UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DOCKET
50-255

April 15, 1997

LICENSEE: Consumers Power Company

FACILITY: Palisades Nuclear Plant

SUBJECT: MEETING WITH CONSUMERS POWER COMPANY TO DISCUSS REACTOR VESSEL FLUENCE EVALUATION

A meeting was held at NRC Headquarters on February 26, 1997, between Consumers Power Company (CPCo) and the NRC to discuss the revised reactor vessel fluence calculation for the Palisades Plant. A list of attendees is provided in Attachment 1. The meeting was requested by CPCo to facilitate technical exchange regarding unresolved issues related to CPCo's reevaluation of reactor pressure vessel (RPV) neutron fluence.

BACKGROUND

The staff concluded in an April 12, 1995, safety evaluation (SE) that the Palisades RPV would reach the pressurized thermal shock (PTS) rule (10 CFR 50.61) screening criteria in late-1999. This conclusion was based on revised material chemistry data provided by CPCo and a previously approved fluence evaluation. CPCo planned to address the issue by annealing the RPV and began submitting portions of its annealing plan in October 1995. Concurrent with pursuit of RPV annealing, CPCo contracted with Westinghouse to evaluate the RPV fluence. CPCo submitted the revised fluence evaluation by letter dated April 4, 1996.

RPV neutron fluence is calculated using a neutron transport code with appropriate neutron cross-section approximations and input data identifying the vessel and internals materials and geometrical configuration. Licensees obtain indirect measurements of RPV neutron fluence to validate these calculations by measuring the activation of dosimeters placed in surveillance capsules that are exposed to the reactor core neutron radiation field during operation. The surveillance capsules are placed in the annular region between the core and the RPV wall and in the reactor cavity. The neutron fluence at a surveillance capsule's location is determined by measuring the dosimeter activation, using the reactor core power history, and an assumed neutron energy spectrum at the capsule's location.

CPCo's fluence reevaluation resulted in a reduction of the RPV fluence by 25 percent. This proposed reduction would result in the Palisades RPV reaching the PTS rule screening criteria in late-2011. This fluence reduction consisted of: (1) an 8 percent reduction resulting from refinements in the physical plant parameter data, (2) a 12 percent reduction due to biasing of the calculated fluence with the results of RPV and reactor cavity dosimetry measurements, and (3) a 5 percent reduction due to adjustment of the neutron energy spectrum.

DF01

The staff issued an interim evaluation on December 20, 1996, that approved the 8 percent reduction due to changes in physical plant parameter data, resulting in the Palisades RPV reaching the PTS rule screening criteria in mid-2003. The staff did not approve the reductions due to the biasing of the calculated RPV fluence or the neutron spectrum adjustment. CPCo requested a meeting with the staff to facilitate technical exchange regarding the basis for the staff's conclusions and unresolved issues.

PRESENTATION/DISCUSSION

The meeting began with a presentation by the staff and the Brookhaven National Laboratory (BNL) contractor. The staff and the BNL contractor summarized the basis for the conclusions in the December 20, 1996, SE and presented additional information from published technical literature in the neutron dosimetry field. The staff's presentation material is provided as Attachment 2. The staff concerns relate to CPCo's bias adjustment of the calculated fluence value and the acceptability of the FERRET code least-squares adjustment of the neutron energy spectrum.

The staff identified that the dosimeter biases varied significantly with neutron energy (high versus low) and dosimeter location (in-vessel versus cavity). The staff identified four discrete mean bias values when dosimeters were grouped together by energy threshold and location, as shown in Figure 4 of Attachment 2. The staff indicated that due to this dependency, selection of different dosimeters could result in a different bias value for the Palisades data. Therefore, the data did not clearly support the calculated bias claimed by Westinghouse and CPCo. The staff also indicated that the results of the biasing calculation for Palisades appeared to be inconsistent with available data for other facilities, including two other US facilities for which the same analysis technique has been used and published foreign data which reported that bias values are independent of reactor type, dosimeter type, and dosimeter location.

With respect to the FERRET least-squares adjustment code, the staff stated that the code has not been reviewed nor approved by the NRC staff, and little information was provided regarding how it accomplishes the neutron spectral adjustment. Without this information, it cannot be established by the staff that the code is applicable to Palisades.

The BNL contractor discussed the differences between the Palisades dosimetry bias values and bias values derived for the Kewaunee and Ginna facilities using the same methodology. The iron and nickel dosimeter values for the other facilities exhibited significant variation in bias mean value and standard deviation, and the derived bias values for the facilities were in relatively good agreement with each other considering uncertainties. The Palisades dosimetry bias values showed relatively little variation in mean value and standard deviation, and the derived bias value did not correlate well with the values derived for the other facilities.

The staff and the BNL contractor summarized the presentation by stating that there is a high confidence level in the calculational methodology and they were, therefore, surprised to see such a significant variation in the overall bias value derived for Palisades when compared to the values derived for the other facilities. Given the high confidence in the calculational methodology, the staff stated that CPCo needs to provide compelling justification for the significant variance in the measured fluence values relative to the calculated values and to the results obtained for other facilities.

Following the staff presentation, Westinghouse, CPCo's contractor, presented a general discussion which it believes supports the calculated bias value. The presentation discussion notes are provided in Attachment 3. Westinghouse stated that its studies identified that a bias exists in the calculational model which results in higher calculated RPV fluence for Palisades relative to other facilities. When this assertion was questioned by the staff, Westinghouse offered to submit additional data to support the contention that a model bias exists.

The NRC reviewers reiterated their concern regarding the significant difference between the bias values derived for Palisades relative to other facilities where the same technique has been applied. Westinghouse and CPCo asserted that the results are all within one standard deviation of the calculated values and should, therefore, be acceptable. The staff questioned the validity of the statistical treatment of the dosimeter data used in the determination of the overall bias value and reiterated the potential effect of a different statistical treatment of the data. CPCo and Westinghouse indicated that they could provide a technical paper to the staff that addresses selection of appropriate dosimeters for inclusion in the biasing calculation.

Westinghouse indicated that the percentage of the neutron spectrum sensed by dosimeters varies with location (i.e., in-vessel versus cavity). They indicated that they had bias data for several plants for multiple dosimeter types that show the mean bias for all of these plants is 0.94 with an 8-percent uncertainty and offered to make these data available to the staff. The BNL contractor stated that if some problem could be shown to exist in the neutron capture cross-section data used in the calculations, this might explain the bias. However, the source of the bias is not clear and could in fact be in the dosimeter measurements.

The NRC reviewers reiterated the position that, based on extensive research and application, the calculational methodology provides a high confidence estimate of RPV fluence exposure. In addition, the reviewers pointed out that as far as is known the use of biasing to effect a reduction in the calculated fluence has not been accepted at any domestic or foreign facility. CPCo was cautioned that the high confidence in the calculational methodology requires that a detailed explanation must be provided to support application of the biasing technique to reduce the calculated RPV fluence exposure. With respect to the FERRET spectral adjustment, the reviewers stated that CPCo needs to provide a physical explanation for any changes made from the spectrum used in the calculation.

SUMMARY

CPCo stated that the meeting had been beneficial in developing a greater mutual understanding of the issues and indicated that it would consider submittal of additional information to support further staff evaluation of the proposed fluence reduction. If there are any questions regarding the information presented in this meeting summary, please contact Robert Schaaf at (301) 415-1312.

ORIGINAL SIGNED BY

Robert G. Schaaf, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-255

Attachments: As stated (3)

cc w/att: See next page

HARD COPY
Docket File PUBLIC PD31 Reading OGC ACRS

E-MAIL w/att 1 only:
SCollins/FMiraglia ATHadani RZimmerman JRoe
DRoss (SAM) BMcCabe BBurgess, RIII
EAdensam (EGA1)

DOCUMENT NAME: G:\WPDOCS\PALISADE\PA970226.MTS
To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	PM:PD31	E	LA:PD31	C	D:PD31
NAME	RSchaaf:db		CJamerson		JHannon
DATE	4/11/97		4/14/97		4/15/97

OFFICIAL RECORD COPY

cc:

Mr. Thomas J. Palmisano
Site Vice President
Palisades Plant
27780 Blue Star Memorial Highway
Covert, Michigan 49043

Mr. Robert A. Fenech, Sr Vice Pres
Nuclear, Fossil, and Hydro Operations
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

M. I. Miller, Esquire
Sidley & Austin
54th Floor
One First National Plaza
Chicago, Illinois 60603

Mr. Thomas A. McNish
Vice President & Secretary
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

Judd L. Bacon, Esquire
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, Illinois 60532-4351

Jerry Sarno
Township Supervisor
Covert Township
36197 M-140 Highway
Covert, Michigan 49043

Office of the Governor
Room 1 - Capitol Building
Lansing, Michigan 48913

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
Palisades Plant
27782 Blue Star Memorial Highway
Covert, Michigan 49043

Drinking Water and Radiological
Protection Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr Blvd
P. O. Box 30630 CPH Mailroom
Lansing, Michigan 48909-8130

Gerald Charnoff, Esquire
Shaw, Pittman, Potts and Trowbridge
2300 N Street, N. W.
Washington DC 20037

Michigan Department of Attorney
General
Special Litigation Division
630 Law Building
P.O. Box 30212
Lansing, Michigan 48909

Mr. Dennis Harrison
U.S. Department of Energy
NE 451
Washington, DC 20585

Mr. Thomas C. Bordine
Manager, Licensing
Palisades Plant
27780 Blue Star Memorial Highway
Covert, MI 49043

MEETING ATTENDANCE

MEETING WITH CONSUMERS POWER COMPANY TO DISCUSS LICENSING ISSUES

May 16, 1996

NAME	AFFILIATION
Robert Schaaf	NRC
Lambros Lois	NRC
Alan Levin	NRC
John Carew	BNL
Dick Smedley	CPCo
Ross Snuggerud	CPCo
John Kneeland	CPCo
Stan Anderson	Westinghouse
John Perock	Westinghouse
John Warren	DOE
Bob Borsum	B&WOG
Althea Wyche	Bechtel
Lynn Conner	STS

ATTACHMENT 1

ENERGY DEPENDENCE OF BIASES. NOT ALL DOSIMETERS POINT IN THE SAME DIRECTION. THUS, FOR A DIFFERENT SET OF DOSIMETERS THE BIAS WOULD BE DIFFERENT

THE SAME METHOD YIELDS DIFFERENT RESULTS FOR DIFFERENT PLANTS. SPECIFICALLY: KEWAUNEE AND GINNA (for $E > 1.0$ MeV) $M/C = 1.13$ and 1.03 RESPECTIVELY

IT IS ASSUMED THAT THE MEASUREMENTS ARE CORRECT WITHOUT ANY JUSTIFICATION FOR THE APPARENT ANOMALIES IN THE MEASUREMENT DATA.

E. POLKE IN A RECENT PAPER COVERING DOSIMETRY OF OVER 20 GERMAN REACTORS (PWRs AND BWRs) REPORTED THAT THE M/C (about 4-8%) IS INDEPENDENT OF REACTOR TYPE, DOSIMETER LOCATION AND TYPE OF DOSIMETER.

THE FERRET CODE HAS NOT BEEN REVIEWED, THUS, WE DO NOT UNDERSTAND WHAT IT DOES AND HOW IT DOES IT. IN PARTICULAR THE RELEVANCE OF THE BUILT IN DATA TO PALISADES OR ANY OTHER PLANT.

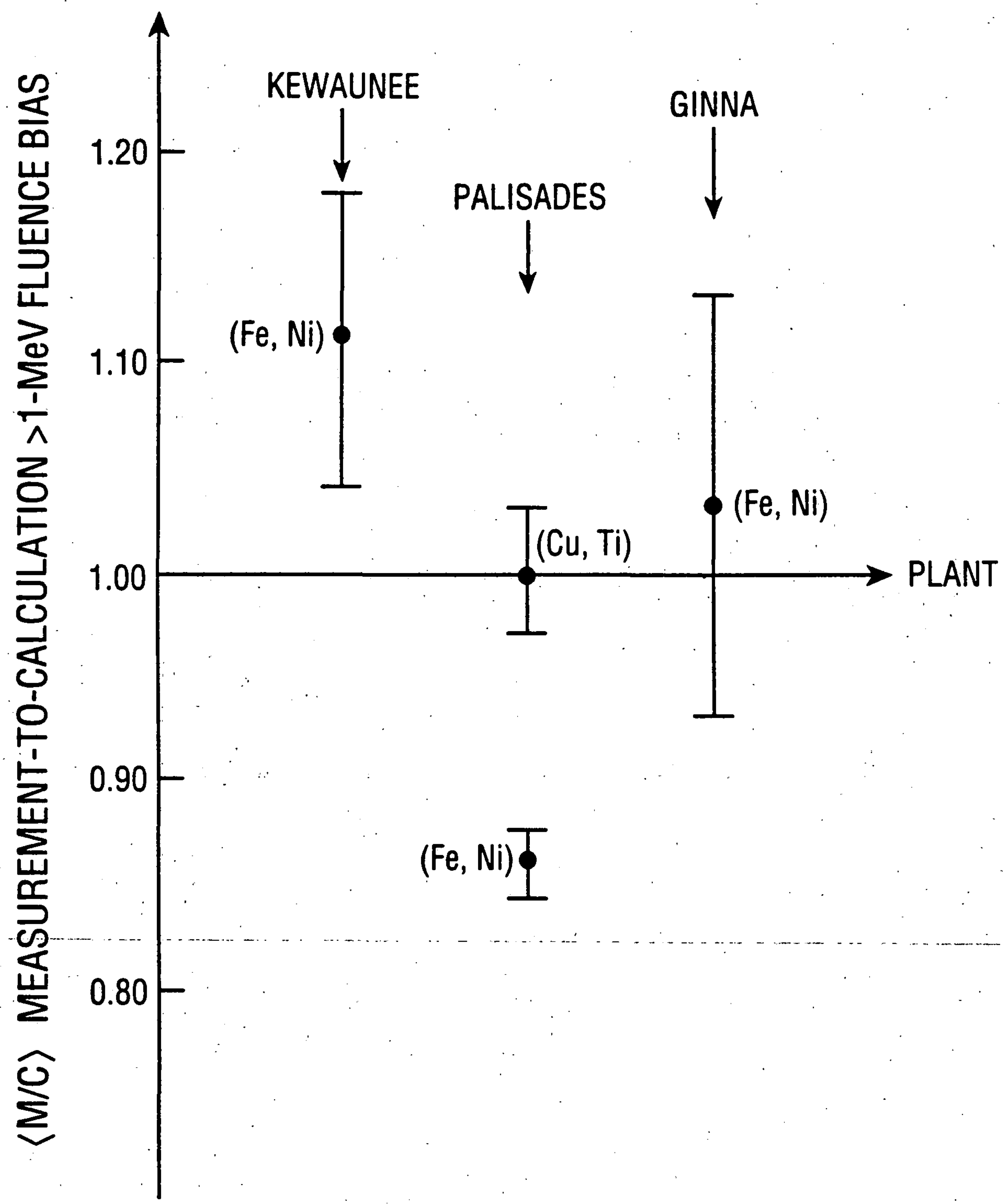
	Dosimeter Threshold Energy (MeV)	
	>4.0	<4.0
In-Vessel	1.00±0.03	0.86±0.02
Cavity	0.91±0.03	0.85±0.07

M/C Data Versus Location and Dosimeter Threshold Energy

Figure 4 Location and Energy Dependence of M/C Bias

Comparison of <M/C> Measurement-to-Calculation Bias Based on Fe-54 and Ni-58 In-Vessel Capsules for Palisades, Ginna and Kewaunee

	Capsule	M/C	<M/C>	σ	Proposed Ferret M/C
Kewaunee					
Fe-54, Ni-58	V	1.16			
	R	1.15			
	P	1.15			
	S	.99			
			1.113	± 0.07	1.13
Ginna					
8 Fe, 1 Ni	V	.86			
6 Fe, 1 Ni	R	1.09			
6 Fe, 1 Ni	T	1.12			
1 Fe, 1 Ni	S	1.05			
			1.03	± 0.10	1.03
Palisades					
1 Fe, 1 Ni	A-240	.88			
1 Fe, 1 Ni	W-290	.87			
	W-290-9	.84			
	W-110	.86			
			0.863	± 0.015	0.83



Carew 1. 2/97 iudv

TOPICS FOR DISCUSSION

1 - General Methodology Considerations.

2 - Measurement Range of Multiple Foil Sensor Set.

3 - Accuracy of Individual Measurements.

4 - Combination of Calculation and Measurements to arrive at a "Best Estimate" projection.

ATTACHMENT 3

GENERAL METHODOLOGY CONSIDERATIONS

1 - Based on 10CFR50.61, evaluation of material embrittlement based on the "Best Estimate" of the neutron fluence (E > 1.0 MeV) is required. Conservative evaluations of the neutron exposure are not required.

2 - The objective of the methodology is to provide spatial distributions of "Best Estimate" Φ(E > 1.0 MeV) with associated uncertainties throughout the beltline region of the pressure vessel.

3 - The methodology should be generally applicable to all reactors. That is, the calculations and measurements should not be treated differently to arrive at a "conservative Best Estimate" for individual reactors.

4 - In developing an overall fluence methodology, the pros and cons of using

1. Calculation alone
2. Dosimetry alone
3. Calculation combined with dosimetry

to produce the "Best Estimate" result must be addressed.

CONCLUSIONS FROM METHODOLOGY DEVELOPMENT

1 - Within the context of an allowable 20% uncertainty in final fluence values, either calculation alone or dosimetry alone may result in an acceptable fluence value, but not the "Best Estimate" value.

  a - Calculation alone is deficient in that biases, either generic or plant specific, cannot be removed and must remain a part of the overall uncertainty.

  b - Dosimetry alone is deficient in that measurements cannot be made at locations of interest resulting in increased uncertainties in fluence projections.

  c - A combination of calculation and measurement affords the best opportunity to remove biases and increase the precision associated with the fluence projections.

2 - To implement a methodology combining calculation and measurement the following must be determined:

  a - Uncertainty associated with the plant specific calculation.

  b - Accuracy and uncertainty associated with the measurement process.

  c - The best method of combining calculation and measurement to arrive at a "Best Estimate" fluence projection with associated uncertainties.

    1. Equal weighting of measurement data.

    2. Spectral coverage weighting of measurement data.

    3. Least Squares weighting of calcs/meas/xsec.

Measurement Range of Multiple Foil Sensor Sets

The typical multiple foil sensor sets included in surveillance capsule and reactor cavity dosimetry packages include detectors employing the following threshold reactions.

Cu-63 (n,α) Co-60
Ti-46 (n,p) Sc-46
Fe-54 (n,p) Mn-54
Ni-58 (n,p) Co-58
U-238 (n,f) F.P.
Np-237 (n,f) F.P.

The use of passive neutron sensors such as those listed above does not yield a direct measure of the energy dependent neutron flux. Rather, the determination of the reaction rate in the individual sensors provides 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. The measured reaction rates observed in the foil materials are related to the energy dependent neutron flux by the following set of equations:

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

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

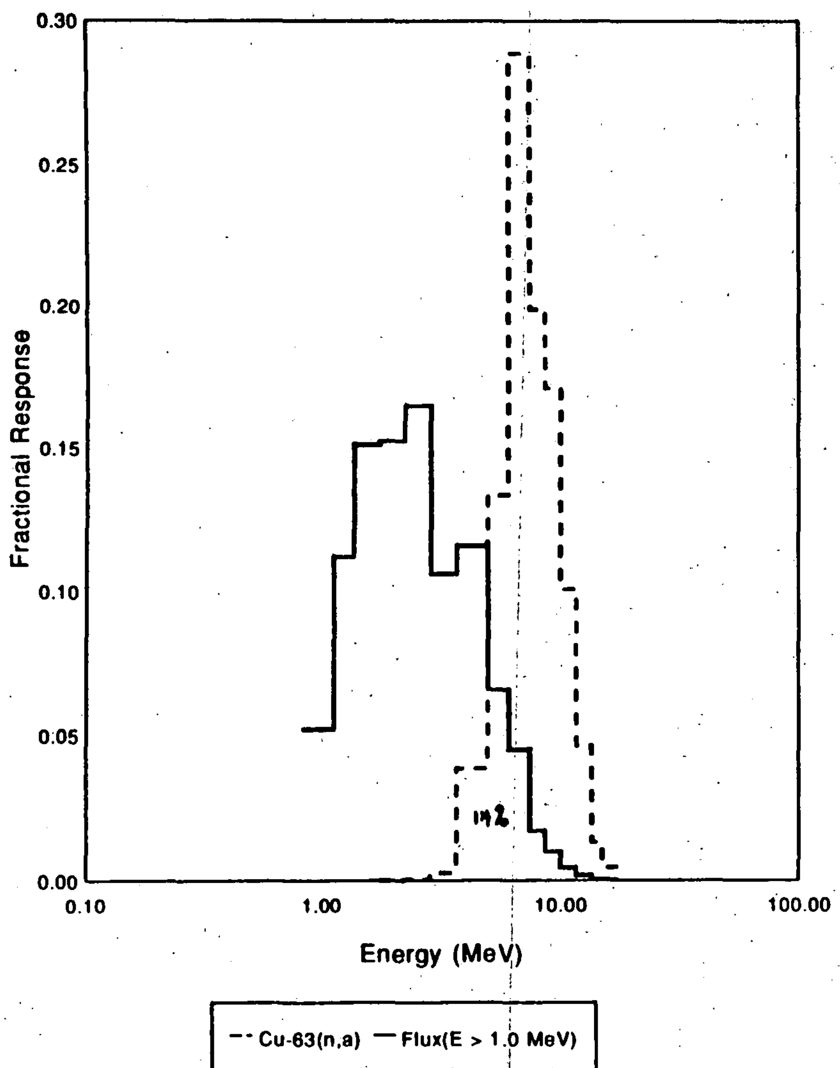
Since the energy dependent reaction cross-sections, σ_{ig}, exhibit different energy thresholds and provide differing energy dependent response to the neutron field, the six target materials listed above provide a sampling of different portions of the energy spectrum; i.e., they do not all measure the same thing.

The following 12 figures provide an illustration of the response range of the individual foils at the Palisades 290° In-Vessel location and the 16° Ex-Vessel position. These figures clearly demonstrate how the lower threshold detectors sample larger fractions of the neutron population above 1.0 MeV.

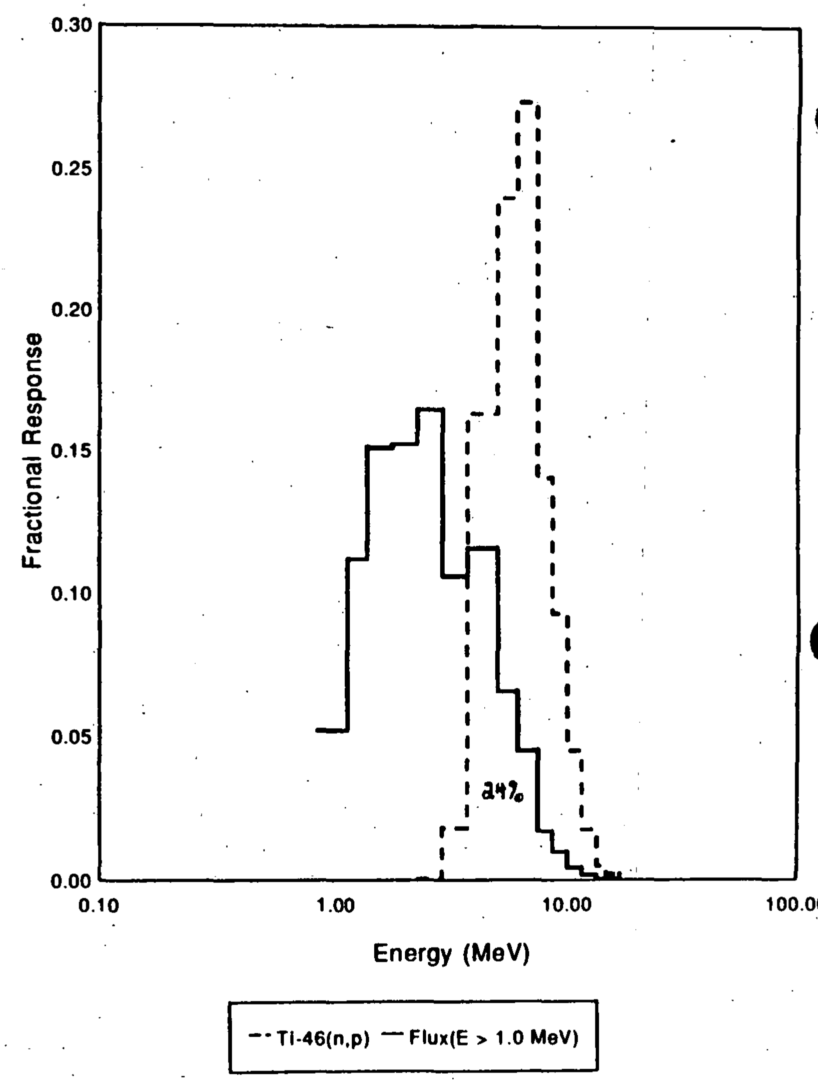
In these figures, the fractional energy response of individual threshold sensors are plotted along with the neutron energy spectrum above 1.0 MeV. From these comparisons it is noted that as the threshold of the sensor drops, the foil samples a larger percentage of the neutron spectrum above 1.0 MeV.

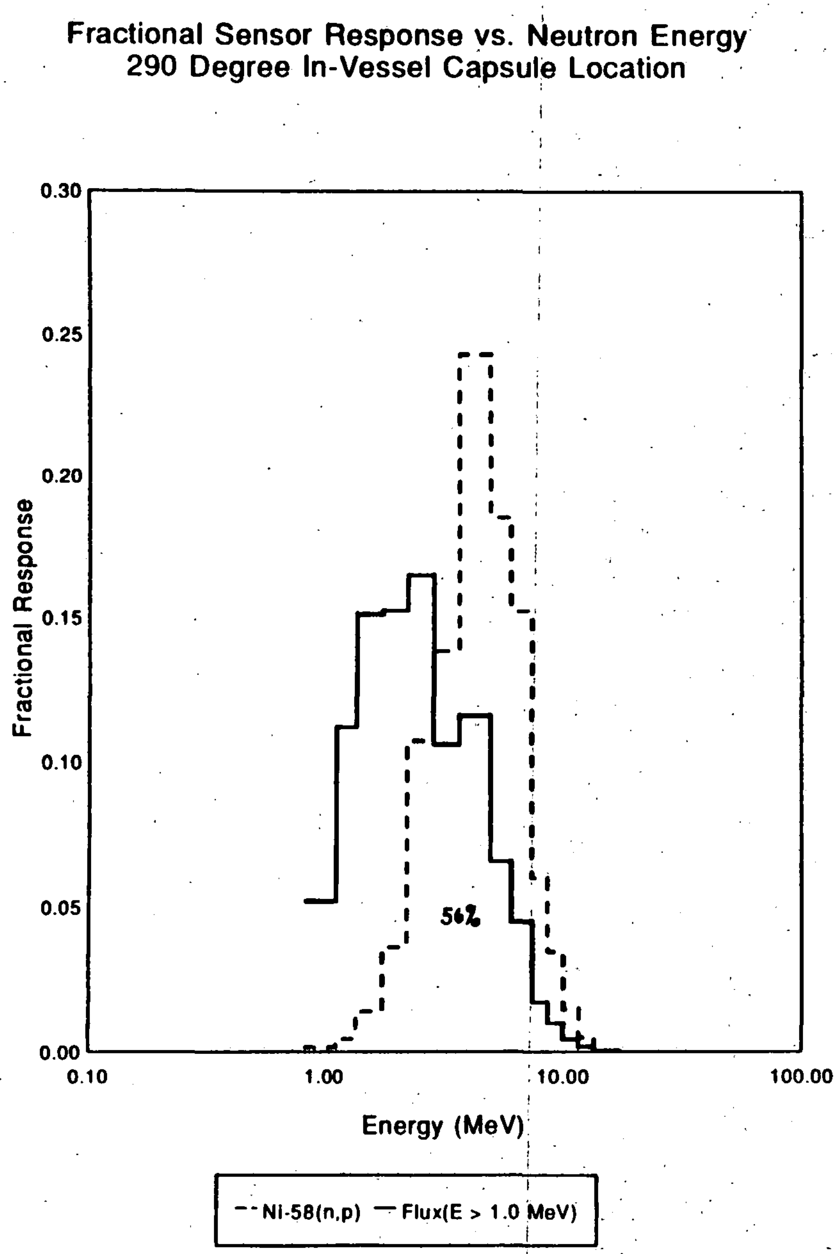
It is the fact that these foils sample different portions of the neutron spectrum that produces the apparent statistical inconsistency observed in Table 7.2-1 of the fluence WCAP-14557, Rev 1. The observed M/C ratios indicate that the calculated neutron distribution exhibits not only an overall bias, but, also, a differing neutron energy distribution.

Fractional Sensor Response vs. Neutron Energy
290 Degree In-Vessel Capsule Location

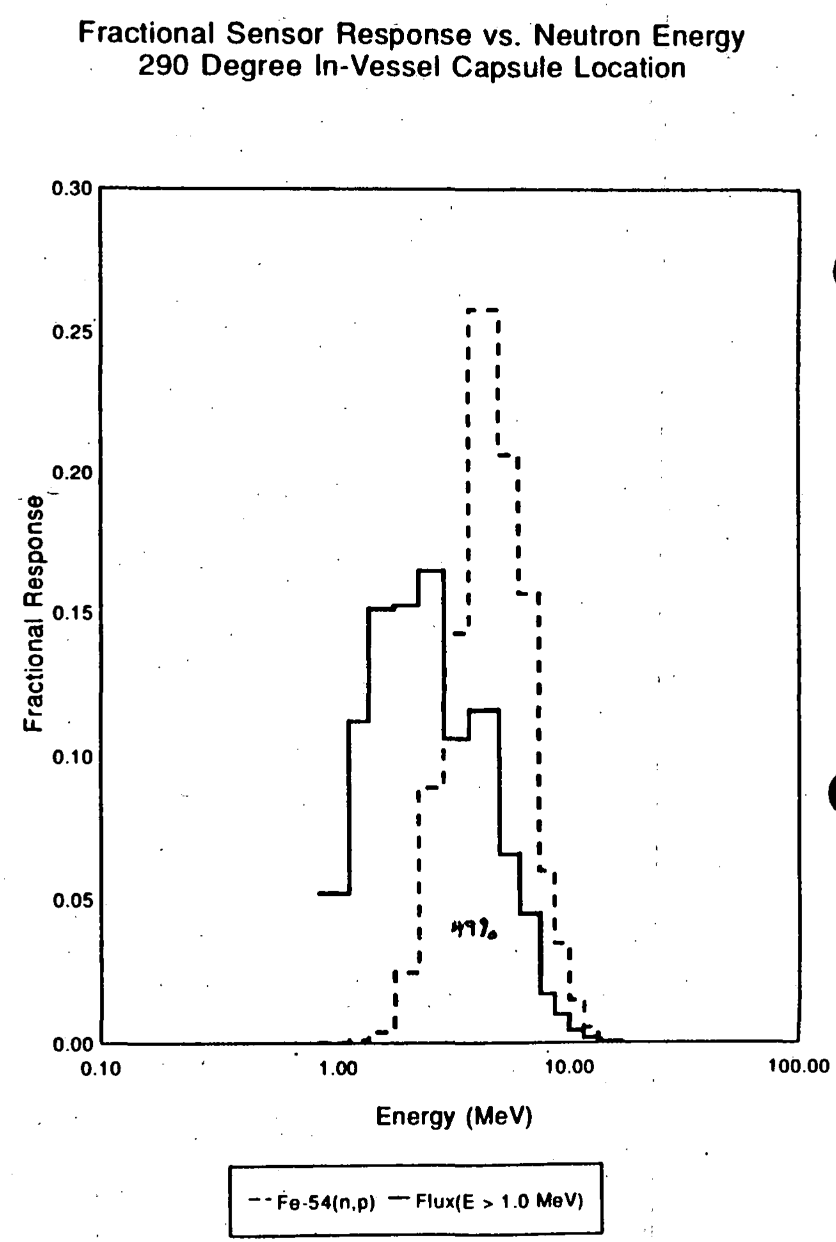


Fractional Sensor Response vs. Neutron Energy
290 Degree In-Vessel Capsule Location

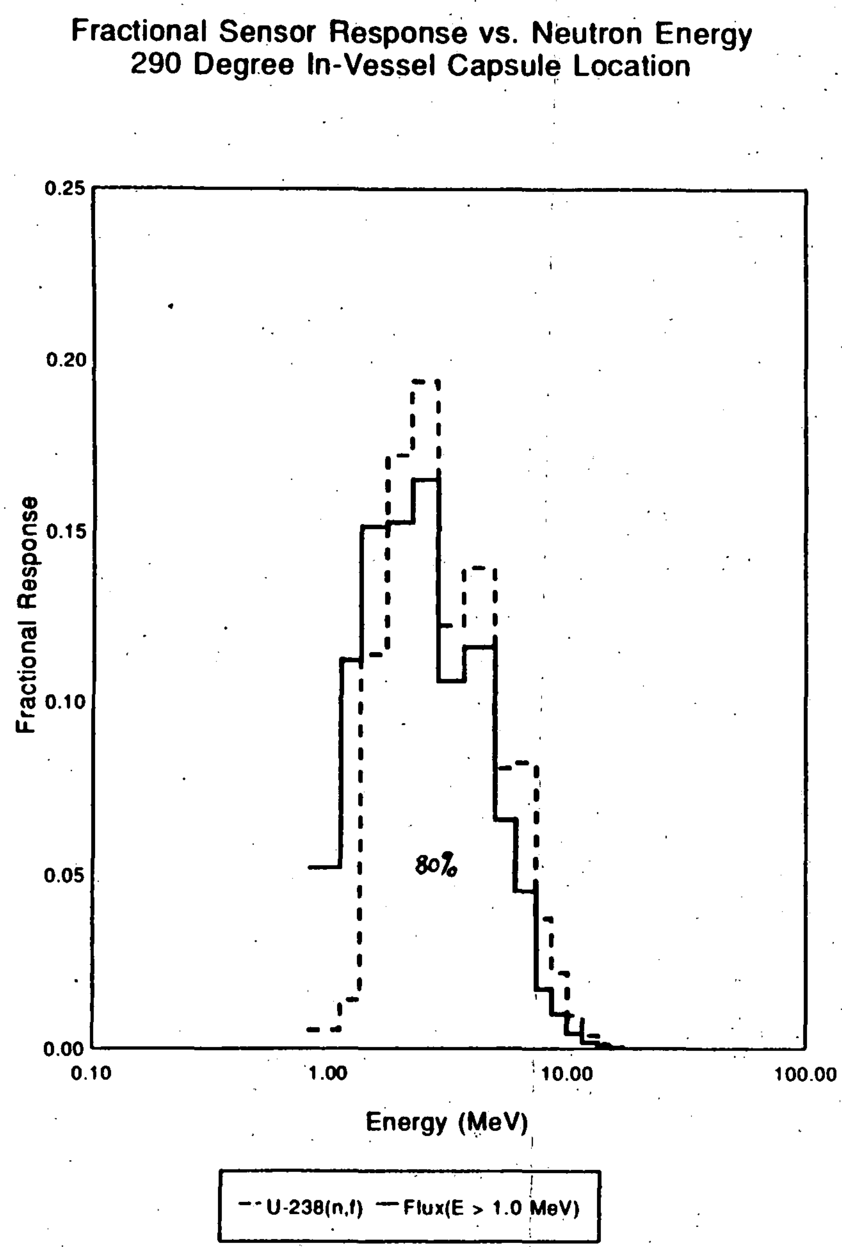


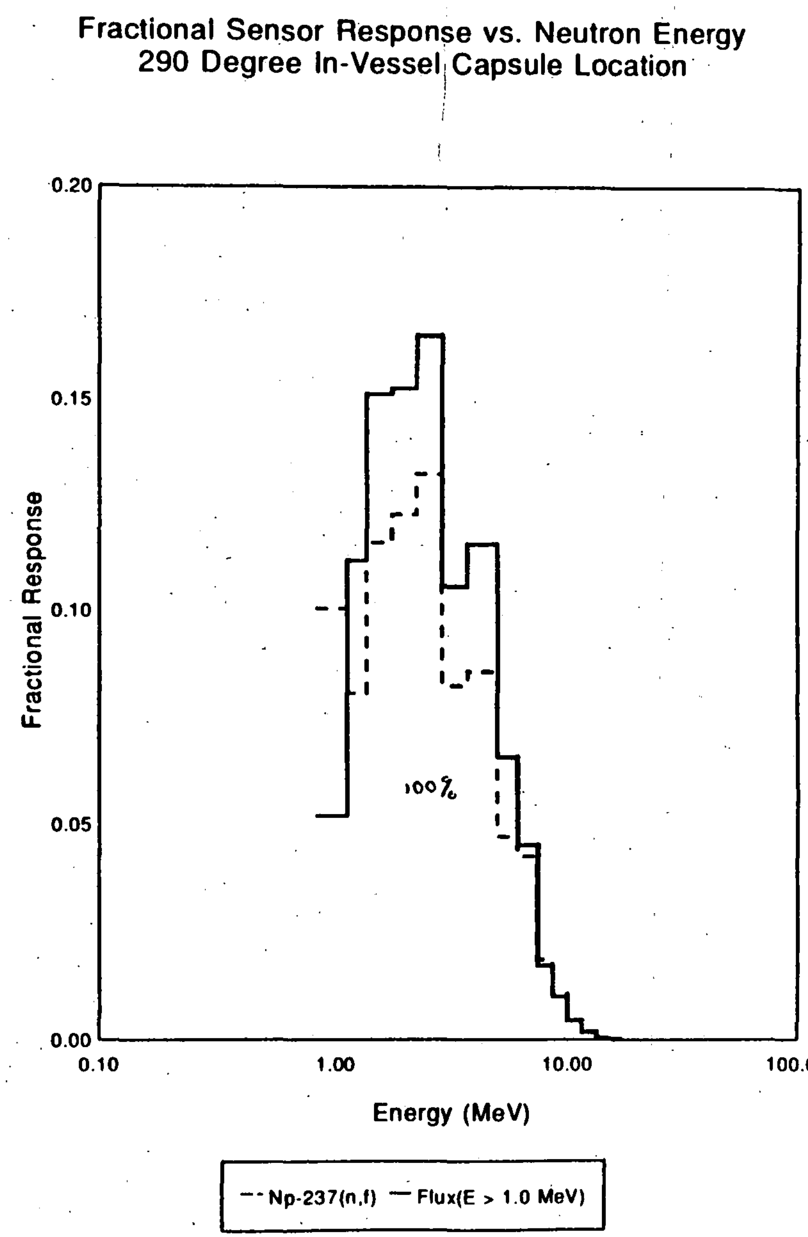


Fractional Sensor Response vs. Neutron Energy
290 Degree In-Vessel Capsule Location
0.30
0.25
0.20
0.15
0.10
0.05
0.00
Fractional Response
56%
0.10
1.00
10.00
100.00
Energy (MeV)
Ni-58(n,p)
Flux(E > 1.0 MeV)

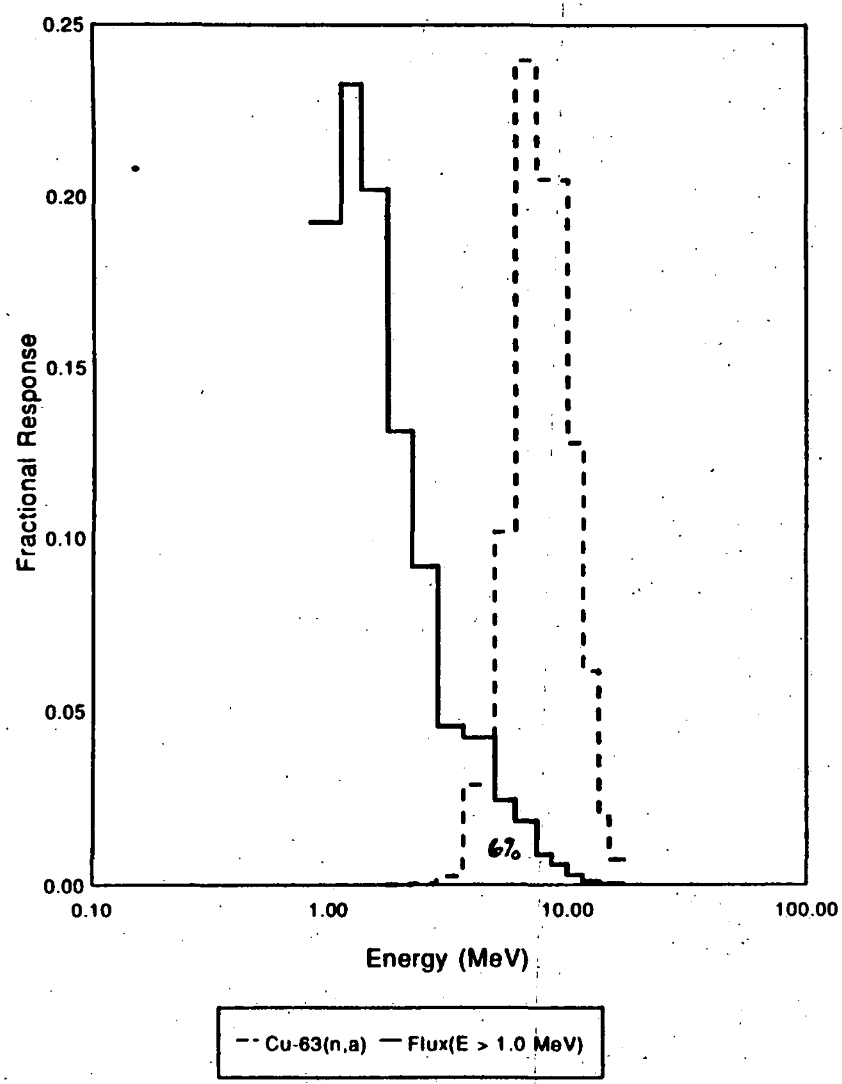


Fractional Sensor Response vs. Neutron Energy
290 Degree In-Vessel Capsule Location
0.30
0.25
0.20
0.15
0.10
0.05
0.00
Fractional Response
49%
0.10
1.00
10.00
100.00
Energy (MeV)
Fe-54(n,p)
Flux(E > 1.0 MeV)

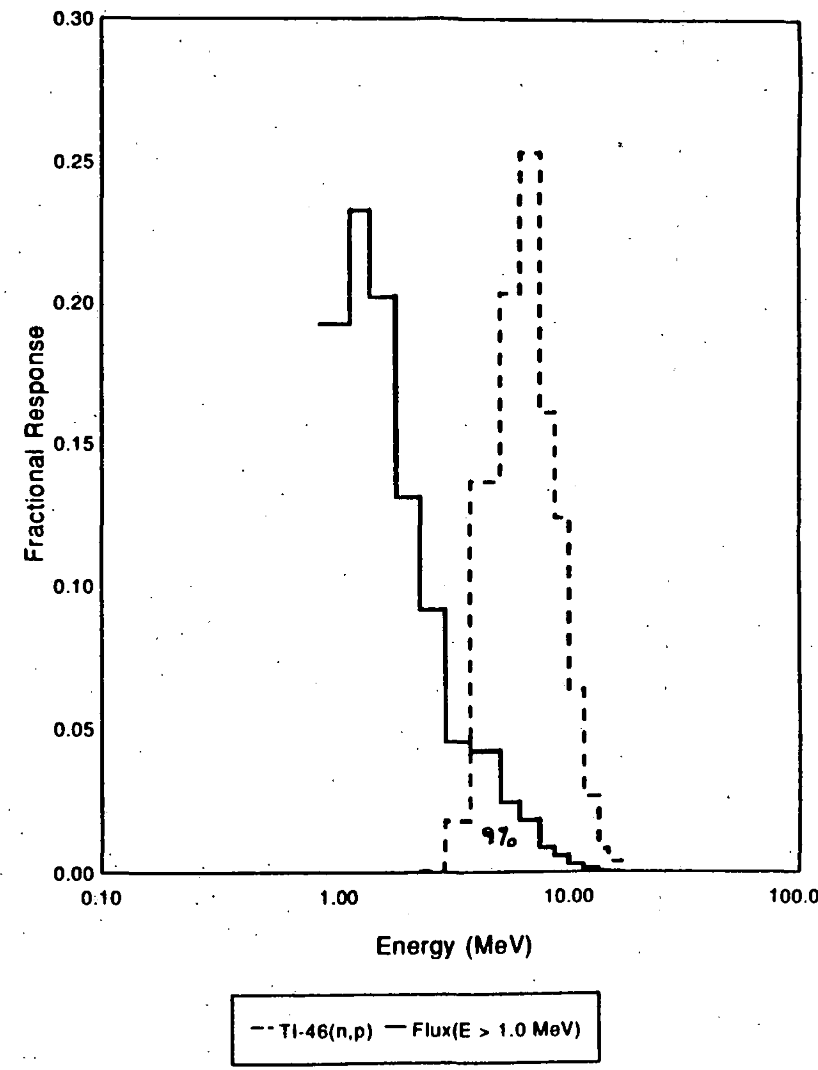

Fractional Sensor Response vs. Neutron Energy
290 Degree In-Vessel Capsule Location
Fractional Response
0.25
0.20
0.15
0.10
0.05
0.00
0.10
1.00
10.00
100.00
Energy (MeV)
80%
U-238(n,f)
Flux(E > 1.0 MeV)


Fractional Sensor Response vs. Neutron Energy
290 Degree In-Vessel Capsule Location
Fractional Response
0.20
0.15
0.10
0.05
0.00
0.10
1.00
10.00
100.00
Energy (MeV)
100%
Np-237(n,f)
Flux(E > 1.0 MeV)

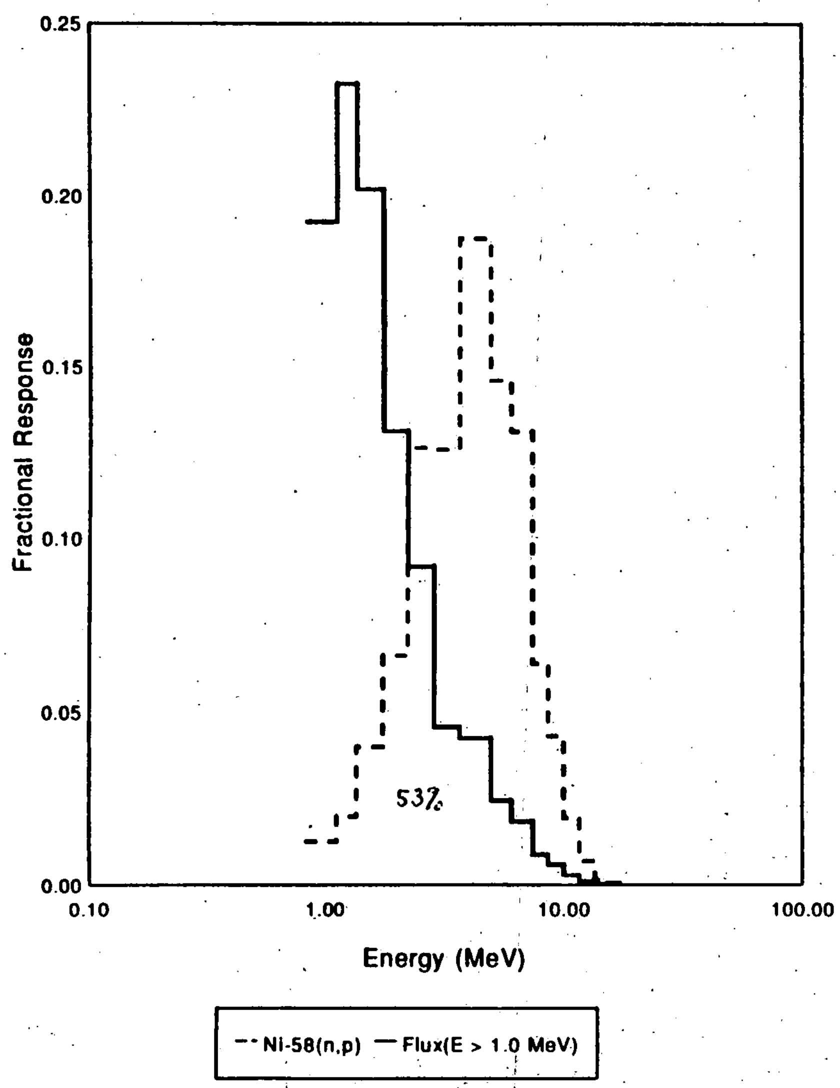
Fractional Sensor Response vs. Neutron Energy
16 Degree Ex-Vessel Capsule Location



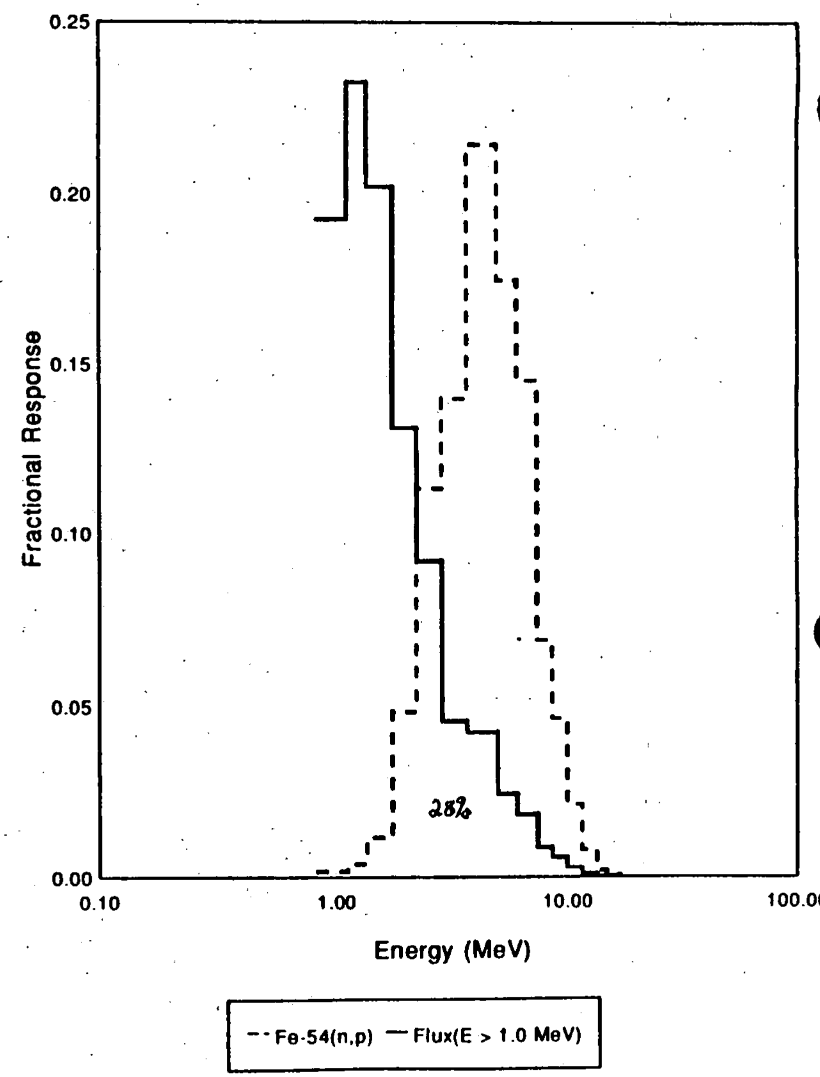
Fractional Sensor Response vs. Neutron Energy
16 Degree Ex-Vessel Capsule Location



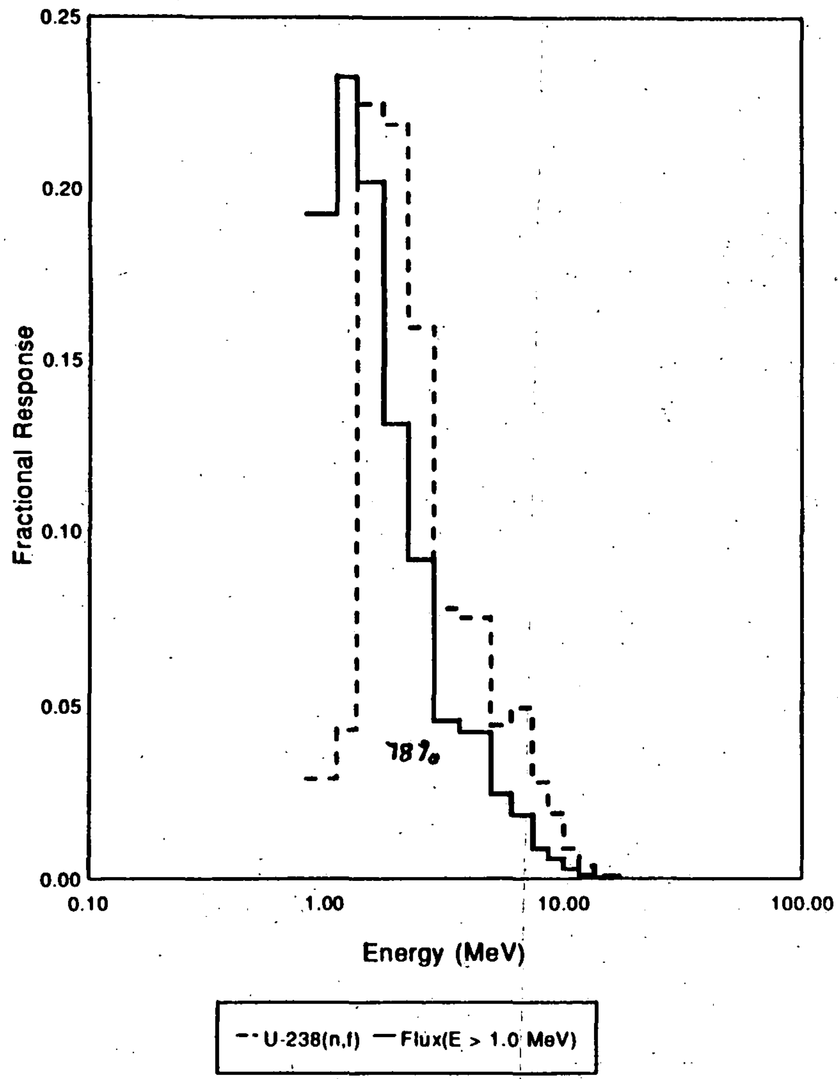
Fractional Sensor Response vs. Neutron Energy
16 Degree Ex-Vessel Capsule Location



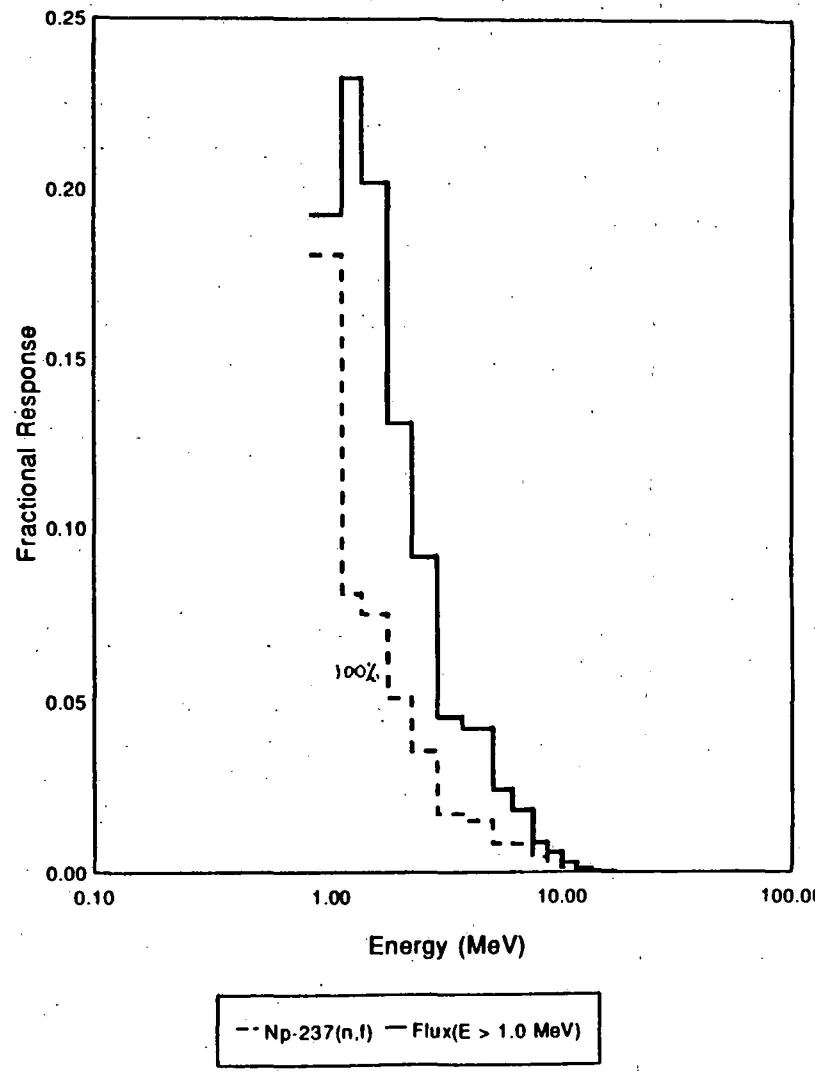
Fractional Sensor Response vs. Neutron Energy
16 Degree Ex-Vessel Capsule Location



Fractional Sensor Response vs. Neutron Energy
16 Degree Ex-Vessel Capsule Location



Fractional Sensor Response vs. Neutron Energy
16 Degree Ex-Vessel Capsule Location



Accuracy of Reaction Rate Measurements

The accuracy of the reaction rate measurements obtained from surveillance capsule and reactor cavity irradiations is assured by utilizing laboratory procedures that conform to ASTM National Consensus Standards for each of the sensors comprising the multiple foil dosimetry sets. In particular, the following standards are applied for the reactions of interest.

Cu-63(n,α)Co-60	ASTM-E-523
Ti-46(n,p)Sc-46	ASTM-E-526
Fe-54(n,p)Mn-54	ASTM-E-263
Ni-58(n,p)Co-58	ASTM-E-264
U-238(n,f)Cs-137	ASTM-E-704
Np-237(n,f)Cs-137	ASTM-E-705
Co-59(n,γ)Co-60	ASTM-E-481

In all cases, the latest available versions of the applicable standard are used in the dosimetry evaluations.

From these standards, it is noted that the expected uncertainties in the measured disintegration rates can be summarized as follows:

Reaction	Precision	Bias
Cu-63(n,α)Co-60	1%	3%
Ti-46(n,p)Sc-46	1%	3%
Fe-54(n,p)Mn-54	1%	3%
Ni-58(n,p)Co-58	1%	3%
U-238(n,f)Cs-137	1%	5%
Np-237(n,f)Cs-137	1%	5%
Co-59(n,γ)Co-60	1%	5%

These uncertainties include the impacts of sample weighing, detector calibration, geometry source/detector geometry corrections, and product nuclide branching ratios.

In determining reaction rates from the measured specific activities, the following additional uncertainties are incurred.

Reaction	Fission Yield	Product Half-life	Competing Reactions
Cu-63(n,α)Co-60		0.02%	
Ti-46(n,p)Sc-46		0.2%	
Fe-54(n,p)Mn-54		0.2%	
Ni-58(n,p)Co-58		0.2%	
U-238(n,f)Cs-137	1%	0.1%	4%
Np-237(n,f)Cs-137	2%	0.1%	1%
Co-59(n,γ)Co-60		0.02%	

After combining all of these uncertainty components, the sensor reaction rates derived from the counting and data evaluation procedures used for surveillance capsule and cavity dosimetry irradiations typically result in the following net uncertainties associated with the data:

Reaction	Reaction Rate Uncertainty
Cu-63(n,α)Co-60	5%
Ti-46(n,p)Sc-46	5%
Fe-54(n,p)Mn-54	5%
Ni-58(n,p)Co-58	5%
U-238(n,f)Cs-137	10%
Np-237(n,f)Cs-137	10%
Co-59(n,γ)Co-60	5%

These uncertainty values are quoted at the 1σ level.

In addition to the use of ASTM National Consensus Standards in the evaluation of sensor reaction rates, over the course of the last 17 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). In all, the following five separate counting comparisons were conducted between 1980 and 1997.

1980 Round robin counting of foil sets irradiated at the Thermal Shield Back (TSB) and Pressure Vessel Face (PVF) positions of the PCA simulator.

1981 Round robin counting of additional foil sets included in the first metallurgical simulated surveillance capsule also irradiated in the PCA benchmark mockup.

These two counting exercises involved direct comparisons with measurements obtained by The Hanford Engineering Development Laboratory (HEDL). At the time of these irradiations HEDL was a prime contractor providing measurement services for the PCA benchmark and was cross-calibrated with NIST and the MOL Laboratory in Belgium.

1985 Counting and evaluation of Ti-46(n,p), Fe-54(n,p), and Ni-58(n,p) certified fluence standards supplied by NIST.

Comparisons with fluence standards involve the determination of the reaction rate of each foil, but also of the spectrum averaged cross-section in the NIST U-235 irradiation facility. Thus, the comparisons with the certified fluence test both the measurement process and the energy dependent reaction cross-section used by the vendor.

1992 Counting of NIST foils irradiated in a reactor cavity dosimetry experiment at the Trojan reactor.

This exercise involved duplicate counting of a subset of irradiated foils by both Westinghouse and NIST to assure adequate cross-calibration of the laboratories so that data could be confidently mixed in the overall fluence evaluations performed by NIST and ORNL.

1996 Irradiation of a set of foils used in Westinghouse cavity dosimetry irradiations at the Materials Dosimetry Reference Facility (MDRF) and subsequent comparison with certified results provided by NIST.

Results of the first four intercomparisons are summarized as follows:

Reaction	[West]/[HEDL] 1980	1981	[WEST]/[NIST] 1985	1992	Average
Cu-63(n,α)Co-60	1.041	1.018		0.969	1.009
Ti-46(n,p)Sc-46	1.036		1.012	1.030	1.026
Fe-54(n,p)Mn-54	1.006	1.008	1.011	1.056	1.020
Ni-58(n,p)Co-58	1.006	0.990	1.028	1.029	1.013
U-238(n,f)Cs-137	1.014	1.014			1.014
Np-237(n,f)Cs-137	1.006	1.017			1.012
Co-59(n,γ)Co-60	1.017	1.017			1.017

Final results of the comparisons from the 1996 irradiations are still pending, but preliminary evaluations support the data comparisons in the preceding tabulation.

The comparisons shown in the preceding table demonstrate that the procedures used by Westinghouse in the determination of reaction rates from both in-vessel surveillance capsule irradiations and ex-vessel cavity irradiations have produced accurate and stable results over a period spanning the last 17 years. The cross-comparisons with HEDL and NIST support the typical uncertainties of 5% for non-fission reactions and 10% for fission reactions that are assigned to Westinghouse reaction rate results.

Further, the certified fluence comparisons performed in 1985, support not only the radiometric counting capability of the Westinghouse Analytical Services Laboratory, but also, demonstrate the accuracy of the Ti-46(n,p), Fe-54(n,p), and Ni-58(n,p) energy dependent reaction cross-sections that are used in the dosimetry evaluations.

Accuracy of Reaction Rate Cross-Sections

The reaction rate cross-sections used in the neutron fluence evaluations were taken from the RSIC DATA LIBRARY COLLECTION DLC-178, "SNLRML Recommended Dosimetry Cross Section Compendium," July, 1994. This data library provides reaction cross-sections and associated uncertainties for 66 dosimetry sensors in common use. These cross-sections were drawn from the most recent cross-section evaluations and they have been compared with each other and evaluated with respect to their accuracy and consistency for spectrum unfolding calculations. The library has been empirically tested for use in fission spectra determination as well as in the fluence and energy characterization of 14 MeV neutron sources.

For sensors of interest to Light Water Reactor (LWR) dosimetry applications, the following uncertainties in the fission spectrum averaged cross-sections were provided in DLC-178:

Reaction	Uncertainty
Cu-63(n,α)Co-60	4.08-4.16%
Ti-46(n,p)Sc-46	4.51-4.87%
Fe-54(n,p)Mn-54	3.05-3.11%
Ni-58(n,p)Co-58	4.49-4.56%
U-238(n,f)Cs-137	0.54-0.64%
Np-237(n,f)Cs-137	10.32-10.97%
Co-59(n,γ)Co-60	0.79-3.59%

Detailed discussions of the contents of the SNLRML library along with the evaluation process for each of the sensors is provided in DLC-178.

The data provided in SNLRML coupled with the certified fluence comparisons discussed earlier demonstrate that reaction rates as well as reaction cross-sections used in the neutron fluence evaluations provide adequate accuracy.

E(95%) [MeV]

Detector	In-vessel 20 Deg	In-vessel 30 Deg	Ex-vessel 6 Deg	Ex-vessel 16 Deg	Ex-vessel 24 Deg	Ex-vessel 26 Deg	Ex-vessel 36 Deg	Ex-vessel 39 Deg
Cu-63(n,a)	5.04E+00	4.72E+00	5.18E+00	5.17E+00	5.17E+00	5.18E+00	5.18E+00	5.18E+00
Ti-46(n,p)	3.93E+00	3.70E+00	3.99E+00	3.98E+00	3.98E+00	3.99E+00	3.98E+00	3.98E+00
Fe-54(n,p)	2.38E+00	2.26E+00	2.08E+00	2.08E+00	2.08E+00	2.08E+00	2.08E+00	2.07E+00
Ni-58(n,p)	2.13E+00	1.87E+00	1.43E+00	1.43E+00	1.43E+00	1.43E+00	1.42E+00	1.42E+00
U-238(n,f)	1.45E+00	1.40E+00	1.12E+00	1.13E+00	1.12E+00	1.13E+00	1.12E+00	1.11E+00
Np-237(n,f)	5.22E-01	4.92E-01	6.74E-02	1.29E-01	6.74E-02	8.86E-02	4.09E-02	2.37E-02

Fractional Flux (E>1.0 MeV) above E(95%)

Detector	In-vessel 20 Deg	In-vessel 30 Deg	Ex-vessel 6 Deg	Ex-vessel 16 Deg	Ex-vessel 24 Deg	Ex-vessel 26 Deg	Ex-vessel 36 Deg	Ex-vessel 39 Deg
Cu-63(n,a)	1.41E-01	9.95E-02	5.75E-02	5.69E-02	5.71E-02	5.77E-02	5.66E-02	5.61E-02
Ti-46(n,p)	2.38E-01	1.69E-01	9.42E-02	9.38E-02	9.41E-02	9.46E-02	9.32E-02	9.26E-02
Fe-54(n,p)	4.94E-01	4.23E-01	2.80E-01	2.82E-01	2.82E-01	2.81E-01	2.81E-01	2.80E-01
Ni-58(n,p)	5.62E-01	5.49E-01	5.33E-01	5.34E-01	5.34E-01	5.33E-01	5.35E-01	5.36E-01
U-238(n,f)	7.97E-01	7.56E-01	7.91E-01	7.84E-01	5.34E-01	7.88E-01	7.94E-01	7.97E-01
Np-237(n,f)	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00

Spectral weighting factors

Detector	In-vessel 20 Deg	In-vessel 30 Deg	Ex-vessel 6 Deg	Ex-vessel 16 Deg	Ex-vessel 24 Deg	Ex-vessel 26 Deg	Ex-vessel 36 Deg	Ex-vessel 39 Deg
Cu-63(n,a)	4.35E-02	3.32E-02	2.08E-02	2.07E-02	2.28E-02	2.10E-02	2.05E-02	2.03E-02
Ti-46(n,p)	7.37E-02	5.65E-02	3.42E-02	3.41E-02	3.76E-02	3.44E-02	3.38E-02	3.35E-02
Fe-54(n,p)	1.53E-01	1.41E-01	1.02E-01	1.03E-01	1.13E-01	1.02E-01	1.02E-01	1.01E-01
Ni-58(n,p)	1.74E-01	1.83E-01	1.93E-01	1.94E-01	2.14E-01	1.93E-01	1.94E-01	1.94E-01
U-238(n,f)	2.47E-01	2.52E-01	2.87E-01	2.85E-01	2.14E-01	2.86E-01	2.88E-01	2.89E-01
Np-237(n,f)	3.09E-01	3.34E-01	3.63E-01	3.64E-01	4.00E-01	3.63E-01	3.62E-01	3.62E-01

Comparison of M/C Ratios

Internal Capsules	Equal Weighting(1)	Spectral Weighting(2)	Least Squares Adjustment(3)
A240	0.953	0.913	0.852
W290	0.907	0.881	0.842
W290-9	0.891	0.858	0.818
W110	0.927	0.895	0.826
6 Deg Cav			
8			
9	0.904	0.893	0.863
11	0.934	0.931	0.922
16 Deg Cav			
8	0.912	0.935	0.883
9	0.850	0.836	0.801
11	0.887	0.870	0.851
24 Deg Cav			
8			
9			
11	0.831	0.797	0.798
26 Deg Cav			
8	0.877	0.883	0.835
9	0.886	0.901	0.864
11	0.885	0.876	0.856
36 Deg Cav			
8			
9			
11	0.833	0.814	0.794
39 Deg Cav			
8	0.872	0.856	0.807
9	0.807	0.760	0.727
11	0.829	0.809	0.794
Average	0.882	0.865	0.831
Std Dev	0.041	0.049	0.044

1. Linear average of M/C ratios of the neutron sensor reaction rates (Table 7.2-1 of WCAP 14557, Rev. 1)
2. Spectral weighting of the neutron sensor reaction rates (Grundl, J., "Derivation of Neutron Exposure Parameters from Threshold Detector Measurements," ASTM STP 1001, pp. 450-459.
3. Least squares adjustment [M/C ratios of the flux (E>1.0 MeV)] (Table 7.1-1 of WCAP 14557, Rev. 1)

In-Vessel Plus Ex-Vessel Data

	Plant 1	Plant 2	Plant 3	Plant 4	Plant 5	Average	std dev	% std dev
Cu-63(n,a)	1.03	1.02	0.89	0.94	0.92	0.96	0.06	6.5
Ti-46(n,p)	1.06	0.99	0.90	0.95	0.94	0.97	0.06	6.1
Fe-54(n,p)	0.93	0.88	0.80	0.90	0.84	0.87	0.05	5.6
Ni-58(n,p)	0.92	0.85	0.84	0.89	0.84	0.87	0.04	4.1
U-238(n,f)	0.93	0.86	0.96	0.92	0.85	0.90	0.05	5.4
Np-237(n,f)	0.97	0.96	0.99	0.93	0.88	0.95	0.04	4.5
Average	0.97	0.92	0.90	0.92	0.88	0.92	0.04	3.9
FERRET	0.92	0.86	0.93	0.92	0.83	0.89	0.04	5.0

	Plant 1	Plant 2	Plant 3	Plant 4	Plant 5	Average	std dev	% std dev
Cu-63(n,a)	1.06	1.10	0.99	1.02	1.05	1.05	0.04	3.9
Ti-46(n,p)	1.08	1.07	1.00	1.03	1.07	1.05	0.04	3.3
Fe-54(n,p)	0.95	0.95	0.90	0.98	0.95	0.95	0.03	3.1
Ni-58(n,p)	0.94	0.92	0.94	0.96	0.96	0.94	0.02	2.1
U-238(n,f)	0.96	0.93	1.07	1.00	0.96	0.98	0.05	5.4
Np-237(n,f)	1.00	1.03	1.10	1.01	1.00	1.03	0.04	4.3
Average	1.00	1.00	1.00	1.00	1.00	1.00		

In-Vessel Data Only

	Plant 1	Plant 2	Plant 3	Plant 4	Plant 5	Average	std dev	% std dev
Cu-63(n,a)	0.98	1.06	1.03	1.09	0.98	1.03	0.05	4.6
Ti-46(n,p)					1.01	1.01		
Fe-54(n,p)	0.95	0.90	0.83	0.97	0.86	0.90	0.06	6.7
Ni-58(n,p)	0.93	0.82	0.93	0.97	0.87	0.90	0.06	6.3
U-238(n,f)	1.07	0.94	0.96	1.06	0.86	0.98	0.09	8.7
Np-237(n,f)	1.12	1.12	1.14	1.10	0.82	1.06	0.14	12.8
Average	1.01	0.97	0.98	1.04	0.90	0.98	0.05	5.3
FERRET	1.04	0.95	0.93	1.03	0.84	0.96	0.08	8.8

	Plant 1	Plant 2	Plant 3.	Plant 4	Plant 5	Average	std dev	% std dev
Cu-63(n,a)	0.97	1.10	1.05	1.05	1.09	1.05	0.05	4.8
Ti-46(n,p)					1.12	1.12		
Fe-54(n,p)	0.94	0.93	0.85	0.94	0.95	0.92	0.04	4.5
Ni-58(n,p)	0.92	0.85	0.95	0.93	0.96	0.92	0.04	4.8
U-238(n,f)	1.06	0.97	0.98	1.02	0.96	1.00	0.04	4.1
Np-237(n,f)	1.11	1.16	1.17	1.06	0.91	1.08	0.10	9.7
Average	1.00	1.00	1.00	1.00	1.00			

Ex-Vessel Data Only

	Plant 1	Plant 2	Plant 3	Plant 4	Plant 5	Average	std dev	% std dev
Cu-63(n,a)	1.05	1.01	0.87	0.92	0.90	0.95	0.08	7.9
Ti-46(n,p)	1.06	0.99	0.90	0.95	0.92	0.96	0.06	6.5
Fe-54(n,p)	0.92	0.87	0.80	0.89	0.83	0.86	0.05	5.5
Ni-58(n,p)	0.92	0.85	0.82	0.87	0.83	0.86	0.04	4.4
U-238(n,f)	0.90	0.84	0.96	0.90	0.84	0.89	0.05	5.5
Np-237(n,f)	0.94	0.91	0.98	0.88	0.89	0.92	0.04	4.4
Average	0.96	0.91	0.89	0.90	0.87	0.91	0.04	3.9
FERRET	0.89	0.83	0.93	0.88	0.83	0.87	0.04	4.9

	Plant 1	Plant 2	Plant 3	Plant 4	Plant 5	Average	std dev	% std dev
Cu-63(n,a)	1.09	1.11	0.98	1.02	1.04	1.05	0.05	4.9
Ti-46(n,p)	1.10	1.09	1.01	1.05	1.06	1.06	0.03	3.1
Fe-54(n,p)	0.96	0.96	0.90	0.98	0.95	0.95	0.03	3.1
Ni-58(n,p)	0.95	0.93	0.92	0.97	0.96	0.95	0.02	2.0
U-238(n,f)	0.93	0.92	1.08	1.00	0.97	0.98	0.06	6.4
Np-237(n,f)	0.97	1.00	1.10	0.98	1.02	1.01	0.05	5.2
Average	1.00	1.00	1.00	1.00	1.00	1.00		