Consumers Power

POWERING MICHIGAN'S PROGRESS

Palisades Nuclear Plant: 27780 Blue Star Memorial Highway, Covert, MI 49043

June 21, 1996

U.S. Nuclear Regulatory Commission
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DOCKET 50-255 - LICENSE DPR-20 - PALISADES PLANT
UPDATED REACTOR VESSEL FLUENCE SUBMITTAL - BALANCE OF THE RESPONSE TO QUESTIONS

On April 4, 1996, Consumers Power Company (CPCo) submitted a reevaluation of the Palisades fluence data. The reevaluation contained a new estimate of when the limiting reactor vessel material will reach the Pressurized Thermal Shock (PTS) screening criteria. On May 15, 1996, CPCo met with the NRC staff to discuss the updated reactor vessel fluence values and their effect on the required 10 CFR 50.61 analysis for the Palisades Plant. At the end of the meeting, the NRC staff provided a list of questions which were discussed during the meeting. They requested that Consumers Power Company respond to those questions within 30 days of the date of the meeting. Those questions were later included as Attachment 4 to the May 31, 1996, NRC letter which summarized the information discussed at the May 15, 1996, meeting.

To assist the staff and to expedite the review of the Palisades pressure vessel fluence reevaluation, we have provided our response in two parts as the information became available. On June 12, 1996, CPCo submitted a partial response to the NRC questions concerning the updated fluence submittal. This submittal contains, as Attachment 1, our response to the balance of the items from the May 31, 1996, NRC letter and questions

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questions that were discussed in subsequent telephone conversations on May 17, 1996, and June 3, 1996.

During a telecon on June 13, 1996, the NRC Project Manager concurred with an extension of the submittal date for the balance of the requested information.

SUMMARY OF COMMITMENTS

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

CSKozup for
Richard W. Smedley
Manager, Licensing

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

Attachment

ATTACHMENT 1

UPDATED REACTOR VESSEL FLUENCE SUBMITTAL - ADDITIONAL INFORMATION

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

RESPONSE TO BALANCE OF QUESTIONS

IN ATTACHMENT 4 TO THE MAY 31, 1996, NRC LETTER

AND MAY 17, 1996, AND JUNE 3, 1996, TELECONS

30 Pages

Updated Palisades Reactor Vessel Fluence Submittal - Additional Information

Request for Additional Information

1.2 *Provide the quantitative effect of each of the changes on the calculated value of the fluence and the effect on the interpretation of the in-vessel and cavity measurements.*

CPCo Response

The values presented in the following tables were derived from explicit calculations, estimates and engineering judgement.

Estimated Impact of Palisades Calculated Fluence Evaluation Changes			
Change	Vessel Inner Radius	In-Vessel Capsules	Ex-Vessel Capsules
Cross-section change from ENDF/B-IV to ENDF/B-VI	1-2% increase depending upon the azimuthal location	2-4% increase depending upon the radial and azimuthal location	15-16% increase depending upon the azimuthal location
Fission spectrum change from ENDF/B-V to ENDF/B-VI	Based on a study for the ORNL PCA benchmark pressure vessel simulator, the change to ENDF/B-VI produces a 1% decrease in the calculated fluence.		
Vessel inner radius increase by 0.12″	3% decrease	No effect	Combined effect produces a 13% decrease, of which, the majority is due to the increased vessel thickness
Vessel thickness increase by 0.29″	No effect	No effect	
Independent cycle specific calculations, there are several factors that are cycle and position dependent	In this table the effect of temperature and axial peaking are discussed separately. The other affected items include core composition, fission spectrum (burn-up), and pin power gradients. These changes have not been quantified, but they have a range of both positive and negative effects, all less than the effect of water temperature.		No effect
Source change from PDQ to SIMULATE-3	The change from PDQ to SIMULATE-3 pin power distributions produces no net effect.		
Use of core midplane peaking factors rather than assembly maximum peaking factors	The treatment of the axial peaking factor at the core midplane produces a 1-6% decrease. The larger decrease occurs in cycles 1 through 4, where the previous calculations assumed an average source for cycles 1 through 5, and the smaller decrease occurs in the later fuel cycles.		

Updated Palisades Reactor Vessel Fluence Submittal - Additional Information

Estimated Impact of Palisades Calculated Fluence Evaluation Changes			
Change	Vessel IR	In-Vessel Capsules	Ex-Vessel Capsules
Cycle specific T_{inlet} and T_{bypass} modeling	This change primarily affects the Cycle 1 and Cycle 2 calculations since the previous analysis assumed an average source for cycles 1 through 5 using a higher inlet and bypass water temperature. To a lesser extent, cycles 9, 10 and 11 were also affected. This change produces a 0-10% decrease depending upon the azimuthal location and cycle.		
Explicit modeling of ex-vessel dosimetry locations	No effect	No effect	3 to 23% increase depending on location.
Increased angular discretization with an S_{16} angular quadrature across the entire energy spectrum	No effect	No effect	Using a higher angular quadrature improves the accuracy of the calculation. The impact is small, 0.5 to 1.5% increase depending on the cavity location.

The large changes in the calculated fluxes for Cycles 1 and 2 are primarily due to the use of cycle specific source strengths versus a Cycle 1-5 average source, reduced bypass and inlet temperatures and, to a lesser extent, the treatment of the axial peaking factors. The table below approximates the magnitudes of these effects in Cycles 1 and 2 for an azimuthal location of 30°.

Calculation Change	Cycle 1 Impact	Cycle 2 Impact
Independent cycle specific sources	-4%	-14%
Cycle specific T_{inlet} and T_{bypass} modeling	-9%	-6%
Cross-section change from ENDF/B-IV to ENDF/B-VI	+2%	+2%
Vessel IR increase by 0.12″	-3%	-3%
Use of core midplane axial peaking factors rather than the assembly maximum	-3%	-5%
Fission spectrum change from ENDF/B-V to ENDF/B-VI	-1%	-1%
Total approximate effect	-17%	-25%

The specific activity of each of the neutron monitors was determined using established ASTM procedures. Having the measured specific activities, the physical characteristics of the sensors, and the operating history of the reactor, reaction rates referenced to full power operation were determined as per the detailed description provided in the response to Request 3.1.

In the reaction rate calculation, the ratio C_j, calculated for each fuel cycle using discrete ordinates transport technology, accounts for the change in sensor reaction rates caused by variations in flux level induced by changes in core spatial power distributions from fuel cycle to fuel cycle. For a single cycle irradiation such as is common with cavity dosimetry, C_j is taken to be 1.0. However, for multiple cycle irradiations, particularly those employing low leakage fuel management, the additional C_j correction must be employed. Thus, changes in the calculated flux at the capsule positions would only impact those capsules irradiated over multiple cycles. These include:

A240	In-vessel accelerated capsule at azimuthal location of 30° withdrawn at the end of Cycle (EOC) 2.
W290	In-vessel wall capsule at azimuthal location of 20° withdrawn at EOC 5.
W110	In-vessel wall capsule at azimuthal location of 20° withdrawn at EOC 10.
Capsule A	Ex-vessel dosimetry at azimuthal location of 6° (270° Ref) irradiated during Cycles 8 and 9.
Capsule O	Ex-vessel dosimetry at azimuthal location of 6° (270° Ref) irradiated during Cycles 10 and 11.
Capsule P	Ex-vessel dosimetry at azimuthal location of 16° (280° Ref) irradiated during Cycles 10 and 11.
Capsule R	Ex-vessel dosimetry at azimuthal location of 26° (290° Ref) irradiated during Cycles 10 and 11.
Capsule S	Ex-vessel dosimetry at azimuthal location of 36° (300° Ref) irradiated during Cycles 10 and 11.
Capsule T	Ex-vessel dosimetry at azimuthal location of 39° (315° Ref) irradiated during Cycles 10 and 11.
Capsule U	Ex-vessel dosimetry at azimuthal location of 24° (330° Ref) irradiated during Cycles 10 and 11.

In addition to the dosimetry irradiated over multiple cycles, the calculated values at the dosimetry locations are used to correct the fission monitors and are used as the input

trial spectrum for FERRET. These effects are addressed in more detail in the responses to Requests 2.3 and 3.4, respectively.

2.1 *Provide a description of the methodology used to calculate the gamma flux and its spectrum in the ^{237}Np and ^{238}U dosimeter locations. Include: core fission gamma transport, capture gamma sources, capture gamma cross sections, thermal flux calculation, associated neutron cross sections, ^{237}Np, ^{238}U (gamma, f) yield, etc.*

CPCo Response

2.1.1 General Discussion

The forward transport calculations for the Palisades reactor model were carried out in R-θ geometry using the DORT two-dimensional discrete ordinates transport theory code and the BUGLE-93 cross-section library. The BUGLE-93 library is a 47 neutron and 20 gamma ray energy group, ENDF/B-VI based, data set produced specifically for light water reactor applications. In these analyses, anisotropic scattering was treated with a P_3 expansion of the scattering cross-sections and the angular discretization was 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 forward calculations were supplied as pin-by-pin power distributions, initial enrichments, and cycle burnups for each fuel assembly in the 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 per fission evaluated at the mean assembly burnup value. The source spectrum was calculated by determining the fraction of fissions occurring 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.

The source from each fuel assembly was spatially located to take into account the varying gaps between fuel assemblies and represents the location of the source from each pin. The source was converted from the X-Y pin geometry to the R-θ DORT geometry by distributing the source over a square area equal to the pitch for each pin.

The neutron fission spectrum used to determine the source spectrum is from the ENDF/B-VI fission spectrum for neutrons for ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu with the standard weighting. The gamma-ray fission spectrum inherent in the BUGLE-93 cross-section library for 3 weight percent enriched ^{235}U weighting was used. Both the $X(n,f)^{137}Cs$ and $X(\gamma,f)^{137}Cs$ used 6.00 percent and 6.27 percent fission yields

for ^{238}U[1] and ^{237}Np[2], respectively.

The thermal neutron flux was calculated using the DORT code employing the ENDF/B-VI cross-sections as part of the 47 neutron energy groups contained in the BUGLE-93 library.

2.1.2 Calculation of Gamma-Ray Fluxes

The calculation of gamma-ray fluxes in a pressurized water reactor involves a number of different steps. First, the reactor core power distribution must be defined. Second, the energy and spatial distribution of the neutron flux must be calculated within the reactor geometry. Third, the energy and spatial distribution of the gamma-ray sources must be calculated. Fourth, the energy and spatial distribution of the gamma-ray flux must be calculated. Finally, the photofission rate at a dosimeter location may be calculated.

The gamma-ray flux was calculated using the DORT code employing the ENDF/B-VI cross-sections contained in the BUGLE-93 library. The coupled neutron/gamma-ray library contains gamma-ray production data from fission, radiative capture, and inelastic scattering reactions for core, water, and structural materials. This gamma-ray production component of the cross-section library results in a space and energy dependent gamma-ray source throughout the reactor geometry.

The source of photofission cross-sections used in the Palisades analysis are provided in: V. V. Verbinski, *et al.*, "Photo Interference Corrections in Neutron Dosimetry for Reactor Pressure Vessel Lifetime Studies," *Nuclear Science and Engineering* (75), page 159, 1980.

2.1.3 Radiation Analysis

Approximately 10 to 15 percent of the recoverable energy in the fission process appears as gamma rays and fission neutron kinetic energy. The majority of this energy is absorbed in the core, the reactor coolant, and the reactor internals structure, where it generates heat.

Gamma rays are born in the reactor in several ways. Prompt gamma rays are emitted at the instant of fission by the fissioning nucleus. Delayed gamma rays are emitted by the decaying fission products. Secondary gamma rays are also produced in the core,

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[1] ASTM Standard E 704-90, "Standard Test Method for Measuring Reaction Rates by Radioactivation of Uranium-238"

[2] ASTM Standard E 705-90, "Standard Test Method for Measuring Reaction Rates by Radioactivation of Neptunium-237"

the decaying fission products. Secondary gamma rays are also produced in the core, reactor coolant, and the reactor structures by neutron interactions other than fission. The two most important of these neutron interaction processes are radiative capture and inelastic scattering.

In radiative capture of neutrons by nuclei, the kinetic energy of the incident neutron and its binding energy in the compound nucleus are emitted as one or more, generally high-energy gamma rays.

In neutron inelastic scattering, part of the energy of the incident neutron is carried off by the scattered neutron, and part is absorbed by the target nucleus, which subsequently emits one or more, generally low-energy gamma rays.

In the present reactor calculations, the core power distribution defines the fission neutron source and the fission gamma-ray source. Neutron transport calculations are required to define the distribution of the other gamma-ray sources both inside and outside the core. Once all the sources of gamma rays are defined, gamma-ray transport calculations are performed to determine the energy and spatial distribution of the gamma-ray flux. In practice, coupled neutron/gamma-ray transport calculations are run wherein gamma-ray sources are represented as transfer matrices in the cross-section set from neutron groups to gamma-ray groups.

It is the neutron-induced gamma-ray sources in the internals themselves which allow the boron concentration in the water to play a role in the calculation of the gamma-ray flux over a fuel cycle. This occurs because the boron in the water competes with the steel for thermal neutrons. When a neutron is captured in the boron in the water, an alpha particle is emitted which is stopped in a very short distance, giving up heat to the water. There are no significant gamma rays produced by this process, which is in marked contrast to the high-energy gamma rays resulting from neutron radiative capture in the structure discussed earlier. Gamma-ray fluxes vary ±15 percent over a fuel cycle due to the change in the boron concentration in the water.

There are three main mechanisms through which gamma rays interact with and deposit energy in the core, the reactor coolant, and the reactor structures. These mechanisms are the photoelectric effect, pair production, and Compton scattering.

In the photoelectric process, the entire energy of the gamma ray is transferred to an orbital electron which is ejected from its shell and emerges from the atom as a photoelectron. The photoelectric process is important only at low gamma-ray energies.

In the process of pair production, a gamma ray interacts with the electric field of atomic electrons or the nucleus. The incident gamma ray is completely annihilated, and its energy is converted into the mass and kinetic energy of an electron-positron pair. Pair production is important only at high gamma-ray energies. For the purposes of understanding reactor internals heating, both the photoelectric effect and pair

production may be regarded as a total absorption of the gamma-ray energy at the point of interaction, which then appears as heat in the material.

The Compton effect is the scattering of a gamma ray by a free electron. The gamma ray imparts energy to the electron and is altered in direction and energy. The energy given to the electron appears as heat in the material. A very important feature of the Compton effect is the fact that, except when the scattering angle is large, the gamma ray emerges from the interaction with a significant fraction of the incident gamma-ray energy. This fact accounts for much of the complexity associated with gamma-ray transport analysis. Compton scattering is the most important gamma-ray interaction process at intermediate gamma-ray energies.

2.1.4 Analytical Methods

The calculation of gamma-ray fluxes used in the Palisades analysis employs methods which are well established in the nuclear industry. For example, NUREG/CR-4827 (WCAP-11173), "Neutron and Gamma-Ray Flux Calculations for the VENUS PWR Engineering Mockup", A. H. Fero, January 1987, describes the analysis of neutron and gamma-ray fluxes in the VENUS PWR engineering mockup benchmark experiment (SCK/CEN Mol, Belgium). This mockup experiment is unique in two ways. It is the first mockup to correctly represent the heterogeneities which exist in the PWR core peripheral fuel assemblies, core baffle, core barrel, and neutron pad. This is accomplished by using low-enrichment fuel pins and a representative PWR fuel assembly geometry (15 x 15) with full thickness type 304 stainless steel reactor internals structures located with representative water gap spacing. The VENUS mockup also represents locally the stair-step geometry of the core periphery. Second, the VENUS mockup is extremely well characterized in terms of as-built dimensions, material compositions, and pin-by-pin core power distributions.

The analysis of VENUS was performed using the methods and procedures used to analyze commercial PWRs. Specifically, two-dimensional discrete ordinates transport theory calculations were performed using the DOT-IIIW code. Calculations were run in both X-Y and R-θ geometries for a 90 degree sector of the mockup. Fixed distributed source calculations were run with a P_3 expansion of the scattering cross-sections and an S_8 order of angular quadrature using the ENDF/B-V ^{235}U thermal fission neutron spectrum and cross-sections from the SAILOR and BUGLE-80 (coupled 47 neutron - 20 gamma-ray energy groups) cross-section libraries which were derived from the VITAMIN-C (coupled 171 neutron - 36 gamma-ray energy groups) cross-section library. The finite height of the VENUS mockup was accounted for in the two-dimensional calculations by the use of group- and zone-dependent DB^2 terms derived from axial leakages calculated in a DOT-IIIW R-Z geometry approximation of the mockup. Calculated results were presented for eight fast-neutron reactions and for gamma-ray energy-deposition rates. Comparisons to measured neutron reaction rates show agreement in the fuel, water, and steel regions (baffle and barrel) that is generally within ±10 percent. The average of comparisons to measured gamma-ray energy-

deposition rates in the baffles and core barrel is 0.96 which is within the variation in the experimental data.

2.2 *Describe the methodology used to incorporate ^{237}Np and ^{238}U dosimetry results into the fast (E>1.0 Mev) neutron flux calculation.*

CPCo Response

See the responses to Requests 3.1 and 3.4.

2.3 *Describe all the corrections and/or adjustments made to the measured dosimetry values.*

CPCo Response

The use of passive monitors such as those listed below does not yield a direct measure of the energy dependent neutron flux at the point of interest. Rather, the activation or fission process is a measure of the integrated effect that the time and energy dependent neutron flux has on the target material over the course of the irradiation period.

Nuclear Parameters Used in the Evaluation of Neutron Sensors

Monitor Material	Reaction of Interest	Detector Response	Product Half-life	Fission Yield (%)
Copper	^{63}Cu (n,α)	E > 4.7 MeV	5.271 y	
Titanium	^{46}Ti (n,p)	E > 4.4 MeV	83.83 d	
Iron	^{54}Fe (n,p)	E > 1.0 MeV	312.5 d	
Nickel	^{58}Ni (n,p)	E > 1.0 MeV	70.78 d	
Uranium-238	^{238}U (n,f)	E > 0.4 MeV	30.17 y	6.00
Neptunium-237	^{237}Np (n,f)	E > 0.08 MeV	30.17 y	6.27
Cobalt-Al	^{59}Co (n,γ)	E > 0.015 MeV	5.271 y	

An accurate assessment of the average neutron flux level incident on the various monitors may be derived from the activation measurements only if the irradiation parameters are well known. In particular, the following variables are of interest:

- The measured specific activity of each monitor,
- The physical characteristics of each monitor,
- The operating history of the reactor,
- The energy response of each monitor, and
- The neutron energy spectrum at the monitor location.

The specific activity of each of the neutron monitors was determined using established ASTM procedures (see Response 3.1). Having the measured specific activities, the physical characteristics of the sensors, and the operating history of the reactor, reaction rates referenced to full power operation were determined as per the detailed description provided in the response to Request 3.1.

In order to correct for ^{235}U content and ^{239}Pu build-in in the ^{238}U sensors as well as for photofission reactions in both of the fission sensors, the ^{238}U and ^{237}Np reaction rates were reduced prior to use in determining the exposure evaluation parameters. These corrections are shown for each dosimeter on the following pages.

These corrections are generic and are applied to fluence analyses performed for other plants, however, the magnitude of the corrections are plant specific and are derived from the calculated fast neutron flux and the thermal neutron and the gamma-ray induced fission rates. The application of the photofission correction to the ^{238}U and ^{237}Np is a change from the previous Palisades analysis. If the photofission correction was neglected, the change to the best estimate pressure vessel fluence would be a 2% increase which is small compared to the total uncertainty associated with the best estimate. Also, neglecting the photofission correction increases the scatter among the M/C bias factor data.

Updated Palisades Reactor Vessel Fluence Submittal - Additional Information

Capsule ID & Dosimetry	Description	^{238}U Correction Factor for ^{235}U Content & ^{239}Pu Build-in	^{238}U Photofission Correction Factor	^{237}Np Photofission Correction Factor
A240 ^{63}Cu (n,α) ^{60}Co Cd ^{54}Fe (n,p) ^{54}Mn ^{58}Ni (n,p) ^{58}Co Cd ^{46}Ti (n,p) ^{46}Sc	In-vessel accelerated capsule at azimuthal location of 30° withdrawn at the end of Cycle 2.	No ^{238}U or ^{237}Np dosimetry		
W290 ^{63}Cu (n,α) ^{60}Co Cd ^{54}Fe (n,p) ^{54}Mn ^{58}Ni (n,p) ^{58}Co Cd ^{46}Ti (n,p) ^{46}Sc ^{238}U (n,f) ^{137}Cs Cd	In-vessel wall capsule at azimuthal location of 20° withdrawn at the end of Cycle 5.	0.842	0.882	No ^{237}Np dosimetry
W290-9 ^{63}Cu (n,α) ^{60}Co Cd ^{54}Fe (n,p) ^{54}Mn ^{58}Ni (n,p) ^{58}Co Cd ^{46}Ti (n,p) ^{46}Sc ^{238}U (n,f) ^{137}Cs Cd ^{237}Np (n,f) ^{137}Cs Cd ^{59}Co (n,γ) ^{60}Co ^{59}Co (n,γ) ^{60}Co Cd	In-vessel wall capsule at azimuthal location of 20° irradiated during Cycle 9.	No correction necessary due to high purity foil and low exposure of the capsule	0.891	0.931
W110 ^{63}Cu (n,α) ^{60}Co Cd ^{54}Fe (n,p) ^{54}Mn ^{58}Ni (n,p) ^{58}Co Cd ^{46}Ti (n,p) ^{46}Sc	In-vessel wall capsule at azimuthal location of 20° withdrawn at the end of Cycle 10.	No corrections necessary ^{238}U dosimeter cadmium cover melted No ^{237}Np dosimetry		
Capsule B ^{63}Cu (n,α) ^{60}Co Cd ^{54}Fe (n,p) ^{54}Mn Cd ^{58}Ni (n,p) ^{58}Co Cd ^{46}Ti (n,p) ^{46}Sc Cd ^{238}U (n,f) ^{137}Cs Cd ^{237}Np (n,f) ^{137}Cs Cd ^{59}Co (n,γ) ^{60}Co ^{59}Co (n,γ) ^{60}Co Cd	Ex-vessel dosimetry at azimuthal location of 16° (280° Ref) irradiated during Cycle 8.	0.961	0.968	0.993

Updated Palisades Reactor Vessel Fluence Submittal - Additional Information

Capsule ID & Dosimetry	Description	^{238}U Correction Factor for ^{235}U Content & ^{239}Pu Build-in	^{238}U Photofission Correction Factor	^{237}Np Photofission Correction Factor
Capsule D ^{63}Cu (n,α) ^{60}Co Cd ^{54}Fe (n,p) ^{54}Mn Cd ^{58}Ni (n,p) ^{58}Co Cd ^{46}Ti (n,p) ^{46}Sc Cd ^{238}U (n,f) ^{137}Cs Cd ^{237}Np (n,f) ^{137}Cs Cd ^{59}Co (n,γ) ^{60}Co ^{59}Co (n,γ) ^{60}Co Cd	Ex-vessel dosimetry at azimuthal location of 26° (290° Ref) irradiated during Cycle 8.	0.951	0.962	0.992
Capsule G ^{63}Cu (n,α) ^{60}Co Cd ^{54}Fe (n,p) ^{54}Mn Cd ^{58}Ni (n,p) ^{58}Co Cd ^{46}Ti (n,p) ^{46}Sc Cd ^{238}U (n,f) ^{137}Cs Cd ^{237}Np (n,f) ^{137}Cs Cd ^{59}Co (n,γ) ^{60}Co ^{59}Co (n,γ) ^{60}Co Cd	Ex-vessel dosimetry at azimuthal location of 39° (315° Ref) irradiated during Cycle 8.	0.914	0.954	0.991
Capsule A ^{63}Cu (n,α) ^{60}Co Cd ^{54}Fe (n,p) ^{54}Mn Cd ^{58}Ni (n,p) ^{58}Co Cd ^{46}Ti (n,p) ^{46}Sc Cd ^{238}U (n,f) ^{137}Cs Cd ^{237}Np (n,f) ^{137}Cs Cd ^{59}Co (n,γ) ^{60}Co ^{59}Co (n,γ) ^{60}Co Cd	Ex-vessel dosimetry at azimuthal location of 6° (270° Ref) irradiated during Cycles 8 and 9.	0.959	0.962	0.992

Updated Palisades Reactor Vessel Fluence Submittal - Additional Information

Capsule ID & Dosimetry	Description	^{238}U Correction Factor for ^{235}U Content & ^{239}Pu Build-in	^{238}U Photofission Correction Factor	^{237}Np Photofission Correction Factor
Capsule J ^{63}Cu (n,α) ^{60}Co Cd ^{54}Fe (n,p) ^{54}Mn Cd ^{58}Ni (n,p) ^{58}Co Cd ^{46}Ti (n,p) ^{46}Sc Cd ^{238}U (n,f) ^{137}Cs Cd ^{237}Np (n,f) ^{137}Cs Cd ^{59}Co (n,γ) ^{60}Co ^{59}Co (n,γ) ^{60}Co Cd	Ex-vessel dosimetry at azimuthal location of 16° (280° Ref) irradiated during Cycle 9.	0.961	0.969	0.993
Capsule K ^{63}Cu (n,α) ^{60}Co Cd ^{54}Fe (n,p) ^{54}Mn Cd ^{58}Ni (n,p) ^{58}Co Cd ^{46}Ti (n,p) ^{46}Sc Cd ^{238}U (n,f) ^{137}Cs Cd ^{237}Np (n,f) ^{137}Cs Cd ^{59}Co (n,γ) ^{60}Co ^{59}Co (n,γ) ^{60}Co Cd	Ex-vessel dosimetry at azimuthal location of 26° (290° Ref) irradiated during Cycle 9.	0.951	0.965	0.992
Capsule N ^{63}Cu (n,α) ^{60}Co Cd ^{54}Fe (n,p) ^{54}Mn Cd ^{58}Ni (n,p) ^{58}Co Cd ^{46}Ti (n,p) ^{46}Sc Cd ^{238}U (n,f) ^{137}Cs Cd ^{237}Np (n,f) ^{137}Cs Cd ^{59}Co (n,γ) ^{60}Co ^{59}Co (n,γ) ^{60}Co Cd	Ex-vessel dosimetry at azimuthal location of 39° (315° Ref) irradiated during Cycle 9.	0.914	0.958	0.992

Capsule ID & Dosimetry	Description	^{238}U Correction Factor for ^{235}U Content & ^{239}Pu Build-in	^{238}U Photofission Correction Factor	^{237}Np Photofission Correction Factor
Capsule O ^{63}Cu (n,α) ^{60}Co Cd ^{54}Fe (n,p) ^{54}Mn Cd ^{58}Ni (n,p) ^{58}Co Cd ^{46}Ti (n,p) ^{46}Sc Cd ^{238}U (n,f) ^{137}Cs Cd ^{237}Np (n,f) ^{137}Cs Cd ^{59}Co (n,γ) ^{60}Co ^{59}Co (n,γ) ^{60}Co Cd	Ex-vessel dosimetry at azimuthal location of 6° (270° Ref) irradiated during Cycles 10 and 11.	0.996	0.964	0.992
Capsule P ^{63}Cu (n,α) ^{60}Co Cd ^{54}Fe (n,p) ^{54}Mn Cd ^{58}Ni (n,p) ^{58}Co Cd ^{46}Ti (n,p) ^{46}Sc Cd ^{238}U (n,f) ^{137}Cs Cd ^{237}Np (n,f) ^{137}Cs Cd ^{59}Co (n,γ) ^{60}Co ^{59}Co (n,γ) ^{60}Co Cd	Ex-vessel dosimetry at azimuthal location of 16° (280° Ref) irradiated during Cycles 10 and 11.	0.996	0.967	0.993
Capsule R ^{63}Cu (n,α) ^{60}Co Cd ^{54}Fe (n,p) ^{54}Mn Cd ^{58}Ni (n,p) ^{58}Co Cd ^{46}Ti (n,p) ^{46}Sc Cd ^{238}U (n,f) ^{137}Cs Cd ^{237}Np (n,f) ^{137}Cs Cd ^{59}Co (n,γ) ^{60}Co ^{59}Co (n,γ) ^{60}Co Cd	Ex-vessel dosimetry at azimuthal location of 26° (290° Ref) irradiated during Cycles 10 and 11.	0.996	0.965	0.993

Capsule ID & Dosimetry	Description	^{238}U Correction Factor for ^{235}U Content & ^{239}Pu Build-in	^{238}U Photofission Correction Factor	^{237}Np Photofission Correction Factor
Capsule S ^{63}Cu (n,α) ^{60}Co Cd ^{54}Fe (n,p) ^{54}Mn Cd ^{58}Ni (n,p) ^{58}Co Cd ^{46}Ti (n,p) ^{46}Sc Cd ^{238}U (n,f) ^{137}Cs Cd ^{237}Np (n,f) ^{137}Cs Cd ^{59}Co (n,γ) ^{60}Co ^{59}Co (n,γ) ^{60}Co Cd	Ex-vessel dosimetry at azimuthal location of 36° (300° Ref) irradiated during Cycles 10 and 11.	0.996	0.961	0.992
Capsule T ^{63}Cu (n,α) ^{60}Co Cd ^{54}Fe (n,p) ^{54}Mn Cd ^{58}Ni (n,p) ^{58}Co Cd ^{46}Ti (n,p) ^{46}Sc Cd ^{238}U (n,f) ^{137}Cs Cd ^{237}Np (n,f) ^{137}Cs Cd ^{59}Co (n,γ) ^{60}Co ^{59}Co (n,γ) ^{60}Co Cd	Ex-vessel dosimetry at azimuthal location of 39° (315° Ref) irradiated during Cycles 10 and 11.	0.996	0.960	0.992
Capsule U ^{63}Cu (n,α) ^{60}Co Cd ^{54}Fe (n,p) ^{54}Mn Cd ^{58}Ni (n,p) ^{58}Co Cd ^{46}Ti (n,p) ^{46}Sc Cd ^{238}U (n,f) ^{137}Cs Cd ^{237}Np (n,f) ^{137}Cs Cd ^{59}Co (n,γ) ^{60}Co ^{59}Co (n,γ) ^{60}Co Cd	Ex-vessel dosimetry at azimuthal location of 24° (330° Ref) irradiated during Cycles 10 and 11.	0.996	0.966	0.993

2.4 *Provide a description of the fractional contribution to the flux from each of the dosimeter reactions.*

CPCo Response

See the response to Request 3.4.

3.1 *Describe the uncertainty analysis for the accelerated capsule, the inner wall capsule and the cavity dosimeters. Include: position, counting, weighing, power history, calibrations, cross sections, etc.*

CPCo Response

3.1.1. General Discussion

The methodology used in the evaluation of the multiple foil dosimetry sets irradiated at the Palisades reactor utilizes a least squares adjustment procedure to produce a best fit among the calculated neutron spectrum at each sensor set location and the set of measured reaction rates from the dosimetry package. In this methodology, uncertainties in the derived exposure rates [ϕ (E > 1.0 MeV), ϕ (E > 0.1 MeV), and dpa] at the measurement locations are dependent on the resultant fit of the adjusted spectrum to the measured data; and include a combination of the uncertainties in measured reaction rates, sensor cross-sections, and the calculated spectrum.

In the analysis performed for the Palisades reactor, both the "best estimate" exposure rates and the associated uncertainties were obtained from the measured reaction rates, dosimetry cross-sections, and calculated neutron spectra by means of the SAND-II/FERRET least squares adjustment procedure. See the response to Request 3.4 for further discussion of the SAND-II/FERRET approach.

The use of an adjustment procedure to evaluate neutron dosimetry from Light Water Reactors is described in ASTM E 944-89, "Standard Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Vessel Surveillance" as follows.

> "3.3.2.1 The algorithms of the adjustment codes tend to decrease the variances of the adjusted data compared to the corresponding input values. The least squares adjustment codes yield estimates for the output data with minimum variances, that is, the 'best' estimates. This is the primary reason for using these adjustment procedures."

In using the adjustment procedure, dosimetry measurements are provided as a set of reaction rates denoted by the following symbols:

$$R_i \quad I = 1,2,...$$

Reaction cross-sections for the dosimetry sensors are obtained from the Sandia National Laboratories Radiation Metrology Laboratory (SNLRML) ENDF/B-VI based evaluated dosimetry cross-section file. The cross-sections for the i^{th} reaction as a function of energy are denoted by the following:

$$\sigma_i(E) \quad I = 1,2,....$$

The calculated neutron spectrum input to the adjustment procedure are obtained on a location specific basis from the results of the cycle specific discrete ordinates transport calculation. The group fluxes from the transport calculation are denoted by the following:

$$\phi_j \quad j = 1,2,...k$$

The uncertainties associated with the measured reaction rates, dosimetry cross-sections, and calculated neutron spectrum are also input to the adjustment procedure in the form of variances and covariances. In the evaluations performed for the Palisades reactor, the assignment of the input uncertainties also follows the guidance provided in the ASTM standard E 944-89.

3.1.2. Uncertainties in the Input to the Adjustment Procedure

3.1.2.1 Measured Reaction Rates

The determination of the individual reaction rates involves laboratory counting procedures, decay corrections to account for the operating history of the reactor, and corrections for competing reactions within the sensors. Each of these facets of the reaction rate determinations are discussed in this section.

3.1.2.2 Counting Procedures

Internal surveillance capsule and ex-vessel reactor cavity dosimetry packages employed at the Palisades reactor consist of comprehensive multiple foil sensor sets that make use of some or all of the following reactions:

$^{63}Cu\ (n,\alpha)\ ^{60}Co$
$^{46}Ti\ (n,p)\ ^{46}Sc$
$^{54}Fe\ (n,p)\ ^{54}Mn$
$^{58}Ni\ (n,p)\ ^{58}Co$
$^{238}U\ (n,f)\ ^{137}Cs$
$^{237}Np\ (n,f)\ ^{137}Cs$
$^{59}Co\ (n,\gamma)\ ^{60}Co$

Following irradiation, the specific activity of each of the irradiated radiometric sensors is determined using the latest version of ASTM counting procedures for each reaction. In particular, the following standards are applicable to the radiometric sensors utilized in LWR programs:

E523	Standard Test Method for Measuring Fast Neutron Reaction Rates by Radioactivation of Copper
E526	Standard Test Method for Measuring Fast Neutron Reaction Rates

	by Radioactivation of Titanium
E263	Standard Test Method for Measuring Fast Neutron Reaction Rates by Radioactivation of Iron
E264	Standard Test Method for Measuring Fast Neutron Reaction Rates by Radioactivation of Nickel
E704	Standard Test Method for Measuring Reaction Rates by Radioactivation of Uranium-238
E705	Standard Test Method for Measuring Reaction Rates by Radioactivation of Neptunium-237
E481	Standard Test Method for Measuring Neutron Fluence Rate by Radioactivation of Cobalt and Silver
E1005	Standard Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance
E181	Standard General Methods for Detector Calibration and Analysis of Radionuclides

Following sample preparation and weighing, the specific activity of each sensor is determined by means of a lithium drifted germanium, Ge(Li), gamma spectrometer. In the case of the multiple foil sensor sets, these analyses are performed by direct counting of each of the individual sensors, or, as is sometimes the case with ^{238}U and ^{237}Np fission monitors from internal surveillance capsules, by direct counting preceded by dissolution and chemical separation of cesium from the sensor.

3.1.2.3 Decay Corrections

Having the measured specific activities, the operating history of the reactor, and the physical characteristics of the sensors, reaction rates referenced to full power operation are determined from the following equation:

$$R = \frac{A}{N_0 \, F \, Y \sum \frac{P_j}{P_{ref}} C_j \, [1 - e^{-\lambda t_j}] \, [e^{-\lambda t_d}]}$$

where:

R	=	Reaction rate averaged over the irradiation period and referenced to operation at a core power level of P_{ref} (rps/nucleus).
A	=	Measured specific activity (dps/gm).
N_0	=	Number of target element atoms per gram of sensor.
F	=	Weight fraction of the target isotope in the sensor material.
Y	=	Number of product atoms produced per reaction.
P_j	=	Average core power level during irradiation period j (MW).
P_{ref}	=	Maximum or reference power level of the reactor (MW).
C_j	=	Calculated ratio of ϕ(E > 1.0 MeV) during irradiation period j to the time weighted average ϕ(E > 1.0 MeV) over the entire irradiation period.
λ	=	Decay constant of the product isotope (1/sec).
t_j	=	Length of irradiation period j (sec).
t_d	=	Decay time following irradiation period j (sec).

and the summation is carried out over the total number of monthly intervals comprising the irradiation period.

In the above equation, the ratio $[P_j]/[P_{ref}]$ accounts for month by month variation of power level within any given fuel cycle as well as over multiple fuel cycles. For the sensor sets utilized in surveillance capsule and reactor cavity irradiations, the half-lives of the product isotopes are long enough that a monthly histogram describing reactor operation has proven to be an adequate representation for use in radioactive decay corrections for the reactions of interest in the exposure evaluations. The ratio C_j, calculated for each fuel cycle using discrete ordinates transport technology, accounts for the change in sensor reaction rates caused by variations in flux level induced by changes in core spatial power distributions from fuel cycle to fuel cycle. For a single cycle irradiation such as is common with cavity dosimetry C_j is taken to be 1.0. However, for multiple cycle irradiations, particularly those employing low leakage fuel management, the additional C_j correction must be employed. This additional correction can be quite significant for internal surveillance capsules that have been irradiated for many cycles in a reactor that has transitioned from non-low leakage to low leakage fuel management.

3.1.2.4 Corrections for Competing Reactions

Prior to using the measured reaction rates in the dosimetry evaluation procedures, additional corrections are made to the ^{238}U measurements to account for the presence of ^{235}U impurities in the sensors as well as to correct for the build-in of plutonium isotopes over the course of the irradiation. These corrections are location and fluence dependent and are derived from the results of the discrete ordinates calculations and, when available, with measurements from paired uranium dosimeters.

In addition to the corrections made for the presence of ^{235}U and the build-in of plutonium isotopes in the ^{238}U fission sensors, corrections are also made to both ^{238}U and ^{237}Np sensors to account for gamma-ray induced fission reactions occurring over the course of the irradiation. These photofission corrections are, likewise, location dependent and are based on the plant specific discrete ordinates transport calculations.

3.1.2.5 Reaction Rate Uncertainties

The overall uncertainty associated with the measured reaction rates used in the evaluation of exposure parameters includes components due to the basic measurement process, the irradiation history corrections, and the corrections for competing reactions in the fission sensors. A matrix of the uncertainties associated with the reactions applicable to the Palisades dosimetry evaluations is as follows:

	COUNTING	DECAY CORRECTION	COMPETING REACTIONS	NET UNCERTAINTY
^{63}Cu (n,α)	3%	2%	0%	4%
^{46}Ti (n,p)	3%	4%	0%	5%
^{54}Fe (n,p)	3%	2%	0%	4%
^{58}Ni (n,p)	3%	4%	0%	5%
^{238}U (n,f)	5%	3%	4%	7%
^{237}Np (n,f)	5%	3%	1%	6%
^{59}Co (n,γ)	3%	4%	0%	5%

In developing this uncertainty tabulation, the counting component is derived from the expected accuracy using the appropriate ASTM standards. The component due to irradiation history (decay correction term) includes the effects of short vs long product half lives, the product yield in the fission monitors, and target abundance in the sensor material. The uncertainties due to competing reactions were based on the assumption that the error in the calculated correction using the plant specific transport results could be as high as 25% and the resultant uncertainty in the net reaction rate is then 25% of the total correction. For example, a 25% uncertainty in a 4% photofission correction to a measured dosimeter activity results in a net 1% additional uncertainty in the derived reaction rate.

In addition to the use of the ASTM standards in the evaluation of sensor reaction rates, over the course of the last 15 years these procedures have been tested via round robin counting exercises included as a part of the NRC sponsored Light Water Reactor

Surveillance Dosimetry Improvement Program (LWR-SDIP) as well as by evaluation of fluence counting standards provided by the National Institute of Science and Technology (NIST). Each of these counting exercises involved evaluation of neutron sensors typical of those used in light water reactor measurement programs. The results of these studies demonstrated that these procedures do, in fact, produce measured reaction rates within the 1σ uncertainties specified in the above tabulation.

A further consistency check on the measured reaction rates from in-vessel surveillance capsule and reactor cavity dosimetry irradiations is obtained from an examination of the several location and reactor dependent data bases built up over many years of performing reactor dosimetry. Examples of these data bases are provided in WCAP-14044, "Westinghouse Surveillance Capsule Neutron Fluence Re-evaluation", E. P. Lippincott, April 1994. The plant data included in these data bases also lend support to the reaction rate uncertainties specified above.

3.1.2.6 Dosimetry Cross-Sections

As noted in Section 3.1.1, the dosimetry cross-sections are taken directly from the SNLRML evaluated dosimetry cross-section data base (DLC-178, RSIC Data Library Collection SNLRML, Recommended Dosimetry Cross Section Compendium, July 1994). Cross-section uncertainties in the form of variances and covariances are provided on this data file along with the basic dosimetry cross-section data.

3.1.2.7 Calculated Spectrum

The uncertainties in the calculated neutron energy spectrum at the location of the individual sensor sets are also input in the form of variances and covariances with the following specifications:

Flux Normalization Uncertainty	30%
Flux Group Uncertainties	
(E > 0.0055 MeV)	30%
(0.68 eV < E < 0.0055 MeV)	58%
(E < 0.68 eV)	104%
Short Range Correlation	
(E > 0.0055 MeV)	0.9
(0.68 eV < E < 0.0055 MeV)	0.5
(E < 0.68 eV)	0.5
Flux Group Correlation Range	
(E > 0.0055 MeV)	6
(0.68 eV < E < 0.0055 MeV)	3
(E < 0.68 eV)	2

It should be noted that the uncertainties listed for the upper energy ranges extend down

to the lower energy range. Thus, the 58% group uncertainty in the second range is made up of a 30% uncertainty with a 0.9 short range correlation and a range of 6, and a second part of magnitude 50% with a 0.5 correlation and a range of 3.

3.1.2.8 Other Uncertainties

Additional uncertainties such as sensor positioning, vessel inner radius, vessel thickness, and water density variations are evaluated based on sensitivity studies using the transport code calculations. These additional uncertainties are included after the dosimetry evaluations are completed and are considered an uncertainty in relating the M/C bias factor to positions that are removed from the measurement locations; i.e., the pressure vessel wall.

3.2 Describe and justify any biases applied on the measured data.

CPCo Response

See response to Request 2.3.

3.3 *Provide a summary of which changes to the updated fluence evaluation were plant specific and which changes were industry generic.*

CPCo Response

Palisades Fluence Evaluation Changes		
Change	Generic	Plant Specific
CALCULATIONS		
Cross-section change from ENDF/B-IV to ENDF/B-VI	✔	
Fission spectrum change from ENDF/B-V to ENDF/B-VI	✔	
Vessel IR increase by 0.12″		✔
Vessel thickness increase by 0.29″		✔
Independent cycle specific calculations		✔
Source change from PDQ to SIMULATE-3		✔
Use of core midplane axial peaking factor		✔
Cycle specific T_{inlet} and T_{bypass} modeling		✔
Explicit modeling of ex-vessel dosimetry locations		✔
Increased angular discretization with an S_{16} angular quadrature across the energy spectrum	✔	
MEASUREMENTS		
Dosimetry cross-section change from ENDF/B-V to ENDF/B-VI	✔	
Photo-fission reaction correction to the ^{238}U and ^{237}Np fission monitors	These corrections are generic and are applied to other plant fluence analyses, however, the magnitude of the corrections are plant specific.	
Independent cycle calculations		✔
Use of ^{137}Cs daughter product reaction for fission monitors	This is generically used for fluence analyses at other plants and was implemented at Palisades during this latest evaluation.	

Relative to Table 3.3, the changes in the calculation that are characterized as generic do not impact the best estimate fluence calculation for the Palisades reactor vessel. If

these changes were not implemented, the calculations and the bias factor would be affected in equal and opposite amounts, leaving the best estimate unchanged.

Relative to the changes in the measurement evaluation that are characterized as generic, only the change in dosimetry cross-sections to an ENDF/B-VI based library has the potential to introduce a bias relative to damage trend curves that have been incorporated into regulations governing reactor vessel integrity. Other changes, such as photofission corrections, have already been incorporated into some of the dosimetry evaluations that are included in the present regulations.

The replacement of ENDF/B-V dosimetry reaction cross-sections to an updated set based on the latest ENDF/B-VI evaluations represents a generic change to surveillance capsule and reactor cavity dosimetry analysis procedures that has the potential to introduce a bias relative to prior evaluations. The net impact of this cross-section update on the derived exposure parameters [(Φ(E > 1.0 MeV), Φ(E > 0.1 MeV), and dpa] depends on the number and type of sensors included in individual foil sets. Experience to date has indicated that the net impact is quite small and generally falls within a range of ± 3% relative to analysis performed with the ENDF/B-V dosimetry cross-sections. A net change of this magnitude falls well within and is consistent with the overall uncertainties associated with dosimetry evaluations. This generic change in dosimetry cross-sections should not produce any significant bias relative to previously developed trend curves that are currently incorporated into the regulations governing pressure vessel integrity.

3.4 Provide additional information on the FERRET and SAND-II computer codes used during the fluence evaluation.

CPCo Response

The FERRET least squares adjustment procedure consists of two parts. The first is a pre-processing step using the SAND-II code to convert the calculated input spectrum to the FERRET group structure and to compute multi-group dosimetry cross-sections for use in FERRET. The second stage is the main FERRET program which performs a log-normal least squares adjustment of the measured data, dosimetry cross-sections, and calculated spectrum to produce the most consistent set using a least squares methodology. The code calculates an output covariance matrix relating the uncertainties in all parameters.

The FERRET code uses a log-normal least squares technique to obtain the best solution to the equations:

$$R_i = \sum_g \sigma_{ig} \phi_g$$

where:
- R_i = A set of measured reaction rates for I sensors.
- σ_{ig} = Multi-group reaction cross-sections for I reactions and g neutron groups.
- ϕ_g = Calculated multi-group neutron spectrum for g groups at the measurement location.

The number of simultaneous equations solved in the adjustment procedure is dependent on the number of sensors contained in the dosimeter set.

Input values supplied to FERRET include these parameters and a covariance matrix relating the uncertainties in each of these parameters.

In the FERRET calculations, multi-group neutron fluxes, reaction cross-sections, and covariance data are treated in a 53 group energy scheme. The calculated neutron spectrum is provided in the multi-group structure used in the transport computations; in this case 47 groups. The dosimetry cross-sections are supplied in 620 groups in the SNLRML library.

The pre-processing done in the SAND-II routine first performs an interpolation of the 47 group input spectrum to provide a 620 group spectrum for use in processing the cross-section library. This 620 group, location dependent, spectrum is then used to weight and collapse the dosimetry cross-sections to the 53 group structure used in FERRET. The 620 group neutron spectrum is, likewise, collapsed to the FERRET group structure.

This pre-processing procedure allows the fine (620) group dosimetry cross-sections to be collapsed to a broad group structure using the actual transport calculated spectrum at the dosimeter location. This, in turn removes any additional uncertainty that could be incurred by using a set of multi-group cross-sections that have been pre-collapsed over an arbitrary spectrum.

Following this pre-processing by the SAND-II code, the least squares adjustment algorithms discussed in ASTM Standard E 944-89 are solved in FERRET to produce best estimate exposure results with associated uncertainties at the measurement location.

In the least squares procedure all of the dosimeters are treated simultaneously to produce the final answer. The degree to which any single measurement contributes to the final solution will depend on the uncertainty in the individual measurement and its energy dependent reaction cross-sections as well as on the spectral coverage of the individual sensor. In general, for surveillance capsule and reactor cavity dosimetry evaluations the importance of individual sensors increases as the reaction threshold decreases and as the activation product half-life increases. This follows from an examination of the calculated spectra at the capsule and cavity locations that shows a large neutron population in the 0.5 MeV to 3 MeV range with a rapidly decreasing population at higher energies. Given the expected energy distribution, it follows that the threshold reactions employed in reactor dosimetry programs would have the following order of importance in the overall solution:

^{237}Np
^{238}U
^{54}Fe
^{58}Ni
^{63}Cu
^{46}Ti

However, since the best estimate exposures are based on solving a series of equations involving all of the dosimeters, it is not possible to assign a fractional contribution from individual sensors to the final best estimate result.