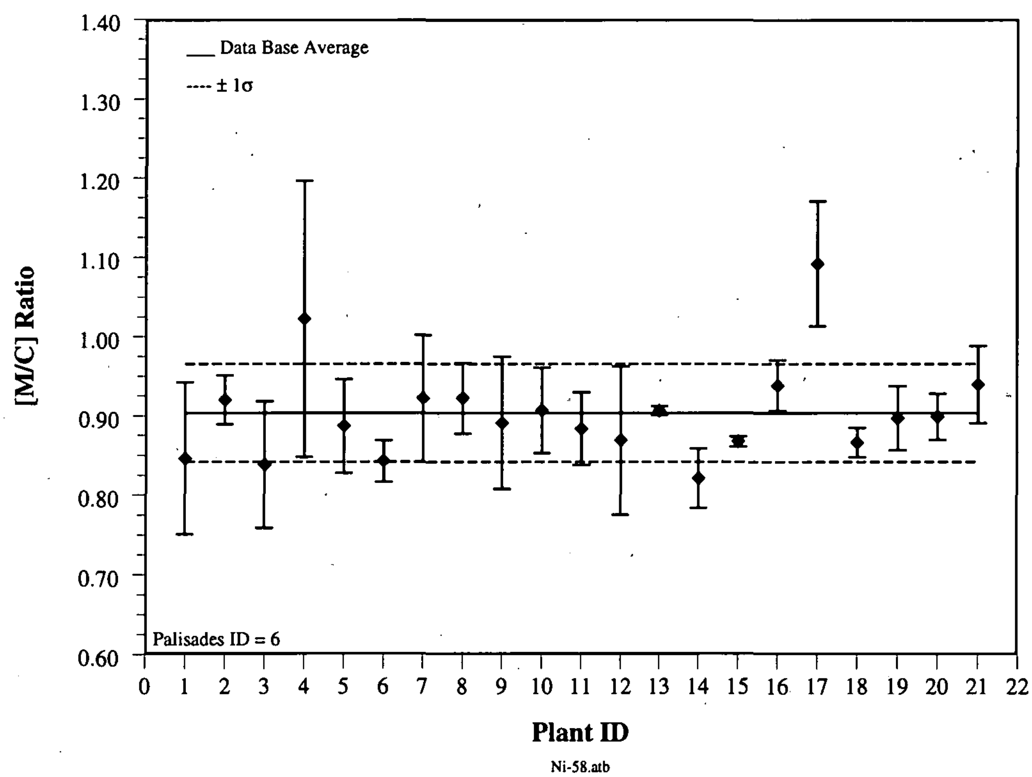


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Figure 3-4

[M]/[C] Ratios from In-Vessel/Ex-Vessel Data Base
^{238}U (n,f) FP

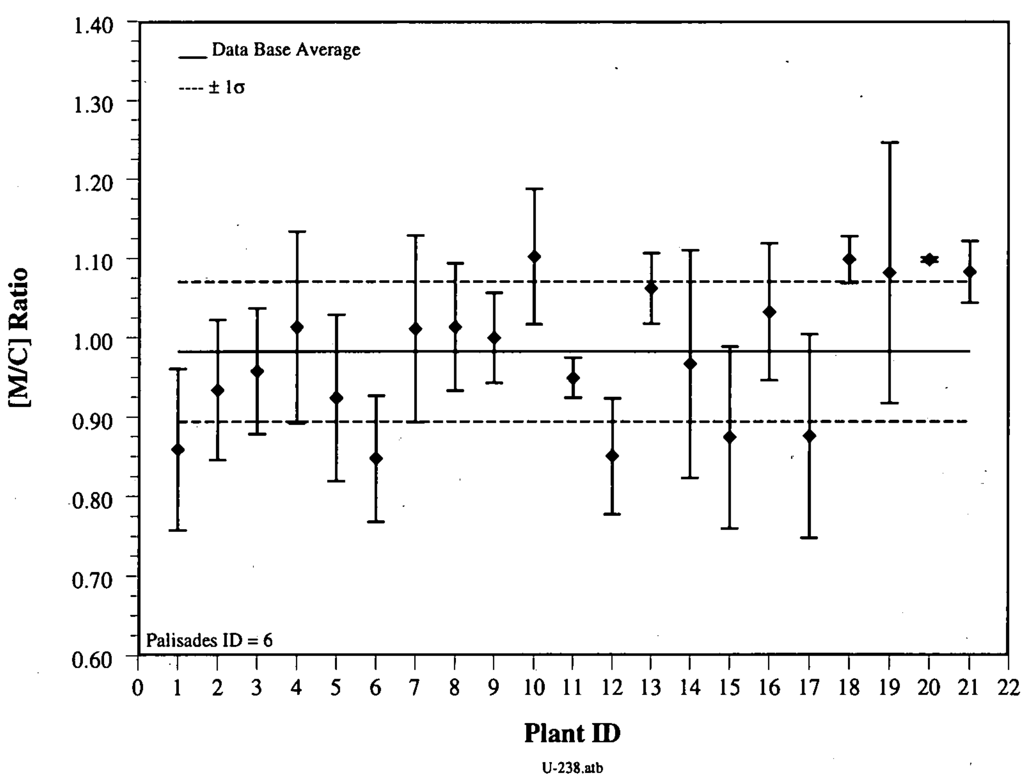

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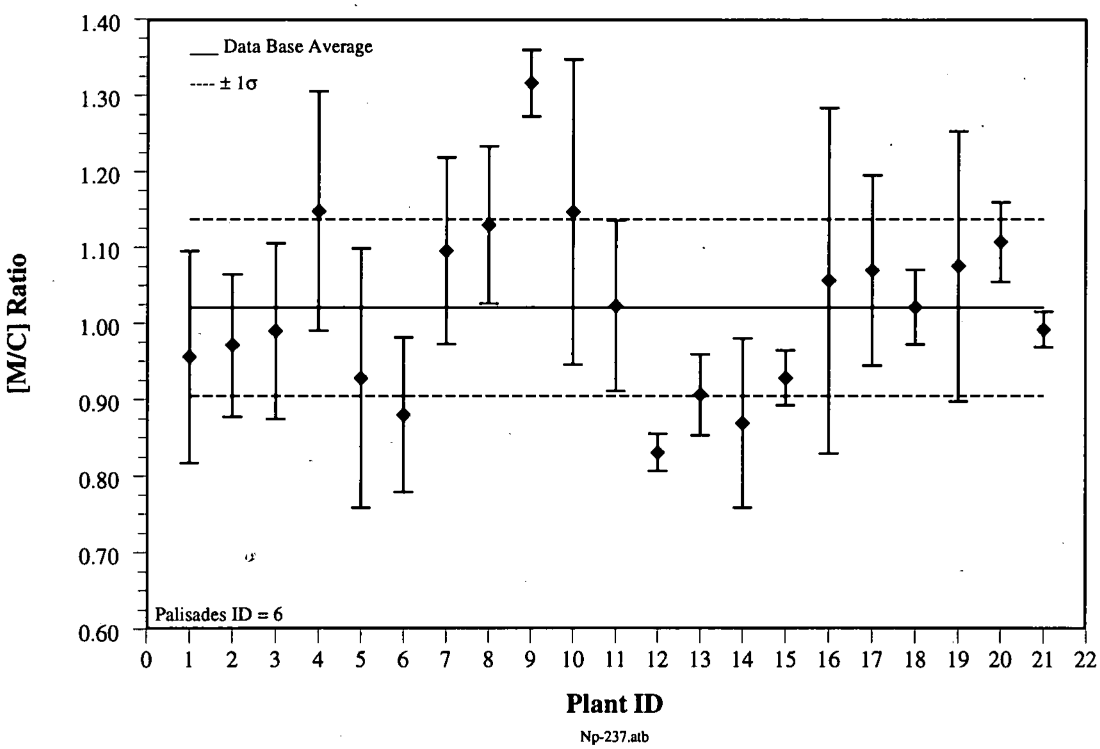
Figure 3-5

[M]/[C] Ratios from In-Vessel/Ex-Vessel Data Base
^{237}Np (n,f) FP



<u>NRC Issue 4:</u> *What are the causes of the [BE]/[C] bias exhibited at Palisades?*

<u>Consumers Energy Response:</u>

The individual effects causing the overall bias observed at Palisades and other plants are to a large extent unknown. The plant has made extensive efforts to quantify and remove the known biases from the calculations. One additional known source of bias is the fouling of flow venturi in the steam generator feed system. Testing has been performed using an ultrasonic flow measurement device which has demonstrated that the plant has been operating at 98% of its rated thermal power. This bias cannot be removed because the historical effect of fouling of the flow venturi is not available. This directly affects the [BE]/[C] bias. Other possible sources of bias include, the calculation of pin powers in low powered peripheral assemblies, as-built core barrel and shroud thicknesses compared to nominal values, coolant temperatures in the bypass region and peripheral fuel assemblies, and undiscovered errors in the cross-sections used in the transport analyses.

In addition there are several differences in the design of Palisades that may contribute to the bias. These differences include, narrow and wide water gaps between fuel assemblies, lower inlet temperature, capsules mounted on the inside diameter of the vessel instead of the outside diameter of the thermal shield, and the lack of a thermal shield in the Palisades design. Errors in the ENDF/B-VI cross-section data base may be exaggerated at Palisades due to the lack of a thermal shield and the location of the in-vessel dosimetry.

Since the identification and quantification of these individual effects is extremely difficult, the use of the plant specific measurements represents the only practical means to quantify the net overall bias in the calculation. Bench marking the calculational methodology to known benchmark problems does not identify or quantify variations or errors in the input to plant specific calculations. The accuracy of the input is limited by the availability and quality of the plant specific documentation.

Table 4-1 represents an estimate of possible biases existing within the calculation of the Palisades reactor vessel fluence. These estimates are based on sensitivity studies done to calculate the calculational uncertainty and engineering judgement.

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Table 4-1

Summary of Potential Sources of Calculational Bias

Potential Bias Source	Possible Magnitude
Fouling of the Feed Water Flow Venturi	+1% to +3%
Peripheral Assembly Pin Powers	±8%
Core Support Barrel Thickness (+1/16)	+2%
Core Shroud Thickness (+1/16 inch)	+2%
Material Compositions & Densities	±4%
Bypass Temperature	0% to +5%
Exterior Core Temperature	0% to +3%
Transport Cross Sections	±8%
Neutron Source (Pu vs. U)	±3%