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APPLICATION OF THE SENSITIVITY AND UNCERTAINTY ANALYSIS SYSTEM <u>LASS</u> TO FUSION REACTOR NUCLEONICS*

by

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ABSTRACT

Sensitivity analysis, as applied to both nuclear design and data uncertainty, has developed into a valuable tool for fusion reactor nuclear analysis. Several such studies have been undertaken with the LASL sensitivity system LASS, which includes as its principal modules SENSIT-1D, ONETRAN, and ALVIN. These modules function in a multigroup environment using standard flux and data interface files for communication. The input multigroup cross-section data and uncertainties are obtained primarily from ENDF/B using the NJOY processing system. In particular cases, the input library can be modified by the ALVIN module to improve consistency with available integral experiments. The primary output from LASS is the uncertainty (or change) in important reactor parameters, as calculated in the SENSIT-1D module. Applications of LASS and its component parts have been made to the Tokamak Fusion Test Reactor (TFTR), the Reference Theta-Pinch Reactor (RTPR), and to an Experimental Power Reactor (EPR). This paper emphasizes the initial assessment of cross-section sensitivity for an EPR design. Nucleonic responses examined include neutron and gamma-ray kerma in the toroidal field coils and Mylar superinsulation, displacement damage and transmutation in the copper of the toroidal field coils, and activation of the outboard dewar. These sensitivities are now being used to narrow the range of uncertainty analyses required to quantitatively assess cross-section adequacy for EPR design calculations. Acceptable target uncertainties in nucleonic design parameters are simultaneously being formulated. Experience at LASL with sensitivity and uncertainty analysis techniques incorporated in LASS has provided convincing evidence of their value for fusion reactor studies. Many of these studies are of a shielding nature; e.g., deep penetrations of high-energy neutrons through steel, lead, boron carbide, and graphite, with responses such as activation and kerma.

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1 INTRODUCTION

There exists an extensive commonality of fusion reactor nucleonics, including of course fusion reactor shielding, with fission reactor shielding. For example, both generally involve moderate-to-deep penetration transport calculations for neutrons and gamma rays, along with kerma responses. Those kermas may be manifested as heating in concrete or a superconducting magnet, or simply as absorbed biological dose. Similar commonality exists in neutron activation, although fusion nucleonics introduces a new class of often dominant reactions; e.g., (n,np), (n,nα), (n,2n), and other threshold reactions producing radioactive nuclides. Interest also arises in the transmutation products, stable and unstable, of such reactions insofar as they affect materials' mechanical properties, electrical conductivity, etc.

Conversely, there are areas of less commonality, where fusion reactors (and even 2.25-pJ neutron irradiation facilities such as the INS[1]) present relatively new if not unique problems. Most such nucleonic interests are in blanket responses, or merely involve new response functions such as radiation damage in superconductors. However, both classes of problems, regardless of commonality, do in general require target accuracies which differ from those required for fission reactor shields. As a case in point, consider the superconducting (S/C) coil, where both radiation damage and heating are required to be known within approximately 10-20%, as a realistic goal. Such accuracies are perhaps consistent with the target goals for fission reactor nuclear heating analysis, but involve an essential difference. In the fusion reactor case, the accuracies are required after an attenuation of approximately 10^6 in the total neutron plus gamma-ray kerma. The pertinent point here is that fusion reactor data requirements may differ from those for fission reactors in two ways: (1) different reactions can be involved in both transport and response cross sections of interest; and (2) the accuracy requirements will in some cases, as for S/C magnet damage, be more stringent.

It is perhaps useful here to review some fusion reactor nucleonic problems in order to demonstrate the foundations for their data requirements. First in order are the intense 2.25-pJ neutron source facilities, where biological dose from primary neutron and secondary prompt gamma rays is of most concern, with activation gamma-ray doses also being important. Here, as with fission reactor biological shields, we are currently content with "factor of 2 or so" target accuracies, but we have hopes for better accuracy in the next generation of fusion devices (after approximately 1985). Shield design is presently underway[1] for the INS and the High Energy Gas Laser Facility (HEGLF), so cross-section requirements are somewhat academic; we make do with the best presently available data. It is worth pointing out that both the INS and HEGLF shield designs involve void penetrations with diameters of the order of one metre, so that streaming and reflection are the principal computational difficulties. Bulk concrete shields are approximately 3-m thick.

Conceptual design efforts in the USA which impact near-term data needs are mainly those evolving about the Experimental Power Reactor (EPR) projected for construction in the mid 1980's. The EPR is a next-generation Tokamak reactor after the Tokamak Fusion Test Reactor (TFTR), for which a thorough data assessment has been performed.[2] Whereas the only nucleonic parameter of crucial interest in the TFTR blanket/shield was activation dose levels after shutdown (biological shielding for primary radiation during operation is roughly comparable in extent and importance to that in HEGLF and INS), the EPR will have numerous important responses to consider. Most important of these are the radiation effects in S/C toroidal field (TF) coils -- the principal substance of this paper. Details are given in Sec. 4 below.

From our experience in fusion reactor analysis we can make some general observations which may interest the shielding community. First, sensitivity analysis is of little use in improving data for the near-term devices INS, HECLF, and TFTR. Even so, almost all detailed design calculations for these devices must be performed in complex geometries by Monte Carlo methods; hence, one-dimensional sensitivity calculations are of limited value. An important point to emphasize in this discussion is the requirement in many cases for two-dimensional discrete-ordinates calculations and associated sensitivity analysis, as well as the need for long-range development of Monte Carlo sensitivity methods. Streaming problems presently dominate the shield design for the EPR, especially for the TF coils. By dominating a design, we mean that the shield may well determine the TF coil magnet diameter, and hence have a major impact on the device cost. Thus, an essential difference immediately occurs from fission reactor shielding; the TF coil shield is ab initio a major plant cost determinant.

A second general observation of interest is that fusion reactor nucleonic analysis provides a fortuitous convergence of two technologies. That is, the sensitivity and uncertainty methods are maturing just as a vital requirement for them is emerging in the form of an embryonic fusion reactor nucleonics technology.

2 NUCLEAR DATA IMPLICATIONS OF FUSION TECHNOLOGY

Fusion reactor technology has introduced new materials and reactions of importance for nuclear analysis. Also, for materials and reactions of common interest in fusion and fission reactor programs, new energy ranges are now of primary interest for fusion. It rapidly becomes clear that fusion technology cannot depend completely on the nuclear data evaluation and assessment programs performed for the fission technologies. Not only are new materials (e.g., W, Cu, Pb mortars), reactions, and energy ranges of interest, but data assessment by sensitivity methods must direct added attention to new high-energy responses and to secondary energy/angular distributions. Consequences of changes in such secondary distributions are already known to be important in fusion systems,[3] and a methodology for their systematic analysis is the subject of a forthcoming paper by Gerstl.[4]

Although fusion technology presents many new demands in the nuclear data area, we are fortunate that it may not be too late to plan and execute an orderly, rational and coordinated program of data

- assessment of requirements
- measurements (differential and integral)
- evaluations.

Having at our disposal not only existing ENDF[5] data, but also fairly well-developed sensitivity and uncertainty analysis methods, we have the opportunity to avoid the pitfalls experienced by the fission reactor data programs, where the cost-effectiveness of the integral experiments was limited by the lack of modern sensitivity-based planning methods. We also feel that failure to develop a coordinated (international) program for fusion technology nuclear data would be invaluable. Initial efforts at defining high priority nuclear data needs within the U. S. A. fusion program have been published,[6] and are being used as the basis for the U. S. A. nuclear data measurement request lists. A review of nuclear data needs can be found in Ref. 7.

3 LASS SYSTEM AND APPLICATIONS

Sensitivity and uncertainty analysis is done within the Los Alamos Scientific Laboratory LASS system, as shown in Fig. 1. Partial cross sections and uncertainties, σ and $\Delta\sigma$, are processed from ENDF into multigroup form by the NJOY system.[8] Because covariance data in ENDF are presently very scarce, most multigroup covariance matrices [Cov] are produced by the COVMAT code described in Appendix A of Ref. 2. Forward and adjoint multigroup neutron/gamma-ray fluxes are computed by the ONETRAN code,[9] using standard cross-section and flux files. The actual calculation of sensitivities and uncertainties is performed by the SENSIT-1D code,[10] using the inputs shown in Fig. 1. Again, standard CCCC (Committee on Computer Code Coordination) interface files are used for cross-section and flux inputs. The ALVIN module for differential and integral data consistency analysis is discussed in detail by Kerpke and Muir,[11] so it will not be considered in this paper.

Applications of LASS components have been made to several fusion facilities, including the Tokamak Fusion Test Reactor (TFTR),[2] the Reference Theta-Pinch Reactor (RTPR),[12,13] and the Intense Neutron Source (INS).[1] These applications have included design sensitivity as well as cross-section sensitivity, both of which use the same basic perturbation theory methodology incorporated in SENSIT-1D. That is, in both cases one is simply performing integrations over phase space of an integrand involving perturbations of the transport operator, δL; either cross-section changes or design changes are simply manifested as the difference operator of the perturbed and unperturbed transport operator. The sensitivity is given in the notation of Ref. 2 as

$$P_{\Sigma_i} = \frac{\langle \phi^*, L_{\Sigma} \phi \rangle_i}{R} .$$

All present day sensitivity studies based upon transport theory derive from the work of [illegible]indowski.[14] Extension and computer coding of his methods have been discussed in detail by Gerstl, Bartine, and others.[2,4,15,16] In the discussions below we follow the terminology and notation of Ref. 2.

In all the applications discussed above, as well as the fusion EPR application which is the principal subject of this paper, we have consistently observed the limitations of one-dimensional analysis. Fusion reactor nucleonic design problems are frequently multidimensional, and the concommitant sensitivities are often most important for these problems. Thus sensitivity methods and code development at LASL is concentrated on multidimensional analysis, along with the secondary energy/angular distribution sensitivities mentioned above.

4 DATA ASSESSMENT FOR A FUSION EXPERIMENTAL POWER REACTOR (EPR)

Assessment of nuclear data needs for a fusion EPR is, of course, design dependent. However, the EPR designs currently extant, as well as later conceptual studies of Ignition Test Reactors (ITR), are generically similar. For example, several conceptual reactor designs use laminated stainless steel/B$_4$C shields. For our current test case we have chosen the EPR design of the Argonne National Laboratory, as described in Ref. 17 and in private communications. The design has two shield assemblies, denoted "inner" and "outer". The inner shield refers to a segment of shielding around the toroidal axis of the torus; i.e., if one considers a major radius through the center of the plasma chamber, the inner shield is toward the origin. Figures 2 and 3 show one-dimensional models based upon radial traverses from the poloidal axis (plasma centerline) through the inner and outer shields, respectively. Observe that the thinner inner shield is of effective but costly stainless steel/B$_4$C, while the thicker outer shield is composed largely of less costly lead mortar. The technical basis for alternative shields is clear if the magnetic field profile is considered: With the D-shaped toroidal field (TF) coils, there exists a relatively large space for the outer shield, whereas the inner shield must be as thin as possible. The latter requirement arises from a desire to maximize the magnetic field in the plasma, and thus maximize power density.[17]

At this point we digress to discuss the general approach used in the EPR data assessment, as well as in the previous assessments.[2] First, a broad ranging sensitivity study is performed simply using the total, scattering (matrix) and absorption cross sections from the transport code cross-section sets. These included neutron interaction, gamma-ray production, and gamma-ray scattering matrices. From the large mass of these survey calculations, which are automated in SENSIT-1D, we then isolate materials, partial cross sections and energy regions of potential interest. This latter step is greatly assisted by computing integral sensitivities. After a semi-quantitative review of the germane cross-section errors, we chose a manageable number of potentially important materials and partial cross sections for more detailed error evaluation. For these we process available covariance data into multigroup form. However, as noted above, error data in ENDF are sparse, so most covariance data need to be evaluated on an *ad hoc* basis. Using such covariance matrices, an uncertainty analysis is performed for the suspect partial cross sections. This paper discusses the results for an EPR through the stage of evaluating and processing covariance data.

In the case of our EPR analysis, error data for C and O were taken from ENDF/B-IV and for Al from a LASL evaluation in ENDF/B format. Data for Fe were a combination of an ORNL evaluation at lower energies, and a LASL evaluation for (n,2n) and (n,n' continuum) cross section errors. Also, the LASL evaluation for Fe combines (n,n'p), (n,n'α), and (n,n'γ) reactions into a "macropartial", which makes possible a more reasonable estimate of the uncertainty in the elastic scattering cross section. These data were then all processed by the NJOY code into the 30-neutron-group structure[18] used for the 30 x 12-group coupled neutron/gamma-ray transport calculations.

4.1 Responses of Interest

Because of the thinner inner shield, radiation effects in the inner TF coils are more critical[17] than in the outer TF coils. However, for access during maintenance the outer structure and TF coil activation are important, as opposed to the inner. Thus, for our analysis we have chosen four radiation effects in the inner TFC, and activation of the stainless steel outer dewar. Specifically, we consider

INNER SHIELD

1) neutron and gamma-ray heating in the TF coil superconductor,
2) neutron and gamma-ray dose to the MYLAR insulation in the TF coils,

3) displacements per atom (dpa) in the Cu matrix of the TF coils, and
4) transmutation of the Cu matrix.

OUTER SHIELD

1) activation of the stainless steel (SS) dewar [e.g., $^{58}Ni(n,p)^{58}Co$ or $^{56}Fe(n,p)^{56}Mn$].

A typical response function is shown in Fig. 4, where we give the neutron and gamma-ray flux-to-dose response for MYLAR. Details of all the response functions, as well as sensitivities, etc., are presented in a forthcoming report.[19] In this paper we present only selected sample results.

4.2 Procedure and Results

All forward and adjoint flux calculations were performed in S_8-P_3, using the models shown in Figs. 2 and 3, and all cross sections were processed from ENDF/B-IV. As a check on proper convergence it was verified that

$$\langle \phi^*, Q \rangle \simeq \langle \phi, R \rangle .$$

Sensitivity profiles, P_{Σ_i}, were then computed for neutron and gamma-ray interactions, as well as for gamma-ray production.

As a sample case, let us consider the total neutron and gamma-ray heating in the inner TF coil. Table I shows the integral sensitivities,

$$S_{\Sigma} = \sum_i P_{\Sigma_i} ,$$

for this response, to SS total cross sections. From this table we find the region(s) in Fig. 2 which contribute most to the sensitivity. It is worth noting that these data also give insight into the sensitivity of the response to design alterations in these regions. From Table I it is clear that the blanket SS regions 6–8 are most important. Also, it can be seen that Fe is the largest contributor to the integral sensitivities, regardless of which region is considered.

Narrowing our example further we show in Table II the component sensitivities for Fe in regions 6–8. Here the sensitivity has been divided into the gain term and loss terms (cf., Ref. 2, App. B for details)

$$P_{\Sigma_i} = - P_{\Sigma^{tot}_{i,loss}} + P_{\Sigma^{scat}_{i,gain}} .$$

Most of the net integral sensitivity is clearly due to scattering. An anomalous appearing result in Table II warrants some discussion; viz, the negative loss term for Σ_a. Because of the idiosyncracies of the transport codes, the (n,2n) and (n,3n) reactions appear as a negative component in the absorption and total cross sections, and as a positive component in the scattering matrix. Thus, by using the transport code cross-section sets for scoping sensitivity analyses, one introduces an artifice in the results. This artifice is also seen in the fact that the scattering loss term, computed by summing diagonals of the scattering matrix, is larger than the total loss term. In fusion reactor sensitivity analysis we often observe this effect, in particular when the sensitivity in the top (approximately 2.25 pJ, or 14 MeV) group is manifestly dominant.

A representative sensitivity profile is shown in Fig. 5, where the sensitivity of the TF coil heating to the Fe scattering cross section was selected. Notice the high sensitivity in the top two groups, with a subsidiary peak below 160 fJ (1 MeV). This general shape is characteristic of all the sensitivity profiles, for all responses and all materials pertaining to this EPR design.

Referring again to Table II, the low sensitivity to the gamma-ray production cross section, $\Sigma_{(n,\gamma)}$, is caused by the relatively short mean free path of the gamma-rays in SS. However, for regions closer to the TF coil the sensitivity increases monotonically.

Turning now to the B_4C component of the shield, Table III presents integral sensitivity results comparable to those of Table I for SS. Here we see that the sensitivity is highest for the outboard regions, where the neutron spectrum is softened somewhat. However, the spatial variation is not nearly as strong as for Fe (cf. Table I). Also, the ^{10}B component of the B_4C does not overwhelmingly dominate the sensitivity as does, for example, Fe in SS. As would be expected, the net integral sensitivity is in all cases negative, because almost any interaction decreases the probability of a neutron's transmission to the TF coil.

The spectrum of neutrons at the inner edge of the TF coil is of some interest, and is shown in Fig. 6. Sensitivity profiles for the B and C cross sections show the same general shape as those for Fe (Fig. 5), with a peak in the top group and another peak in the 16-160 fJ (100 keV-1 MeV) region. For ^{10}B, however, the sensitivity to the total cross section is of comparable magnitude in the two peaks, and the lower peak is much broader. This high sensitivity at the lower energy peak is due in part to the neutron spectrum, which shows this same peak at all positions in the shield regions 8-10. One can conclude that even though these lower energy neutrons have lower transmission probabilities to the TF coil, they are so prevalent in the spectrum as to be a major contributor to the flux reaching the TF coil. A more quantitative explanation of this phenomenon can be gleaned from the X and Ψ functionals in Ref. 20.

Table IV shows the individual cross section integral loss and gain terms for the ^{10}B in region 17, the region with highest sensitivity. Here the integral loss term is positive because the $^{10}B(n,2n)$ cross section is very small.

As a final example from our detailed sensitivity analysis[19] of the EPR, consider the sensitivity of heating in the TF coils to the cross sections in the TF coil region itself. The response here is in the inboard edge (first mesh interval) of the TF coil, while the sensitivity is to cross sections in the entire region 24. Table V shows a very low sensitivity to all neutron cross sections except for Cu. This is to be expected because interactions in the TF coil itself do not significantly alter the probability of a neutron contributing to heating at the inboard edge of the coil. Although it is of somewhat academic interest (because of the precision with which gamma-ray interaction cross sections are known), the relatively high negative sensitivity to Cu gamma-ray interaction cross sections is as expected. Similarly, the integral sensitivity to the gamma-ray kerma response function has been observed to be a large positive value for Cu.

5. CONCLUSIONS

Several major conclusions have been reached thus far in our sensitivity and uncertainty analyses of fusion reactors. First, the wide ranging survey calculations, using transport-code cross sections, have provided a rapid and thorough coverage of all materials and regions of potential interest. This has proven to be an effective way of eliminating the need for further analyses of many partial cross-section sensitivities. From a pragmatic viewpoint, these partials are of interest only if they provide significant contributions to the total sensitivity, and have significant errors associated with them. In the case of the EPR study, we have now processed the partials of interest into multigroup form, and are currently performing uncertainty analyses.

A major effort in sensitivity and uncertainty analyses stems from the need for detailed covariance matrices. The lack of these data in either ENDF or individual laboratory files has been a serious deterrent to complete uncertainty determinations for the EPR. On a hopeful note, however, we expect that the error file data produced for our EPR study will provide the major expected uncertainty information. Especially valuable are the data for Fe and C, materials which figure prominently in present shield design concepts. In addition, error files for ^{7}Li, ^{9}Be and possibly ^{11}B will be forthcoming shortly.

TABLE I

NEUTRON INTEGRAL SENSITIVITY, S_{Σ_T}, OF THE INNER TFC NUCLEAR HEATING RESPONSE TO THE TOTAL CROSS SECTIONS OF STAINLESS STEEL COMPONENTS

Component	Region					Total
	6-8	12	14	16	23	
Cr	-0.601	-0.212	-0.188	-0.150	-0.014	-1.168
Mn	-0.103	-0.034	-0.031	-0.026	-0.005	-0.202
Fe	-2.430	-0.868	-0.767	-0.602	-0.058	-4.775
Ni	-0.526	-0.182	-0.164	-0.140	-0.010	-1.022
Mo	-0.091	-0.033	-0.030	-0.024	-0.008	-0.187
TOTAL	-3.801	-1.330	-1.180	-0.944	-0.097	-7.375

TABLE II

PARTIAL AND NET NEUTRON INTEGRAL SENSITIVITIES OF THE INNER TFC NUCLEAR HEATING RESPONSE

TO THE Fe COMPONENT IN STAINLESS STEEL REGION 6-8

Neutron Cross Section, Σ_x	Integral Loss Term	Integral Gain Term	Integral Net, I_n
Σ_a	- 0.17	----	0.17
Σ_s	12.98	10.[illegible]	- 2.[illegible]
Σ_T	12.[illegible]1	10.33	- [illegible]
$\Sigma_{(n\to\gamma)}$	---	0.014	0.0[illegible]

TABLE III

NEUTRON INTEGRAL SENSITIVITY, S_T, OF THE INNER TFC NUCLEAR HEATING RESPONSE TO THE TOTAL CROSS SECTIONS OF B_4C COMPONENTS

Component	Region				Total
	11	13	15	17	
^{10}B	−0.221	−0.269	−0.375	−0.543	−1.409
C	−0.166	−0.190	−0.243	−0.196	−0.796
TOTAL	−0.387	−0.459	−0.618	−0.739	−2.205

TABLE IV

PARTIAL AND NET NEUTRON INTEGRAL SENSITIVITIES OF THE INNER TFC NUCLEAR HEATING RESPONSE TO CROSS SECTIONS OF THE ^{10}B COMPONENT IN B_4C REGION 17

Cross Section, Σ_x	Integral Loss Term	Integral Gain Term	Integral Net, S_{Σ_x}
Σ_a	0.41	——	-0.41
Σ_s	0.81	0.68	-0.13
Σ_T	1.22	0.68	-0.54
$\Sigma_{(n,\gamma)}$	——	0.0009	0.0009

TABLE V

NEUTRON AND GAMMA-RAY SENSITIVITIES, S_{Σ_T},
OF THE FIRST INTERVAL OF TFC NUCLEAR HEATING RESPONSE
TO TFC MATERIAL CROSS SECTIONS

Component	Neutron S_{Σ_T}	Gamma S_{Σ_T}
Cr	-0.000	-0.020
Mn	-0.001	-0.003
Fe	-0.003	-0.086
Ni	0.001	-0.019
Mo	-0.003	-0.004
Cu	-0.065	-0.150
Nb	-0.011	-0.015
Ti	0.001	-0.004
He	0.008	-0.002
TOTAL	-0.073	-0.306

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Fig. 2. One-dimensional model of the FER inner blanket/shield.

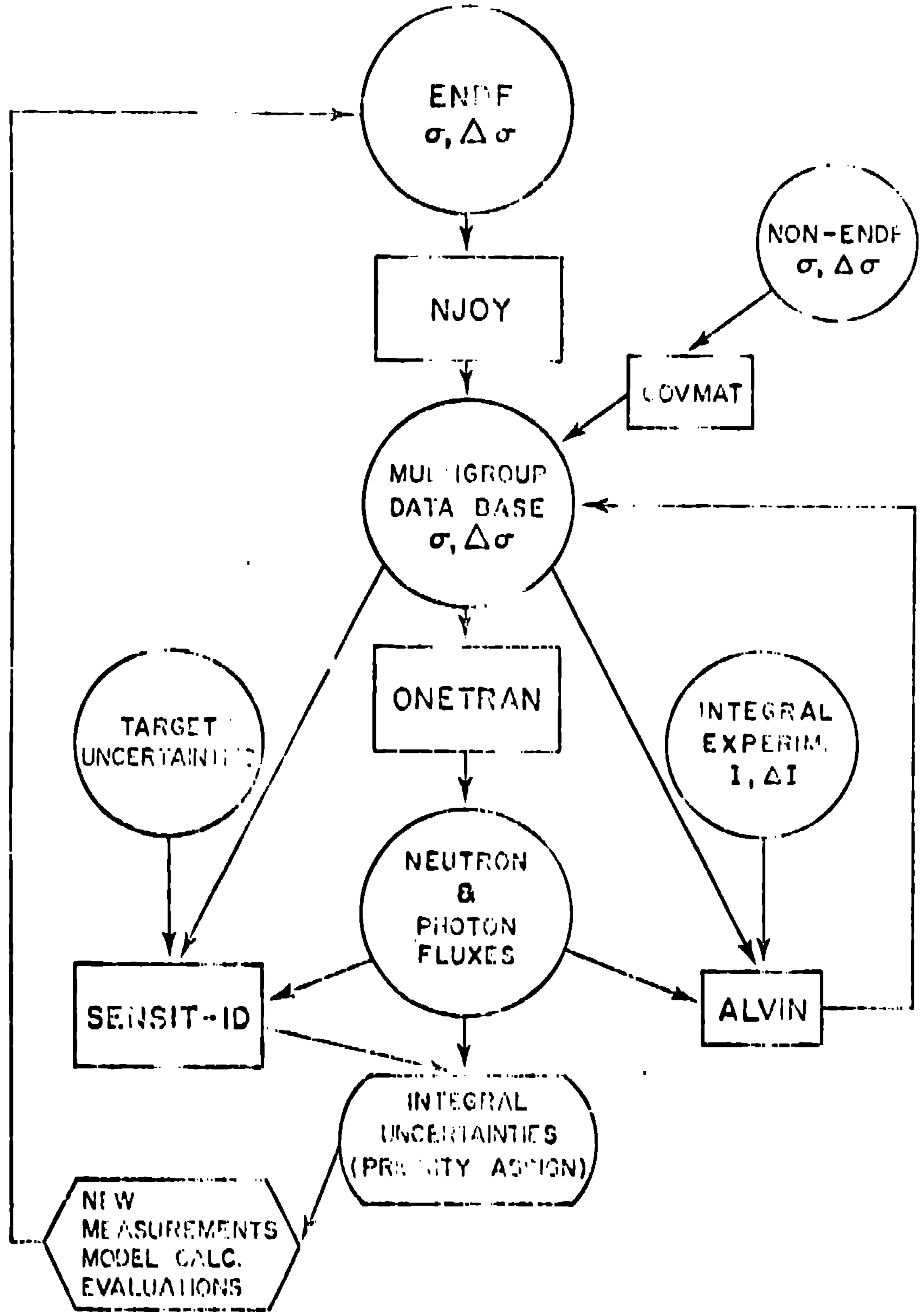
Fig. 3. One-dimensional model of the FER outer blanket/shield.

Fig. 4. Response function for absorbed dose in MYLAR insulation in the TF coils.

Fig. 5. Sensitivity of the maximum neutron plus gamma-ray heating in the TF coils to all scattering cross sections of Fe.

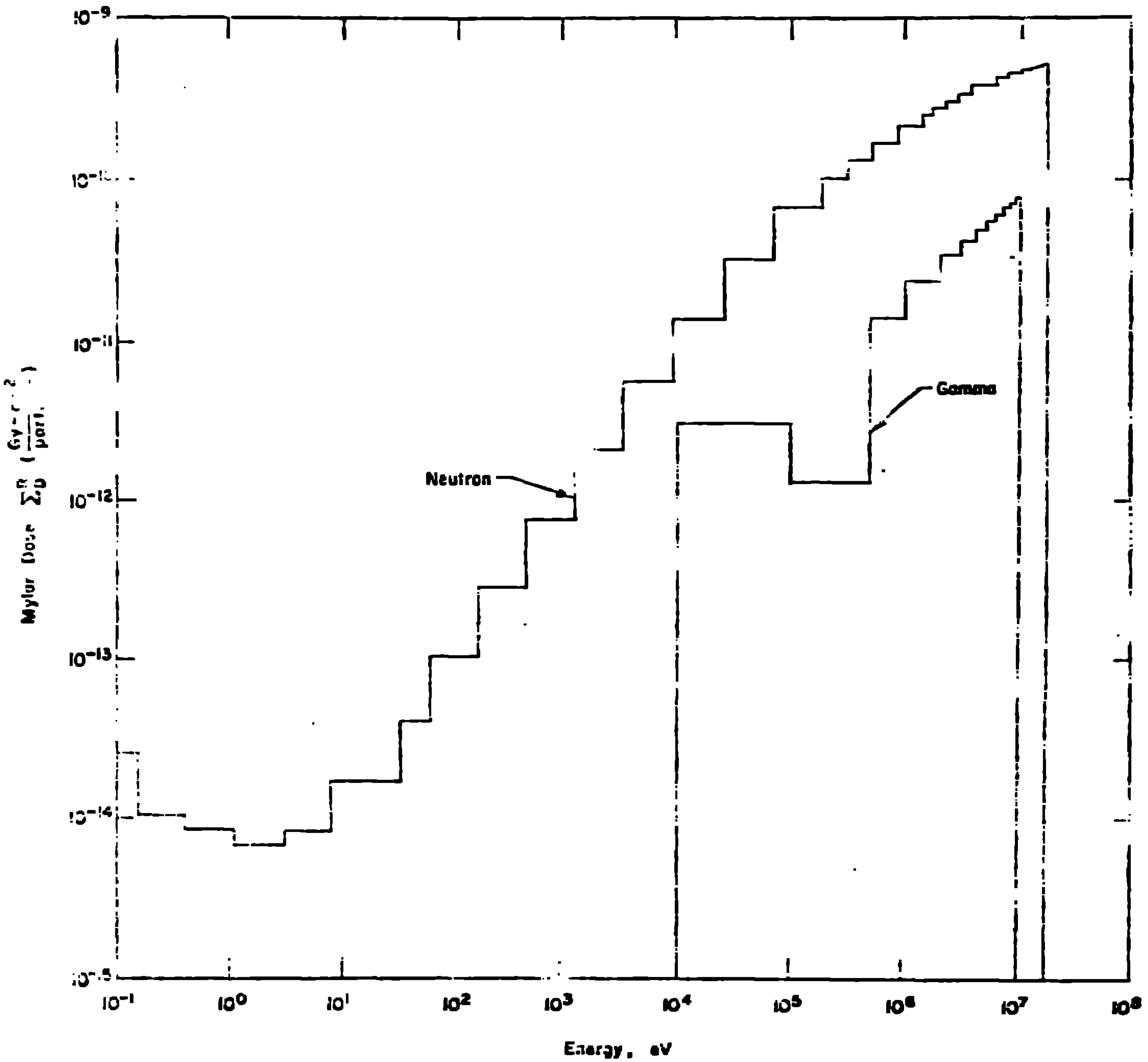
Fig. 6. Neutron flux spectrum at the inner edge of the TF coil (radius = 3.47 m).

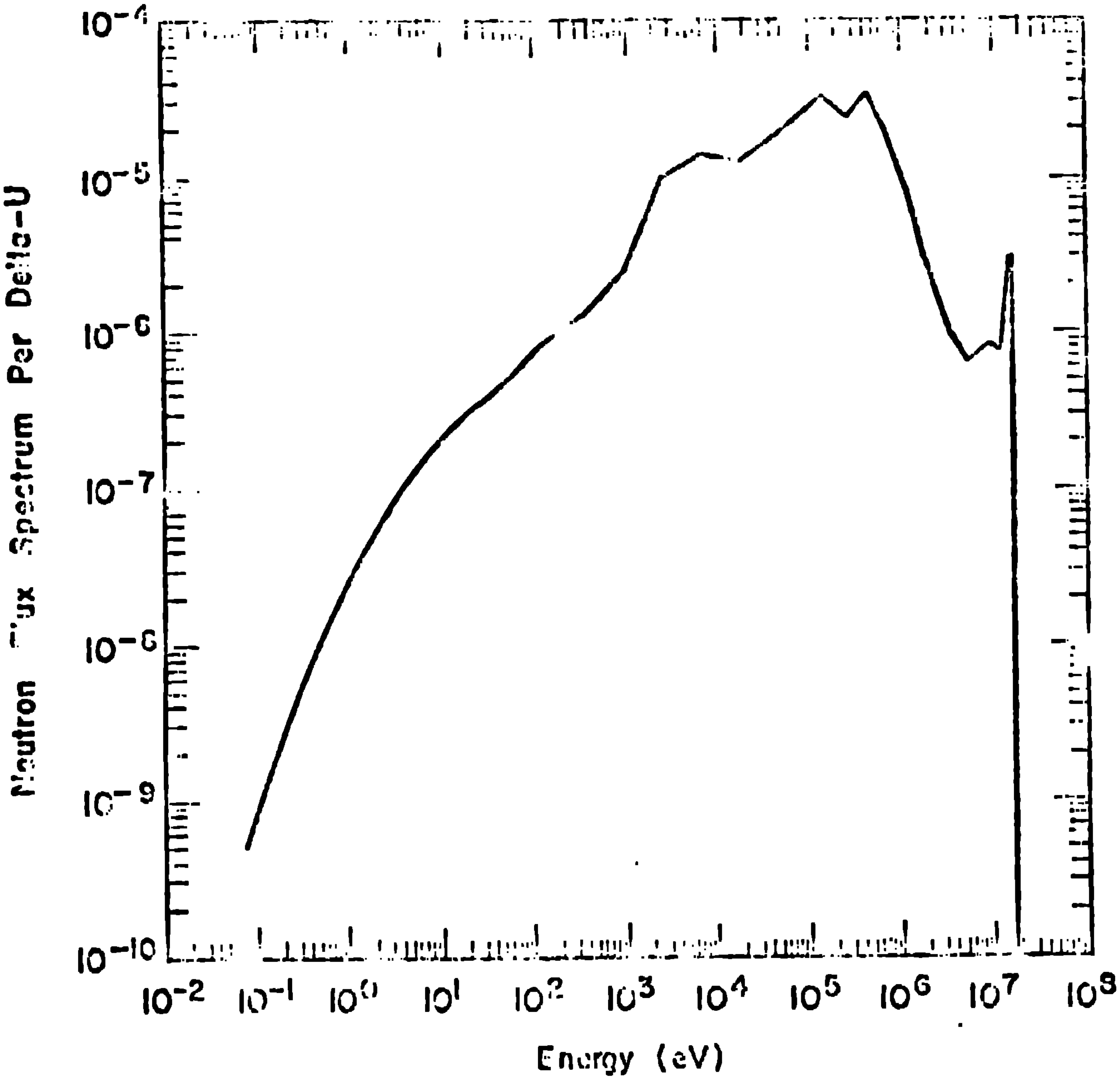
LASS SYSTEM FLOW DIAGRAM



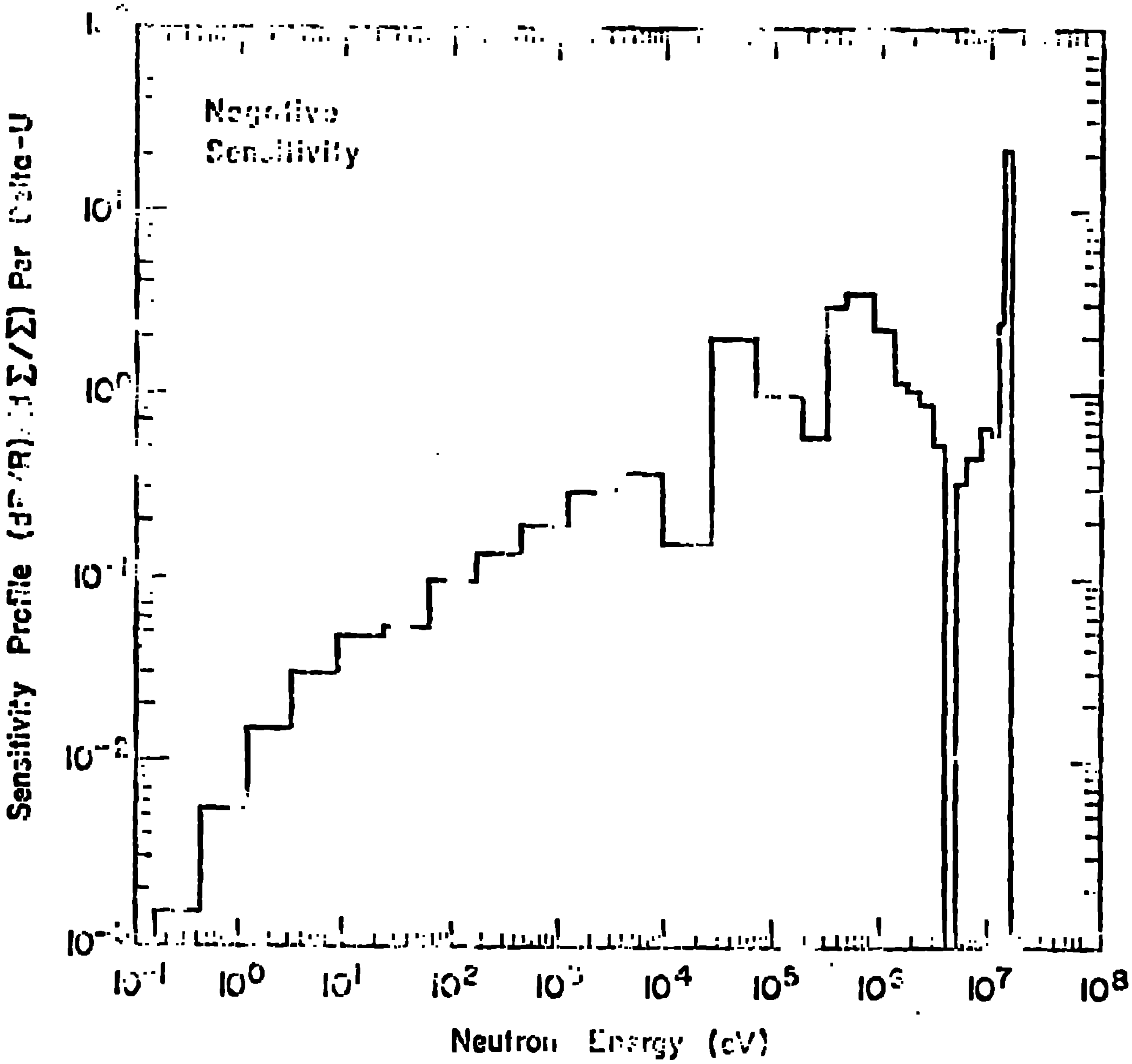
Region No.	Region Material	Radius (cm)
		0.0
1	PLASMA	210.0
2	VACUUM	240.0
3	1st WALL S. S.	241.0
4	1st WALL S. S.	242.0
5	1st WALL S. S.	244.0
6	BLANKET S. S.	254.0
7	BLANKET S. S.	264.0
8	B[illegible] S. S.	272.0
9	VACUUM	27[illegible].0
10	S. S.	276.0
11	B_4C	281.0
12	S. S.	2[illegible]1.0
13	B_4C	297.0
14	S. S.	307.0
15	B_4C	315.0
16	S. S.	325.0
17	B_4C	33[illegible].0
18	S. S.	335.0
19	TFC INNER S. S.	337.0
20	VACUUM, TUBING, ETC.	339.8
21	THERMAL SHI[illegible]	340.7
22	VACUUM	343.2
23	TFC [illegible] (S. S.)	345.7
24	TFC	350.7
25	S. S.	355.7
26	TFC	360.7
27	TFC	416.2
28	SUPPORT CYLINDER	44[illegible].0
29	OHC	46[illegible].0

REGION NO.	REGION MATERIAL	RADIUS (cm)
		0.0
1	[illegible]	210.0
2	VACUUM	240.0
3	1st WALL S. S.	241.0
4	1st WALL S. S.	242.0
5	1st WALL S. S.	244.0
	BLANKET S. S.	254.0
7	BLANKET S. S	264.0
8	BLANKET S. S.	272.0
9	VACUUM	275.0
10	S. S.	276.0
11	GRAPHITE	281.0
12	GRAPHITE	286.0
13	GRAPHITE	291.0
14	S. S.	296.0
15	LEAD MORTAR	301.0
16	LEAD MORTAR	311.0
17	LEAD MORTAR	321.0
18	LEAD MORTAR	331.0
19	LEAD MORTAR	341.0
20	LEAD MORTAR	351.0
21	LEAD MORTAR	356.0
22	LEAD MORTAR	361.0
23	S. [illegible]	366.0
24	[illegible]	370.0
25	VACUUM	440.6
26	TFC DEWAR	442.6
27	VACUUM	445.1
28	THERMAL SHIELD	446.0
29	VACUUM	448.5
30	T[illegible] BOBBIN	451.0
31	TFC	45[illegible].0
32	TFC	461.0
33	TFC	466.0
34	TFC	521.5
35	[illegible]	529.8
36	TFC [illegible]	532.3
37	VACUUM	534.8
38	THERMAL SHIELD	535.7
39	VACUUM	538.2
40	TFC DEWAR	540.2


Mylar Dose
Neutron
Gamma
10^{-1}
10^{0}
10^{1}
10^{2}
10^{3}
10^{4}
10^{5}
10^{6}
10^{7}
10^{8}
Energy, eV
10^{-9}
10^{-11}
10^{-12}
10^{-13}
10^{-14}



Neutron Flux Spectrum Per Delta-U
10^-4
10^-5
10^-6
10^-7
10^-8
10^-9
10^-10
10^-2
10^-1
10^0
10^1
10^2
10^3
10^4
10^5
10^6
10^7
10^8
Energy (eV)



Negative
Sensitivity
Neutron Energy (eV)