WCAP-16083-NP Rev. 0
Project Number 694

March 30, 2005

WOG-05-161

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

**Subject:** **Revision to WCAP-16083-NP Rev. 0, "Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry," (TAC No. MC3974) (MUHP-7550)**

Reference:

1. WOG Letter, F. Schiffley to Document Control Desk, "Transmittal of WCAP-16083-NP, Rev. 0, Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry," WOG-04-389, July 30, 2004.

In July 2004, the Westinghouse Owners Group submitted WCAP-16083-NP, Rev. 0, "Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry," for approval (Reference 1).

Attachment 1 to this letter contains the NRC comments on WCAP-16083-NP, Rev. 0 and the response to the NRC comments. Attachment 2 contains revisions to WCAP-16083-NP, Rev. 0 that incorporate the response to the NRC comments. These revisions to WCAP-16083-NP, Rev. 0 were discussed in a meeting between the NRC staff and Westinghouse that was held on December 16, 2004.

The approved version of WCAP-16083, that will be issued following receipt of the NRC Safety Evaluation, will incorporate the changes contained in Attachment 2 and will be issued as WCAP-16083-NP-A, Rev. 1.

If you require further information, feel free to contact Mr. Steve DiTommaso in the Westinghouse Owners Group Program Management Office at 412-374-5217.

Sincerely,

Steven M. DiTommaso for

Frederick P. "Ted" Schiffley, II
Chairman, Westinghouse Owners Group

mjl

Attachments

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March 30, 2005
WOG-05-161

cc: Girija Shukla, NRC (2L, 2A) (via Fed Ex)
J. Andrachek, W
S. Anderson, W
WOG Materials Subcommittee
WOG Program Management Office
WOG Steering Committee

Attachment 1

NRC Comments on WCAP-16083-NP, Rev. 0 and the Response to the NRC Comments

NRC Comment:

WCAP-16083NP, Request for Clarification

The purpose of this letter is to obtain clarifying information from the applicant regarding acceptability of the subject topical report (TR) submitted for NRC review.

This TR is a straight forward application of a least squares adjustment method to determine best estimate values. It would be difficult to find any faults with a least squares method in treating measured and calculated values. However, this TR is dedicated to reactor dosimetry and as such should account for the underlying physics and known phenomenological behavior of reactor dosimetry.

The TR failed to address the large discrepancies between calculated and measured capsule dosimetry. These discrepancies are much larger than the typical uncertainties as from material density, geometry, cross sections, dosimeter activation measurement, etc. The staff believes that the major source of uncertainty is the capsule location, particularly in older plants with poor historical records and plants outfitted with surveillance capsules after startup.

There are many examples of older surveillance capsule analysis reports recording +/- 30 - 40% deviations of measured to the calculated values. A specific surveillance capsule report records three capsules from the same plant, one at +30%, one at -30%, and one in reasonable agreement with the calculated value. Such differences cannot be attributed to ordinary uncertainties which are generally in the neighborhood of 10-15%.

The staff believes that this phenomenon should be explicitly addressed and resolved. The results of a best estimate evaluation using a process which ignores this phenomenon would lead to erroneous results. The staff is reluctant to initiate review of this report, which appears technically correct in the least squares method but seems to be seriously flawed in its physics content.

There exist statistical techniques to characterize each capsule before a best estimate value is calculated using the proposed method. If such discrepancies are folded into the best estimate value, both the best estimate and the associated uncertainty will be in error. The staff in its effort to avoid the effect of this phenomenon requires calculated values (for licensing actions) rather than the best estimate. The reason is that a benchmarked code should produce better results for vessel fluence (since sources, dimensions and densities are known with some accuracy) without the surveillance capsule measurement adjustment. Likewise, for applications which involve exposure of a Charpy sample and the associated irradiation, the measured values should be correct. However, using measured dosimetry for vessel fluence estimates could lead to errors.

The staff feels that this kind of TR is useful and has repeatedly requested that FERRET be submitted for review. However, based on the limitations outlined above, the staff requests that the TR be expanded to account for the reactor dosimetry as previously described before initiating its review.

(Note: the FERRET code was characterized by covariant adjustments which are not present in this version).

Response:

A new section will be added to WCAP-16083-NP that addresses the comments above. This is included as Attachment 2.

Attachment 2

Revisions to WCAP-16083-NP, Rev. 0

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## LIST OF TABLES

## 4.0 FERRET SENSITIVITY STUDIES

The information discussed in this section is intended to provide an understanding of how the composition of the multiple foil sensor sets and the input values for the uncertainties in the calculated neutron spectrum and measured reaction rates impact the results of the least squares analysis in terms of both the magnitude and uncertainty of the adjusted~~best estimate~~ spectrum.

The threshold foils comprising typical LWR sensor sets respond to different portions of the neutron energy spectrum. These multiple foil measurements should be thought of as a set of partial measurements of the flux or fluence rather than as a group of complete and independent determinations. Of particular interest for weighting and averaging threshold detector measurements is the spectrum coverage of the individual foils which recognizes that such measurements do not form an equivalent observation set and hence are not easily matched to the principle of maximum likelihood.

This response is highlighted in Figures 4-1 and 4-2 for the neutron spectra characteristic of an in-vessel surveillance capsule location and an ex-vessel dosimetry location. The graphical representations shown in Figures 4-1 and 4-2 provide response profiles for the $^{63}$Cu(n,α), $^{54}$Fe(n,p), $^{238}$U(n,f), and $^{237}$Np(n,f) threshold reactions, as well as for the neutron flux (E > 1.0 MeV). The $^{46}$Ti(n,p) and $^{58}$Ni(n,p) reactions exhibit behavior similar to the $^{63}$Cu(n,α), $^{54}$Fe(n,p) reactions, respectively.

From Figures 4.1 and 4.2, it is evident that the response of the higher threshold reactions exhibits significantly different behavior than does the neutron flux (E > 1.0 MeV), while the fission monitor response shows a better match to the spectral behavior of the neutron flux. This behavior suggests that in order to validate a calculation of the neutron flux (E > 1.0 MeV), significant spectral weighting of measured reaction rates should be included in the comparisons. The least squares approach allows this spectral weighting to be included in a rigorous manner. The data from Figures 4-1 and 4-2 also indicate that the makeup of the foil set could have an impact on the final results of the dosimetry comparisons.

### 4.1 COMPOSITION OF THE MULTIPLE FOIL SENSOR SET

In order to assess the impact of the makeup of the sensor set on the final solution of the least squares adjustment, a parametric study was performed for a typical LWR dosimetry data set. In the parametric study, the calculated neutron spectrum and uncertainty was held constant along with the uncertainties associated with the measured reaction rates. The base case consisted of an evaluation including all of the threshold reactions while the variations in the analysis were accomplished by deleting foil reactions individually and in combination. The 11 cases analyzed in the parametric study are summarized as follows:

## 4.3 OPERATING POWER REACTOR COMPARISONS

In addition to the sensitivity studies described above, the transport methodology and the least squares dosimetry evaluations have been extensively compared with data from operating power reactors. These comparisons are intended to provide support for the validation of the transport calculation itself as well as validation for the uncertainties assigned to the results of those calculations.

One concern that has been raised in regard to measurement/calculation comparisons from operating power plants is that the uncertainties in the positioning of the dosimetry within the reactor could lead to large uncertainties in the measurement/calculation data base that would, in turn, tend to reduce the value of these comparisons. The data comparisons provided in this section are intended to demonstrate that the dosimetry locations within operating power reactors are known within the manufacturing tolerances and that, when calculations and dosimetry processing are completed on a consistent basis, the uncertainties in the measurement/calculation data base are less than those associated with the stand alone calculation.

In Section 2.3 of this report, it was noted that the combination of benchmarking comparisons and analytical sensitivity studies resulted in an evaluated uncertainty of 13% (1σ) in the calculation of neutron flux or fluence (E > 1.0 MeV). Based on the analytical sensitivity studies using allowable manufacturing tolerances for the reactor components as limits, the component of the uncertainty in measurement/calculation comparisons due to miss-positioning of surveillance capsule dosimetry is in the range of 4-5% (1σ).

In order to demonstrate the validity of these uncertainty assessments, comparisons using the methodologies described in Section 2.0 of this report were applied to a large measurement data base consisting of 104 surveillance capsule dosimetry sets withdrawn from 29 reactors having Westinghouse or Combustion Engineering as the Original Equipment Manufacturer (OEM) of the NSSS. The data base includes surveillance capsules attached to the pressure vessel wall, capsules mounted on the external surfaces of thermal shields or neutron pads, and accelerated capsules mounted on the core barrel. The irradiation times included in the data base range from 1 to more than 20 fuel cycles. The number of capsules withdrawn from individual reactors ranges from two to six. The overall data base includes five basic Westinghouse internals configurations and two Combustion Engineering internals designs.

In Table 4-1, the comparisons of the adjusted results with the original calculations of the neutron flux (E > 1.0 MeV) are provided as [adjusted]/[calculated] (A/C) ratios for each of the 104 surveillance capsule dosimetry sets included in the data base. Also included in the tabulation are average A/C values for the 29 individual reactors and for the data base as a whole. From the data listed in Table 4-1, it is noted that the overall data base average A/C is 0.99 with an associated standard deviation of 7%. This data shows that the stand alone transport calculations are essentially unbiased. Further, the 7% standard deviation associated with the A/C data base is approximately half of the 13% uncertainty assigned to the calculation alone. It is evident from this data base that lack of knowledge of dosimeter positioning does not introduce large scatter and correspondingly high uncertainty into the comparisons of dosimetry results with calculations.

Although the data comparisons listed in Table 4-1 compare the results of the least squares adjustment with the original calculated neutron flux (E > 1.0 MeV), similar conclusions regarding the effects of dosimetry positioning can be drawn from a comparison of [measurement]/[calculation] (M/C) ratios for individual foil reactions. These comparison applicable to the 104 capsule data base are listed in Table 4-2 along with the previously discussed

A/C results from the adjustment analysis. From Table 4-2, it is noted that none of the comparisons for the individual foil reactions show any excessive scatter that could be attributed to a large uncertainty in the dosimeter positioning.

It must be emphasized that a key factor in completing the comparisons of calculation and measurement is that, for all data points, both the transport calculations and the dosimetry evaluations must be done using the same methods, cross-sections, and basic nuclear data. Use of older published data based on different methods and assumptions will lead to distortions in the data base. This could be due to one or more of the following:

1 - Changes in treatment of the core source (conservative vs best estimate)
2 - Changes in transport cross-sections (ENDF/B-IV vs ENDF/B-VI)
3 - Changes in dosimetry cross-sections (ENDF/B-IV vs ENDF/B-VI)
4 - Changes in dosimetry evaluation methods (spectrum averaged cross-sections vs least squares adjustment)

These artificial distortions could introduce data scatter that would lead to a misinterpretation of the validity of the data base.

Consider the following example for Reactor 15 from Table 4-1:

The Reactor 15 data base consists of 5 surveillance capsule withdrawals during the 20 year interval between 1985 and 2004. The original documentation of each of those capsule analyses was based on the methodology and basic nuclear data accepted at the time of the evaluations. These are summarized as follows:

1 - Completed in 1985.
Spectrum averaged cross-section dosimetry evaluation.
Design basis power distribution, intended to be conservative.
ENDF/B-IV transport cross-sections (SAILOR).
ENDF/B-V dosimetry cross-sections.

2 - Completed 1988.
Spectrum averaged cross-section dosimetry evaluation
Plant specific power distribution, intended to be best estimate.
ENDF/B-IV transport cross-sections (SAILOR).
Axial peaking based on core power distribution.
ENDF/B-V dosimetry cross-sections.

3 - Completed 1991.
Least squares dosimetry evaluation.
Plant specific power distribution, intended to be best estimate.
ENDF/B-IV transport calculations (SAILOR).
Axial peaking based on core power distribution.
ENDF/B-V dosimetry cross-sections.

4 - Completed 1998.
Least squares dosimetry evaluation.
Plant specific power distribution, intended to be best estimate.
ENDF/B-VI transport cross-sections (BUGLE-96).
Axial peaking based on core power distribution.
ENDF/B-VI dosimetry cross-sections (SNLRML).

5 - Completed 2004.
Least squares dosimetry evaluation.
Plant specific power distribution, intended to be best estimate.
ENDF/B-VI transport cross-sections (BUGLE-96).
Axial peaking based on 3D synthesis.
ENDF/B-VI dosimetry cross-sections (SNLRML).

The methodology used for the latest evaluations for Reactor 15 are identical to those described in Section 2.0 of this report.

The following tabulation provides a comparison of the M/C or A/C ratios taken from the original reports for each of the capsule withdrawals from Reactor 15 with the latest A/C comparisons based on the methodology described in Section 2.0.

| Capsule | Original M/C or A/C | Latest A/C |
|---|---|---|
| 1 | 0.86 | 0.93 |
| 2 | 1.03 | 0.94 |
| 3 | 1.11 | 0.96 |
| 4 | 0.98 | 0.98 |
| 5 | 0.95 | 0.95 |
| Average | 0.99 | 0.95 |
| % std dev | 9.4 | 2.0 |

This comparison shows that the use of consistent methodologies for calculation and dosimetry evaluation reduces the standard deviation in the data base by almost a factor of five. This, clearly demonstrates that changing methodologies can introduce significant scatter in the data base and highlights the importance of re-evaluating dosimetry from previously withdrawn capsules each time a new surveillance capsule from a given reactor is analyzed.

Table 4-1

Data Base Comparison for 104 In-Vessel Dosimetry Sets from 29 Reactors

| Reactor | A/C Ratio – Neutron Flux (E > 1.0 MeV) | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | Capsule 1 | Capsule 2 | Capsule 3 | Capsule 4 | Capsule 5 | Capsule 6 | Average | % std |
| 1 | 1.10 | 1.01 | 1.05 | 0.93 | | | 1.02 | 7.0 |
| 2 | 1.03 | 0.96 | 0.97 | 0.92 | | | 0.97 | 4.7 |
| 3 | 1.00 | 0.89 | 0.86 | | | | 0.92 | 8.0 |
| 4 | 1.03 | 0.94 | 1.05 | 1.03 | | | 1.01 | 4.9 |
| 5 | 1.08 | 0.99 | 0.94 | | | | 1.00 | 7.1 |
| 6 | 1.01 | 0.99 | 0.97 | 0.95 | | | 0.98 | 2.6 |
| 7 | 0.94 | 1.06 | 1.06 | | | | 1.02 | 6.8 |
| 8 | 1.01 | 1.02 | 1.02 | 1.01 | | | 1.02 | 0.6 |
| 9 | 0.99 | 1.04 | 1.01 | 0.93 | | | 0.99 | 4.7 |
| 10 | 0.85 | 1.03 | 1.04 | 0.97 | | | 0.97 | 8.7 |
| 11 | 1.16 | 0.92 | | | | | 1.04 | 16.3 |
| 12 | 0.95 | 1.02 | 0.91 | 0.99 | | | 0.97 | 4.9 |
| 13 | 1.01 | 0.99 | 0.93 | 0.84 | 0.96 | | 0.95 | 7.0 |
| 14 | 0.97 | 0.98 | 0.95 | | | | 0.97 | 1.6 |
| 15 | 0.93 | 0.94 | 0.96 | 0.98 | 0.95 | | 0.95 | 2.0 |
| 16 | 0.94 | 0.97 | 0.93 | 0.81 | 0.88 | | 0.91 | 6.9 |
| 17 | 0.94 | 0.86 | 0.89 | 1.04 | 0.85 | | 0.92 | 8.5 |
| 18 | 0.94 | 0.91 | | | | | 0.93 | 2.3 |
| 19 | 1.06 | 0.94 | 0.88 | 0.99 | | | 0.97 | 7.9 |
| 20 | 0.98 | 0.96 | 1.01 | 1.04 | | | 1.00 | 3.5 |
| 21 | 1.07 | 1.02 | 1.14 | 1.04 | | | 1.07 | 4.9 |
| 22 | 1.04 | 1.07 | | | | | 1.06 | 2.0 |
| 23 | 0.99 | 1.01 | | | | | 1.00 | 1.4 |
| 24 | 1.06 | 1.01 | 1.07 | | | | 1.05 | 3.1 |
| 25 | 1.05 | 0.95 | 1.06 | | | | 1.02 | 6.0 |
| 26 | 1.13 | 1.05 | 0.97 | | | | 1.05 | 7.6 |
| 27 | 0.99 | 0.96 | 1.01 | 0.98 | 1.08 | 1.09 | 1.02 | 5.3 |
| 28 | 1.12 | 0.94 | 0.99 | | | | 1.02 | 9.1 |
| 29 | 0.93 | 1.14 | | | | | 1.04 | 14.3 |
| Average | | | | | | | 0.99 | 7.0 |

Table 4-2

Summary Comparison for Individual Foil Reactions

| Reaction | Data Base Average M/C or A/C | % std dev |
|---|---|---|
| $^{63}Cu(n,\alpha)^{60}Co$ | 1.08 | 7.2 |
| $^{54}Fe(n,p)^{54}Mn$ | 0.98 | 7.9 |
| $^{58}Ni(n,p)^{58}Co$ | 0.98 | 7.9 |
| $^{238}U(n,p)FP$ | 1.01 | 11.4 |
| $^{237}Np(n,p)FP$ | 1.04 | 11.7 |
| $\phi(E > 1.0\ MeV)$ | 0.99 | 7.0 |