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Sensitivity Analysis

ORNL CROSS-SECTION SENSITIVITY ANALYSIS APPLICATIONS
FOR RADIATION SHIELDING*

D. E. Bartine,^a F. R. Mynatt,^a E. M. Oblow,^a R. L. Childs,^b
J. V. Pace,^b W. W. Engle,^a J. R. Knight,^b
T. M. Sims^c

ABSTRACT

An historical approach to the need for cross-section sensitivity analysis of radiation shielding problems is developed. The sensitivity function used in this work is then defined. The information generated in this type of sensitivity analysis may be used to identify the most important cross sections for a given problem and to correlate integral experiments with design problems. These sensitivities may also be incorporated with estimated cross-section uncertainties to predict the resulting uncertainty in a calculation, and to indicate the need for evaluation and measurement of specific cross sections. The application of this approach for the identification of important cross sections is illustrated for a one-dimensional sodium-iron system and for a two-dimensional reactor vessel head streaming problem. An initial application of sensitivity analysis to the correlation of an integral experiment with a design problem is also presented.

INTRODUCTION

The solution to radiation shielding problems generically involves deep-penetration, multiple-collision calculations. Since a basic goal of radiation shielding calculations is to provide protection for personnel and instrumentation, there is a need for a reasonable degree of confidence in the results of these calculations. The historical approach to satisfy this need has been to perform an experiment related to the shielding problem. The extent of agreement between experimental and calculated

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^aNeutron Physics Division, Oak Ridge National Laboratory.

^bComputer Sciences Division, Union Carbide Corporation, Nuclear Division, at Oak Ridge National Laboratory.

^cOperations Division, Oak Ridge National Laboratory.

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results is then assumed to reflect the accuracy of the calculations performed for the original problem. Once confidence is established, similar shielding problems may then be calculated to the same degree of confidence without the necessity for experimental verification. This approach has been quite useful in the calculation of many problems and in the verification of calculational techniques such as Monte Carlo and discrete-ordinates methods. However, application of the experimental approach depends on the similarity of the important features of the shielding problem and the experimental configuration. Therefore, the first priority in the design of an integral experiment is to determine what is important in the calculation of the given shielding problem and to ensure that similar features are incorporated in the experimental configuration.

Basic Theory

The work in cross-section sensitivity at Oak Ridge was initiated to determine which cross sections are important for a given shielding calculation, and to provide an estimate of the uncertainty in the calculation due to uncertainties in the cross sections utilized. Cross-section sensitivity analysis refers to a determination of how sensitive a calculated quantity or result is to the microscopic cross-section data utilized in the calculation. Previous work in this field has employed both direct substitution methods (1-4) and variational and perturbation theory (5-9). The sensitivity function used in the work discussed here (10,11) represents an application of the earlier perturbation approaches (5-7). The sensitivity, S , of a result, R , from slab or spherical discrete-ordinates transport calculation to a particular cross section, x , at energy E may be expressed for non-multiplying medium as

$$S_x(E) = \left[\int_{\vec{r}} d\vec{r} \int_{\vec{\Omega}} d\vec{\Omega} \rho_x(\vec{r}) \left\{ -\phi^*(\vec{r}, \vec{\Omega}, E) \sigma_x(E) \phi(\vec{r}, \vec{\Omega}, E) \right. \right. \\ \left. \left. + \int_{\vec{\Omega}'} d\vec{\Omega}' \int_{E'} dE' \phi^*(\vec{r}, \vec{\Omega}', E') \sigma_x(\vec{\Omega} \cdot \vec{\Omega}', E \rightarrow E') \phi(\vec{r}, \vec{\Omega}, E) \right\} \right] \div R, \quad (1)$$

where

$\phi^*(\vec{r}, \vec{\Omega}, E)$ = the unperturbed adjoint flux,

$\phi(\vec{r}, \vec{\Omega}, E)$ = the unperturbed forward flux,

$\sigma_x(E)$ = the cross section for reaction type x,

$\sigma_x(\vec{\Omega} \cdot \vec{\Omega}', E \rightarrow E')$ = the differential scattering cross section from angle $\vec{\Omega}$ to $\vec{\Omega}'$ and from energy E to E' for reaction type x,

$\rho_x(\vec{r})$ = the atomic number density,

R = the integrated response.

All of the above quantities are defined for specific problem under consideration.

R is defined as

$$R = \int_{\vec{r}} d\vec{r} \int_{\vec{\Omega}} d\vec{\Omega} \int_E dE D(\vec{r}, \vec{\Omega}, E) \phi(\vec{r}, \vec{\Omega}, E) \quad (2)$$

where $D(\vec{r}, \vec{\Omega}, E)$ is the response function (for example, a flux-to-dose conversion factor), and the source for the adjoint calculation.

The sensitivity definition of Eq. (1) serves to identify the most important cross sections for a given problem. A log-log plot of a sensitivity function per unit lethargy vs energy is referred to as a sensitivity profile and indicates cross-section importance as a function of the energy of the incident particle. This "fingerprint" of the calculation has many potential uses since it quantitatively focuses attention on the important cross-section data. Equation (1) can also be integrated over the incident energy E to give the total sensitivity of the problem to the cross sections for reaction type x. The effect of Legendre series truncation for a multigroup P_L cross-section set may also be evaluated by examining sensitivity as a function of the number of expansion terms. The information generated in the sensitivity analysis can then be applied to correlate integral experiments with design problems and, in conjunction with estimated cross-section uncertainties, to predict the resulting uncertainty in the calculation of a design problem or an integral experiment. An examination of these predicted uncertainties can then be made to determine the need for specific cross-section evaluation and measurements.

ONE-DIMENSIONAL APPLICATIONS

This method of cross-section sensitivity analysis as incorporated in the SWANLAKE (10) code, has been applied to a wide range of shielding problems calculated with the one-dimensional discrete-ordinates transport code ANISN (12). In a one-dimensional multigroup discrete-ordinates formulation, using the definitions of flux and transfer cross section from ANISN (see ref. 10), Eq. (1) becomes

$$S_{x,g} = \left[-\sum_j \rho_{x,j} \Delta V_j \sum_k \phi_{j,k,g}^* \sigma_{x,j,g} \phi_{j,k,g} \Delta \Omega_k \right. \\ \left. + \sum_j \rho_{x,g} \Delta V_j \sum_{g'} \sum_{\ell=0}^{\infty} \phi_{j,g'}^{*\ell} \sigma_{x,j,g \rightarrow g'}^{\ell} \phi_{j,g}^{\ell} \right] / R, \quad (3)$$

where

$\phi_{j,k,g}$ = the unperturbed forward flux for energy group g in spatial interval j at discrete angle k ,

ΔV_j = the volume for spatial interval j ,

$\Delta \Omega_k$ = the weight given to the flux at angle $\hat{\Omega}_k$ in the Gaussian quadrature representation of the angular flux,

$\phi_{j,g}^{\ell}$ = the ℓ th moment of the Legendre series expansion in L terms of the angular flux for energy group g in spatial interval j .

The first major application of the SWANLAKE code was to perform a sensitivity analysis on a problem of particular interest to LMFBR design -- the problem of calculating neutron penetration through large thicknesses of sodium followed by iron. This study was originally performed with a 50-energy-group library of cross sections (13). The results of this study led to the formation of a 128-group library in which the group boundaries were selected to fit the important structure in the cross sections. The 128-group calculation gave a transmitted dose a factor of two higher than the 50-group calculation (14). This effect is attributed to better resolution of important valleys in the cross sections, especially the 0.3-MeV sodium minimum. It is encouraging to note that the overall sensitivities

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did not change substantially with the change in group structure, although more detail is available from the profiles for the 128-group calculation.

The Na-Fe system studied consisted of 15 ft of sodium followed by 3 ft of iron. The spectrum of neutrons incident on the sodium was the spectrum calculated to be leaking from the FTR radial shield (15). The response calculated was the neutron tissue dose, as defined by a Snyder-Neufeld response function (16), at the exit face of the iron. The cross sections used were ENDF/B-III cross sections, MAT 1156 for Na and MAT 1180 Mod 1 for Fe.

A summary of the results from the Na-Fe sensitivity study is given in Table 1. The sensitivities can be interpreted as the predicted change (decrease, since all are negative) in the tissue dose per percent increase in the cross section (or per percent increase in density). It is apparent from the table that the tissue dose exiting the iron is highly sensitive to the sodium total cross section, of which nearly 88% of the sensitivity is attributable to elastic scattering and 12% to inelastic scattering (about 8% to the first level). Figure 1 presents a sensitivity profile showing the sensitivity of the tissue dose to the sodium cross sections. The peaks in the profile generally correspond to minima in the sodium elastic (and therefore total) cross section, for example the minima at 1.9, 0.5, and 0.3 MeV. The sensitivities in Fig. 1 are very high (a value of above 1.0 on the profile is generally considered significant, and the sensitivity of the 0.3 MeV minimum alone accounts for about 37% of the total sensitivity. The sensitivity above 1 MeV is extraordinarily high considering the small number of source neutrons in this energy range. This is due largely to the moderating characteristics and thickness of the sodium slab (15 ft), which result in the high-energy source particles having a greater probability of entering the iron slab with sufficiently high energy to penetrate and contribute to the dose. The lower energy cutoff at 24.5 keV is due to the fact that this is the location of a very deep and broad iron minimum, below which the iron greatly attenuates neutrons. Also, the dose response function decreases quite rapidly for lower-energy neutrons. The sensitivity profile for iron cross sections

is presented in Fig. 2. Both Table 1 and Fig. 2 show that this problem is much less sensitive to the iron total cross sections than to the sodium cross sections. This can be explained by the fact that the majority of the attenuation occurs in the sodium; the iron filters out the neutrons below the window energies so that the transmitted neutrons have had only a few collisions in the iron. As would be expected, the iron sensitivity profile is dominated by the 24-keV minimum.

It was previously mentioned that the sensitivity analysis provides information concerning the effect of the Legendre series expansion for the problem. For this problem it was predicted that diffusion theory and transport theory give essentially the same tissue dose through the sodium, but that diffusion theory is inadequate for the calculation of the dose through the combined Na-Fe system. This prediction was confirmed by recalculation.

Cross-section sensitivity analysis has also been applied to determine the important cross sections for one-dimensional calculations of coupled neutron-gamma transport in air (17) and concrete (18) systems, and neutron transport in a preliminary CTR blanket (19). The analysis of the air problem included estimates of the uncertainty in the total neutron-gamma-ray tissue dose from preliminary cross-section error files given in the evaluations of the nitrogen (20) and oxygen (21) cross sections. A re-measurement of the nitrogen (n, α) cross section, especially in the energy range of 5-MeV nitrogen elastic minimum, was indicated as the most important data improvement for the reduction of this uncertainty.

TWO-DIMENSIONAL APPLICATIONS

The latest advance in this technological area at ORNL has been the development and testing of the VIP-SWANLAKE code for cross-section sensitivity analysis of calculations made with the two-dimensional discrete ordinates code DOT (22). The two-dimensional multigroup discrete ordinates equation analogous to Eq. (3) for the one-dimensional case is

$$S_{x,g} = \left[-\sum_j \rho_{x,j} \Delta V_j \sum_{\ell=0}^L \sum_{m=0}^{\ell} \phi_{j,g}^{*\ell,m(2\ell+1)} \sigma_{x,j,g} \phi_{j,g}^{\ell,m} + \sum_j \rho_{x,j} \Delta V_j \sum_{g'} \sum_{\ell=0}^L \sum_{m=0}^{\ell} \phi_{j,g'}^{*\ell,m} \sigma_{x,j,g \rightarrow g'}^{\ell} \phi_{j,g}^{\ell,m} \right] \quad (4)$$

where $\phi_{j,g}^{\ell,m}$ is a spherical harmonic expansion of the flux,

$$\equiv \sum_k \phi_{j,g,k} A_k^{\ell,m} \Delta \Omega_k \quad ,$$

with

$$A_k^{\ell,m} = \left[\frac{2}{1-\delta_{0,m}} \frac{(\ell-m)!}{(\ell+m)!} \right]^{\frac{1}{2}} P_{\ell}^m(\eta_k) \cos(m\psi_k) \quad ,$$

η_k is a polar direction cosine for angle k , and ψ_k is an azimuthal angle for angle k .

The primary difference between Eqs. (3) and (4) lies in the treatment of the negative or loss term for cross section x . Any angular flux representation is used for the one-dimensional treatment of this term, while the Legendre moments of the flux are used in the two-dimensional equation. The angular flux is the most complete representation of the information available from a discrete ordinates calculation, but the sheer bulk of the data for a two-dimensional calculation makes this approach impractical. For example, the forward and adjoint flux moments for the FTR sensitivity analyses which follows may both be stored on the same 9 track tape.

It would require six 9 track tapes to store just the forward angular flux for the same calculation. A corresponding increase in time would also be required for determining the sensitivities from the angular flux as opposed to the scalar flux. Recent testing for a deep sodium penetration problem and for an infinite air calculation (including a first-collision source term) indicate the applicability of the VIP-SWANLAKE flux moments approach. Further testing will be required to ascertain if significant limitations exist for this approximation.

The first realistic two-dimensional shielding problem to be analyzed by the VIP-SWANLAKE approach is the calculation of neutron and gamma tissue doses at the maintenance deck of the FTR (23,24). The geometry for the DOT calculations is shown in Fig. 3. The sensitivity analysis did not extend to the full model, but only the head compartment region above $z = 631.2$ cm. The DOT calculations were performed with a 21-18 coupled neutron-gamma-ray library, and an S_{166} quadrature suitable for streaming in annular gaps and P_3 scattering were employed. The peak doses calculated at the maintenance deck as shown in Fig. 3 were 0.6 mrem/hr and 0.4 mrem/hr, respectively, using the Henderson (25) neutron and gamma-ray flux-to-dose conversion factors. The sources for the adjoint calculations were input at the spatial position on the maintenance deck where the doses peaked in the forward calculations.

A sensitivity summary for this problem is presented in Table II. Sensitivities are given for all materials representing a spatial region and for especially important elements in a given region. It is clear that the calculated neutron dose is only significantly sensitive to the materials surrounding and above the N_2 streaming gap, namely the SA508 head and Z ring and the borated polyethylene top shield and its SS304 cover. The cross sections for iron in the SA508 and for hydrogen in the borated polyethylene are especially important. Elastic scattering in hydrogen appears to be especially significant for this problem since back-scatter near the deck sharply reduces the change of contribution to the neutron dose. The profile representing the neutron dose sensitivity to all cross sections for this calculation is given in Fig. 4. The profile

peaks in the 1 to 0.1 MeV region just below the iron inelastic threshold at 0.86 MeV, and drops off significantly below the iron 24 keV minimum.

Table II also indicates that the gamma dose is most sensitive to neutron transport in the SA508 and to gamma-ray transport through the SS304 and SA508 steels. Since sensitivities are calculated for every position in the cross-section transfer matrix, including secondary gamma-ray production by neutrons on a neutron energy group to gamma energy group basis, it is possible to determine where the gamma rays which contribute to the calculated dose are produced, and by what material. Table III indicates that the most important gamma-ray sources for this problem are produced by the iron in the SA508 region and by the boron in the borated polyethylene. A more detailed study indicates that about 31% of the gamma dose is due to gamma-ray production in the 7 to 8 MeV group, which contains strong iron (n,γ) production lines at 7.63 and 7.65 MeV. This gamma production mechanism for iron was responsible for 38% of the gamma dose calculated in the air-concrete study (18). Virtually all the gamma dose due to boron production is a result of the (n,γ) 0.48-MeV line. The importance of these two production mechanisms for the gamma tissue dose is verified by the gamma transport sensitivity profile shown in Fig. 5, where the peak sensitivities occur in the 7 to 8 MeV and 0.4 to 0.6 MeV energy groups.

The 2-D problem analysis presented here not only indicates the important cross sections for the neutron and gamma-ray doses, but also presents a clear enough picture of the important features of the calculation to be useful in shielding design. In such applications, however, it is important to bear in mind the limitations of perturbation theory for predicting changes in the calculated result. The results for this case indicate a seemingly unfortunate trade-off due to boration of the polyethylene shield, since the predicted influence of boron in decreasing the neutron dose is less than the predicted gamma dose increase due to boron (n,γ) production. Since the bulk of the neutron dose and gamma production sensitivity occurs in the SA508 region, it seems reasonable to consider the positioning of some borated polyethylene within the nitrogen gap in

order to obtain a lower neutron flux entering the SA508. A redesign and recalculation of the upper head region would be necessary to evaluate these observations.

CORRELATION OF INTEGRAL EXPERIMENTS WITH DESIGN PROBLEMS

The results obtained from cross-section sensitivity analysis may be used for the correlation of integral experiments with design problems. The basic concern in this approach is to identify the important cross sections for the calculation of the design problem, and then to ensure that results obtained for the integral experiment are subject to similar sensitivities. The problem used to describe this technique is that of the effects of nickel cross sections on calculations through the lower axial shield (14). This problem was of interest because of the relatively large uncertainties associated with the cross sections available for this element, and the desire to employ an inconel reflector in the FTR. The 50-group cross-section set employed at ORNL for FFTF design support calculations was used to perform forward and adjoint ANISN calculations through the slab geometry shown in Fig. 6. A stainless steel energy-dependent neutron damage function was applied as the response function in zone 8, the first interval in the FTR grid plate located approximately 28 in. below the lower-axial reflector. The sensitivity of the calculated stainless steel damage to nickel cross sections was then calculated as a function of spatial zone, and is given in Table IV. The sensitivities for the individual spatial zones integrated over energy are given in Table IV. The zones preceding and including the grid-plate interval in which the damage was calculated, the sensitivities are negative; for zones following the grid-plate interval the sensitivities are positive. This is explained by the fact that increasing the cross section in the zones preceding the grid plate increases attenuation in those zones and thereby decreases the number of neutrons reaching the grid plate and causing damage. On the other hand, increasing interactions in the zones following the grid-plate interval tends to increase the damage because neutrons are reflected back into the grid-plate interval. The overall sensitivity to nickel cross sections is 1.08, which is considered to be low based on previous sensitivity studies, but may still be significant in view of the

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existing uncertainties in nickel cross sections and the need for accurate FTR calculations. The damage sensitivity profile to the nickel cross sections in the reflector region (zone 3) is given in Fig. 7. The profile peaks at about 0.3 and 0.5 MeV, but does not exceed 1.0, which is an arbitrary standard used to judge high sensitivity. Essentially identical trends are observed when the sensitivity analysis considers the cross sections of all the elements included in Inconel (see Fig. 8). This indicates that insofar as the reflector region is concerned, the calculated grid plate damage is primarily sensitive to the nickel cross sections. The damage sensitivity to nickel cross sections in the thick grid plate shield (region 6 contains 75% stainless steel 316, which is 13% by atom Ni) is shown in Fig. 9, and the total sensitivity to nickel cross sections in all zones is given in Fig. 10. The similarity of the total sensitivity profile to those for zones 3 and 6 is apparent. Below the peak at 0.3 MeV the total sensitivity curve is like that for zone 6; above 0.5 MeV it is much like that for zone 3. Another interesting observation is that all the sensitivity curves show a peak at 24 keV, at the large iron cross-section minimum. Thus the decrease of interactions in iron for neutrons of this energy increases the relative probability of interactions in nickel and as a result increases the sensitivity of the damage to nickel cross sections at that energy. The peak is higher in zone 6 than in zone 3 because of the greater abundance of iron in zone 6 and also because of the softer spectrum.

In an experiment performed at the Oak Ridge Tower Shielding Facility (26), 2 $\frac{1}{2}$ -in. and 5-in. slabs of Inconel were positioned immediately behind the Westinghouse spectrum modifier (see Fig. 11) and neutron measurements were made at various locations beyond the configurations. Calculations of neutron transport through the experimental configuration were then performed with ANISN using the same cross sections employed for the FTR calculations. It was then desired to correlate the calculations for the Inconel experiment with the FTR calculations. If neutron damage were measured and the spectrum properly identified, the experiment-to-calculated ratios would give directly the error in the FTR calculations. A sensitivity calculation was performed for the TSF configuration with a

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hypothetical "damage function" detector. These sensitivity profiles, shown in Fig. 12, do not resemble the sensitivity plot obtained in the FTR calculation. The plot for the 5-in. slab comes closest, but it shows a much higher sensitivity to high-energy cross sections than the plot for the FTR configuration showed. The fact that spectrometer measurements were made in the experiment provides additional information content. Through sensitivity analysis this additional information can be used to construct an experimental comparison adjusted to the FTR sensitivity function. In order to do this, the experiment must be "close" to the design configuration in the sense that the adjustment cannot create sensitivity in a particular energy range which has little or no sensitivity in the experiment. This technique, however, allows wider-range use of the experiment and provides a clear correlation between experiment comparison and design analysis uncertainty.

In this demonstration of the sensitivity approach which can correlate the FTR calculation with the experiment, more detailed sensitivity studies were performed for the experimental configuration - a sensitivity function being generated for each of several energy bins over the energy range covered by the neutron spectrometer. Six energy bins were selected, and a nickel sensitivity calculation was performed for each bin for each of the 5-in. Inconel slab configuration. The resulting flux sensitivity curve in each energy bin was then normalized to the FTR damage sensitivity curve in the same energy bin (each bin covered five energy groups). The normalized flux sensitivity curve is compared with the FTR damage sensitivity curve in Fig. 13. Except for the two lowest energy bins, the sensitivity plots for the two cases are very similar. The weighting factors required to normalize the experimental sensitivity profile to the profile for the FTR design calculation are given in Table V. These factors may then be used as a weighted response function for the experimental results. For this integral response the experiment has approximately the same sensitivity profile as the design problem and is, hence, prototypic in the same sense that one would achieve with an experiment having the same source configuration, and response function as the design problem. The disagreement between experiment and analysis then applies directly to the design problem, provided that the errors in the

calculation of the experiment are due to uncertainties in the cross sections for which the weighted response function was determined. The sample application given here as a demonstration of the correlation technique was performed before the availability of the two-dimensional sensitivity code. Due to the limitations of one-dimensional calculations, a full implementation of the procedure described above must be performed using comparisons of two-dimensional calculations and experimental results.

CONCLUSIONS

The cross-section sensitivity analysis procedure illustrated in this paper is capable of indicating qualitatively and quantitatively the relative importance of cross sections (and therefore the physical processes they represent) for one- and two-dimensional calculations as a function of reaction type and energy. This method can also be applied to the correlation of integral experiments with shielding design problems so that the disagreement between experimental measurements and calculations has a direct interpretation for the design problem. These sensitivities may also be incorporated with estimated cross-section uncertainties to predict the resulting uncertainty in a calculation, and to indicate the need for evaluation and measurement of specific cross sections.

It should be noted that much of this work is still in a developmental state. In particular, techniques are being developed to facilitate the use of ENDF error files for the prediction of design problem uncertainties, and to obtain and test error files using integral experiments. Additionally, sensitivity analysis routines are being developed for one-dimensional reactor core problems.

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Table 1. Sensitivity of Neutron Tissue Dose at Exit Face
of Na-Fe Configuration to the Indicated Cross Sections

Reaction	Iron	Sodium
Elastic Scattering	-3.2055	-14.2121
Inelastic scattering	-0.0117	- 1,9658
Absorption	-0.1989	- 0.0320
Total	-3.4174	-16.2103
First-level inelastic	-0.0103	- 1.2642
Second-level inelastic	-0.0003	- 0.2191
.....		
Sixth-level inelastic		- 0.1141
Continuum inelastic	-0.0003	
(n,2n')	0	0
(n,n') α	-1.2×10^{-7}	
(n,n')p	-6.0×10^{-7}	
(n, γ)	-0.1986	- 0.0123
(n,p)	-0.0003	- 0.0125
(n, α)	0	- 0.0071

**Table II. Sensitivities of Calculated Peak Neutron and Gamma
Tissue Doses at the FTR Maintenance Deck to the
Indicated Cross Sections**

Cross Sections	Sensitivity of Neutron Tissue Dose	Sensitivity of Gamma-Ray Tissue Dose	
		Neutron Cross Sections	Gamma Cross Sections
SS 304 (Fe)	-0.88 (-0.52)	-0.50 (-0.26)	-0.93 (-0.65)
Borated Polyethylene (H) (C) (B)	-4.16 (-3.38) (-0.67) (-0.11)	-0.27 (-0.12) (-0.02) (-0.13)	-0.22 (-0.05) (-0.16) (-0.01)
SA508 (Fe)	-3.37 (-3.14)	-1.75 (-1.58)	-0.80 (-0.78)
Na	-0.03	-0.05	-0.00
A	-0.00	-0.00	-0.00
CaSiO ₃	-0.02	-0.02	-0.00
N ₂	-0.01	-0.01	-0.00
Magnetite Concrete	-0.02	-0.01	-0.00
Inconel 600	-0.01	-0.03	-0.00
SS410	-0.01	-0.01	-0.00
SA516	-0.02	-0.03	-0.00
Sum of All Cross Sections	-8.53	-2.68	-2.05

**Table III. Percent of Calculated Henderson (25) Gamma
Tissue Dose Due to Gamma Rays Produced by
the Indicated Materials**

Material Region Description	SS304 (Fe)	Borated Polyethylene 30.9 (H) (B)	SA508 (Fe)	Magnetite Concrete (Fe)
% Gamma Dose	4.2 (2.5)	(2.6) (28.3)	54.6 (49.9)	5.1 (2.7)
Material Region Description	All head region materials	Gamma rays incident at z = 631.2 cm, Fig		
% Gamma Dose	98.4	1.6		

Table IV. Sensitivity of Neutron Damage to FTR Grid Plate
to Nickel Cross Sections in Various Spatial Zones
(Refer to Fig. 6)

Zone No.	Region	Sensitivity Integrated Over Energy
1	Core	-3.07×10^{-2}
2	Insulator	-6.57×10^{-3}
3	Reflector	-3.84×10^{-1}
4		-6.36×10^{-2}
5		-2.96×10^{-3}
6	Grid Plate Shield	-5.93×10^{-1}
7		-9.43×10^{-3}
8 ^a		-2.97×10^{-5}
9	Grid Plate	$+8.08 \times 10^{-3}$
10		$+2.08 \times 10^{-3}$
11		0
1-11		-1.08×10^0

^aThis zone of grid plate for which the nickel cross-section sensitivity of the neutron damage to the grid plate was calculated.

Table V.. Weighting Factors for Experiment-to-Design
Sensitivity Normalization

Energy Bin No.	Energy Range (MeV)	Weighting Factor
1	12.2-4.49	0.0
2	4.49-1.65	0.17
3	1.65-0.608	0.34
4	0.608-0.224	0.53
5	0.224-0.0525	0.29
6	0.0525-0.015	0.063

Fig. 1. Sensitivity of Calculated Tissue Dose Exiting an Na-Fe Configuration to Sodium Cross Sections.

Fig. 2. Sensitivity of Calculated Tissue Dose Exiting an Na-Fe Configuration to Iron Cross Sections.

Fig. 3. Geometry for DOT Calculations of FTRF Assembly.

Fig. 4. Sensitivity of Calculated Henderson (25) Neutron Tissue Dose to Cross Sections for Materials in the FTR Head Region.

Fig. 5. Sensitivity of Calculated Henderson (25) Gamma Tissue Dose to Gamma Transport Cross Sections for Materials in the FTR Head Region.

Fig. 6. Configuration of the FTR Lower Axial Shield Used for One-Dimensional Slab Calculations of Neutron Damage to the FTR Grid Plate.

Fig. 7. Sensitivity of Calculated FTR Grid Plate Damage to Nickel Cross Sections in Reflector Region (Zone 3).

Fig. 8. Sensitivity of Calculated FTR Grid Plate Damage to Inconel Cross Sections in Reflector Region (Zone 3).

Fig. 9. Sensitivity of Calculated FTR Grid Plate Damage to Nickel Cross Sections in Zone 6.

Fig. 10. Sensitivity of Calculated FTR Grid Plate Damage to Nickel Cross Sections in All Zones.

Fig. 11. Experimental Configuration for Spectrum Modifier Plus 5-in. Inconel.

Fig. 12. Comparison of Damage Sensitivities to Nickel Cross Sections for 2½-in. and 5-in. Inconel Configurations.

Fig. 13. Comparison of FTR Grid-Plate Damage Sensitivity to Nickel Cross Sections with Scaled Flux Sensitivity for 5-in. Inconel Configuration.

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Fig. 1. Sensitivity of Calculated Tissue Dose Exiting an Na-Fe Configuration to Sodium Cross Sections.

2.
Fig. 2. Sensitivity of Calculated Tissue Dose Exiting an Na-Fe Configuration to Iron Cross Sections.

Fig. 1. Geometry for DOT Calculations of FFTF Assembly.

LMFBR SENSITIVITY 21N-186 X

SENSITIVITY PER UNIT LETHARGY $((DR/R) / (D\Sigma/\Delta))$

F4

SENS

21N-18G

XSECTS -

GAMMAS

SUM ALL

ZONES

Fig. 6. Configuration of the FTR Lower Axial Shield Used for One-Dimensional Slab Calculations of Neutron Damage to the FTR Grid Plate.

Fig. 7. Sensitivity of Calculated FTR Grid Plate Damage to Nickel Cross Sections in Reflector Region (Zone 3).

Fig. 8. Sensitivity of Calculated FTR Grid Plate Damage to Inconel Cross Sections in Reflector Region (Zone 3).

Fig. 9. Sensitivity of Calculated FTR Grid Plate Damage to Nickel Cross Sections in Zone 6.

Fig. 10. Sensitivity of Calculated FTR Grid Plate Damage to Nickel Cross Sections in All Zones.

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DIMENSIONS ARE IN INCHES EXCEPT AS NOTED

NOTE: MINIMUM OF 8-in. Li PAR. ON TOP OF CONFIGURATION

Fig. 2. Experimental Configuration for Spectrum Modifier Plus 5-in. Inconel, ~~1/4 in. Boron~~

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Fig. 12. Comparison of Damage Sensitivities to Nickel Cross Sections for 2-1/2-in. and 5-in. Inconel Configurations.

Fig. 1/3. Comparison of FTR Grid-Plate Damage Sensitivity to Nickel Cross Sections with Scaled Flux Sensitivity for 5-in. Inconel Configuration.