

THE APPLICATION OF SENSITIVITY THEORY TO CROSS SECTION
ADJUSTMENT AND INTEGRAL EXPERIMENT ANALYSIS

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A thesis submitted for the degree of Ph.D in the
Faculty of Science and Engineering

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November 1984



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SYNOPSIS

Sensitivity theory has been used to computationally design and analyse two integral experiments, associated with a non fissile and a fissile fusion reactor. The sensitivities of certain performance parameters have been obtained for the reactors and their associated integral experiments and these have been used in conjunction with some assumed covariance matrices to perform a standard uncertainty analysis. In addition, the two integral experiments have been remodelled using three dimensional methods to correctly account for neutron leakage. Ultimately the purpose of the above is to adjust cross sections so that we can determine the behaviour of the fusion reactors more accurately. The work presented here represents the first (computational) half that is needed for such a cross section adjustment. With this information, most of the partials in the 100 neutron energy group structure of the ENDF/B-4 cross section set of the LI6, LI7, F19, PB, FE, U238 and U235 isotopes can be adjusted.

In chapter 1 the principles of fusion are given and the benefits of ion beams are discussed. In addition, the benefits of using a coupled fission fusion reactor are outlined.

In chapter 2 the theory and ideas behind sensitivity analysis is discussed. The chapter starts with an outline and history of sensitivity theory and next a derivation of the sensitivity function is presented. Finally, a general sensitivity theory section is given, which discusses the meaning of the adjoint

flux, the meaning of the sensitivity equation, and the sensitivity methodology and sensitivity definitions.

In chapter 3 the calculational tools are discussed with an outline of the physics and mathematics behind each computer code. In addition, the cross section data sources and the codes used to extract the cross section information are outlined.

Chapter 4 gives the theory and ideas behind the cross section adjustment procedure. The relevant equation is derived from first principles and discussed. Also included here is an outline of integral experiments and their uses.

Chapter 5 gives a summary of the physical and mathematical model designs of the two fusion reactors, whilst chapter 6 gives a detailed breakdown of the responses and sensitivities for the two reactors. These results give information as to the behaviour of the two reactors and forms the basis for the designing of the integral experiments.

Chapters 7 and 8 give the methods behind the design and construction of the non fissile and fissile integral experiments, respectively. The neutronic changes recorded in passing from the reactor design to the integral experiment are given and each integral experiment is compared to its fusion reactor using some important sensitivity profiles to see how and why the radiation environments differ.

Chapter 9 details the uncertainty analysis of the reactors and experiments. It discusses uncertainty analysis in general and derives the necessary equation. The analysis presented here deals with the standard cross section uncertainties only and does

not deal with the energy or angular distributions of secondary neutrons or the uncertainties introduced by modelling and computational methods. The covariance matrices were constructed from information contained in the ENDF files and by choosing a range of suitable correlation values. The total uncertainty in a certain performance parameter is shown for the different strengths of chosen correlation. The uncertainty analysis shows that most of the errors are below 10%.

In chapter 10 the results from a 3-dimensional modelling on the two integral experiments is presented as well as the sensitivities of $\text{Ni}(n,p)$ and $^{232}\text{Th}(n,f)$ foils placed at various spatial positions in the assemblies. Comparisons are made between the sensitivities obtained by the 3-dimensional and 1-dimensional computational codes and it is seen that the similarity between the profiles is quite good. This leads to the important result that the 1-dimensional radiation environment is roughly the same as that calculated by the 3-dimensional code. Thus foil measurements made by an experimentalist at the central positions of the integral experiments can be used with confidence with the sensitivities obtained from the 1-dimensional code (which is very cheap and easy to run) for the cross section adjustment scheme.

Chapter 11 lists the main conclusions and recommendations and briefly summarises all the work performed in this thesis.

TO MY PARENTS,
FOR ENCOURAGING ME ALL THESE YEARS.

ACKNOWLEDGMENTS

I should like to thank Dr. T.D Beynon for initially suggesting the field of work and for many interesting discussions during the course of the project.

I am grateful to Dr. E.H Smith for help during the earlier part of this work and to Dr. D.J Picton for help during the latter part, particularly with my many computational problems. I would like to thank Dr. N.P Taylor for help at various times and to Dr. B.S Sim for help during most of the time.

I am indebted to the Dept. of Physics film analysis group for the use of there IBM 4341 and to Dr. L Lowe and to Dr. F McDonald for help during the year. Most particularly I would like to acknowledge the help and assistance of the film analysis group computer operators. They have shown great tolerance, even during the 'difficult periods' when it may have seemed that the submission of an1, an2, an3.... jobs would never cease.

Finally, I would like to acknowledge the financial support of the Science and Engineering Research Council during the course of this work.

CONTENTS

Chapter 1 Introduction

1.1 Introduction	1
1.2 The fusion process	1
1.3 Magnetic and inertial confinement	3
1.4 The mixed fission fusion reactor	7

Chapter 2 General Sensitivity Theory

2.1 History and developement of sensitivity theory	11
2.2 Derivation of the Sensitivity Function	14
2.3 Overview and Definitions	19
2.4 The Adjoint Flux	23
2.5 Discussion and meaning of the Sensitivity Equation	24

Chapter 3 Modelling codes and data sources

3.1 Introduction	28
3.2 ANISN	28
3.3 MORSE	31
3.4 SWANLAKE	37
3.5 ENBAL2	41
3.6 Cross sections and sources	43

Chapter 4 Cross section adjustment and Integral Experiment theory

4.1 Introduction	48
4.2 Derivation of adjustment formula	50
4.3 Discussion of adjustment formula	54
4.4 Integral experiment analysis	56
4.5 Cross section adjustment results	60

Chapter 5 Description of real and computational non fissile and fissile reactor

5.1 Introduction	63
5.2 Non fissile real reactor description	63
5.3 Non fissile reactor computational description	65
5.4 Fissile reactor real description	67
5.5 Fissile reactor computational description	70

Chapter 6	Results from Sensitivity Analysis performed on non fissile and fissile reactor	
6.1	Introduction	77
6.2	Reactor performance parameters	77
6.3	Results from sensitivity analysis on non fissile reactor	80
6.4	Results from sensitivity analysis on fissile reactor	87
Chapter 7	Results and design of the non fissile integral experiment	
7.1	Introduction	116
7.2	Overview	116
7.3	Design of the non fissile integral experiment	118
7.4	Further tuning between profiles	129
Chapter 8	Results and design of the fissile integral experiment	
8.1	Introduction	152
8.2	Overview	152
8.3	Design of the fissile integral experiment	154
8.4	Further tuning between profiles	166
Chapter 9	Theory and Results of Uncertainty Analysis	
9.1	Introduction	185
9.2	Basic covariance formulation	188
9.3	Estimation of cross section covariance matrices	191
9.4	Source of cross section uncertainties	195
9.5	Discussion and results of uncertainty analysis	195
Chapter 10	Three dimensional Sensitivity	
10.1	Introduction	207
10.2	Overview and methodology	209
10.3	Non fissile 3-dimensional sensitivity	212
10.4	Fissile 3-dimensional sensitivity	216
Chapter 11	Conclusions and further work	233
Appendix 1	Modified SWANLAKE input data	241
Appendix 2	Some cross sections	246
Appendix 3	The meaning of the adjoint flux	250
Appendix 4	Some definitions and explanations	255

Figures and tables are located at the end of the chapters
in which they are referred to.

CHAPTER 1

INTRODUCTION

1.1 INTRODUCTION

The first practical demonstration of a working fusion device was presented on the 1st November 1952. A mixed fission-fusion weapon, with an explosive power equal to that of 10 megatonnes of TNT, was detonated over the Eniwetok atoll, in the Marshall Islands. Since that time and particularly in the last 20 or so years, the amount of interest and research into the harnessing and utilisation of controlled fusion has increased quite dramatically. It is the purpose of this chapter to detail the physical processes involved in fusion and to discuss how they can be incorporated in a mixed fission-fusion reactor. The advantages of such a system are also examined.

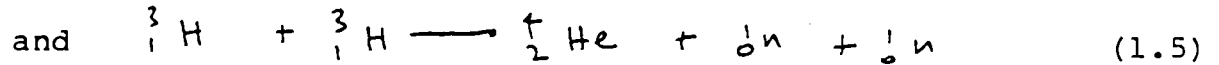
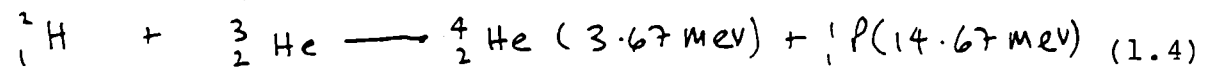
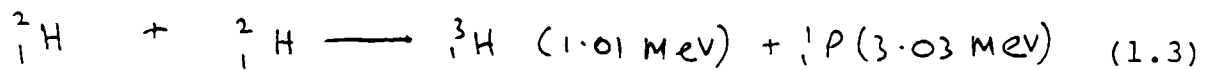
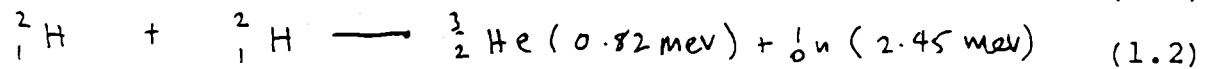
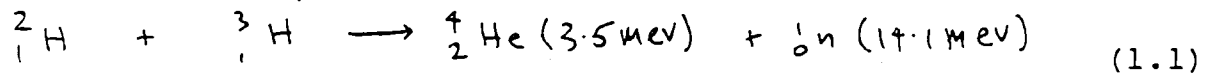
1.2 THE FUSION PROCESS

The basis of a controlled thermonuclear reactor is a fusion reaction in which two light nuclei are fused together to form a heavier nucleus. The reaction is most usually exothermic with the release of a large amount of energy. Unlike the fission process, in which the neutron has no Coulomb barrier^{*} to overcome before entering the fissile nucleus, the two nuclei in the fusion process are charged ions and large amounts of energy must be imparted to them in order to produce a reaction. Therefore, such fusion reactions can only occur at high temperatures and densities where the physical state of matter becomes a plasma, a collective body of electrons and light ions.

* SEE PAGE 255

The most promising set of fusion reactions involve hydrogen isotopes, since these will have the smallest Coulomb barrier.

These reactions are;

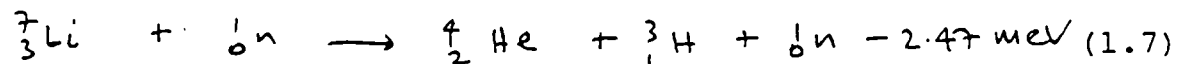
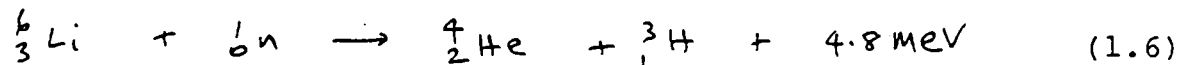


and these are the five reactions requiring the lowest plasma temperatures.* The rate of a thermonuclear reaction depends on the product of the cross section and relative velocity of the two interacting species averaged over the velocity distribution. The energy gain per fusion reaction has been tabulated (Keefe 1982) for a variety of fusion reactions in table 1. It is seen that the DT reaction requires the least initial energy investment ($kT=4 \text{ KeV}$) and provides the greatest energy multiplication, 4400. It must be remembered that in an operating reactor, these gains will be much reduced due to the many inefficiency factors in any practical system, and the need to operate at energies substantially above the threshold shown. Therefore the DT reaction will pave the way for the first set of fusion reactors.

For fusion to work as a practical source of energy, for the DT mixture, physics demands that two conditions be satisfied simultaneously. 1) The temperature should be in the region $kT=20 \text{ keV}$ and. 2) That the hot plasma should be adequately confined, that is, it should satisfy the Lawson criterion (Lawson 1957). This states that the product, $n\tau$, of the number density n and the confinement time τ should lie close to $10^{15} \text{ s cm}^{-3}$.

These conditions ensure that a sufficiently large number of

interactions occur , and so enough fuel will be burned, in the time that the plasma temperature and density are both adequately high to produce useful amounts of output energy. For a reactor operating on the DT process, the deuterium is in abundance in sea water in the form of heavy water (D₂O) and can be extracted chemically. However, tritium is an unstable isotope with a half life of about 12 years, and so not occurring in nature will have to be obtained from other sources. The most popular method, and the one considered in all this work, is to breed tritium directly in the fusion reactor. Two such tritium breeding reactions occur in the natural isotopes of lithium. These are,



The lithium-6 and lithium-7 isotopes constitute 92.58% and 7.42% of natural lithium, respectively. Reaction (1.6) has a high cross section for neutrons with energies in the thermal region, whilst (1.7) has a high cross section for fast neutrons, and a threshold at 2.8 Mev.

A reactor operating on the DT fusion process has been used as the basis for all this work. In particular for all the subsequent neutronics calculations using particle transport codes, a 14.1 MeV neutron source was used as the energy spectrum. It is fortunate that, because the tritium is bred in the reactor, the raw materials actually consumed are deuterium and lithium, both of which are abundant.

1.3 MAGNETIC AND INERTIAL CONFINEMENT

Research into magnetically confined fusion is now at such an advanced state mainly due to it becoming a declassified subject in 1958. In contrast, inertial confinement fusion was only

declassified in 1972, though because of the weapons link, some target designs still remain classified. Magnetic confinement provides a magnetic pressure that constrains the kinetic motions of the plasma whilst inertial confinement compresses and heats up a fuel pellet in a very short time, such that a significant fraction of the plasma burns before disassembly occurs.

The Lawson criterion, $n\tau$, is roughly the same for both confinement systems but the individual values of n and τ needed is dramatically different. Reactors based on electromagnetic schemes are likely to operate with an n value (total ion density) in the range 10^{20} to 10^{22} m^{-3} and a τ value (confinement time) of 10 to 0.1 sec. In contrast, the n and τ quantities for an inertial system are about 10^{31} to 10^{32} m^{-3} and 10^{-10} to 10^{-11} sec, respectively.

Descriptions of systems using magnetic and inertial confinement may be found in detail elsewhere, for example, Ribe(1975) and Ribe(1977). As an example of a magnetically confined reactor, the Tokamak contains its plasma in a toroidal shape, within a toroidal first wall container. The plasma forms the secondary of a transformer, with the primary coils being on the outside of the reactor structure. A high current passed through these coils causes an induced current to flow in the plasma, which generates Ohmic heating, and contains the plasma by the magnetic field so produced. The plasma in this state is unstable and so a coil is wound around the torus to produce an additional toroidal field, which induces more stability. The resultant magnetic field lines, a combination of these two, are helical in shape around the plasma. The coils in a Tokamak reactor need to handle pulses of extremely high currents and are

usually constructed of superconducting material and so must be maintained at temperatures close to absolute zero with the use of cryogenics. This low temperature in the coils is maintained only a short distance (typically 2 or 3 metres) from the first wall. Hence, very high temperature gradients can result from this, and coupled with the high neutron flux, causes an environment in which few structural materials will function. The temperature gradients in an inertial reactor are much reduced since there is no need for any magnetic system with its associated cryogenics. Such a fusion reactor operating by inertial confinement, would be cylindrical in shape, with a first wall, a blanket region and some type of radiation shield. Glass encapsulated pellets of a frozen deuterium-tritium mixture are propelled into the mid reactor chamber from the top. High power lasers or intense beams of heavy ions are focussed onto each of the pellets. This absorption of energy causes ablation of the surface of the pellet, which in turn invokes Newton's third law and causes a pressure wave that implodes the pellet. This implosion heating raises the temperature at the centre of the pellet sufficiently high to reach ignition temperature for a thermonuclear burn. The burn spreads outward through the pellet until it reaches the surface, when the pellet then disassembles. The heat generated in the reactor is used to raise steam for the turbines in the usual way.

Although the magnetic confinement scheme is at such an advanced state, and indeed scientific breakeven is expected this decade, from either the Tokamak fusion test reactor (TFTR) at Princeton or the Joint European Torus (JET) at Culham, inertial confinement does seem to offer certain advantages.

A reactor based on inertial confinement will have considerably less engineering demands placed on it than that of a magnetic Tokamak system (Beynon 1982). As an example, the containment vessel design must handle the high wall loading and the neutron and X-ray flux incident on its surface. For the inertial case, the design of such a system is largely independent from the thermonuclear ignition and burn of the plasma, resulting in a wide choice of design, scale, shape, refueling facilities, blanket and structural material. For the magnetic reactor, however, such as a Tokamak, the above parameters must be consistent with the constraints associated with being in the toroidal and poloidal field coils and operating in a high magnetic field whose source must be shielded. In short, the magnetic field holding the plasma is dependent on the physical characteristics of the containing vessel, whereas for the inertial system no such complication occurs.

The means to achieve inertial confinement can be broken up into four types of beam; laser, electron, light ion and heavy ion. Both reactors discussed in this work use heavy ion beams to initiate fusion. Although the neutronics calculations do not depend on the method by which fusion occurred the reactors have been designed specially to use heavy ion beams. This comes about because of the physical characteristics of the two reactor systems.

One immediate advantage of using ion beams is that many decades of successful design and operation of ion accelerators has occurred. Therefore, a technological base now exists for extrapolation to inertial systems for scientific and engineering breakeven. Also the efficiency of lasers (typically 1%) is much

lower than the efficiency of accelerators (typically 25-30%) in converting electrical energy into a beam of particles. Electron beams can be produced with high efficiency from diodes, driven by discharging a coaxial line, which is both cheap and simple. However, the electron range in targets is too long for this type of beam to be of any use. Following the invention of several types of ion diode, some research has gone into the possibility of using light ions accelerated across a single gap. As with lasers, the operation at high repetition rates represents difficulties. Additionally for light ions there is the difficulty of focussing the beam down to the small size required to ignite a target pellet. With the above considerations in mind, it appears that inertial confinement fusion remains one of the most promising.

1.4 THE MIXED FISSION FUSION REACTOR

The hybrid fission fusion reactor, a system containing both fissile and fusion fuels, offers the opportunity for fusion to make an early and significant contribution to the world's energy needs (Rose 1981 and Abdel-Khalik 1983). The idea of using non fission sources to produce fissile material by transmutation of the fertile isotopes of thorium and uranium dates back to the 1950s (Lawson 1955).

The idea is to surround the fusion reaction region with a blanket of fertile material so that the fusion neutrons would convert the fertile isotopes ^{232}Th or ^{238}U to ^{233}U or ^{239}Pu , respectively. For a DT fusion reactor, the blanket should also allow for tritium breeding to make the reactor self sufficient. The fissile material produced in the hybrid can be burned in fission reactors or it can be partially burned in situ releasing

considerably more energy than that generated by fusion. So hybrids can nicely couple the 'fast neutron rich but energy poor' DT fusion process with the 'energy rich but neutron poor' fission process. Neutron multiplication in the hybrid blanket through $(n,2n)$, $(n,3n)$, and $(n,\text{fission})$ reactions makes it possible for the total number of breeding captures per DT fusion neutron to be considerably larger than unity. This means that even if hybrids are made to breed their own tritium fuel, large quantities of fissile materials can still be produced per unit of fusion energy.

The fusion driven hybrid system has certain generic neutronic characteristics that are rather different from a fission reactor. While fission breeders produce 2.5 to 3 neutrons per fission, 1 neutron is required to sustain the reaction while a second is used to replace the fissile atom consumed. So less than one neutron per fission reaction remains to account for leakage, parasitic losses and to produce a net surplus of fissile material. Fusion hybrid reactors on the other hand, have a subcritical fission blanket and the need to sustain the chain reaction is absent. Also, very high energy neutrons can yield 4 to 5 neutrons per fission, so affording a cascading effect of neutrons for fuel production. These generic differences account for the greater fuel production potential of fusion fission systems as compared to fission systems alone.

The most attractive feature of the hybrid concept is that it may allow fusion to become a viable source of energy much sooner. The fission energy produced in the supported fission reactors and in the hybrid blanket itself makes it possible to relax the fusion gain requirements in the hybrid.

As will be seen in chapter 8, one of the integral experiments that has been designed is of a cylindrical hybrid fission fusion reactor. This reactor has been designed (Sim 1984) specifically to utilise the relaxation of pellet gain requirements by an increased energy multiplication for a fission fusion system. The pellet gain used in this reactor was only considered to be a factor 8. Contrast this with HIBALL (Badger 1981), which is the other reactor which has had an integral design made of it. This contains no fissile nuclides, and is a pure fusion reactor with a pellet gain of 83. However, this does not seem realistic, since target gains are turning out to be lower than originally hoped (Lawson 1983).

The fission fusion hybrid can be used for either breeding fissile fuel or for energy production. As far as breeding fissile fuel is concerned, the fission fusion hybrid should far exceed the capability of the fast breeder (LMFR) as a fuel source (Youseff 1979). In this way the hybrid has a high potential as a means to greatly extend energy from uranium sources.

	ENERGY RELEASE PER FUSION (MeV)	THRESHOLD PLASMA TEMPERATURE (keV)	MAXIMUM ENERGY GAIN PER FUSION
${}^2_1\text{H} + {}^3_1\text{H} \rightarrow {}^4_2\text{He} + \text{N}$	17.6	4	4400
${}^2_1\text{H} + {}^2_1\text{H} \rightarrow {}^3_2\text{He} + \text{N}$	3.2	50	70
${}^2_1\text{H} + {}^2_1\text{H} \rightarrow {}^3_1\text{H} + \text{P}$	4.0	50	80
${}^2_1\text{H} + {}^3_2\text{He} \rightarrow {}^4_2\text{He} + \text{P}$	18.3	100	180
${}^6_3\text{Li} + \text{P} \rightarrow {}^3_2\text{He} + {}^4_2\text{He}$	4.0	900	6
${}^6_3\text{Li} + {}^2_1\text{H} \rightarrow {}^7_3\text{Li} + \text{P}$	5.0	>900	6
${}^6_3\text{Li} + {}^2_1\text{H} \rightarrow {}^3_1\text{H} + \text{P} + {}^4_2\text{He}$	2.6	>900	3
${}^6_3\text{Li} + {}^2_1\text{H} \rightarrow {}^4_2\text{He} + {}^4_2\text{He}$	22.0	>900	22
${}^7_3\text{Li} + \text{P} \rightarrow {}^4_2\text{He} + {}^4_2\text{He}$	17.5	>900	18
${}^{11}_5\text{B} + \text{P} \rightarrow {}^4_2\text{He} + {}^4_2\text{He} + {}^4_2\text{He}$	8.7	300	30

Table 1. Some fusion reaction energy gains.

CHAPTER 2

GENERAL SENSITIVITY THEORY

2.1 HISTORY AND DEVELOPEMENT OF SENSITIVITY THEORY.

At least as regards radiation shielding, cross section sensitivity analysis started life as an attempt to quantify the reasons for the accuracies requested in cross section measurements. Much of the work, particularly in the early days, has come from the Radiation Shielding division at Oak Ridge, from where the one dimensional sensitivity analysis code, SWANLAKE, was acquired. An obvious technique for finding the sensitivity of a detector response to a given cross section phenomenon is to calculate the response for two different sets of cross sections, and from the resultant obtain the functional derivative of the response. This 'brute force' or direct substitution approach has been used since the subject of cross section sensitivity was first raised. Hansen and Sandmeier (1967) in their early work use this method to calculate the effects of high energy neutron penetration in air due to nitrogen cross sections.

The first attempt to estimate the change in a result due to changes in input data, without resorting to recalculation, was made by Usachev (1956) when he proposed a perturbation theory to calculate reactivity changes due to cross section changes in a critical reactor system. He later developed this theory (Usachev 1964) so that changes in other parameters, such as breeding ratio, in a fast reactor could be found and this was later

extended by Gandini (1967) so that an even larger range of functions could be studied.

As far as fusion work is concerned, the direct method has been employed by Tobias and Steiner (1973(a),1973(b),1974) to observe the change in the tritium breeding in a breeding blanket when gross changes are made to the $6\text{Li}(n,\alpha)\text{T}$, $7\text{Li}(n,\alpha n')\text{T}$ and $\text{Nb}(n,2n)$ cross sections. Markowskii et al (1975) also used this method in fissile and non fissile blankets to look at the sensitivity of fission and tritium breeding rates to the secondary neutron spectra arising from $^{238}\text{U}(n,\text{fission})$ and $7\text{Li}(n,n' \alpha)\text{T}$ reactions.

The main drawback in the direct method is that large changes in cross sections must be made to ensure that the change in response is considerably larger than any computational fluctuations. It is also very expensive in computing time to investigate individual changes in large numbers of cross sections.

It was therefore recognized early in the game that it would be advantageous to adapt linear perturbation theory to finding cross section sensitivities. Both Bartine et al (1974a) and Gerstl et al (1975) have discussed the problem in some detail and they and others(eg. Alsmiller et al (1974), Simmons et al (1976)) have presented results of interest in fusion reactor design, using their own sensitivity codes, SWANLAKE and SENSIT-1D, respectively. With either of these codes it is only necessary, as will be shown, to run a forward and an adjoint calculation, obtain the fluxes and analyse the sensitivity of a response (where the source in the adjoint run is the macroscopic response activation rate) to any cross sections.

The whole game of sensitivity theory in its present version has come about really from the work achieved by the original people at Oak Ridge by Bartine, Mynatt and Oblow in 1973. Since then, methods and definitions have been refined and improved. In the last six years or so much active work has occurred in the evaluation of uncertainties and the formatting and acquisition of covariance files. In particular, in the last three years the area has seen very active work in integral experiment analysis and the way in which it can be coupled to differential data to subsequently improve error bands. There are a large number of papers on sensitivity analysis and related subjects, of which only a few will be mentioned here.

Much work has come from Oak Ridge ;Alsmiller (1974), Oblow (1976), Weisbin (1975), and Alsmiller (1976) are all papers which deal with general sensitivity and its applications. Bartine (1973) and Oblow (1973) give detailed derivations of the sensitivity function and discuss the meaning of sensitivity, and Oblow (1975) gives an alternative derivation of the sensitivity function in terms of functional derivatives. Following on from Oblows paper, Dubi and Dudziak (1981) obtain the higher order terms in the sensitivity equation by using a differential approach. Gerstl (1977) gives the first detailed uncertainty analysis work using sensitivity profiles and covariance matrices. The more recent references deal with two dimensional calculations (Santoro 1981) and the accuracy of one dimensional calculations (Seki 1979). Cross section uncertainty analysis has been extensively developed over the years and Weisbin (1978), Reupke (1978), Gerstl et al (1980), Gerstl (1980) and Youseff (1982) give good accounts of this.

2.2 DERIVATION OF THE SENSITIVITY FUNCTION

The mathematical derivation of the sensitivity equation is now presented. In the next section the ideas behind sensitivity theory and the meaning of the sensitivity equation are explained. This derivation is taken from a variety of sources, mostly that of Bartine, Alsmiller, Oblo (1973) but also from Reupke (1978) and Bartine, Oblo, Mynatt (1974b). All this work considers steady state conditions only, so the derivation begins with the time independent Boltzmann equation for a general medium containing both a fixed and a fissile source.

$$\begin{aligned} \underline{r} \cdot \nabla \phi(r, E, \Omega) + \Sigma_t(r, E) \phi(r, E, \Omega) &= S(r, E, \Omega) \\ + \iint dE' d\Omega' \Sigma_s(r, E' \rightarrow E, \Omega' \rightarrow \Omega) \phi(r, E', \Omega') & \\ + \frac{\chi(E)}{4\pi} \iint_{4\pi} dE' d\Omega' \nu(E') \Sigma_f(r, E') \phi(r, E', \Omega') & \end{aligned} \quad (2.1)$$

The symbols are defined as,

$\phi(r, E, \Omega)$ = time independent angular flux,

$\Sigma_t(r, E)$ = total macroscopic cross section at space point r for particles of energy E .

$\Sigma_s(r, E' \rightarrow E, \Omega' \rightarrow \Omega) dE d\Omega$ = differential macroscopic scattering cross section which describes the probability per unit path that a particle with an initial energy E' and an initial direction Ω' undergoes a scattering collision at r which places it into a direction that lies in $d\Omega$ about Ω with a new energy in dE about E .

$S(r, E, \Omega) dE d\Omega$ = external source particles emitted per unit volume and time at the space point r with energies in dE about E and directions which lie in $d\Omega$ about Ω .

$\underline{r} \cdot \nabla \phi(r, E, \Omega) dE d\Omega$ = net convective loss per unit volume at point r

with energies in $d\mathcal{E}$ about E which lie in $d\mathcal{L}$ about $\mathcal{L}$.

$\chi(E) d\mathcal{E}$ = fraction of fission neutrons emitted having energies in $d\mathcal{E}$ about E .

The 4π term comes in because fission is assumed isotropic. This equation is now expressed in the more compact operator notation,

$$\hat{H}(r, E, \mathcal{L}) \phi(r, E, \mathcal{L}) = S(r, E, \mathcal{L}) \quad (2.2)$$

We now perturb the cross sections of nuclides contained in the reactor system and the perturbed equation becomes,

$$\hat{H}_p(r, E, \mathcal{L}) \phi_p(r, E, \mathcal{L}) = S_p(r, E, \mathcal{L}) \quad (2.3)$$

In an analogous fashion we introduce the adjoint Boltzmann equation, which is defined as that function $\phi^*(r, E, \mathcal{L})$ which satisfies,

$$\begin{aligned} -\underline{\mathcal{L}} \cdot \nabla \phi^*(r, E, \mathcal{L}) + \Sigma_t(r, E) \phi^*(r, E, \mathcal{L}) &= S^*(r, E, \mathcal{L}) \\ + \iint d\mathcal{E}' d\mathcal{L}' \Sigma_s(r, E \rightarrow E', \mathcal{L} \rightarrow \mathcal{L}') \phi^*(r, E', \mathcal{L}') \\ + \nu(E) \Sigma_f(r, E) \iint d\mathcal{E}' d\mathcal{L}' \frac{\chi(E')}{4\pi} \phi^*(r, E', \mathcal{L}') \end{aligned} \quad (2.4)$$

with the notation having the same meaning as for the forward case and $S^*(r, E, \mathcal{L})$ is defined as the adjoint source. In operator notation, the perturbed and unperturbed forms for this are written as,

$$H^*(r, E, \mathcal{L}) \phi^*(r, E, \mathcal{L}) = S^*(r, E, \mathcal{L}) \quad (2.5)$$

and

$$H_p^*(r, E, \mathcal{L}) \phi_p^*(r, E, \mathcal{L}) = S_p^*(r, E, \mathcal{L}) \quad (2.6)$$

Now, the quantity $H_p^*(r, E, \mathcal{L}) \phi_p^*(r, E, \mathcal{L})$ is added and subtracted to equ(2.5) and equ(2.6), and by rearranging and equating we get,

$$\begin{aligned} H_p^*(r, E, \mathcal{L}) [\phi_p^*(r, E, \mathcal{L}) - \phi^*(r, E, \mathcal{L})] = \\ [H^*(r, E, \mathcal{L}) - H_p^*(r, E, \mathcal{L})] \phi^*(r, E, \mathcal{L}) + [S_p^*(r, E, \mathcal{L}) - S^*(r, E, \mathcal{L})] \end{aligned} \quad (2.7)$$

Equation (2.7) is now multiplied throughout by $\phi_p(r, E, \mathcal{L})$, the

perturbed forward flux to yield,

$$\begin{aligned} & \phi_p(v, \epsilon, \Omega) H_p^*(v, \epsilon, \Omega) [\phi_p^*(v, \epsilon, \Omega) - \phi^*(v, \epsilon, \Omega)] = \\ & \phi_p(v, \epsilon, \Omega) [H^*(v, \epsilon, \Omega) - H_p^*(v, \epsilon, \Omega)] \phi^*(v, \epsilon, \Omega) + [S_p^*(v, \epsilon, \Omega) - S^*(v, \epsilon, \Omega)] \phi_p(v, \epsilon, \Omega) \end{aligned} \quad (2.8)$$

Now define a quantity called a response, which is the basic result of a solution to a problem and is assumed to be a flux integrated quantity.

$$R = \int \Sigma_R(\xi) \phi(\xi) d\xi \quad (2.9)$$

where $\Sigma_R(\xi)$ = response function which relates the flux to the integrated response being studied, and ξ denotes all phase space. It can be shown,

$$R = \iiint \phi(v, \epsilon, \Omega) S^*(v, \epsilon, \Omega) dv d\epsilon d\Omega = \iiint \phi^*(v, \epsilon, \Omega) S(v, \epsilon, \Omega) dv d\epsilon d\Omega \quad (2.10)$$

$$R = \iiint \phi_p(v, \epsilon, \Omega) S_p^*(v, \epsilon, \Omega) dv d\epsilon d\Omega = \iiint \phi_p^*(v, \epsilon, \Omega) S_p(v, \epsilon, \Omega) dv d\epsilon d\Omega \quad (2.11)$$

and using these identities in equ(2.8) yields equ(2.12),

$$\begin{aligned} & \phi_p(v, \epsilon, \Omega) H_p^*(v, \epsilon, \Omega) [\phi_p^*(v, \epsilon, \Omega) - \phi^*(v, \epsilon, \Omega)] = \\ & \phi_p(v, \epsilon, \Omega) [H^*(v, \epsilon, \Omega) - H_p^*(v, \epsilon, \Omega)] \phi^*(v, \epsilon, \Omega) + \phi_p(v, \epsilon, \Omega) [\Sigma_{vp}(v, \epsilon) - \Sigma_v(v, \epsilon)] \end{aligned} \quad (2.12)$$

Now define the quantity ΔR , the difference between the perturbed and unperturbed responses as,

$$\begin{aligned} \Delta R &= R_p - R \\ &= \iiint [\Sigma_{vp}(v, \epsilon) \phi_p(v, \epsilon, \Omega) - \Sigma_v(v, \epsilon) \phi(v, \epsilon, \Omega)] dv d\epsilon d\Omega \end{aligned} \quad (2.13)$$

The idea here is to replace the $\phi(v, \epsilon, \Omega)$ term in equ(2.13) by $\phi_p(v, \epsilon, \Omega)$. By using the basic definition of an adjoint operator, namely,

$$\iiint \phi_p(v, \epsilon, \Omega) H_p^*(v, \epsilon, \Omega) \phi^*(v, \epsilon, \Omega) dv d\epsilon d\Omega = \iiint \phi^*(v, \epsilon, \Omega) H_p(v, \epsilon, \Omega) \phi_p(v, \epsilon, \Omega) dv d\epsilon d\Omega \quad (2.14)$$

and by subsequent rearrangement we get that,

$$\phi_p(v, \epsilon, \Omega) H_p^*(v, \epsilon, \Omega) [\phi_p^*(v, \epsilon, \Omega) - \phi^*(v, \epsilon, \Omega)] = \phi_p(v, \epsilon, \Omega) \Sigma_{vp}(v, \epsilon) - \phi(v, \epsilon, \Omega) \Sigma_v(v, \epsilon) \quad (2.15)$$

The RHS of equ(2.15) is equivalent to the kernel of equ(2.13), which in turn is equivalent to the RHS of equ(2.8). Hence,

$$\Delta R = \iiint \phi_p(v, \epsilon, \Omega) \left[H^*(v, \epsilon, \Omega) - H_p^*(v, \epsilon, \Omega) \right] \phi^*(v, \epsilon, \Omega) dv d\epsilon d\Omega \\ + \iiint \left[\Sigma_{vp}(v, \epsilon, \Omega) - \Sigma_v(v, \epsilon, \Omega) \right] \phi_p(v, \epsilon, \Omega) dv d\epsilon d\Omega \quad (2.16)$$

Equation (2.16) is exact and is valid for any specification of the perturbed cross sections. Here, the perturbed differential cross sections will be specified to be proportional to the unperturbed differential cross sections in all regions of phase space where the perturbed and unperturbed differential cross sections are assumed to be unequal. Thus,

$$\Sigma_{sp}(v, \epsilon \rightarrow \epsilon', \Omega \rightarrow \Omega') = C \Sigma_s(v, \epsilon \rightarrow \epsilon', \Omega \rightarrow \Omega') \quad (2.17)$$

where C is a constant which is independent of $v, \epsilon, \epsilon', \Omega$ and Ω' in all regions of phase space in which $C \neq 1$. The equation may be applied to any partial differential cross section with $C=1$ for all other partial differential cross sections. The total collisional cross section can be defined similarly,

$$\Sigma_{Tp}(v, \epsilon) = C \Sigma_T(v, \epsilon) \quad (2.18)$$

and if ΔC is now defined by,

$$\Delta C = C - 1 \quad (2.19)$$

then ΔC may be written as

$$= \frac{\Sigma_{sp}(v, \epsilon \rightarrow \epsilon', \Omega \rightarrow \Omega') - \Sigma_s(v, \epsilon \rightarrow \epsilon', \Omega \rightarrow \Omega')}{\Sigma_s(v, \epsilon \rightarrow \epsilon', \Omega \rightarrow \Omega')} = \frac{\Sigma_p(v, \epsilon) - \Sigma_s(v, \epsilon)}{\Sigma_s(v, \epsilon)} \quad (2.20) \\ = \frac{\Sigma_{Tp}(v, \epsilon) - \Sigma_T(v, \epsilon)}{\Sigma_T(v, \epsilon)} = \frac{v_p \Sigma_{Tp}(v, \epsilon) - v \Sigma_T(v, \epsilon)}{v \Sigma_T(v, \epsilon)}$$

and so ΔC is the fractional change in the cross sections and is to be independent of $v, \epsilon, \epsilon', \Omega$ and Ω' in all regions of phase space where it is unequal to zero.

Writing out the $H^*(v, \epsilon, \Omega)$ operator terms in equ(2.16) into their explicit form, with the ΔC terms included, yields,

$$\frac{\Delta R}{\Delta C} = \iiint \left[\iint d\epsilon' d\Omega' \phi_p(v, \epsilon, \Omega) \Sigma_s(v, \epsilon \rightarrow \epsilon', \Omega \rightarrow \Omega') \phi^*(v, \epsilon', \Omega') \right. \\ \left. + \phi_p(v, \epsilon, \Omega) v(\epsilon) \Sigma_{fx}(v, \epsilon) \iint_{4\pi} d\epsilon' d\Omega' \frac{x(\epsilon')}{4\pi} \phi^*(v, \epsilon', \Omega') \right] \quad (2.21)$$

$$- \phi_p(v, E, \Omega) \sum_{Tx} (v, E) \phi^*(v, E, \Omega)] dv dE d\Omega \\ + \iiint \phi_p(v, E, \Omega) \sum_{rx} (v, E) dv dE d\Omega$$

which is called the sensitivity density function. A sensitivity profile is defined by normalising this function to the response R. This normalization allows the sensitivity profile to reflect the fractional sensitivity of the result R, to reactions of type $\sum_x(E)$ at energy E. Note that because linear perturbation theory is being used here, the obtained results will be symmetric. So with a negative ΔC the sensitivity results will be the same, except different in sign. The integrations in the above equation are to be performed only over the perturbed regions of phase space in which $\Delta C \neq 0$. The above equation gives the change in R, ΔR , for a given ΔC and is still exact for the assumed form of the perturbed cross sections. However, the right side cannot be readily evaluated because $\phi_p(v, E, \Omega)$ depends on the choice of ΔC and can only be determined by solving the perturbed transport equation. To overcome this problem, linear perturbation is invoked and says that for small changes in cross sections, $\phi_p(v, E, \Omega)$ may be approximated by $\phi(v, E, \Omega)$. In this approximation ΔR is proportional to ΔC and may readily be evaluated for a complete range of values of ΔC . The results in this work will be given in terms of $(\Delta R/R)/\Delta C$, and therefore will be independent of the magnitude of ΔC . So, finally, the sensitivity function equation is given by,

$$\rho_x(E) = [\Delta R/R / \Delta \sum_x(E) / \sum_x(E)] = \iiint \left[- \phi(v, E, \Omega) \sum_{Tx} (v, E) \phi^*(v, E, \Omega) \right. \\ + \iiint dE' d\Omega' \phi(v, E, \Omega) \sum_{sx} (v, E \rightarrow E', \Omega \rightarrow \Omega') \phi^*(v, E', \Omega') \\ + \phi(v, E, \Omega) v(E) \sum_{fx} (v, E) \left(\iiint dE' d\Omega' \frac{x(E')}{4\pi} \phi^*(v, E', \Omega') \right) \Big] dv dE d\Omega \\ + \iiint \phi(v, \Omega, E) \sum_{vx} (v, E) dv dE d\Omega / R \quad (2.22)$$

where x denotes the cross section nuclide species that is being perturbed. If it is a fissile nuclide or if it is the same reaction cross section that formed the response being studied,

then the fissile term and direct term, are invoked respectively, in the sensitivity equation. It is this equation that is used by the SWANLAKE code in its sensitivity calculations.

2.3 OVERVIEW AND DEFINITIONS

The goal of sensitivity analysis in its widest possible sense in radiation shielding is an attempt to determine quantitatively how sensitive a calculated integral response is to the input data used for the transport calculation and to provide an estimate of the uncertainty in the calculation due to uncertainties in the cross sections utilised. Such input data may concern either cross section data, geometry specifications (design data), methods approximations or any other input required to perform a transport calculation.

The sensitivity function used in the work described here uses linear first order perturbation theory as presented in the sensitivity function derivation. The linear perturbation approach to cross section sensitivity has all the attractions of simplicity, elegance and ease of calculations. Two central questions are addressed in performing such a sensitivity analysis, the answers to both being essential ingredients in a general sensitivity theory.

- 1) How can the contribution of an input parameter to the solution of a particular problem be quantitatively measured?

- 2) How will the solution change as a result of change in such a parameter?

The answer to the first question is analytic in nature, involving a description of the role played by each parameter in determining a particular solution with an eye toward understanding the relationship between parameter and solution. The answer to the second question involves methods for determining the character of the new solution resulting from a change in the input data. Thus this aspect of the theory is predictive in nature.

It is assumed throughout this work that sensitivity questions are properly stated only after all problem specifications are provided, including sources, geometry, materials and detector responses. Therefore sensitivity questions for which quantitative answers are to be provided are strictly problem dependent. Also assumed here is the adequacy of linear predictive approaches for estimating changes in the problem results arising from data perturbations. Since linear perturbation theory is invoked in sensitivity theory, it is assumed that the need for sensitivity analysis arises in part from the vast amounts of data used in the calculations and the lack of specific information about the importance of any particular data element in arriving at the result.

The basic result of a solution to a problem will be assumed to be a flux integrated quantity or more simply a response. Mathematically this is,

$$R = \int_{\xi} \Sigma_R(\xi) \phi(\xi) d\xi$$

where $\phi(\xi)$ is the angular flux solution of the Boltzmann transport equation, which in operator notation can be written as,

$$L\phi(\xi) = S(\xi)$$

where $S(\xi)$ = external source, L = Boltzmann operator, ξ = point in phase space and is a function of the conventional independent variables r, Ω and E ; and $\xi_R(\xi)$ is the response function which relates the flux to the integrated response being studied. The meaning of the term sensitivity will be derived from a mathematical connection between the result R and the cross section data used to solve a specific problem. This connection will be established using the adjoint flux $\phi^*(\xi)$ which is a solution to the adjoint Boltzmann equation,

$$L^* \phi^*(\xi) = S^*(\xi)$$

where $S^*(\xi)$ is the adjoint source and L^* is the adjoint Boltzmann operator, which for a suitable choice of boundary conditions (Lewins 1965) satisfies the well known inner product relationship,

$$\langle \phi, L^* \phi^* \rangle = \langle \phi^*, L \phi \rangle$$

where the braces denote integration over all ξ space.

The crux of the sensitivity definition is that to connect the problem cross section data included in the operator, L , to the final result, the adjoint source must be chosen to be the functional derivative of the result with respect to the flux, which for this case gives,

$$S^*(\xi) = \xi_R(\xi)$$

The sensitivity of the result R to cross section data can now be defined by the following inner product relationship involving the adjoint flux,

$$R_x \equiv \langle \phi^*, L_x \phi \rangle$$

Here L_x is some operator in the subset $[L_x]$ of the Boltzmann operator, L , whose definition and domain determine what input

cross sections are being studied. The term $L_x \phi$ is a source of radiation arising from the operation denoted by L_x . Therefore R_x represents an adjoint weighted production rate integrated over all phase space. The R_x is referred to as the integrated sensitivity of the result to the cross sections included in L_x . Treating the integrand of the inner product of the previous equation as a density function describing the adjoint weighted production rate per unit volume in phase space, we can define $R_x(\xi)$, a differential sensitivity function by,

$$R_x(\xi) = \phi^*(\xi) L_x \phi(\xi)$$

As a final note in this overview, the accuracy of linear perturbation theory should be addressed (Bartine, Mynatt, Oblow 1974a). To this end, the range of linearity of a chosen problem has been studied. The non fissile reactor (described later) was chosen for this work. Table 1 presents a comparison between predicted and calculated changes in the $Li6(n,\alpha)T$ reaction rate resulting from density type perturbations in the indicated cross sections (a uniform percent change in the cross sections at all energies). The predicted sensitivities shown were obtained from the SWANLAKE code by multiplying the predicted percent change in the $Li6(n,\alpha)T$ reaction rate due to a 1% change in a specific cross section set by the percent change selected for that cross section set. The calculated values were obtained by making the indicated changes in the cross sections and rerunning the original ANISN calculation. The agreement between prediction and calculation is reasonably good for small changes in the total cross sections but becomes poorer for larger changes, with the SWANLAKE prediction being systematically higher than the direct calculation. The apparent random differences at 5% perturbation may be due to rounding errors in the calculation in the ANISN

runs. A further (and final) problem with perturbation theory that the obtained sensitivities are very specific for the source and detector combination. The technique requires a repetition of the adjoint calculation for each shift of the detector in space and energy. However, for the direct method we can obtain this information with only an extra forward mode calculation, although this will only be for one energy and for one cross section set. To put this into more mathematical terms, the response R is a functional in a three dimensional phase space defined by S , D and Σ . The perturbation approach makes a cut in the domain of Σ whilst keeping S and D fixed, whereas the direct method involves a cut which keeps S and Σ constant but gives results over the full domain of D .

2.4 THE ADJOINT FLUX

The interpretation of the adjoint flux as a function describing the importance of particles in contributing to the final result is the underlying physical basis of the general approach to sensitivity analysis. Many arguments can be used to justify such an interpretation (Lewins 1965) but for this discussion the use of the adjoint flux Greens function will offer the justification for its use in this context. Consider a steady state source $Q(r, \epsilon, \lambda)$ in a subcritical system. The forward flux in this system can be obtained by solving the normal transport equation. Suppose that a neutron detector is present and it has a response proportional to the macroscopic cross section, $\Sigma_d(v, \epsilon)$, of the detector nuclide. The adjoint source is thus $\Sigma_d(v, \epsilon)$ and the adjoint equation is

$$L^* \phi^* = \Sigma_d(r, \epsilon)$$

Now multiplying the forward equation by the adjoint flux and the adjoint equation by the forward flux, and integrating over all phase space yields the identity,

$$\langle \phi, L \phi \rangle - \langle \phi, L \phi \rangle = \langle Q \phi^* \rangle - \langle \xi_d \phi \rangle$$

The two terms on the left hand side of the above equation are, by definition, equal and so

$$\langle Q \phi^* \rangle = \langle \xi_d \phi \rangle$$

The forward source has been left undefined but could be a unit source at some phase space point described by a delta function,

$$\delta(r - r_0) \delta(E - E_0) \delta(\Omega - \Omega_0)$$

which constructs the adjoint problem Greens function. Using this source we get,

$$\phi^*(r_0, \Omega_0, E_0) = \langle \xi_d, \phi \rangle$$

with the right side of the equation being a measure of the detector response to the unit source. Hence the adjoint flux is proportional to the detector response due to a neutron originating at the phase space point (r_0, Ω_0, E_0) . So it is in this sense that the adjoint flux can be considered to be a measure of the importance of a neutron in contributing to the response of a detector. Indeed, this definition of the adjoint flux is in agreement with the surface boundary condition, used in the Boltzmann equation, of zero outgoing adjoint flux, because a neutron crossing the free surface boundary in an outward direction is lost to the system and has no importance.

2.5 DISCUSSION AND MEANING OF THE SENSITIVITY EQUATION

In the sensitivity equation, (2.22), certain terms can be identified. The first term is the collisional loss term. It is the relative loss, weighted by the adjoint flux, due to removing the particle from the phase space point (r, E, Ω) upon collision. The second term is the scattering gain term. This represents the gain in sensitivity when the particle is transferred to the phase space point (r, E', Ω') upon a scattering collision at (r, E, Ω) . That is, it represents the collective gain in sensitivity after emerging from such collisions at other energies and angles. The third term is the fission gain term and represents the gain in sensitivity due to neutrons appearing with energy E' following a fission event at energy E . The fourth term is the direct term. This represents the gain in sensitivity at (r, E, Ω) due to the direct increase of the reaction rate causing the response. It is not importance weighted. This term allows us to evaluate the sensitivity of the response function itself. The collisional loss, the scattering gain and the fission gain terms are known as the indirect terms and they reflect the flux perturbations via the transport operator resulting from a perturbation in the cross sections. Their algebraic sum gives the total indirect effect. It should be noted that in the sensitivity equation the direct term is separated from the indirect one. This allows for the fact that the integral parameter may be in a different physical form to the transport medium. For example, this can occur when the integral parameter is the fission rate in a thin foil of enriched U238 and it is embedded in a slab of natural U₂O and the U238(n,fission) reaction is perturbed.

If the sensitivity function is integrated over angle and space, the remaining function of energy is referred to as a sensitivity profile for the particular cross section represented

by $\Sigma_x(E)$. For the case where the sensitivity profiles for all reaction processes for a particular nuclide are added together, the resulting sensitivity function is referred to as the sensitivity profile. However, careful consideration must be given to cases when the reaction cross section $\Sigma_x(E)$ is a composite cross section. Ideally though, for simplicity and to gain the maximum information, $\Sigma_x(E)$ should always be chosen as a partial reaction cross section of one exclusive type. The sensitivity profile defined by

$$\rho_x^R(E) = \left(\Delta R / R \right) / \left(\Delta \Sigma_x(E) / \Sigma_x(E) \right)$$

can be interpreted as the fractional (percentage) change of the design parameter R , per fractional (percentage) change of the cross section $\Sigma_x(E)$ at energy E . The SWANLAKE code perturbs the cross sections by unity percent. It is also useful to define an integral cross section sensitivity

$$S_{\Sigma_x}^R = \sum_{\text{all } E} \rho_x^R(E)$$

that reduces the sensitivity information to one number. Here $S_{\Sigma_x}^R$ can be interpreted as the percent change of the design parameter of interest, $\Delta R / R$, resulting from a simultaneous 1% increase of the group cross sections $\Sigma_x(E)$ in all energy groups E , assuming full correlation (that is, a correlation of +1) between all energy groups to exist. The sensitivity profile is plotted per unit lethargy to remove the energy group dependence. Detailed descriptions can be found on the meaning of sensitivity profiles in the papers by Bartine, Alsmiller, Oblow (1974a) and Bartine, Oblow, Mynatt (1974b).

Cross Section Perturbation	SWANLAKE predicted change in ${}^6T$ (%)	Direct predicted Change in ${}^6T$ (%)
5% Li6 total	1.14765	1.35194
10% Li6 total	2.29543	1.94885
50% Li6 total	11.47715	8.94036
100% Li6 total	22.95430	15.30536
5% Li7 total	0.46713	0.44934
10% Li7 total	0.93426	0.98277
50% Li7 total	4.67132	4.28043
100% Li7 total	9.34263	8.03091
5% Pb total	2.47664	1.78352
10% Pb total	4.95328	4.43547
50% Pb total	24.76640	18.29640
100% Pb total	49.53280	28.59070

TABLE 1. Predicted versus calculated variation of the
Li6(n,alpha)T reaction to cross section
perturbations.

CHAPTER 3

MODELLING CODES AND DATA SOURCES.

3.1 INTRODUCTION

This chapter gives the details on the computer codes used for modelling the behaviour of neutrons in various assemblies. The main codes used for calculating the particles dynamics were ANISN, a one dimensional discrete ordinates program, and MORSE, a three dimensional Monte Carlo code. Both forward and adjoint versions are discussed. The SWANLAKE code was used to calculate all the sensitivity information, and is central to the work performed in this thesis. ENBAL2 was used to calculate kerma factors for certain isotopes. Finally, a discussion of the cross section data source is given and the codes used to generate the multigroup cross section sets are briefly mentioned.

3.2 ANISN

ANISN (Engle 1967), an acronym for anisotropic S_n code, is a discrete ordinates code which solves a finite differenced version of the steady state particle transport equation in systems with one dimensional spatial symmetry. The steady state situation occurs when particle production balances loss and so the time dependence can be eliminated. The code can be run in either the forward or adjoint solution mode. Different problems or cases may be stacked and solved in the same run. Depending on the

cross section set used, ANISN will generate the neutron and photon field flux for all energy groups, spatial positions and angles. ANISN allows for general anisotropic scattering and offers a wide variety of boundary conditions, criticality search and source options. The code assumes that particles move in straight lines at constant velocity between collisions, with some background medium. It can therefore only be used for particles which experience short range interactions with matter eg neutrons and photons.

The advantage of only solving problems which have one dimensional spatial symmetry is that the number of phase space dimensions reduces from six to three. There is now only one energy, one distance and one angle variable. The three different symmetry options available are infinite slab, infinite cylinder and a sphere. Either a fissile or a fixed source or a subcritical combination of both can be used as the particle source. For the fixed source, the above three geometries may be run with either a distributed source (isotropic) or a shell source (anisotropic).

The discrete ordinates form used by ANISN is obtained by discretising the space, energy and angle variables of the Boltzmann transport equation. No detailed mathematics or derivations will be given here, only an outline of the discrete ordinates method. Both Bell and Glasstone (pp 214-251, 1970) and Duderstadt and Martin (pp 422-474, 1979) give good accounts of the discrete ordinates method. For an excellent discussion on discrete ordinates procedures and their limitations, see the paper by Lathrop (1972).

In the discrete ordinates formulation the energy variable is simplified by making use of the multigroup approximation with the transport equation being averaged over energy ranges or bins

$$E_{g+1} < E < E_g \quad g = 1, 2, 3 \dots G.$$

and the detailed variation of cross sections is replaced by suitable averages. The ordinates form of the angular variable is handled by evaluating the transport equation at a set of M discrete directions

$$\mu_m \quad m = 1, 2, 3 \dots M$$

and these directions are equivalent to a set of points upon a unit sphere. Because the Boltzmann equation contains both an angular derivative and an angular integration, the range of μ is divided into intervals

$$W_m = \mu_{m+1/2} - \mu_{m-1/2} \text{ where } -1 < \mu_{1/2} < \mu_{3/2} \dots < \mu_{M+1/2} < +1$$

and $W_1, W_2, \dots, W_M$ can be identified as the angular mesh volumes or quadrature weights. These weights account for the numerical integrations over angles. In slab and sphere geometries the subscript m labels only the one angular coordinate whereas in cylindrical geometry the direction of motion must be specified by two angles, so that the m variable labels a pair of coordinates. The spatial variable is put into the form of discrete points and the equation is evaluated at the positions

$$r_1 \ r_2 \ r_3 \ \dots \ r_{IM}$$

in the intervals

$$\Delta r_i = r_{i+1/2} - r_{i-1/2} \text{ with } 0 < r_{1/2} < r_{3/2} \dots < r_{IM+1/2}$$

and the flux is assumed constant across the spatial interval.

The domain of the variables μ and r has now been divided into

MMxIM boxes of size $W_m \times \Delta r_i$ and the discrete ordinates form of the Boltzmann equation involves evaluating the flux inside and on the edges of these mesh boxes. The adjoint ANISN run has all the options as that of the forward run. In an adjoint run any energy, velocity or angular dependent storage arrays and cross section sets are reversed. This reverse is with respect to the energy group number. The fluxes (angular and scalar) and fixed sources are also reversed and so any information printed by interval and/or angle for group 1 actually refers to group IGM, and group 2 refers to group IGM-1 etc. The angle dependent information (angular flux and shell source) must be interpreted as if it were reflected through the origin. For example, an angular flux printed for any angle μ should be interpreted as the angular flux for angle $-\mu$. The cross sections are reversed with respect to group number and the transfer coefficients are reversed with respect to the direction of transfer. Therefore $6_{1 \rightarrow 2}$ in the forward case first becomes $6_{2 \rightarrow 1}$, reversed with respect to transfer direction and then $6_{IGM-1 \rightarrow IGM}$, reversal with respect to group number, in the adjoint case.

3.3 MORSE

The MORSE (Emmett 1975) Monte Carlo program is a flexible neutron and photon transport code, capable of analysing neutron, gamma and coupled neutron-gamma transport problems in either a forward or an adjoint solution mode. The main features of the code are its ability to use standard multigroup cross sections in the ANISN format and its use of combinatorial geometry which enables complex 3-dimensional models to be constructed. The code can also handle time dependence for both shielding and

criticality problems, albedo options at any material boundary, the use of several types of importance sampling and it also holds routines for use in debugging problems.

The MORSE code is composed of 5 distinct modules; random walk, source, cross section, combinatorial geometry, and the analysis module. The source module reads the source geometry, the angular and energy distribution of source particles and assigns the initial coordinates (g, r_0, Ω_0, E_0) to initiate the tracking of the source particle. The random walk module traces the source particle from birth, and for subsequent collisions until death occurs, using the Monte Carlo method. This method uses a random sampling technique to obtain an overall picture of the particle distribution in a system. The variables (r, E, Ω, t) , which define the state of a particle, are obtained before and after each collision by using a statistical sampling technique. With the combinatorial geometry module, the code can model complicated three dimensional shapes using the Boolean algebra form of intersections, differences and body unions. This module also calculates the distance travelled in the present zone and determines the next zone to be entered. The cross section module reads the cross sections of each element, calculates macroscopic cross sections of a mixture, determines group to group transfer probabilities and determines the probabilities and angles of scattering for each group to group transfer.

In following the history of a neutron, one must decide its initial source energy and direction, choose the distance travelled before having a collision and select the collision type. The outcome of the collision is then determined and tracking continued or terminated. Each of these stages is

governed by a probability law and the basic principle of a Monte Carlo code is the way in which these probability laws are sampled. Many forms of the integral transport equation can be obtained from the multigroup transport equation, but it was decided by the authors of MORSE to use the emergent particle density equation because the source particles would be introduced according to the natural distribution rather than the distribution of first collisions. The emergent particle density $\chi^g(v, \Omega, t)$ is defined as the density of particles leaving a source or emerging from a real collision. The emergent particle density equation is given by (Duderstadt 1979)

$$\chi^g(v, \Omega, t) = S^g(v, \Omega, t) + \sum_{g' \rightarrow g} C_{g' \rightarrow g}(v, \Omega' \rightarrow \Omega) T_{g'}(v' \rightarrow v, \Omega) \chi^{g'}(v', \Omega', t) \quad (3.1)$$

$$\text{where } v' = v - R\Omega \quad t' = t - R/v'$$

$$C_{g' \rightarrow g}(v, \Omega' \rightarrow \Omega) = \sum_{g'=1}^g \int d\Omega' \frac{\xi_{g' \rightarrow g}^g(v, \Omega' \rightarrow \Omega)}{\xi_t^{g'}(v)} = \text{COLLISION OPERATOR} \quad (3.2)$$

$$T_{g'}(v' \rightarrow v, \Omega) = \int_0^\infty dR \xi_t^{g'}(v) e^{-\beta_{g'}(v, R, \Omega)} = \text{TRANSPORT OPERATOR} \quad (3.3)$$

$$\beta_{g'}(v, R, \Omega) = \int_0^R \xi_t^{g'}(v - R'\Omega) dR' = \text{OPTICAL THICKNESS} \quad (3.4)$$

so the random walk based on the integral emergence particle density equation introduces a particle into the system according to the source function. The particle travels to the site of its first collision as determined by the transport kernel. Its weight is modified by the non absorption probability and a new energy group and flight direction are selected from the collision kernel. The transport and collision kernels are applied successively, determining the particles emergent phase space coordinates corresponding to the second, third etc collision sites until the random walk is terminated due to the reduction of the particles weight below some cut off value or because the

particle escapes from that portion of phase space associated with a particular problem. For example, escape from the system, slowing down below an energy cutoff or exceeding some arbitrary age cut off. With fission occurring in a system, the outgoing particle energy is sampled from the fission spectrum and the neutron is assumed to be emitted isotropically. The fission born neutrons are then themselves tracked after their weight has been modified by $\nu \Sigma_f / \Sigma_{TOTAL}$.

The actual code implementation of the random walk procedure is accomplished by approximating the integrals implied in the collision and transport integral operators by the sum

$$X_g(v, \Omega, t) = \sum_{n=0}^{\infty} X_g^n(v, \Omega, t) \quad (3.5)$$

where $X_g^n(v, \Omega, t) d\Omega$ = the emergent particle density of particles emerging from its n^{th} collision and having phase space coordinates group $g, v, d\Omega$ about Ω , at time t . We have

$$X_g^0(v, \Omega, t) = S_g(v, \Omega, t) \text{ and that}$$

$$X_g^n(v, \Omega, t) = C_{g' \rightarrow g}(v, \Omega' \rightarrow \Omega) T_{g'}(v' \rightarrow v, \Omega') X_{g'}^{n-1}(v', \Omega', t') \quad (3.6)$$

thus the source coordinates, group g_0, v_0, Ω_0 , time t_0 , are selected from $S_{g_0}(v, \Omega, t)$ and a flight distance R is picked

$$\Sigma_{t_0}^{g_0}(v) e^{-\beta_{g_0}(v, R, \Omega_0)} \quad (3.7)$$

to determine the site for the first collision r , and the particles age $t_1 = t_0 + R/V_{g_0}$. The probability of scatter is

$$\Sigma_s^{g_0}(r_i) / \Sigma_t^{g_0}(r_i) \quad (3.8)$$

All particles are forced to scatter and their weights are modified with this probability. A new group g , is selected according to the distribution

$$\int_{4\pi} d\Omega \sum_s^{g_0 \rightarrow g_1} (r, \Omega_0 \rightarrow \Omega) / \xi_s^{g_0}(v_1) \quad (3.9)$$

and then a new direction is determined from

$$\sum_s^{g_0 \rightarrow g_1} (r, \Omega_0 \rightarrow \Omega) / \sum_s^{g_0 \rightarrow g_1} (v_1) \quad (3.10)$$

The process is repeated until the particle history is terminated.

MORSE contains two different types of scoring, the so called point and track detectors. The point detector employs the next event estimator which scores the probability of the next event being at the detector from each collision point without actually entering the detector. The track detector scores continuously along the particles flight path and the zone flux is determined by recording the product of particle track length and particle weight at every collision point and boundary crossing for each flux requested zone.

Frequently, an important but rarely occurring event is required to be known accurately and to avoid the use of lots of computer time, the event reaction rate can be increased by variance reduction techniques. MORSE contains four; importance sampling, path length stretching (exponential transform), splitting and Russian roulette. In the importance sampling scheme, the collision and transport kernel are distorted in such a way that the number of particles entering any region is proportional to the importance of that region. In the exponential transform scheme the transport kernel is biased to increase the number of histories in the important region by artificially increasing the distance travelled between collisions when the neutron travels toward the zone of interest. For the Russian roulette and splitting schemes, the particle is split if it enters an unimportant region, Russian roulette is then played to see if the

particles history is to be terminated or continued.

Like the ANISN adjoint, the MORSE adjoint has all the options as that of the forward run. Before being used properly it was first tested relative to the ANISN adjoint for a geometry of concentric spheres, for the same system materials and boundary conditions. Iron and lithium were used as the zone materials and the adjoint source was the macroscopic $\text{Li6}(n,\alpha)T$ reaction rate. In order to get good statistics the zone (not interval) MORSE and ANISN fluxes were compared and the agreement was very good in the zone containing the adjoint source but poor in zones which were spatially small or distant from the adjoint source zone. Many adjunction case histories would be needed here to get good statistics.

The emergent adjunction density equation, which corresponds to the transport of adjunctions and provides the basis for the adjoint random walk, is derived in a similar way to the forward case. A detailed derivation of the MORSE adjoint equations can be found in pp 265-279 of the MORSE manual (Emmett 1975). MORSE normalises the input adjoint source and so saves the units inherent in the forward problem detector response. All output quantities must be multiplied by this source normalisation factor. The normalised adjoint source can be thought of as the average over the group width of the fractions of particles to be emitted in the natural distribution. The cumulative distribution function so obtained is printed in the output. The energy of the source adjunction is obtained by choosing a random number and sampling from this distribution function. MORSE reverses all the energy and angular dependent arrays for an adjoint case and automatically handles any changes needed in the variance

biasing.

3.4 SWANLAKE

The SWANLAKE code presented here was used to determine all the sensitivity information. The original version was written and coded by Bartine et al (1973) at Oak Ridge National Laboratory for their shielding studies work. It only included the effects of collisional loss and inscattering gain and so the code used at Birmingham has been extensively modified. It now includes effects from the direct term, due to a perturbation in the response function itself, and from upscattering gain due to having fissile nuclides in the system. The present version also includes the option to store the sensitivity profiles (per unit lethargy) by writing it to disk or tape. Another amendment was that of including a fast input/output routine which speeds up the information exchange between the scratch discs and the core. This results in a saving of time of about 10%. The input data needed to run this modified version is given in appendix 1. SWANLAKE was also modified slightly for use in the 3D sensitivity calculations, when using the fluxes from MORSE. These modifications are discussed in chapter 10.

SWANLAKE calculates the sensitivity of a result (such as a dose or reaction rate) obtained from a one dimensional discrete ordinates solution of the Boltzmann transport equation to the cross sections used in the calculation. The main function of the code is to read in the forward and adjoint group angular fluxes, $\phi_g(r, \Omega)$ and $\phi_g^*(r, \Omega)$ respectively, that have been generated previously by two ANISN runs. It then constructs the scalar flux matrix (referred to as the $\phi\phi^*$ matrix) consisting of terms of the

form $\phi_i \phi_j^*$, which are obtained from the group angular fluxes by integrating over all angles and the spatial region of interest. Corresponding positions of the matrix and the cross section scattering table are multiplied together to yield the sensitivity matrix $\phi \phi^* \hat{T}$. The $\hat{T}$ refers to multiplication by cross section, interval volumes, Legendre coefficients and quadrature weights.

The user has a choice as to how many and which zones are to be perturbed in the system. However, the spatial and energy information - such as zone boundaries and distances and energy group boundaries - must correspond exactly in the ANISN and SWANLAKE runs. The perturbation zones or intervals are chosen by making the nuclide number density of the nuclide containing the cross section non zero in that particular zone. The code can be assumed to be giving a 1% cross section (or density) perturbation in those zones that contain a non zero number density. In order for the code to calculate the sensitivities numerically, the point sensitivity function, previously defined, has a discrete ordinates formulation applied to it. This can be found at pp 8-11 in the SWANLAKE manual (Bartine et al 1973). With this formulation and dividing the spatial integration into NZONE spatial zones, the multigroup discrete ordinates form for the sensitivity function becomes,

$$\begin{aligned}
 P_x(I) = & \left\{ \sum_{Iz=1}^{NZONE} \rho_x(Iz) \sum_{J=IzI}^{Izf} VOL(J) \left[- \sum_{K=1}^{NOA} \phi^*(JKI) \sigma_{xT}(I) \phi(JKI) WT(K) \right. \right. \\
 & + \sum_{K=1}^{NOA} \phi^*(JKI) \sigma_{xq}(I) \phi(JKI) WT(K) + \sum_{\ell=0}^{lmax} \sum_{I'=I}^{Iqm} \phi^*(J,I') \sigma_x^\ell(I \rightarrow I') \phi^*(J,I) \\
 & \left. \left. + \sum_{K=1}^{NOA} \phi(JKI) \gamma(I) \sigma_{fx}(I) WT(K) \sum_{I'} \phi^*(J,K,I') X(I') \right] \right\} \quad (3.11)
 \end{aligned}$$

$$\left\{ \sum_{Iz=1}^{NZONE} \rho_r(Iz) \sum_{I=IzI}^{IZF} VOL(J) \sum_{K=1}^{NOA} \sigma_r(I) \phi(J, K, I) WT(K) \right\} / R.$$

$$\text{where } R = \sum_{J=1}^{INT} \sum_{l=0}^{l_{max}} \sum_{I=1}^{I_{lm}} D^l(J, I) \phi^l(J, I) \quad (3.12)$$

where $Q_x(IZ)$ is the number density for cross section x in spatial zone IZ , and IZI and IZF are the initial and final spatial intervals in spatial zone IZ , $VOL(J)$ is the volume of interval J and the I' index on the fissile term above cycles for both upscatter and downscatter. This equation represents the basic calculation performed in SWANLAKE. In practice, however, the above equation is calculated as the product of the density, the cross section matrix and the $\phi \phi^*$ matrix. The above equation is rearranged and simplified to give

$$\begin{aligned} \rho_x(I) = & \left\{ \sum_{Iz=1}^{NZONE} \rho_x(Iz) \left[-\sigma_{xT}(I) Z^{l=0}(3, I) \right. \right. \\ & + \sum \sum \sigma_x(I \rightarrow I) Z^l(4+I-I, I) + \nu(I) \sigma_{fx}(I) Z^{l=0}(2, I) \\ & \left. \left. - \sigma_a(I) Z^{l=0}(1, I) \right] + \sum_{Iz=1}^{NZONE} \rho_x(Iz) \sigma_x(I) Z^{l=0}(3, I) \right\} / R. \end{aligned} \quad (3.13)$$

where Z is the $\phi \phi^*$ matrix and corresponds to the format of the cross section matrix for ANISN. This separation of the cross section set from the $\phi \phi^*$ matrix as shown in the equation above, gives the obvious advantage that the $\phi \phi^*$ matrix need only be calculated once and then stored. In this way many different cross section partials (which must be of the same nuclide type though) can be analysed in one SWANLAKE run by stacking the datafiles. Though this separation does necessitate a larger peripheral storage capability, it does save considerably on computer time.

The fissile sensitivity modification used here basically breaks down the fission term into upscatter and downscatter fission. These two terms are then added, by energy group, into the full sensitivity matrix and then stored in column 2 of this matrix. This position corresponds to the $\nu(E) \xi_f(E)$ terms in the cross section matrix. The crux of the modification is that the angular adjoint flux is now given direct energy dependence, so that now a complete ϕ/ϕ^* matrix can be constructed. The fissile modification in this form, due to R.H. Curtis (1978) makes use of previously wasted space and the program needs no additional storage. In the original SWANLAKE the ϕ/ϕ^* matrix had the form of fig 1, with the corresponding cross section matrix given by fig 2. However, with upscatter cross sections the matrix becomes that shown in fig 3 and the full ϕ/ϕ^* matrix, complete with ϕ^* having its direct energy dependence is shown in fig 4. The code first forms a separate fissile sensitivity matrix, which may be printed out for inspection, containing both upscatter and downscatter sensitivity terms. These fissile sensitivity terms are then summed by energy group and the result placed in column 2 of the original sensitivity matrix, giving the sensitivity of the response to perturbations in the $\nu(E) \xi_f(E)$ term at each energy. Finally these terms are summed over all groups and added to all the non fissile sensitivity terms to give the total sensitivity of the result.

The direct effect modification allows the inclusion of sensitivity gain when the operator cross section being perturbed is also the reaction rate causing the response. It is always a positive term. The terms are added by energy group and spatial zone into the total collisional loss part (column 3) of the sensitivity matrix. The direct effect terms are macroscopic

reaction rates of the response and must be input for every energy group (in order of decreasing energy) and for each separate spatial zone, by the user.

3.5 ENBAL2

The rate of energy deposition by neutrons in materials is a quantity of considerable importance in various areas of applied neutron physics. In order to calculate this heating rate, the neutron scalar flux (fluence) must first be computed and then use is made of fluence - to - kerma factors (normally referred to as kerma factors), kerma being an acronym for kinetic energy release in materials. The program ENBAL2 (Taylor 1978), an acronym for energy balance, was used for this purpose. The neutron kerma factor $K(E)$ for a particular nuclide is defined by

$$K(E) = \sum_i \sigma_i(E) \bar{E}_i(E) \quad (3.14)$$

where $\sigma_i(E)$ is the microscopic cross section for the neutron reaction type i and $\bar{E}_i(E)$ is the average energy deposition in a reaction of type i involving an incident neutron of energy E . The sum over i is over all reaction types in a given nuclide.

This kerma factor is used to calculate the neutron heating rate in this nuclide at a position r by

$$H(r) = \int_{\text{all } E} N(r) K(E) \phi(r, E) dE \quad (3.15)$$

where $N(r)$ is the atomic density at position r for this nuclide and $\phi(r, E)$ is the neutron scalar flux at position r , energy E . Using multigroup notation the equation becomes

$$H(r) = \sum_g N(r) K^g \phi^g(r) \quad (3.16)$$

where $\phi^g(r)$ is the group scalar flux in energy group g , and K^g is the kerma factor averaged over energy group g . The sum over g is over all energy groups.

In the calculation of kerma factors, the important quantity to be evaluated is the average energy deposited per reaction, $\bar{E}_d(E)$, with the cross sections being available from nuclear data libraries. The energy deposition in a reaction is manifested by the recoil kinetic energy of the target nucleus. So for the purpose of nuclear heating it is sufficient to consider the initial and final states in an interaction, the process involved in the reaction itself not being of consequence. In addition if charged particles are produced then they will contribute to the total energy, and it is assumed that they locally deposit their energy at the point of interaction. Energy released as gamma ray energy is not involved in the neutron heating total and this energy is assumed lost. Multigroup gamma kerma factors must be generated if gamma ray interactions are to be included, provided the cross sections for Compton scattering, pair production and the photoelectric effect are available.

In order to obtain an expression for the neutron kerma factor for a nuclide, the rate of energy deposition in four types of neutron interaction needs to be considered. These are elastic scattering, inelastic scattering, charged particle production and radiative capture. Other, more complex, reaction types can be adequately represented by the expressions obtained for the above four. No further mathematics is given here. A detailed derivation and description for the ENBAL2 code can be found in the ENBAL2 manual, Dept of Physics report, paper no. 81/02 by N.P Taylor.

For this work ENBAL2 was used to generate the adjoint sources for ANISN heating calculations and to calculate the direct effect terms needed in the subsequent heating sensitivity calculations. A slight modification to the code was needed to write the partial kerma factors to a dataset, where they were then acted on by a short code to produce the partial heating reaction rates in a form suitable for the SWANLAKE code. The code was also used to calculate the total kerma factors for flourine, U238 and U235, these not being available in the DLC37F cross section library.

3.6 CROSS SECTIONS AND SOURCES.

Cross section data used in this work for discrete ordinates codes and Monte Carlo codes are obtained from various cross section libraries, containing information in special formats for all nuclides likely to be needed. For the multi-group discrete ordinates codes this data is already in a group averaged form and contains the expansion coefficients for the Legendre expansion of scattering cross sections. Examples of such cross section libraries are DLC2(RSIC Feb 1972) containing 100 group neutron data for sixty-four nuclides, prepared from ENDF/B-3 primary data with activation cross sections, and DLC37F (RSIC July 1974) containing coupled 100 neutron and 21 gamma group data, prepared chiefly from ENDF/B-4 primary data. Both libraries have expansion coefficients up to a P order of 8. Information on the ENDF library can be found in the report by Drake et al (1970).

The group averaged data in the libraries above is obtained by the use of a processing code which takes cross sections defined at energy points (point data) and performs the group averaging

and calculation of the Legendre expansion coefficients. Such a processing code is XLACS, one of the programs contained in the AMPX suite of codes, which uses point energy cross section data from ENDF/B-4, a large library of evaluated cross sections, energy and angular secondary distributions and atomic form factors. The cross sections in ENDF/B-4 are drawn from a large number of sources, from various laboratories around the world, some the results of experimental measurements and some theoretical predictions. Any inaccuracies, of course, contained in the data are carried through into the results calculated by the transport codes. ENDF/B-4 is the last full cross section library to be published since 1974 and contains covariance files for nitrogen, carbon and oxygen. The next version, ENDF/B-5, has yet to be released by RSIC(Radiation and Shielding Information Centre), and will contain, apart from the more accurate fissile cross sections in the range 2-15 Mev, a large number of covariance matrices for materials which are important in fusion reactors.

On the ENDF/B-4 tape each element or material is identified by an MAT number. Each element MAT number has various MF sections associated with it, for the different kinds of cross section information. For example MF=1 contains general information, MF=3 contains the point cross sections and MF=14 contains the gamma angular distributions. In addition all the cross sections, total, semi-partial or partial are labelled by an MT number, for example, MT=1 is the total cross section (sum of all partial cross sections), MT=16 (n, 2n) reactions and MT=102 is the (n, gamma) radiative capture cross sections. The MT number can cater for up to the 40th nuclear excited state as well as the continuum itself. In this work the ENDF/B-4 cross section data source was

used, so that the DLC37F 100 neutron and 21 gamma energy library was used in all transport calculations. The DLC37F library is held on tape and is retrieved via the program GETEX. The cross sections so retrieved are then directly usable by transport codes.

For the sensitivity analysis work certain cross sections were required in their partial cross section form. To obtain these the codes XLACS and NITAWL, from the AMPX suite, had to be run (Green 1976). AMPX is a modular system for producing coupled multigroup neutron gamma cross section sets. Basic neutron and gamma cross section data for AMPX are obtained from the ENDF/B-4 libraries.

The first module to run is XLACS. This is a program that calculates weighted multigroup neutron cross sections from interfacing directly with ENDF/B-4 data. XLACS is designed to produce full energy range neutron cross section libraries and provisions are included for treating fast, resonance and thermal ENDF/B-4 data in a single calculation. Energy group structure and expansion orders used to represent differential cross sections can be averaged over an arbitrary user supplied weighting function or over any of several built in weighting functions. XLACS produces what is known as a master cross section library.

The second module to run is NITAWL which interfaces directly with the master cross section library from XLACS to cater for resonance self shielding and to generate the group dependent library in the form of transfer matrices. In its turn NITAWL produces a working cross section library which can couple directly to either ANISN, MORSE or SWANLAKE. So the NITAWL module, which represents a relatively large portion of the AMPX

suite, collects data into workable arrangements for the use of transport codes. One of the array options in NITAWL is the production of a sensitivity library such as required for the SWANLAKE program. In this work they take the role of cross section operators and they are multiplied by $\phi\phi^*$ in the manner shown in the sensitivity equation.

For the heating calculations the DLC37F kerma factor set was used. However no neutron kerma factor data was available for F19, U238 or U235. So these values were obtained using the partial cross sections in question and the program ENBAL2. The Q values for the partial reactions, including all the inelastic levels, is obtainable from file 3 of the material. The gamma ray kermas for F19, U238 and U235 were not generated because the multigroup cross sections for Compton scattering, pair production and the photoelectric effect were not readily available for easy use.

		$-\phi_1 \phi_1^*$	$\phi_1 \phi_1^*$		
		$-\phi_2 \phi_2^*$	$\phi_2 \phi_2^*$		EMPTY.
		$-\phi_3 \phi_3^*$	$\phi_3 \phi_3^*$	$\phi_1 \phi_2^*$	
				$\phi_2 \phi_3^*$	
					$\phi_1 \phi_3^* \dots$
$\vdots$	$\vdots$	$\vdots$	$\vdots$	$\vdots$	$\vdots$

FIG. 1

G_{a_1}	$\gamma_1 G_{f_1}$	G_{t_1}	$G_{1 \rightarrow 1}$		EMPTY.
G_{a_2}	$\gamma_2 G_{f_2}$	G_{t_2}	$G_{2 \rightarrow 2}$	$G_{1 \rightarrow 2}$	
G_{a_3}	$\gamma_3 G_{f_3}$	G_{t_3}	$G_{3 \rightarrow 3}$	$G_{2 \rightarrow 3}$	$G_{1 \rightarrow 3} \dots$
$\vdots$	$\vdots$	$\vdots$	$\vdots$	$\vdots$	$\vdots$

FIG. 2

G_{a_1}	$\gamma_1 G_{f_1}$	G_{t_1}	$G_{1 \rightarrow 1}$	$G_{2 \rightarrow 1}$	$G_{3 \rightarrow 1} \dots$
G_{a_2}	$\gamma_2 G_{f_2}$	G_{t_2}	$G_{2 \rightarrow 2}$	$G_{1 \rightarrow 2}$	$G_{3 \rightarrow 2} \dots$
G_{a_3}	$\gamma_3 G_{f_3}$	G_{t_3}	$G_{3 \rightarrow 3}$	$G_{2 \rightarrow 3}$	$G_{1 \rightarrow 3} \dots$
$\vdots$	$\vdots$	$\vdots$	$\vdots$	$\vdots$	$\vdots$

FIG. 3

$-\phi_1 \phi_1^*$	$\phi_1 \phi_1^*$	$\phi_2 \phi_1^*$	$\phi_3 \phi_1^* \dots$
$-\phi_2 \phi_2^*$	$\phi_2 \phi_2^*$	$\phi_1 \phi_2^*$	$\phi_3 \phi_2^* \dots$
$-\phi_3 \phi_3^*$	$\phi_3 \phi_3^*$	$\phi_2 \phi_3^*$	$\phi_1 \phi_3^* \dots$
$\vdots$	$\vdots$	$\vdots$	$\vdots$

FIG. 4

CHAPTER 4

CROSS SECTION ADJUSTMENT AND INTEGRAL EXPERIMENT THEORY

4.1 INTRODUCTION

Since the main aim of this work is to adjust 100 energy group dependent ENDF/B-4 neutron partial cross sections, this chapter will be presented in some detail. The work presented here forms the first (computational) half of the cross section adjustment scheme. The chapter first discusses cross section adjustment and the ideas behind it and then the equation used for adjustment is derived in detail. Next is a discussion on integral experiments and the ideas behind them. The chapter finishes with a discussion on what results have been obtained and how the results can be used as part of the adjustment scheme.

The systematic adjustment of cross sections, based on the results of integral experiments, was first proposed at the 1964 Geneva conference (Cecchini 1964 and Humi 1964). Since then several groups have developed such procedures and have applied them to generate improved sets of group constants for practical reactor calculations. Reviews of these procedures, and summaries of the mathematics involved in cross section adjustment, are given in papers by Campbell et al (1970) and Pendlebury (1971).

The whole idea of cross section adjustment came about because of the inadequacy of existing cross section tables. Usually, graphs and tables of cross section data gives information in the form of a smoothed out curve which often neglects to give the experimental uncertainties and the density of the measurements. Also, quite often no details on the methods of evaluation are given. In more recent years, however, particularly with the ENDF

set of data, evaluated cross sections have been more fully documented, though without detailed cross section uncertainties.

Although a detailed description of the actual evaluation process may be found in the articles of Schmidt (1967) and Parker (1967), a brief outline will be given here. In considering a certain cross section, all available data are first put onto a graph. Some of this data may be mutually inconsistent and so the experimentalists quoted errors should be disregarded and some data may even need to be rejected. Since all the measured values lie in a limited band, then a smooth curve $\hat{\sigma}(\epsilon)$ through the centre of this band should represent the actual function $\sigma(\epsilon)$. The half width of the band should roughly represent the standard deviation of a single measurement, $\Delta \hat{\sigma}(\epsilon)$. Over those energy ranges where no experimental data are available, the cross sections are interpolated either intuitively or by assuming a functional form derived from nuclear theory. In judging the quality of the estimate $\hat{\sigma}(\epsilon)$ for a cross section, the density of measurements, $w(\epsilon)$, should also be considered. For instance, if within a narrow energy interval $\Delta\epsilon$ (in which the cross section can be assumed to be a linear function of the energy) there are $n [= w(\epsilon)\Delta\epsilon]$ measurements, the average value of the cross section in this interval is determined to within $\pm \Delta \hat{\sigma}(\epsilon) / \sqrt{n}$.

Without detailed cross section uncertainties, the designer has to estimate the likely spread of the cross sections, but care must be taken here as the calculated results of system parameters may be influenced significantly. However, the physically measured values of these same system parameters constitute a further source of information on the cross sections. These

parameters are certain functionals, defined by the transport equation, of the cross sections of the isotopes in which the system under consideration is composed. So integral experiments should most definitely be used in the determination of the cross section curves. Also, as a final point, it should be noted that in some cases integral parameters can be measured more accurately than the cross sections themselves. So with this in mind, it can be said that the role of integral experiments in this work comes about not just because of the inadequacy of the cross section evaluation procedures, but also because they can furnish such accurate results.

So, after the above discussion, a derivation is now presented of adjusting group energy data using the method of sensitivities on differential and integral data.

4.2 DERIVATION OF ADJUSTMENT FORMULA

The derivation presented here is taken principally from papers published by the Jerusalem group, Pazy (1974), Yeiven (1978), and a paper by Campbell (1970). To make the following discussion more meaningful, the variables will be referenced in terms of known quantities wherever possible.

Consider a continuous physical quantity $\sigma(E)$ (ENDF point energy cross section set), and a set of N discrete quantities, computationally determined reaction rates, γ_i , which are functionals of $\sigma(E)$; $\gamma_i = \gamma_i[\sigma(E)]$. The response functions can be any type of reaction rate; activation detectors, proportional counters with multichannel readouts. The index i labels one

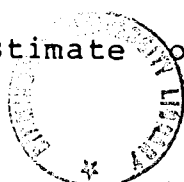
response function (one foil or one channel readout) at one spatial position. The function $\sigma(E)$ has been extensively measured throughout the range of interest with the number of measurements per unit interval (the density of measurements) being $w(E)$. Each of the N responses has also been experimentally measured, and found to have a mean value $\hat{\gamma}_i$ with an error (standard deviation) of $\Delta\hat{\gamma}_i$. The aim here is to derive the best value of $\sigma(E)$, that is, to derive the new set of ENDF cross sections, from all the direct and integral measurements.

Firstly, the energy structure is put into the multigroup form corresponding to the ENDF format. Divide the range of E into small intervals $\Delta E_1, \Delta E_2, \dots$ and approximate $\sigma(E)$ by a piecewise constant function, equal to σ_α in the interval ΔE_α . Also the average of the measurements of $\sigma(E)$ in the interval ΔE_α is given as $\hat{\sigma}_\alpha$, with the standard deviation of a single measurement by $\Delta\hat{\sigma}_\alpha$ and the number of measurements in the interval by n_α .

Assuming that the measurements of $\sigma(E)$ in different energy intervals and those of the γ_i are independent of each other, and that they are normally distributed about their respective true (unknown) values, then the probability of simultaneously obtaining the experimental estimates $\hat{\sigma}_\alpha$ and $\hat{\gamma}_i$ is given by

$$P = C \exp \left\{ -\frac{1}{2} \left[\sum_{\alpha} \left(\frac{\sigma_\alpha - \hat{\sigma}_\alpha}{\Delta\hat{\sigma}_\alpha} \right)^2 n_\alpha + \sum_i \left(\frac{\gamma_i - \hat{\gamma}_i}{\Delta\hat{\gamma}_i} \right)^2 \right] \right\} \quad (4.1)$$

where C is a normalisation constant. Here the γ_i should be considered as functions of the σ_α . The fundamental proposition of the method of least squares is that the best estimate of the σ_α consists of those values $\bar{\sigma}_\alpha$ for which the probability P is maximal. So the best estimate of the (approximated) $\sigma(E)$ is



obtained by minimising the exponential term of the above.

$$Q'' = \sum_{\alpha} \left(\frac{G_{\alpha} - \hat{G}_{\alpha}}{\Delta \hat{G}_{\alpha}} \right)^2 w_{\alpha} + \sum_i \left(\frac{\gamma_i - \hat{\gamma}_i}{\Delta \hat{\gamma}_i} \right)^2 = \text{minimum.} \quad (4.2)$$

In the limit of infinitesimal energy intervals, this equation reduces to

$$Q' = \int \left[\frac{G(E) - \hat{G}(E)}{\Delta \hat{G}(E)} \right]^2 w(E) dE + \sum_i \left(\frac{\gamma_i[G] - \hat{\gamma}_i}{\Delta \hat{\gamma}_i} \right)^2 = \text{minimum} \quad (4.3)$$

The best estimate of $G(E)$ is the solution to this equation. In order to derive an explicit solution to the above minimization problem, the response functions must first be linearized. This is where sensitivity theory is invoked; we assume that the responses calculated by adjusted and unadjusted cross sections are linearly dependent on their cross sections, respectively. The functional derivative $D_i(E)$ is defined as

$$\gamma_i[G] - \gamma_i[\hat{G}] \sim \int D_i(E) [G(E) - \hat{G}(E)] dE. \quad (4.4)$$

In the parlance of sensitivity analysis, the unperturbed response is that calculated by the unadjusted ENDF cross sections, and the perturbed response is that calculated by the adjusted ENDF cross sections. It does not matter that the adjusted cross sections and responses are unknown at this time because the sensitivity profile is calculated with the unadjusted ENDF cross section data. That is, we computationally calculate the sensitivity of the response R (a $\text{Ni}(n,p)$ foil at one position, say) to all the ENDF energy dependent partial cross sections of isotopes contained in the system. For convenience we choose

$$f(E) = \frac{G(E) - \hat{G}(E)}{\Delta \hat{G}(E)} \cdot [w(E)]^{1/2} \quad (4.5)$$

Now, the extreme right hand side of eq (4.3) is rearranged, using eq (4.5) to give,

$$\gamma_i[\hat{G}] \sim \gamma_i[\hat{G}] + \frac{\int D_i(E) \Delta \hat{G}(E) f(E) dE}{[W(E)]^{1/2}} \quad (4.6)$$

With substitution into eq (4.3) we get the quadratic form that Q' may be approximated by

$$Q = \int f^2(E) dE + \sum_i \left[\left(G_i(E) f(E) dE - C_i \right)^2 \right] \quad (4.7)$$

$$\text{where } G_i(E) = \frac{\Delta \hat{G}(E) \cdot D_i(E)}{[W(E)]^{1/2} \cdot \Delta \hat{\gamma}_i} \quad (4.8)$$

$$\text{and } C_i = \frac{\hat{\gamma}_i - \gamma_i[\hat{G}]}{\Delta \hat{\gamma}_i} \quad (4.9)$$

The variation of Q with respect to $f(E)$ is

$$\delta Q = 2 \int \left[f(E) + \sum_i \left[\int G_i(E') f(E') dE' - C_i \right] G_i(E) \right] \delta f(E) dE \quad (4.10)$$

For that $f(E)$ that minimises Q , this variation vanishes for any arbitrary $\delta f(E)$. The function being sought is thus the solution to the integral equation

$$f(E) + \int \sum_i G_i(E) G_i(E') f(E') dE' = \sum_i C_i G_i(E) \quad (4.11)$$

This is a symmetric Fredholm equation with a degenerate kernel.

Its solution can be reduced to that of a system of N linear equation. Defining,

$$X_i = C_i - \int G_i(E) f(E) dE \quad (4.12)$$

$$\text{and } A_{ij} = \int G_i(E) G_j(E) dE \quad (4.13)$$

Equation (4.11) can be written as

$$f(E) = \sum_j X_j G_j(E) \quad (4.14)$$

which on multiplying by $G_i(E)$, and integrating over E , becomes,

$$X_i + \sum_j A_{ij} X_j = C_i \quad (4.15)$$

Solving these equations for the X_i and substituting the result in

eq (4.14) , we obtain

$$f(\epsilon) = \sum_{ij} B_{ij} C_i G_j(\epsilon) \quad (4.16)$$

Where the matrix B is the inverse of I+A. The improved estimate of $\hat{G}(\epsilon)$, by definition in eq (4.5), is therefore

$$\bar{\hat{G}}(\epsilon) = \hat{G}(\epsilon) + \frac{\Delta \hat{G}^2(\epsilon)}{w(\epsilon)} \sum_{ij} B_{ij} \frac{(\gamma_i - \gamma_i[\hat{G}])}{\Delta \hat{\gamma}_i} \cdot \frac{D_j(\epsilon)}{\Delta \hat{\gamma}_j} \quad (4.17)$$

with

$$(B^{-1})_{ij} = \delta_{ij} + \frac{1}{\Delta \hat{\gamma}_i \Delta \hat{\gamma}_j} \int \frac{\Delta \hat{G}^2(\epsilon) \cdot D_i(\epsilon) \cdot D_j(\epsilon) d\epsilon}{w(\epsilon)} \quad (4.18)$$

To make the derivation clearer, the explicit dependences will be given. The integral quantities considered in this work, such as tritium breeding and nuclear heating, are functionals of the elastic, inelastic, (n,2n) etc partial cross sections. Denoting the partial cross section species by α and the isotope by the label μ , eq (4.17) becomes

$$\bar{\hat{G}}_{\alpha\mu}(\epsilon) = \hat{G}_{\alpha\mu}(\epsilon) + \frac{\Delta \hat{G}_{\alpha\mu}^2(\epsilon)}{w_{\alpha\mu}(\epsilon)} \sum_{ij} B_{ij} \frac{(\gamma_i - \gamma_i[\hat{G}])}{\Delta \hat{\gamma}_i} \cdot \frac{D_{j\alpha\mu}(\epsilon)}{\Delta \hat{\gamma}_j}$$

with

$$(B^{-1})_{ij} = \delta_{ij} + \frac{1}{\Delta \hat{\gamma}_i \Delta \hat{\gamma}_j} \sum_{\alpha\mu} \frac{\Delta \hat{G}_{\alpha\mu}^2(\epsilon) D_{i\alpha\mu}(\epsilon) D_{j\alpha\mu}(\epsilon) d\epsilon}{w_{\alpha\mu}(\epsilon)}$$

4.3 DISCUSSION OF ADJUSTMENT FORMULA

A few remarks should be made about eq (4.17), the derived adjustment formula. Two limiting cases of the result can be considered. Firstly, re-express the density of ENDF differential measurements, $w(\epsilon)$, as

$$w(\epsilon) = k w_0(\epsilon)$$

where k is a positive parameter and $w_0(\epsilon)$ function, normalised in some arbitrary fashion

The case of poor integral data is represented that is, the experimental errors are large number of differential data measurements is n integral ones. In this case we are left with

$$G_i \sim \hat{G}_i$$

This is as expected, since the adjusted set is the same as the unadjusted set. In any case, integral data, the adjusted set is at least n than the unadjusted one. The case of good case characterised by $k \rightarrow 0$ and $\Delta \hat{r}_j \rightarrow 0$. This case of their being many integral response measurements. With $k \rightarrow 0$, eq (4.17) tends to X'_i are the solutions of

$$\sum_j A'_{ij} X'_j = C_i$$

The A'_{ij} here are defined just as the A_{ij} in eq (4.17) calculated with $k=1$. The limiting value of eq becomes

$$\bar{G}(\epsilon) = \hat{G}(\epsilon) + \frac{\Delta \hat{G}^2(\epsilon)}{w_0(\epsilon)} \sum_{ij} B'_{ij} \frac{(\hat{r}_i - \gamma_i[\hat{G}])}{\Delta \hat{r}_i}$$

with

$$(B'^{-1})_{ij} = \frac{1}{\Delta \hat{r}_i \Delta \hat{r}_j} \int \frac{\Delta \hat{G}^2(\epsilon)}{w_0(\epsilon)} \rho_i(\epsilon) \rho_j(\epsilon) d\epsilon$$

The basic motivation behind this adjustment procedure is that good integral data is available and so it is of principle importance that the adjusted cross sections are independent of the normalization k of the $w(E)$.

A second remark concerns the assumption that $\hat{\sigma}_i$ and $\hat{\gamma}_j$ are all normally distributed, which was a necessary condition for the validity of the least squares solution. However, this is not the case, and a more general derivation of the result, independent of assuming a statistical distribution for $\hat{\sigma}_i$ and $\hat{\gamma}_j$, is based on the Gauss-Markov theorem (Beynon 1978).

A third is the adjustment by sensitivities. In the past there has been much criticism of the least squares adjustment procedure, particularly because it seems that the adjustments are arbitrary and spread over seemingly irrelevant cross sections. Indeed H. Goldstein (1977) has called them a 'practice of questionable value'. However, with the sensitivities approach (Yeiven 1978), this problem is done away with because only cross sections of substantial sensitivities (more precisely, cross sections the sensitivities to which are substantial) are modified and those of small sensitivities are not. So, it is not really the method but the poor integral data which is questionable.

4.4 INTEGRAL EXPERIMENT ANALYSIS

The early 14 Mev neutron driven integral experiments afforded rough confirmation of the relatively crude nuclear data and calculational tools then in use. In more recent years, an increasing number and variety of such experiments have been

performed and modern calculation experiment comparisons have been undertaken. Indeed, during the 1950s, 1960s and 1970s the number of new 14.1 Mev neutron driven integral experiments has increased one, two and eight fold, respectively. With this increase in the number of integral experiments certain advantages appear; the calculational base is tested over a wider range of conditions, the learning process leads to designs of greater quality and there is greater scope for realistic error evaluation in using the discrepancies between results on similar experiments. One of the earliest and well documented experiments is the Weale experiment (Weale 1961) in which a 14.1 Mev neutron source was surrounded by a massive natural uranium metal blanket. Reaction rates were measured throughout the system. Using this same design, to validate the neutronic analysis of hybrid fission fusion blankets, Haight et al (1976) calculated these same reaction rates using various cross section libraries.

At Birmingham, a 1 m diameter sphere of LiF has been neutronically analysed and had its sensitivity profiles derived (Beynon 1976) and has subsequently been built (Perkins et al 1981 and Scott 1982) and experimentally measured. Other experiments have been designed and built. A tritium production design performed in 1954 at Los Alamos (Wyman 1972) has been extensively studied (Muir 1976). More recently, Seki and Maekawa have made extensive calculations (1977b) and measurements (1977a) of fission rate distributions in an integral experiment made of slabs of lithium, uranium and graphite. Perhaps the most recent work in this area is by Kumar et al (1983). They are setting up a set of neutronics investigations to assess the feasibility of testing reactor blanket ideas using integral experiments in slab geometry. One idea being pursued is to use actinide wastes as

neutron multipliers.

For the purpose of neutronic design, the basic criterion is that the integral experiment approximate reasonably well, in a neutronics sense, the engineering design of interest (Oblow 1976 and Beynon 1978). Obviously the integral experiment should not exactly duplicate the design, otherwise the design prototype itself would serve as the experiment and the saving in cost and advantages of working in a clean geometry would be lost. The similarity of design is desirable in that we need to guarantee that the neutronicallly important governing cross sections respond in the same way for the two systems. This requires that the sensitivity profiles be fairly well tuned or matched. An illustration of this is when the computational cross section adjustment is carried out in a broad energy group structure. For this case, the similarity of spectrum between the integral experiment and reactor must be assured in order for the adjusted cross sections to be applicable to the fusion reactor. An integral experiment should also be built on the basis that it eases or reduces calculational difficulties (Reupke 1978). This comes about because we wish to reduce as much as possible the contribution of discrepancies between computationally calculated and experimentally determined results due to calculated systematic errors. We use as many energy groups, spatial intervals and angular bins as possible and in this respect try to limit the number of nuclides in the experiment to reduce data storage requirements and to permit the above analysis in fine detail.

The important thing in tuning via sensitivity profiles is to get the shape of the profiles to correspond as closely as

possible. The two systems will then respond in the same way to perturbations. It is more important to get the shape of the profiles to correspond than the absolute magnitudes. For example, having an integral experiment with a sensitivity profile the same shape as that of a fusion reactor, but with twice the overall sensitivity, means simply that although changes of cross sections in the systems will act in the same way, the changes will have twice as much effect in the integral experiment than in the reactor. The criterion that the direction of change be similar is most important. The choice of system parameters, with which to do the tuning, should be dictated by either how important they are or by the ease in which the data of the parameters can be obtained. In the present work such parameters are tritium breeding, for both isotopes, neutron and gamma heating, gas production and displacements per atom. Both fissile and non fissile cases had the above integral parameters calculated and the relevant sensitivities obtained. The fissile case had the additional calculation of the fissile parameters.

For both fissile and non fissile cases, the integral experiment and the reactor had as many profiles matched as possible, to see what the overall tuning looked like. This tuning was done after the design of the integral experiment and as will be shown later, the actual design of the integral experiment was by some lesser sensitivity profile matching, perhaps by just one or two profiles. This had to be done since the number of profiles calculated was quite large and it was clearly unreasonable to try and tune the designs to each profile.

4.5 CROSS SECTION ADJUSTMENT RESULTS

This section deals with the results obtained and their uses in the cross section adjustment scheme. It should first be mentioned that no actual cross section results can be given because the integral experiments of the designs presented here have not yet been physically built. However, results have been obtained in this thesis which form the first (computational) half of the adjustment process. Once the actual (real) integral experiment results (reaction rates) become available, the computational results presented in this thesis can be used in conjunction with these results to adjust ENDF/B-4 group cross sections. Indeed, the results so obtained in this thesis will allow an experimentalist to adjust, by the least squares sensitivity method, the 100 energy group dependent ENDF/B-4 neutron partial cross sections of Li6, Li7, Fe, F19, Pb, U238 and U235. However, not all the partials for these elements are included. In particular, the option does not exist for any of the inelastic level scattering partials to be adjusted. The connection here is obviously that it is only those partials which have had their sensitivities obtained to various parameters that can be adjusted using this scheme.

As mentioned in an earlier section, a response must be chosen so that the sensitivity of this reaction rate to all the partial cross sections to be adjusted can be obtained. The main criterion for choosing this response is that it should be available for an experimentalist to work with. For the computational calculations there is no point in choosing responses that would be difficult to measure or those which are

exotic, expensive and difficult to obtain. For this reason (Scott 1984) the responses $\text{Ni}(n,\text{proton})$ and $^{232}\text{Th}(n,\text{fission})$ were chosen. A substantial amount of time, energy and expenditure has been invested into obtaining these sensitivity results and profiles; examples of which are breeding rates, nuclear heating and gas production. These were chosen for the work in tuning a fusion reactor to an integral experiment and represent very important system parameters. In theory the responses and sensitivities from these parameters could be used in the cross section adjustment procedure but in practice the reaction rates will either be too low to measure (eg tritium breeding) or virtually impossible to get an accurate reading of (eg nuclear heating). For this reason, none of the reactor parameters used for tuning have been included in the cross section adjustment procedure. A large effort in this work has been towards making the subsequent experimental measurements as easy and as error free as possible.

The cross section adjustment measurements (sensitivities and responses) have been performed on the two integral experiments. It would make no sense to do the analysis on the fusion reactors because they cannot be built to do the experimental analysis on. The results have been obtained by placing the $\text{Ni}(n,p)$ and $^{232}\text{Th}(n,\text{fission})$ foils at various positions in the integral experiments. For each foil, at each chosen spatial position, the sensitivity of this foil to the partial cross sections making up the experiment have been computationally obtained. This response and sensitivity calculation has occurred for both the 1-dimensional case, using the discrete ordinates code ANISN, and the 3-dimensional case, using the Monte Carlo code MORSE. The foils were placed at the centre and off centre positions in the

MORSE geometries, to see how the sensitivity profiles change towards the outer edge. In this way space dependent information can be obtained to see if the sensitivity profiles change substantially towards the integral experiment edges. If they do, then the experimentalist should not make any measurements there, since the obtained sensitivities and corresponding data adjustments will not be relevant to the fusion reactor design. The details and results from this work can be found in chapter 10.

As far as agreement between responses in ANISN and MORSE are concerned, a foil placed in the centre of the MORSE geometry should give the closest result possible to that of an equivalent ANISN one. In MORSE, the foils had an area of 25 cm^2 ($5 \times 5 \text{ cm}^2$) and a thickness of 0.3 cm , whilst for the fissile case the area was 1 cm^2 ($1 \times 1 \text{ cm}^2$). The foil nuclei number density is that of the natural element. For the equivalent ANISN runs, the nuclei number density was adjusted so that the foil would present the same mean free path thickness of foil to a neutron.

It should be noted that both the MORSE and ANISN sensitivities of the $\text{Ni}(n,p)$ and $^{232}\text{Th}(n,\text{fission})$ foils are available for cross section adjustment, though due to time considerations only a few selected partials can be adjusted using the MORSE results, since only those few have their sensitivities obtained.

DESCRIPTION OF REAL AND COMPUTATIONAL NON FISSILE AND FISSILE REACTOR

5.1 INTRODUCTION

This chapter handles the details of the hybrid fusion reactors. In this work two such reactors are considered, a fissile and a non fissile one. The chapter deals with the background to each reactor and the real description of it. Finally, the computational model used in analysing the reactor design via the 1-dimensional code ANISN is described.

5.2 NON FISSILE REACTOR REAL DESCRIPTION

The non fissile reactor described throughout this thesis is really a reactor design called HIBALL, an acronym for heavy ion beams and lithium lead, by Badger et al (1981). The objectives of the HIBALL study are an investigation into the system characteristics of a fusion reactor based on a heavy ion beam accelerator as the driver facility and to identify problems which will need investigation in the future. The report cited above goes into considerable detail in describing the HIBALL reactor; target design, the heavy ion beam driver, beam transmission and final focussing, the cavity design with vacuum systems, materials considerations and neutronics are all dealt with. However, only a general physical overview of the HIBALL reactor will be given here.

The main operating parameters of HIBALL are given in table 1. A DT fusion power of 8,000 MW has been chosen, which results in a net electrical output of 3,768 MW. HIBALL utilises 4 cylindrical reactor chambers of height 10 m, each fired at a repetition rate of 5 Hz. The whole plant layout is shown in figure 1 and the HIBALL reactor is shown in figure 2. The heavy ion beams are Bi^{+2} at 10 GeV and the total energy of each pulse is 4.8 MJ. This energy is delivered in a reactor chamber to the target by 20 beams with a maximum electrical current of 2.5 $\mu\text{A}/\text{beam}$. The 20 beams are propagated through the gas vapour atmosphere of a reaction chamber which contains no more than 10^{11} atoms of Pb vapour cm^{-3} before each shot. The targets are made from lead and lead-lithium surrounding a cryogenic layer of deuterium and tritium. The 4.8 MJ pulse of Bi^{+2} ions is assumed to give a target gain of 83 and a total DT yield of about 400 MJ. With an accelerator efficiency of 26.7%, the fusion gain (driver efficiency times target gain) is 22. If it is assumed that all the beam hits the target and the power requirements of the coolant pumps and vacuum system are included, then this results in a recirculating power fraction of 12.3%.

The first metallic wall, made of HT-9 ferritic steel, has a density of 7.8 g cm^{-3} and a thickness of 1 cm, and is 7 m from the target. Ferritic steel is an alloy composed of 85.25 wt% iron and 11.5 wt% chromium. It is protected from the target x rays, ions and neutrons by an array of porous tubes made of braided SiC through which $\text{Pb}_{83}\text{Li}_{17}$ eutectic is flowing. SiC and $\text{Pb}_{83}\text{Li}_{17}$ have nominal densities of 3.17 and 9.4 g cm^{-3} , respectively. These tubes, called INPORT tubes, an acronym for inhibited flow porous tube concept, occupy 33% of the 2 m thick blanket region in front of the first wall, yielding an effective

thickness of 66 cm. The SiC tubes are protected from the neutrons and ion debris by a thin coating of $Pb_{83}Li_{17}$ flowing through the porous tube wall. This coating is partially vaporised on each shot and recondenses on the tubes in the 0.2 secs between shots. So besides serving as the coolant and tritium breeder, the $Pb_{83}Li_{17}$ coolant is utilised for first wall protection. A 0.4 m thick reflector composed of 90% ferritic steel structure and 10% $Pb_{83}Li_{17}$ coolant is used. The reactor also utilises a 3.5 m thick concrete shield. The shield consists of 95% ordinary concrete and 5% water coolant.

The maximum coolant temperature is 500 °C, insuring that the Pb pressure in the cavity is less than 10^{-5} Torr (normalised to 0 °C). The tritium breeding ratio is 1.25 and the low solubility of tritium in $Pb_{83}Li_{17}$ results in less than 100 grams of inventory in the blanket and fuel processing system. The low solubility also allows all tritium extraction to be done in the cavity itself. Hence, the bred tritium is removed through the vacuum system along with the unburned tritium from the target. The effective 66 cm thickness of the INPORT blanket, composed of 98% $Pb_{83}Li_{17}$ and 2% SiC, reduces the radiation damage in the first structural wall to 2.7 DPA per full power year, meaning that damage equivalent to 5 MW yr m⁻² will be accumulated over 30 calendar years.

5.3 NON FISSILE REACTOR COMPUTATIONAL DESCRIPTION

Although the non fissile reactor is essentially cylindrical in shape, there is also coolant and structure in the top and bottom of the reactor, and it was decided to include this and model it

in a spherical geometry. The materials and amounts and densities used are exactly the same as those given in the HIBALL report. The nuclide densities of the materials is shown in table 2 and a schematic representation of the model is shown in figure 3.

The 1-dimensional discrete ordinates code ANISN was used to perform neutronics and photonics calculations, yielding time averaged integrated results. The following computational description applies for the input to the ANISN code only, as no other modelling was made of the reactors. Since spherical geometry was used, the results represent the conditions at the central plane of the reactor. The DLC37F cross section set was used with the 100 energy groups being employed for the neutron transport and 21 energy groups for the gamma ray transport. The angular flux was represented by a fourth order quadrature (S_4) and the anisotropy of elastically scattered neutrons by a third order Legendre expansion (P_3). The source was considered to be an isotropic, monoenergetic source at 14.1 Mev. The use of more accurate source representations, accounting for spectrum softening and neutron multiplication in the plasma is not dealt with here but a discussion on it can be found in the paper by Beynon et al (1979). One hundred spatial intervals and five spatial zones were utilised. The first two zones near the centre of the reactor contain the plasma source and a 'void' containing a low density (10^{-20}) of first wall material. The plasma source is considered to be contained in a sphere of radius 8 cm. The next zone contains the SiC and $Pb_{83}Li_{17}$ eutectic, and the fourth zone contains the first wall. The outer most zone contains the reflector. Since no concrete shield has been included in the modelling, a 50% albedo was imposed on the reactor outer surface. This means that 50% of all incident particles are reflected

isotropically back into the system. Since the geometry is spherical the boundary condition at the centre of the sphere is perfectly reflecting.

5.4 FISSILE REACTOR REAL DESCRIPTION

A natural path to take in designing a fissile fusion reactor is to simply add a fissile blanket to the non fissile design. This has been done, and a complete sensitivity analysis was performed, though because of space restrictions no results will be given here. A spherical zone of natural uranium, 10 cm thick, was placed between the source and the tritium breeding zone. This fissile region was varied until an energy multiplication of 5 was obtained, which resulted in a blanket thickness of 4.5 cm. However, this design had to be discarded because of the high rate of plutonium production. It was found that just over 30% of the neutrons were producing ^{239}Pu from ^{238}U . The reason for this is that the inelastic, (n,2n) and (n,3n) reactions in the lead substantially soften and backscatter neutrons into the fissile region, thereby producing much plutonium.

A reactor has been designed, hereby called the fissile reactor, which eliminates the $\text{Pb}_{83}\text{Li}_{17}$ eutectic and uses Li_2O as its breeding material. This final fissile reactor design, designed by Dr. B.S Sim (1984) has a blanket energy multiplication of 5 and uses fusion pellets with the very low gain of 8. Since a detailed and complete description of the building and analysis of the reactor is given in Dr. Sim's thesis, only a brief outline will be given here. No attempt will be made to discuss his intermediate fissile designs. Only the physical and computational forms of the final fissile reactor

will be mentioned.

Very briefly, the steps leading to the final cylindrical fissile reactor are as follows. A homogeneous fissile fusion reactor was modelled using the 1-dimensional code ANISN. It contained Li_2O as its breeding material, uranium as its fissile material and a reflector composed mostly of iron. The material amounts were varied until an energy multiplication of 5 was achieved. Taking this design as a standard, it was remodelled in the homogeneous form with the 3-dimensional Monte Carlo code MORSE. Equivalent answers were obtained when a high order quadrature (S_{16}) was used in ANISN. Next, still using the 3-dimensional analysis, it was remodelled in cylindrical geometry with the materials in the heterogeneous form that they would have in a real reactor. In this way the reactor is correctly modelled. The result was a big drop in reactor performance parameters resulting from the increased neutron leakage around the heterogeneous structures. To avoid structural problems the top and bottom 'caps' of the reactor had their uranium and Li_2O removed, leaving only the reflector. To counteract the large neutron leakage, the fissile region was increased by 70%, the thickness of the tritium breeding zone was doubled and the lithium in this zone was enriched to 25% in the Li6 isotope.

The whole idea behind the designing of this fissile reactor is that we wish to examine the consequences of performance when only low gain (10-20) deuterium tritium reactor grade pellets are used as targets. In contrast, the non fissile HIBALL reactor uses pellet designs of thermonuclear gain 83. These may, or may not, prove ultimately feasible. To design the fissile reactor a wall loading of 0.1 Wm^{-2} and an energy multiplication of 5 was chosen.

Assuming that 80% of the fusion energy is given to neutrons and that the heavy ion beam driver is pulsed at 2.5 Hz, a pellet gain of about 8 is required. One reactor chamber with these parameters will generate 175 MW(th). Hence, using a similar beam storage system, as in HIBALL, a 10 chamber system with a 25 Hz heavy ion driver will generate 1.75 GW(th). The system operating parameters of a single chamber are given in table 3 and reactor blanket parameters in table 4.

The reactor is cylindrical in geometry with a total diameter (without biological shield) of 6.5 m and a blanket thickness of 1.5 m. The coolant used throughout the reactor is helium, being largely benign to neutrons and chemically very inert. Helium also has a high heat capacity and requires less pumping power for a given performance. The first wall occurs at 5 m and is composed of uranium carbide (UC) clad in type-316 stainless steel. Uranium carbide has been used because of its high thermal conductivity, high melting temperature (2775 K) and because of the high uranium content in the material. Type-316 stainless steel is chosen as structural material throughout the reactor, mainly because it is relatively cheap and has excellent fabricating and welding properties. The other contender was HT-9 ferritic steel, which although would have had a longer structural lifetime than 316-stainless steel, it is considerably more expensive. The tritium breeding zone is composed of Li_2O , type-316 stainless steel and helium coolant. The choice of lithium oxide in its solid form as the breeding material comes about because it has a high melting temperature (~ 1973 K) and a high lithium content. The tritium recovery, in the form of T_2O , is also relatively easy. The outer 36 cm of the reactor is the reflector region which contains type-316 stainless steel with

helium coolant.

5.5 FISSILE REACTOR COMPUTATIONAL DESCRIPTION

The fissile reactor was modelled in a cylindrical geometry with an angular quadrature of 4 (S_4) and a Legendre expansion of 3 (P_3). Sixty spatial intervals and five spatial zones were used. A U238 fission spectrum was generated from XLACS and used in all the fissile ANISN runs. To account for an outer shield, which has'nt been included in the neutronics, the outer boundary condition was set to an albedo of 50%. As for the sphere geometry, the central boundary condition for a cylinder is that of a perfect reflector. The nuclide densities of the reactor materials are given in table 5 and a schematic diagram of the reactor is given in figure 4. Since all the materials are mixed homogeneously to form distinct spatial zones, no detailed structures are modelled and the neutronics results for the reactor must be considered as averages. A monoenergetic 14.1 Mev isotropic source was used to model the neutrons escaping from the DT plasma. The plasma is a cylinder of radius 8 cm. As in all this work, the 121 energy group DLC37F set was used for the cross sections. The zone next to the plasma zone contains a low nuclide density (10^{-20}) of first surface materials. Moving away from the source, the next zone, zone 3, contains the fissile blanket, 15.8 cm thick, whilst zone 4 contains the tritium breeding zone, 143.2 cm thick which has its lithium enriched to 25% in the Li6 isotope. The outer zone, zone 5, acts as a reflector and is 36 cm thick. The total diameter of the cylindrical reactor, without biological shield is 6.5 m.

Problems were found with this model in trying to achieve a satisfactory particle balance. In order to get an acceptable

neutron convergence in the ANISN code, 50 outer iterations had to be used. The problem stemmed from the fact that the neutron balance in the void region (zone 2) was converging only slowly, due to some neutrons from the fission zone travelling back towards the source and because the mesh structure in this zone was too coarse.

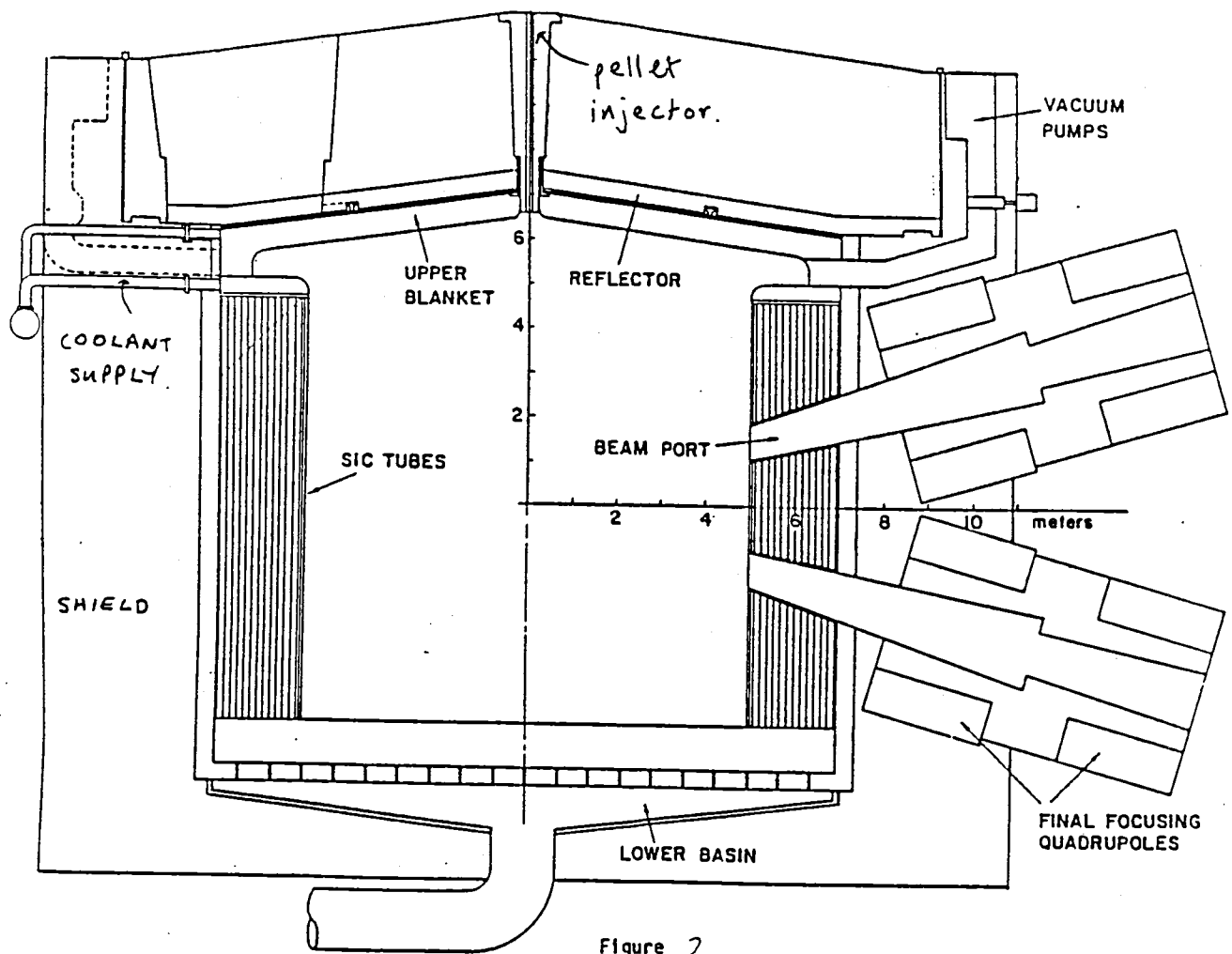


Figure 2.

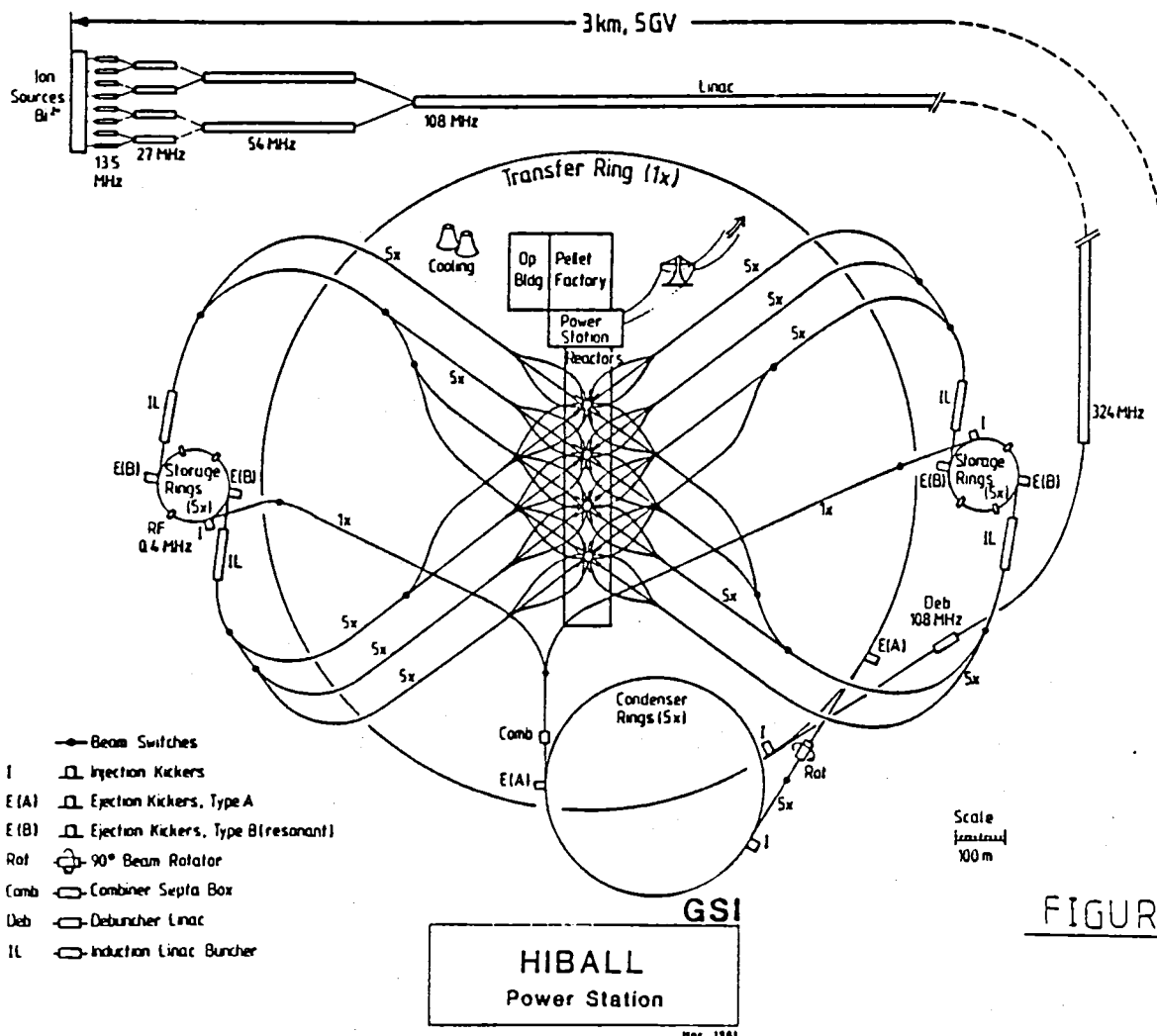


FIGURE 1.

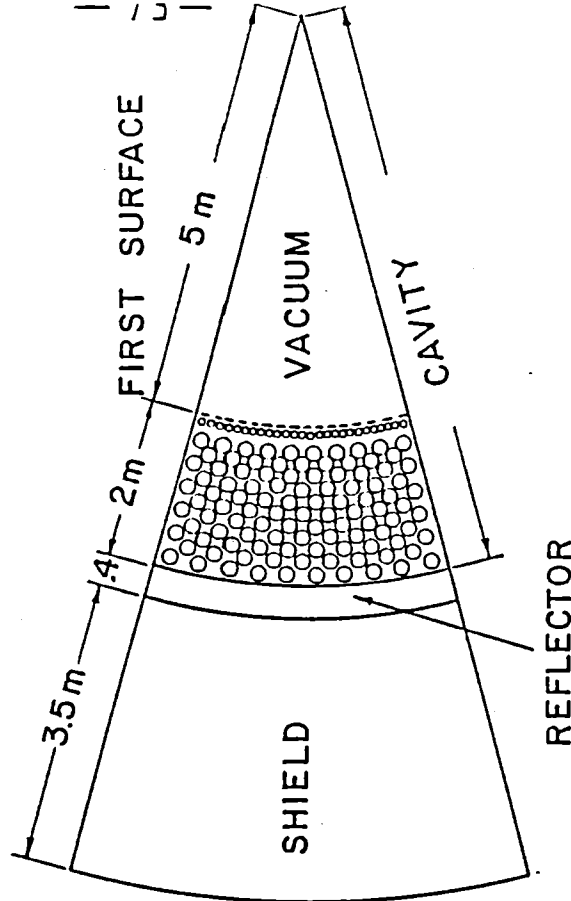


Fig. 3 HIBALL reactor cavity model.

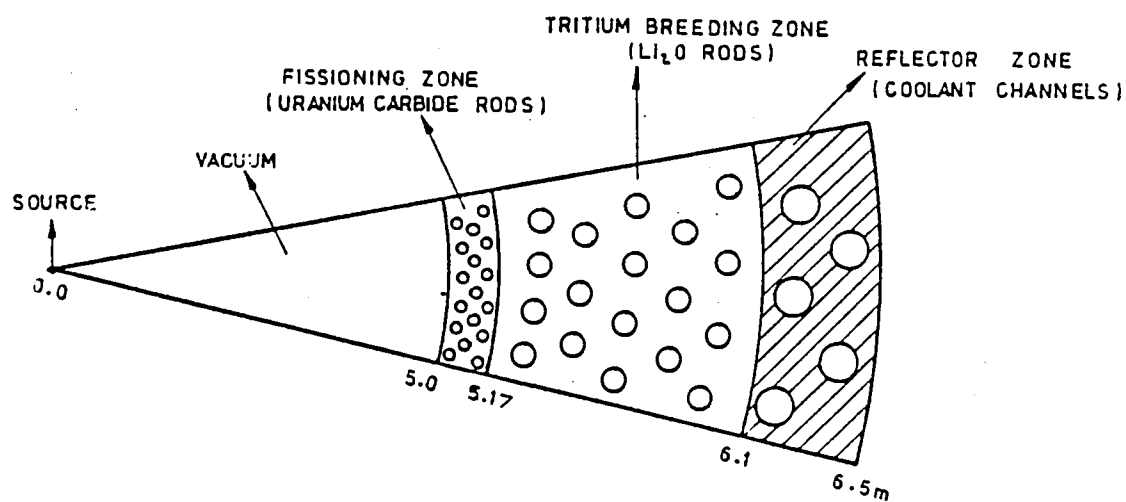


FIGURE 4. FISSIONABLE REACTOR

Table 2.

Nuclide Densities Used in Neutronics Analysis of HIBALL Fusion Reactor

Region	Constituent Elements	Nuclide Density (nuclei/b cm)
Blanket [98 v/o Li ₁₇ Pb ₈₃ + 2 v/o SiC] (.33 density factor)	⁶ Li	0.00040
	⁷ Li	0.00505
	Si	0.00095
	C	0.00095
	Pb	0.02661
Ferritic Steel First Wall (1.0 density factor)	Fe	0.07174
	Cr	0.01725
	Ni	0.00066
	Mo	0.00081
	V	0.00046
	Si	0.00069
	Mn	0.00071
	C	0.00130
	W	0.00022
Reflector [90 v/o ferritic steel + 10 v/o Li ₁₇ Pb ₈₃] (1.0 density factor)	Fe	0.06456
	Cr	0.01553
	Ni	0.00060
	Mo	0.00073
	V	0.00041
	Si	0.00063
	Mn	0.00064
	C	0.00117
	W	0.00020
	⁶ Li	0.00004
	⁷ Li	0.00051
	Pb	0.00272

GENERAL PARAMETERS OF HIBALL

	4/29/81	5/8/81	6/30/81
Average DT power - 17.6 MeV/fusion (MW)	8000	8000	8000
Target power (MW)	7873.6	7920	7920
Target multiplication	0.984	0.99	0.99
Target yield (MJ)	400	400	396
Ion beam pulse energy (MJ)	4.8	4.8	4.8
Overall accelerator efficiency (%)	26.5	26.5	26.7
Target gain	83	83	83
Fusion gain ng	18.5	18.5	22
Blanket multiplication	1.27	1.274	1.274
Total nuclear thermal power (MW)	10149	10193	10193
Gross power		10233	10233
Gross thermal efficiency (%)	42	42	42
Gross electrical output (MW)	4168	4278	4298
Recirculating power fraction	0.2	0.2	0.123
Net electrical output (MW)	3309	3309	3768
# of chambers	4	4	4
Chamber repetition rate (Hz)	5	5	5
Chamber geometry	cylindrical cylindrical cylindrical		
Chamber diameter (m)	14	14	14
Chamber height (m) (at vacuum wall)	10	10	10
Chamber alloy	HT-9	HT-9	HT-9
# of beam ports per chamber	20	20	20
Breeding material	Pb ₈₃ Li ₁₇	Pb ₈₃ Li ₁₇	Pb ₈₃ Li ₁₇
Breeding ratio	1.25	1.25	1.25

TABLE 1.

[FISSILE REACTOR]

TABLE 4 REACTOR BLANKET PARAMETERS

Fissioning Zone	Fertile Material	Uranium Carbide (UC)
	Enrichment	Natural
Clad and Tube Material	Clad and Tube Material	Type-316 stainless steel
	Cooling and Operating Pressure	Helium 5MPa
	First Surface Radius	5 m
	Zone Thickness	0.17 m
Dimension	UC rod O.D.	0.015 m
	Clad O.D.	0.0151 m
	Tube I.D.	0.0202 m
	Tube O.D.	0.0212 m
	Pitch	0.0231 m
	Length	10 m
	Number of Rods	11,520
Tritium Breeding Zone	Tritium Breeding Material	Lithium Oxide (Li ₂ O)
	Enrichment	25% enriched Li6
	Clad and Tube Material	Type-316 Stainless Steel
	Coolant and Operating Pressure	Helium 5MPa
	First Zone Surface Radius	5.17 m
	Zone Thickness	0.93 m
Dimension	Li O Rod O.D.	0.04 m
	Clad O.D.	0.0405 m
	Tube I.D.	0.0445 m
	Tube O.D.	0.0465 m
	Pitch	0.0537 m
	Length	10 m
	Number of Rods	7,200
Reflector Zone	Structural Material	Type-316 Stainless Steel
	Coolant and Operating Pressure	Helium 5MPa
	Coolant Channel I.D.	0.015 m
	Pitch	0.044 m
	Length	10 m
	Number of Coolant Channels	9,720
Chamber Top Plate and Reflector	Structural Material	Type-316 Stainless steel, 90%
	Coolant v/o and Operating Pressure	Helium 5MPa 10%
	Plate Thickness	0.05 m
	Reflector Thickness	0.4 m
	Height at Chamber Centreline	6.5 m
	Structural Material	Type-316 Stainless Steel, 90%
	Coolant v/o and Operating Pressure	Helium 5MPa, 10%
	Plate Thickness	0.05 m
Chamber Bottom Plate and Reflector	Structural Material	Type-316 Stainless Steel, 10%
	Coolant v/o and Operating Pressure	Helium 5MPa, 10%
	Reflector Thickness	0.4 m
Height at Chamber Centreline		6.5 m

TABLE 3

SYSTEM OPERATING PARAMETERS OF A SINGLE REACTOR CHAMBER (FISSILE REACTOR)

Fusion Power	40 MW
Thermal Power	175 MW (th)
Gross Electrical Power	66.5 MW (e)
Net electrical Power	46.6 MW (e)
Target Gain (G)	8
Ion Beam Pulse Energy (E)	2 MJ per pulse
Driver Repetition Rate (f)	2.5 Hz
Driver Efficiency (η_0)	30%
Efficiency Of Thermal Plant (η_a)	38%
Recirculated Power Fraction	30%
Blanket Energy Multiplication	5.1
First Wall Neutron Loading	0.1 MW m ⁻²
First Wall Thermal Loading	41 KW m ⁻²

TABLE 5.
NUCLIDE DENSITIES USED IN FISSILE REACTOR

Region	Constituent Elements	Nuclide Density (nuclei/b cm)
<u>Fissile region</u>		
45% UC, 45% He, 10% Stainless steel	U235	1.068E-4
	U238	1.453E-2
	C	1.463E-2
	Ni	9.850E-4
	Cr	1.575E-3
	Fe	5.642E-3
	Mn55	1.750E-4
	Mo	1.260E-4
	He4	4.734E-4
<u>Breeding region</u>		
20% Li ₁₀ , 75% He, 5% Stainless steel, Li is 25% Li6, and 75% Li7	Ni	4.925E-4
	Cr	7.875E-4
	Fe	2.821E-4
	Mn55	8.750E-5
	Mo	6.300E-5
	He4	7.890E-4
	Li6	4.058E-3
	Li7	1.217E-2
	OL6	8.116E-3
<u>Reflector</u>		
90% Stainless steel, 10% He	Ni	8.865E-3
	Cr	1.418E-2
	Fe	5.078E-2
	Mn55	1.575E-3
	Mo	1.134E-3
	He4	1.052E-4

CHAPTER 6

RESULTS FROM SENSITIVITY ANALYSIS PERFORMED ON NON FISSILE AND FISSILE REACTOR

6.1 INTRODUCTION

The chapter starts with a discussion on the reactor parameters used for the sensitivity analysis and how to compute the profiles. Next, the details of the results obtained from a 1-dimensional neutronic analysis on the non fissile reactor (NFR) and the fissile reactor (FR) are given. Important reactor parameters are presented but the effort was directed into obtaining the most important fluxes (forward and adjoint) and sensitivity profiles to explain neutronic processes in the reactors.

6.2 REACTOR PERFORMANCE PARAMETERS

A large part of the work presented here has been concerned with deriving the relevant sensitivity profiles. A great deal of time, effort and energy has gone into obtaining these profiles and coefficients from the SWANLAKE code. All the sensitivity profiles given here have been stored permanently on tape and are readily available. In performing this detailed sensitivity analysis, seven important reactor design parameters have been chosen. These parameters give information as to how a system behaves. The group wise sensitivity coefficients were obtained for each of the design parameters to each of the group wise partial cross sections of all important elements. The context of

important here means those elements which are contained in both the reactor and its associated integral experiment. This is justified since the most important thing is that the sensitivity profiles for the same reactor parameters should be as close as physically possible.

It should be pointed out that the sensitivity analysis presented in this thesis only deals with the neutronic behaviour of reactor systems and does not consider photon ray processes. The seven parameters are,

- 1) The tritium breeding ratio from $\text{Li6}(n,\alpha)\text{T}$,
- 2) The tritium breeding ratio from $\text{Li7}(n,n',\alpha)\text{T}$,
- 3) The neutron and gamma nuclear heating rates,
- 4) Average atomic displacement rates,
- 5) Gas production rates,

and for the fissile reactor and integral experiment,

- 6) The $\text{U238}(n,f)$ fission reaction rate,
- 7) The $\text{U235}(n,f)$ fission reaction rate.

The nuclear heating in the blanket will directly influence the electric power output and the heat transfer design, and therefore one needs a more accurate value for this quantity. The first wall lifetime will be strongly affected by the DPA and gas production rates in the first wall material, which could be the determining factor for the choice of the first wall material and wall loading. The sensitivity results presented here are as complete as they can be, since the fissile and direct terms are now properly accounted for. However, there are some gaps in the analysis which should be mentioned. For the heating sensitivity, the analysis is carried out only for the heat deposited due to nuclear reactions and takes no account of the decay heat

contribution. The total kerma factor for any element will be equal to the sum of the kerma factors for the partial reactions. The percentage contribution of each partial kerma to the total kerma varies with the energy group value. So, the group dependent kerma factor must be calculated separately for each partial. There is no direct effect contribution on any $Pb(n,3n)$ partials in the heating sensitivity analysis. This is because the calculated kerma factors were negative. Also no direct effects have been included on any of the total inelastic partials for any reactor parameter. This was done because the indirect part of the sensitivity coefficient indicates the degree of flux perturbation due to cross section changes and also because no single Q value can adequately represent the production of a kerma factor from the total inelastic cross section. Therefore, depending on the reactor parameter, results given for some of the inelastic partials are not complete. To calculate the kerma factors for the fission cross sections it was assumed that only the fission fragments deposited their energy locally, and so contributed to the total heating. The Q value used was 168 MeV. The extra 32 MeV is not lost to the system but deposits its energy elsewhere. The Q values used for the heating work, and obtained from the ENDF/B-4 library, are shown in table 1.

For the gas production sensitivity work, the parameter chosen was $Fe(n,p)$. This is because the $Fe(n,p)$ reaction was found to be amongst the highest for all the system types. Also the reaction gives useful information on the behaviour of the high energy flux. The gas production response was taken over the whole reactor system. For the DPA work, no direct terms have been included at all, due to there being no information on the energy dependent DPA cross sections of the partials. The DPA

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sensitivities for Li6, Pb and stainless steel (SS) have been chosen for analysis.

6.3 RESULTS FROM SENSITIVITY ANALYSIS ON NON FISSILE REACTOR

The results presented here, the fluxes, reaction rates and sensitivities have all been obtained from the 1-dimensional transport code ANISN. The reactor performance parameters, heating, gas production and displacements per atom (DPA) are shown in table 2. These reaction rates have been calculated using the fluxes from ANISN and the DLC37F activity cross sections. The gas production rate (APPM/year) was calculated by

$$\sum_i k_1 k_2 \left(\sum_g \sum_{i,g} \phi^{i,g} \right) V_i S / 10^{24} \sum_i N_i V_i$$

and the DPA (per year) was calculated by

$$\sum_i k_1 \left(\sum_g \sum_{i,g} \phi^{i,g} \right) V_i S / 10^{24} \sum_i N_i V_i$$

where $k_1 = 3.154 \times 10^7$ secs,

$$k_2 = 10^6$$

V_i = volume of interval mesh i

S = number of source neutrons per second

$\phi^{i,g}$ = scalar flux for group g, spatial interval i

N_i = nuclide density of material in interval i

The energy integrated, zone sensitivities for certain parameters and partials are shown in tables 3-10. The partials chosen are those which will subsequently be adjusted. The forward scalar flux is shown in figures 1-4 at various spatial positions in the reactor. In traversing the lithium breeding blanket, the 14.1 MeV flux drops by a factor 100, the absorption resonance centred at 285 KeV deepens (fig 3) whilst most of the flux in the energy region 0.1-1.0 MeV retains its energy

dependent shape. The structure in figure 4 is due mostly to the absorption in Fe, since this profile is taken inside the Fe reflector. In particular the absorptions at 1 keV and 9 keV are noticeable, as is the strong absorption valley centred at about 18-20 keV. The flux saddle between 3 MeV and about 12 MeV is due to two effects. Firstly, the charged particle production cross sections (n,p), (n,alpha) etc have high values here with the subsequent loss of neutron flux and secondly, and most importantly, the flux in this energy range can only get there by having one or two elastic scattering events. However, since the Pb(n,2n) and Pb inelastic cross sections are equally important at high energies, the elastic scattering event is not quite so common.

Some selected sensitivity profiles are shown in figures 5-8. Only four profiles have been chosen, but these do give important radiation transport information and indicate quite clearly how sensitivity profiles can be used to gain neutronic information from a system. To be able to fully analyse these profiles, the energy dependences of the cross sections in the reactor and the detector responses must be known. To this end, the elastic, inelastic and absorption cross sections are presented in appendix 2 for all isotopes analysed in this work. It should be pointed out that in all this analysis the energy description of cross sections is in terms of bins of finite width energy. A midpoint energy could be chosen but it is more meaningful to quote the energy group boundaries. This method has been used throughout in presenting results. The sensitivity of $\text{Li6}(n,\alpha)\text{T}$ to $\text{Li6}(n,\alpha)\text{T}$ is shown in figure 5. This was the profile used for designing the non fissile integral experiment. The peak at about 280 keV is a combination of there being a large amount of

- 52 -

flux and because the $\text{Li6}(n,\alpha)\text{T}$ reaction has a large absorption resonance centred at this energy. The total integrated sensitivity coefficient for this partial is 21.6%. This means that increasing the $\text{Li6}(n,\alpha)\text{T}$ reaction by 100% increases the amount of tritium breeding via the $\text{Li6}(n,\alpha)\text{T}$ reaction by 21.6%. The slight increase in the sensitivity profile at lower energies (1 eV-10 eV) is due to some extra breeding in the reflector zone, where the flux is very soft and where there is an albedo of 50% at all energies on the outer reactor surface. The sensitivity profile is negative, although quite small, for the top three energy values. The reason for this is that although the flux is high here, the $\text{Li6}(n,\alpha)\text{T}$ cross section is low and so there is a small direct term. The collisional loss term is equally important here since a 14.1 MeV neutron has a potential multiplicative reaction, and removing a neutron to breed tritium at 14.1 MeV decreases the chance of multiplicative reactions. Hence the direct and collisional loss terms are roughly the same at these energies. Figure 6 gives the sensitivity of $\text{Li6}(n,\alpha)\text{T}$ to $\text{Pb}(n,2n)$. It is a very high profile and is necessarily positive since any increase in the multiplying, energy-degrading $\text{Pb}(n,2n)$ reaction must increase the $\text{Li6}(n,\alpha)\text{T}$ reaction rate. It is clear from this graph that the SWANLAKE code does sum up high energy to low energy neutron transfers back into the energy group that the collision occurred in. The dotted line on the ordinate has no significance and is just a function of the type of plotter used. The sensitivity of $\text{Li7}(n,\alpha\ n')\text{T}$ to $\text{Li7}(n,\alpha\ n')\text{T}$ is shown in figure 7. Clearly any increase in the $\text{Li7}(n,\alpha\ n')\text{T}$ partial cross section will increase the $\text{Li7}(n,\alpha\ n')\text{T}$ reaction rate. This profile is very high, with an integrated sensitivity coefficient

of 97.7%. In general any coefficient greater than 10%, for a 100% cross section perturbation, indicates a region of high sensitivity. The reason for this value being much higher than the $\text{Li6}(n,\alpha)\text{T}$ sensitivity is that the $\text{Li7}(n,\alpha\text{ n}')\text{T}$ reaction produces an additional neutron which may then go into further breeding. The $\text{Li7}(n,\alpha\text{ n}')\text{T}$ reaction has a threshold at about 2.8 MeV and so the sensitivity of this reaction rate to any other partial cross section is zero below this energy. Figure 8 gives the sensitivity of total Pb heating to the total Fe cross section. Since the total cross section is being perturbed, the sensitivity profile contains the effects of all the partial cross sections. A dotted line on the high energy side of a histogram bar indicates that an increase in that group would cause an increase in the reaction rate. It should be noted that peaks in the sensitivity profile for deep penetration transport problems frequently correspond to local minima in $\sum_{\tau}$, and valleys frequently correspond to energy regions where $\sum_{\tau}$ is relatively high. The profile can be seen to be broken up into a positive and negative half. The reason for this is the energy dependent structure of the Pb total kerma factor. The kerma factor has a high energy bias, via the $\text{Pb}(n,2n)$ and $\text{Pb}(n,3n)$ reactions, and below their thresholds via the Pb inelastic cross section, to its threshold energy of about 0.58 MeV. Below this, the elastic and (n,γ) reactions produce a low, monotonically decreasing total kerma factor. The reason for the sign switch is that above 0.58 MeV any Fe reactions occurring may soften neutrons into energies above 0.58 MeV and thus still being able to contribute to the Pb total heating via the Pb inelastic partial. A detailed analysis shows that the Fe elastic partial is responsible for this latter effect. The Fe inelastic contributes

positively to the Pb heating to an energy of 6.1-6.7 MeV, below which its mean secondary neutron energy is below 0.58 MeV, the Pb inelastic threshold. The positive peak centred at about 0.45-0.37 MeV is due to a peak in the Fe elastic partial which is offsetting the loss of the Pb inelastic partial, to give a sensitivity gain in Pb(n,gamma) and Pb elastic cross sections. Below the sensitivity peak, where the profile becomes negative, the Fe elastic and Fe(n,gamma) reactions compete for neutrons. The Fe(n,gamma) reaction necessarily removes neutrons from the system whereas the Fe elastic cross section softens neutrons into lower energy groups. There are a couple of positive sensitivity points at these energies but they are quite small. They indicate energies where the Fe elastic downscattering cross section is more dominant than the Fe(n,gamma). Such regions are at 0.3-0.4 keV and 7-8 keV. The peak at 1 keV is the Fe(n,gamma) absorption resonance. The other structures, at about 0.15 keV, 0.4 keV, 7 keV and 40 keV are resonance windows of which the origins cannot be traced.

Observing the zone sensitivities, it is clear that Pb is the dominant material in this reactor. Indeed, as later analysis will show, it plays a similar role to uranium in the fissile reactor. Because of its high nuclide density and the large amount of it in this reactor, its macroscopic Pb(n,2n), Pb(n,3n) and Pb inelastic cross sections are high. These reactions tend to downscatter neutrons into the energy region 2 MeV-50 keV. The Pb(n,2n) reaction has a threshold at 6.7-7.4 MeV. With an interaction in group 1, the 2 neutrons start appearing at 6.1-6.7 MeV, peak at about 1.4-1.5 MeV, and then decrease steadily till about 50 keV, after which the secondary energy cross section becomes quite small. At lower interaction energies

energies, the $(n,2n)$ peak moves toward lower energies and the secondary energy curve tends to flatten off. The total inelastic cross section peaks at 6.1-6.7 MeV, with the secondary neutrons from this peaking in the range 1-3 MeV. In table 3 and table 4, the sensitivities of $\text{Li6}(n,\alpha)T$ and $\text{Li7}(n,\alpha n')T$ to all partials are given. Adding up the zone sensitivities for an isotope gives the total integrated sensitivity coefficient for the isotope. The $\text{Li6}(n,\alpha)T$ response has a total (all isotopes) sensitivity coefficient of 0.81. The Pb contributes 60% of this sensitivity, primarily through the elastic and $(n,2n)$ cross sections, showing just how important it is to a low energy biased reaction rate. The Li6 isotope contributes 27%, Li7 11% and the Fe 2%. All the reactors and experiments analysed here have shown this insensitivity to Fe. With a high energy response, such as $\text{Li7}(n,\alpha n')T$ in table 3, the situation is different. Any interaction at high energies removes those high energy neutrons from having a possible $\text{Li7}(n,\alpha n')T$ reaction. The total (isotopic) sensitivity coefficient is nearly zero, this being due to the strong Pb negative effects in downscattering being equalled by the $\text{Li7}(n,\alpha n')T$ direct gain term. The Li6 and Fe are fairly insensitive-the Li6 because of its low concentration and the Fe because of the cancellation effects of the Fe elastic cross section against its other partials.

The heating zone sensitivities of tables 5-8 show that in all heating calculations the (n,γ) term decreases the heating although not usually by any considerable amount. This is because most of the heating parameters are high energy biased except for Li6 which is dominated by the $\text{Li6}(n,\alpha)T$ partial. If these heating parameters had been coupled neutron gamma responses then more interesting effects would have occurred, particularly if the

response had a high energy neutron and gamma ray dependence. In this case there would be positive sensitivity for the (n,gamma) reaction, mathematically because the sensitivity downscattering gain term of the equation is invoked, which can come about when a low energy neutron, having a low probability in contributing to the response, is replaced by a high energy gamma ray, which has a high probability in contributing to the response. The sensitivity of Fe and Li7 neutron heating is negative to the Pb(n,2n) partial because even though it is a multiplicative process, it is taking out high energy neutrons, from the high energy biased Fe and Li7 heating rates, and replacing them by low energy ones. The integrated sensitivity coefficient of the Fe heating reaction to the Pb elastic partial is -181%. This indicates that the Fe heating response is indeed very high energy biased and that the Pb elastic cross section degrades neutrons in energy by more than just one or two energy groups. The sensitivity of total gas production via the Fe(n,p) reaction to Fe(n,p) is 96%. All other sensitivities for this response are negative, especially those due to Pb elastic and Pb(n,2n), -357% and -356%, respectively, indicating the high energy nature of Fe(n,p). Contrast this with a low energy biased response function, such as Li6(n,alpha)T, where these two sensitivities are about 8 times lower. Table 4 and table 6 are essentially the same because the Li6(n,alpha)T reaction dominates the Li6 neutron nuclear heating. The Li6(n,p) reaction is now positive because it contributes to the heating through its direct term. The sensitivities of stainless steel have also been obtained. The macroscopic cross sections of the total amount of stainless steel in the first wall and reflector were used as the adjoint source. The steel is extremely sensitive to the Pb elastic partial-the

DPA decreases by 165% for a 100% increase in the Pb elastic partial. This high sensitivity originates from group 1. The Pb elastic partial is softening 14.1 MeV neutrons very successfully in the lithium breeding zone to energies where the DPA response function is a lot smaller. The sensitivity of steel DPA to $Pb(n,2n)$ is slightly less negative. This is due to the fact that, although the $(n,2n)$ reaction softens neutrons further down in energy than the elastic partial, two neutrons are produced which cause more damage than if only one neutron had been produced.

6.4 RESULTS FROM SENSITIVITY ANALYSIS ON FISSILE REACTOR

The essential neutronics difference between the non fissile and fissile reactor is that the latter is cylindrical and has uranium rather than Pb as its neutron multiplication medium. The streaming term on the transport equation is different for spherical and cylindrical geometries, resulting in different neutronic behaviours. The reactor performance parameters for this reactor are given in table 11, and the energy integrated zone sensitivities in tables 12-21. Because of the fissile modification to the sensitivity equation in SWANLAKE, the fissile terms can now be included correctly.

In figures 9-11, the forward scalar flux is plotted against energy for various spatial positions. It should be noted that in the ANISN representation of a cylinder, the flux has dimensions of neutrons per unit length per cm per sec. For this reason the cylindrical fluxes will have a higher magnitude than fluxes obtained from a spherical geometry. This is because the cylinder

is considered infinite in length, as it must do to fulfil its symmetry condition. ANISN cannot calculate the leakage from the ends of the cylinder. By comparing figures 2 and 10 it can be seen that the fissile material is keeping the flux very hard. In particular the amount of flux in the energy range from about 3 MeV to 13 MeV has increased dramatically. Compared to the non fissile flux, the amount of low energy flux is very small indeed, due mostly to fast fissions occurring in the $U^{238}(n,f)$ which keep the flux hard, but also because of the absorption by $Li^6(n,\alpha)T$ which is now 25% enriched and the $Fe(n,\gamma)$ partial cross section. The small blip or spike centred about 8 MeV in figure 9 is due to a resonance in the U fission cross sections and indicates 2nd chance fission.* Figure 10 gives the flux in the middle of the lithium zone. The amount of 14.1 MeV flux is now less than the flux in the 10-100 keV energy range. The valley at about 200 keV is the 285 keV absorption resonance in $Li^6(n,\alpha)T$. The two small valleys to its right are due to the O^{16} elastic partial and the $U^{238}(n,f)$ cross section, respectively. The steady increase in the amount of low energy flux is clearly seen. Figure 11 gives the flux in the middle of the reflector region. The large amount of structure is due to the elastic and (n,γ) partial cross sections of Ni and Cr but mostly due to Fe. There is some 14.1 MeV flux penetrating to these regions but the amount is fairly small.

The scalar adjoint flux has been plotted in figures 12-14. The energy representation here is in the adjoint form and so the flux should have its energy dependence reversed for it to be in the correct form. In this discussion the energy group representation as seen in the graphs is used. Figures 12 and 13 give the adjoint flux for positions in the fission and tritium

*SEE PAGE 256

breeding zones, respectively, for a $U238(n,f)$ adjoint source located throughout the fission zone. Figure 14 corresponds to figure 12 except that the adjoint source is now the $U235(n,f)$ macroscopic cross sections. The adjunctions of this system give a measure of the eventual importance of the birth of a neutron in contributing to the chosen adjoint source. The word eventual is used here since the system has sufficient criticality, 0.26, that fission repeat processes will be important. The qualitative description of the structure seen in figures 12 and 14 is due to the $U238$ isotope, by virtue of its greater isotopic abundance. The resonance structure seen at high energies is due to the $U238$ isotope (n,γ) absorption valleys and peaks, and the adjoint flux forms a reverse image of this. For instance, a neutron born in the $U238(n,\gamma)$ absorption peak centred at 6.4-8.3 eV has a high chance of being absorbed and a low chance of contributing to the adjoint source. The deepest valley in figures 12 and 14 corresponds to this (n,γ) resonance. Figure 12 has a slight rise at low energies and this is due to the increasing $U238(n,f)$ reaction. A neutron born at this energy obviously has more chance of its subsequent interaction going via the $U238(n,f)$ reaction. Figure 13 shows that the adjoint flux has an energy cutoff at positions removed from the adjoint source. This cutoff occurs at progressively lower energies the further the adjunctions are from the detector. Physically the reason for this is that the neutrons are finding it very difficult to pass the non fission absorption reactions and so contribute to the fissile adjoint source. The slight dip in adjoint flux in figure 13 at the 10-100 eV energy range is due to the non fissile absorption at 285 KeV in the $Li6(n,\alpha)T$ partial.

The main differences between figures 12 and 14 is that the $U235(n,f)$ adjoint source gives a higher adjoint flux. The reason is the energy dependent form of the adjoint sources and the repeat fission processes. In this system the forward flux peaks at roughly 1-500 Kev which favours the $U235(n,f)$ adjoint source, since $U238(n,f)$ has a threshold at about 1.8 MeV. A neutron below this energy can have a reaction in $U235(n,f)$. It is a direct process. However, for a neutron at these energies to have a reaction in $U238(n,f)$, it must first have a reaction in $U235(n,f)$ and then a subsequent reaction in $U238(n,f)$. This is not a direct process and so is less likely to happen. Hence, the lower adjoint flux for the $U238(n,f)$ adjoint source.

Some selected sensitivity profiles are shown in figures 15-20. These show important particle processes and, additionally in figures 15, 17 and 18, both direct gain and fissile gain terms have been invoked in the sensitivity equation to produce these profiles. The sensitivities of $U238(n,f)$ in the fissile blanket to the $U238(n,f)$ and $U235(n,f)$ cross sections is shown in figure 15 and 16. The peak in figure 15 corresponds to the product of the fission spectrum and a sharply falling $U238(n,f)$ cross section having a maximum here. The small spike in the energy range 8.2-7.4 MeV occurs only in the fissile region. This is due to a small peak in the fissile cross sections for both isotopes around this energy region. The $U238$ isotope is dominant in this spike because of the larger amounts of it. The peak indicates 2nd chance fission. The sensitivity is very high at the highest energy, reflecting the fact that the fissile cross section and the amount of flux is at its highest here. Figure 16 has essentially the same shape as figure 15 except that the sensitivities are much lower because of the smaller amount of the

U235 isotope. The reason for the positive sensitivity is solely due to the fission gain term in the sensitivity equation. Neutrons are downscattered to lower energies with a subsequent U235(n,f) fission event occurring causing the upscattering sensitivity gain. The sensitivity peak at about 50-500 keV is due to the fact that this is where most of the flux resides, and so the chance of an upscattering gain occurring in this energy range is a maximum. The sensitivity profile does continue to lower energies but becomes very small ($< 10^{-7}$) because of the sharply decreasing value of the fission spectrum and the rapid attenuation of the forward flux. The fission spectrum shape in the fissile sensitivity gain term can just be seen in figure 16 at about 3-6 MeV, before being swamped by the large amounts of forward flux at energies below this.

Figure 17 gives the sensitivity of the U235(n,f) reaction to the U238(n,f) cross section. Its shape is essentially that of figure 15 because the same fission spectrum is being used, though the sensitivities are slightly lower because here the response function is not the same as the operator partial and so there is no direct gain term. The effect of the direct gain term can be seen in that the total coefficient of the sensitivity of U235(n,f) to U238(n,f) is 43%, whilst that of the sensitivity of U238(n,f) to U238(n,f) is 114%. This comes about because the direct gain term integrated over all energies and zones is 100%. Figure 18 gives the sensitivity of U235(n,f) to the U235(n,f) cross section. The shapes of figures 16 and 18 are virtually the same, though figure 18 has much higher sensitivities because of its direct gain term. The U235 resonance effects do exist in the sensitivity profile of figure 18 but they have been cutoff because of their low sensitivities. The direct gain term can be

seen to be making a large contribution to the second chance fission as seen in the spike at energies about 8 MeV.

Figure 19 gives the sensitivity of $\text{Li6}(n,\alpha)\text{T}$ to $\text{Li6}(n,\alpha)\text{T}$ and shows a profile that is positive except for the very lowest energy groups, from thermal energies to about 18 eV. However, these sensitivities are small and the overall sensitivity coefficient, 0.144, is fairly large and positive. The steep high energy slope from about 7 MeV to 200 keV is a combination of a peaking fission spectrum and an increasing $\text{Li6}(n,\alpha)\text{T}$ cross section. The highest peak is the usual 285 keV absorption resonance of $\text{Li6}(n,\alpha)\text{T}$. Additionally, two absorption valleys can be seen on this slope at energies of about 1.8-2.0 MeV and 0.45-0.5 MeV. Both valleys can be identified with absorption resonances in the O16 elastic scattering cross section. These resonances enhance the removal of neutrons from the interaction energy group and increase the amount of subsequent downscatter. The drop in sensitivity comes about because of the reduction in flux—there are now less neutrons with which to breed at this energy. The drop is quite pronounced because of the large amounts of oxygen. The valley at 1.8-2.0 MeV is, however, really quite a small effect but appears deep because the peak slightly to its right, centred at 2.2-2.5 MeV, is a local minimum in the O16 elastic partial. The effect of this minimum is to act as a window in deep penetration problems so that the flux tends to build up in them. Hence, the sensitivity peak at 2.2-2.5 MeV, showing that more breeding is occurring at this energy. The two absorption valleys can also be seen in figure 10. The lower energy sensitivity falls off rapidly because of the rapid attenuation of the forward flux at these energies.

The final profile, figure 20, gives $\text{Li7}(n, \alpha n')T$ to $\text{Li7}(n, \alpha n')T$. The sensitivity is very high, with a total sensitivity coefficient of 1.0. Figure 19 indicates the large amounts of lithium. The contribution of the oxygen to the soft spectrum is quite large. The inelastic and elastic cross sections are most important. The sensitivity of $\text{Li6}(n, \alpha n')T$ is strongly negative in those energy regions where the flux is high. This is because the removal of these energies by an elastic scattering reduces the chance of a fission occurring, most particularly for the ^{235}U isotope. The sensitivity of $\text{Li6}(n, \alpha n')T$ is small and positive, since downscattering the spectrum into the lower energy regions where $\text{Li6}(n, \alpha n')T$ has a large cross section necessitates a higher breeding rate. The sensitivity of $\text{Li6}(n, \alpha n')T$ indicates that the downscattering is removed from the potential fission reactions. Even so the sensitivity is still positive. The sensitivity is negative in the 0.8-1.0 MeV region because there is a small peak in the cross section. This analysis is further discussed in the next section. The dynamics of the fissile reactor is dominated by Li7 , mainly because of the large amounts of ^{235}U and also because of the important high energy $\text{Li7}(n, \alpha n')T$ partials. This can be seen from the sensitivities given in tables 12-21. The ^{56}Fe and ^{59}Fe take little part in the particle spectrum corresponding to low sensitivities. The effect of ^{59}Fe lies somewhere between these two 'extremes', coming through because of its 25% enrichment and importance via $\text{Li6}(n, \alpha n')T$ at energies below 1 MeV.

Li6		Li7	
Elastic	0.0	Elastic	0.0
(n,2n) alpha	-3.696	(n,2n)	-7.252
(n,n') alpha	-1.471	(n,2n) alpha	-8.723
(n,gamma)	7.252	(n,n' alpha) T	-2.466
(n,proton)	-2.733	(n,gamma)	2.032
(n,alpha) T	4.786	(n,deuterium)	-7.760
F19		U238	
Elastic	0.0	Elastic	0.0
(n,2n)	-10.43	(n,2n)	-6.04
(n,n') alpha	-4.878	(n,3n)	-11.46
(n,n') proton	-7.993	(n,fission)	168.00
(n,gamma)	6.601	(n,gamma)	4.804
(n,proton)	-4.036	Fe	
(n,deuterium)	-5.768	Elastic	0.0
(n,tritium)	-7.557	(n,2n)	-11.2
(n,alpha)	-1.523	(n,n') alpha	-7.6
U235		(n,n') proton	-10.2
Elastic	0.0	(n,gamma)	7.803
(n,2n)	-5.306	(n,proton)	-2.731
(n,3n)	-12.147	(n,deuterium)	-7.96
(n,fission)	168.0	(n,tritium)	-11.93
(n,gamma)	6.54	(n,helium 3)	-10.53
Pb		(n,alpha)	0.392
Elastic	0.0		
(n,2n)	-6.733		
(n,3n)	-14.1		
(n,gamma)	5.415		

TABLE 1. Q values used on partials for heating sensitivity work (MeV).

<u>Average DPA/FPY</u>	type-316	Li ₇ Pb ₈₃
Tritium zone	-	35.8
First wall zone	3.8	-
Reflector zone	0.9	0.52
<u>Average gas production in SS-316 APPM/yr</u>		
	H	He
Tritium zone	-	-
first wall zone	1.6	0.08
Reflector zone	0.2	0.01
Li6(n,alpha)T	1.1703	
Li7(n,n' alpha)T	2.963E-2	
Ni neutron heating	1.2232E+3	
Cr neutron heating	1.9143E+4	
Fe neutron heating	8.6363E+4	
C neutron heating	1.7789E+5	
Pb neutron heating	6.0328E+5	
Li6 neutron heating	5.7418E+6	
Li7 neutron heating	8.3101E+5	
V neutron heating	3.5108E+3	
Ni gamma heating	2.5942E+4	
Cr gamma heating	5.4811E+5	
Fe gamma heating	2.5292E+6	
C gamma heating	1.6967E+4	
Pb gamma heating	8.0960E+6	
Li6 gamma heating	1.9387E+3	
Li7 gamma heating	2.4540E+4	
V gamma heating	1.3700E+4	
Mn gamma heating	2.3820E+4	
Si gamma heating	3.2764E+4	
W gamma heating	4.5310E+4	

TABLE 2. Non fissile reactor parameters.
(calculated with the P₃S₄ approximation).

SENSITIVITY COEFFICIENTS OF L17(N,N' ALPHA)T
TO L16,PB,FE,L17 PARTIAL CROSS SECTIONS
NON FISSION REACTOR

	ZONE 4	ZONE 5	ZONE 6	SUM
L16 ELASTIC	-1.0941E-03	0.0	0.1843E-07	-1.0933E-03
L16 (N,2N)ALPHA	-5.0224E-03	0.0	-4.7607E-08	-5.0225E-03
L16 (N,N')ALPHA	-1.1869E-02	0.0	-6.7793E-08	-2.1914E-02
L16 (N,GAMMA)	-8.7656E-07	0.0	-1.0079E-11	-8.7657E-07
L16 (N,PROTON)	-7.4539E-04	0.0	-1.1421E-08	-7.4540E-04
L16 (N,ALPHA)T	-2.6292E-03	0.0	-3.9282E-08	-2.6293E-03
L16 INELASTIC	-1.6891E-02	0.0	-1.1540E-07	-1.6891E-02
PB ELASTIC	7.7494E-03	0.0	1.5794E-05	7.7652E-03
PB (N,2N)	-7.9629E-01	0.0	-8.1807E-06	-7.9629E-01
PB (N,3N)	-2.1508E-02	0.0	-1.5061E-07	-2.1508E-02
PB (N,GAMMA)	-1.0057E-04	0.0	-1.5096E-09	-1.0057E-04
PB INELASTIC	-1.2652E-01	0.0	-2.5642E-06	-1.2652E-01
FE ELASTIC	0.0	9.4450E-05	1.6817E-04	2.6262E-04
FE (N,2N)	0.0	-9.9169E-06	-3.0137E-05	-4.0049E-05
FE (N,N')ALPHA	0.0	-7.0687E-08	-2.1654E-07	-2.8722E-07
FE (N,N')PROTON	0.0	-8.7172E-07	-2.6513E-06	-3.5230E-06
FE (N,GAMMA)	0.0	-1.3604E-08	-3.6120E-08	-4.9724E-08
FE (N,PROTON)	0.0	-4.8394E-06	-1.3752E-05	-1.8592E-05
FE (N,DEUTERIUM)	0.0	-4.4957E-07	-1.3522E-06	-1.8088E-06
FE (N,TRITIUM)	0.0	-9.1160E-09	-2.7950E-08	-3.7066E-08
FE (N,HELIUM 3)	0.0	-6.7204E-09	-2.0592E-08	-2.7312E-08
FE (N,ALPHA)	0.0	-1.3516E-06	-3.9108E-06	-5.2704E-06
FE INELASTIC	0.0	-2.1993E-05	-7.9504E-05	-1.0158E-04
L17 ELASTIC	-1.4737E-03	0.0	0.8214E-07	-1.4728E-03
L17 (N,2N)	-1.5807E-03	0.0	-1.4834E-08	-1.5807E-03
L17 (N,N')ALPHA	-2.3923E-03	0.0	-1.9546E-08	-2.3923E-03
L17 (N,ALPHA N')T	9.7669E-01	0.0	1.2390E-04	9.7681E-01
L17 (N,GAMMA)	-8.4970E-07	0.0	-1.8070E-11	-8.4971E-07
L17 (N,DEUTERIUM)	-7.2492E-04	0.0	-6.2049E-09	-7.2493E-04
L17 INELASTIC	-1.3857E-02	0.0	-5.0795E-08	-1.3857E-02

TABLE 3.

SENSITIVITY COEFFICIENTS OF L16(N,ALPHA)T
TO L16,PB,FE,L17 PARTIAL CROSS SECTIONS
NON FISSION REACTOR

	ZONE 4	ZONE 5	ZONE 6	SUM
L16 ELASTIC	9.2838E-03	0.0	-1.5971E-06	9.2022E-03
L16 (N,2N)ALPHA	4.3465E-05	0.0	9.4449E-09	4.3474E-05
L16 (N,N')ALPHA	-8.7289E-04	0.0	-7.3172E-10	-7.8594E-04
L16 (N,GAMMA)	-2.9380E-05	0.0	-1.6988E-07	-2.9550E-05
L16 (N,PROTON)	-6.1125E-05	0.0	-2.9752E-09	-6.1128E-05
L16 (N,ALPHA)T	1.9944E-01	0.0	1.6782E-02	2.1622E-01
L16 INELASTIC	-8.2943E-04	0.0	9.7131E-09	-8.2942E-04
PB ELASTIC	4.7443E-01	0.0	5.4999E-03	4.7993E-01
PB (N,2N)	9.9823E-02	0.0	2.0852E-05	9.9843E-02
PB (N,3N)	2.8291E-02	0.0	7.4555E-07	2.8292E-02
PB (N,GAMMA)	-8.5869E-02	0.0	-2.2567E-04	-8.5295E-02
PB INELASTIC	-9.5209E-05	0.0	7.1922E-06	-8.8016E-05
FE ELASTIC	0.0	1.1508E-02	5.3473E-02	6.4981E-02
FE (N,2N)	0.0	1.9236E-05	7.6389E-05	9.5626E-05
FE (N,N')ALPHA	0.0	-9.3076E-09	-1.7696E-08	-2.7004E-08
FE (N,N')PROTON	0.0	-1.1871E-07	-1.7530E-07	-2.9409E-07
FE (N,GAMMA)	0.0	-8.5121E-03	-4.1390E-02	-4.9932E-02
FE (N,PROTON)	0.0	-9.4822E-06	-3.5429E-05	-4.4911E-05
FE (N,DEUTERIUM)	0.0	-9.7839E-07	-3.5555E-06	-4.5330E-06
FE (N,TRITIUM)	0.0	-2.1322E-08	-7.6834E-08	-9.8156E-08
FE (N,HELIUM 3)	0.0	-1.5662E-08	-5.6464E-08	-7.2126E-08
FE (N,ALPHA)	0.0	-2.5143E-06	-9.3088E-06	-1.1823E-05
FE INELASTIC	0.0	9.4824E-05	2.1268E-04	3.0742E-04
L17 ELASTIC	1.0243E-01	0.0	2.4954E-05	1.0246E-01
L17 (N,2N)	1.6898E-04	0.0	3.7438E-08	1.6901E-04
L17 (N,N')ALPHA	3.3350E-04	0.0	4.9374E-08	3.3355E-04
L17 (N,ALPHA N')T	-9.0434E-03	0.0	3.4125E-08	-9.0414E-03
L17 (N,GAMMA)	-4.1923E-04	0.0	-2.2094E-06	-4.2144E-04
L17 (N,DEUTERIUM)	-7.0694E-04	0.0	-1.5945E-08	-7.0696E-04
L17 INELASTIC	-5.6461E-03	0.0	6.9554E-07	-5.6454E-03

TABLE 4.

SENSITIVITY COEFFICIENTS OF PB NEUTRON HEATING
TO LI6,PB,FE,L17 PARTIAL CROSS SECTIONS
NON FISSION REACTOR

	ZONE 4	ZONE 5	ZONE 6	SUM
LI6 ELASTIC	-1.5875E-02	0.0	-1.9562E-05	-1.5895E-02
LI6 (N,2N)ALPHA	-1.9445E-05	0.0	3.2639E-09	-1.9441E-05
LI6 (N,N')ALPHA	-1.3858E-06	0.0	-1.6035E-09	-1.3874E-06
LI6 (N,GAMMA)	-6.3253E-05	0.0	-1.2707E-09	-6.3254E-05
LI6 (N,PROTON)	-5.3378E-02	0.0	-6.7153E-05	-5.3445E-02
LI6 (N,ALPHA)T	-1.1708E-03	0.0	-2.5339E-09	-1.1708E-03
LI6 INELASTIC	-2.0831E-05	0.0	1.6595E-09	-2.0829E-05

PB ELASTIC	5.1357E-02	0.0	7.6473E-04	5.0592E-02
PB (N,2N)	8.7208E-02	0.0	1.6514E-05	8.7225E-02
PB (N,N')ALPHA	-1.1088E-02	0.0	-2.0153E-08	-1.1088E-02
PB (N,GAMMA)	-1.7301E-02	0.0	-1.5498E-05	-1.7317E-02
PB INELASTIC	-2.2433E-01	0.0	-2.5610E-05	-2.2435E-01

FE ELASTIC	0.0	7.5514E-04	2.3594E-03	3.1146E-03
FE (N,2N)	0.0	7.6202E-06	2.2992E-05	3.0613E-05
FE (N,N')ALPHA	0.0	-2.0739E-08	-6.3860E-08	-8.4600E-08
FE (N,N')PROTON	0.0	-1.4212E-07	-4.7572E-07	-6.1783E-07
FE (N,GAMMA)	0.0	-1.9962E-04	-8.0706E-04	-1.0067E-03
FE (N,PROTON)	0.0	-4.7954E-06	-1.4875E-05	-1.9670E-05
FE (N,DEUTERIUM)	0.0	-4.6649E-07	-1.4368E-06	-1.9025E-06
FE (N,TRITIUM)	0.0	-9.9752E-09	-3.0578E-08	-4.0553E-08
FE (N,HELIUM 3)	0.0	-7.3341E-09	-2.2488E-08	-2.9822E-08
FE (N,ALPHA)	0.0	-1.2423E-06	-3.8601E-06	-5.1025E-06
FE INELASTIC	0.0	-1.9455E-04	-7.2368E-04	-9.1843E-04

LI7 ELASTIC	-1.5614E-01	0.0	-1.8154E-04	-1.5632E-01
LI7 (N,2N)	-8.6158E-05	0.0	1.2854E-08	-8.6145E-05
LI7 (N,2N)ALPHA	-5.9466E-04	0.0	1.0812E-08	-5.9465E-04
LI7 (N,ALPHA N')T	-9.0055E-03	0.0	4.3852E-08	-9.0054E-03
LI7 (N,GAMMA)	-1.7905E-05	0.0	-2.1613E-08	-1.7926E-05
LI7 (N,DEUTERIUM)	-7.1446E-04	0.0	-6.4906E-09	-7.1447E-04
LI7 INELASTIC	-2.3866E-02	0.0	-3.6825E-06	-2.3870E-02

TABLE 5.

SENSITIVITY COEFFICIENTS OF LI6 NEUTRON HEATING
TO LI6,PB,FE,L17 PARTIAL CROSS SECTIONS
NON FISSION REACTOR

	ZONE 4	ZONE 5	ZONE 6	SUM
LI6 ELASTIC	1.6707E-02	0.0	2.6337E-05	1.6734E-02
LI6 (N,2N)ALPHA	5.5163E-04	0.0	3.9337E-08	5.5167E-04
LI6 (N,N')ALPHA	2.2878E-03	0.0	2.7296E-07	2.2880E-03
LI6 (N,GAMMA)	-2.8870E-05	0.0	-1.4528E-07	-2.9015E-05
LI6 (N,PROTON)	3.9202E-05	0.0	6.6927E-09	3.9208E-05
LI6 (N,ALPHA)T	2.9509E-01	0.0	1.4801E-02	2.1989E-01
LI6 INELASTIC	2.8394E-03	0.0	3.1230E-07	2.8397E-03

PB ELASTIC	4.4755E-01	0.0	4.7345E-03	4.5228E-01
PB (N,2N)	9.2581E-02	0.0	1.4202E-05	9.2595E-02
PB (N,N')ALPHA	1.0989E-02	0.0	4.9526E-07	1.0990E-02
PB (N,GAMMA)	-8.1074E-02	0.0	-1.9070E-04	-8.1264E-02
PB INELASTIC	-6.4522E-03	0.0	4.4840E-06	-6.4477E-03

FE ELASTIC	0.0	1.0199E-02	4.5784E-02	5.5983E-02
FE (N,2N)	0.0	1.3113E-05	5.1831E-05	6.4947E-05
FE (N,N')ALPHA	0.0	-6.7105E-09	-1.3354E-08	-2.0065E-08
FE (N,N')PROTON	0.0	-8.2484E-08	-1.2703E-07	-2.0952E-07
FE (N,GAMMA)	0.0	-7.5607E-03	-3.5680E-02	-4.3241E-02
FE (N,PROTON)	0.0	-6.5707E-06	-2.4435E-05	-3.1006E-05
FE (N,DEUTERIUM)	0.0	-6.6833E-07	-2.4201E-06	-3.0085E-06
FE (N,TRITIUM)	0.0	-1.4510E-08	-5.2110E-08	-6.6619E-08
FE (N,HELIUM 3)	0.0	-1.0660E-08	-3.0301E-08	-4.0961E-08
FE (N,ALPHA)	0.0	-1.7300E-06	-6.3793E-06	-8.1092E-06
FE INELASTIC	0.0	6.6530E-05	1.3558E-04	2.0211E-04

LI7 ELASTIC	9.1551E-02	0.0	1.2740E-05	9.1564E-02
LI7 (N,2N)	1.5713E-04	0.0	2.5615E-08	1.5715E-04
LI7 (N,2N)ALPHA	3.0015E-04	0.0	3.3450E-08	3.0019E-04
LI7 (N,ALPHA N')T	-9.0324E-03	0.0	2.5514E-08	-9.0323E-03
LI7 (N,GAMMA)	-4.1340E-04	0.0	-1.9021E-06	-4.1530E-04
LI7 (N,DEUTERIUM)	-7.0359E-04	0.0	-1.0074E-08	-7.0360E-04
LI7 INELASTIC	-6.1702E-03	0.0	3.9560E-07	-6.1698E-03

TABLE 6.

SENSITIVITY COEFFICIENTS OF FE NEUTRON HEATING
TO LI6,PB,FE,LI7 PARTIAL CROSS SECTIONS
NON FISSILE REACTOR

	ZONE 4	ZONE 5	ZONE 6	SUM
LI6 ELASTIC	-4.2043E-02	0.0	-0.9565E-04	-4.2938E-02
LI6 (N,2N)ALPHA	-1.3951E-04	0.0	-4.4408E-07	-1.3995E-04
LI6 (N,N')ALPHA	-2.2885E-03	0.0	-3.7815E-06	-2.2923E-03
LI6 (N,GAMMA)	-1.8006E-06	0.0	-5.7175E-08	-1.8577E-06
LI6 (N,PROTON)	-0.8004E-05	0.0	-1.1790E-07	-8.8922E-05
LI6 (N,ALPHA)T	-7.3277E-02	0.0	-2.7583E-03	-7.6035E-02
LI6 INELASTIC	-2.4280E-03	0.0	-4.2256E-06	-2.4323E-03
PB ELASTIC	-1.7865E+00	0.0	-2.0767E-02	-1.8073E+00
PB (N,2N)	-3.1924E-01	0.0	-9.5903E-04	-3.2020E-01
PB (N,3N)	-2.4746E-02	0.0	-2.5636E-05	-2.4771E-02
PB (N,GAMMA)	-2.9294E-02	0.0	-7.5152E-04	-3.0146E-02
PB INELASTIC	-4.5045E-01	0.0	-1.3528E-03	-4.5180E-01
FE ELASTIC	0.0	1.4419E-02	2.0705E-01	2.2147E-01
FE (N,2N)	0.0	-5.5119E-04	-2.5437E-03	-3.0949E-03
FE (N,N')ALPHA	0.0	1.6142E-05	7.4292E-05	9.0434E-05
FE (N,N')PROTON	0.0	4.1498E-05	1.8508E-04	2.2658E-04
FE (N,GAMMA)	0.0	-2.5804E-03	-3.5801E-02	-3.8381E-02
FE (N,PROTON)	0.0	1.8591E-03	8.9043E-03	1.0763E-02
FE (N,DEUTERIUM)	0.0	2.2277E-03	1.0801E-02	1.3028E-02
FE (N,TRITIUM)	0.0	1.5127E-04	7.1094E-04	8.6211E-04
FE (N,HELIUM 3)	0.0	5.9540E-07	2.6088E-06	3.2042E-06
FE (N,ALPHA)	0.0	7.7386E-04	3.7070E-03	4.4808E-03
FE INELASTIC	0.0	-6.6390E-03	-3.8162E-02	-4.4801E-02
LI7 ELASTIC	-4.4109E-01	0.0	-9.3919E-03	-4.5049E-01
LI7 (N,2N)	-5.7063E-04	0.0	-1.8104E-06	-5.7244E-04
LI7 (N,2N)ALPHA	-1.4582E-03	0.0	-2.7080E-06	-1.4610E-03
LI7 (N,ALPHA N')T	-1.9378E-02	0.0	-3.3048E-05	-1.9411E-02
LI7 (N,GAMMA)	-2.1395E-05	0.0	-7.8183E-07	-2.2177E-05
LI7 (N,DEUTERIUM)	-1.0613E-03	0.0	-1.1687E-06	-1.0625E-03
LI7 INELASTIC	-5.6878E-02	0.0	-1.9026E-04	-5.7066E-02

TABLE 7.

SENSITIVITY COEFFICIENTS OF LI7 NEUTRON HEATING
TO LI6,PB,FE,LI7 PARTIAL CROSS SECTIONS
NON FISSILE REACTOR

	ZONE 4	ZONE 5	ZONE 6	SUM
LI6 ELASTIC	-1.4221E-02	0.0	-1.5080E-05	-1.4236E-02
LI6 (N,2N)ALPHA	-7.9483E-05	0.0	2.0690E-09	-7.9481E-05
LI6 (N,N')ALPHA	-1.2958E-03	0.0	-5.1137E-09	-1.2958E-03
LI6 (N,GAMMA)	-7.8306E-07	0.0	-1.3282E-09	-7.8439E-07
LI6 (N,PROTON)	-6.3466E-05	0.0	-1.1123E-09	-6.3467E-05
LI6 (N,ALPHA)T	-2.6357E-02	0.0	-4.4294E-05	-2.6401E-02
LI6 INELASTIC	-1.3753E-03	0.0	-3.0447E-09	-1.3753E-03
PB ELASTIC	-4.0030E-02	0.0	4.0210E-04	-3.9628E-02
PB (N,2N)	-1.8879E-01	0.0	3.9635E-06	-1.8878E-01
PB (N,3N)	-1.6748E-02	0.0	-7.7407E-08	-1.6748E-02
PB (N,GAMMA)	-9.6392E-03	0.0	-7.6057E-06	-9.6469E-03
PB INELASTIC	-2.6750E-01	0.0	-2.5045E-05	-2.6752E-01
FE ELASTIC	0.0	6.1323E-04	1.7528E-03	2.3660E-03
FE (N,2N)	0.0	4.0107E-06	1.2867E-05	1.7606E-05
FE (N,N')ALPHA	0.0	-2.8108E-08	-0.6780E-08	-1.1490E-07
FE (N,N')PROTON	0.0	-2.1818E-07	-7.3502E-07	-9.5320E-07
FE (N,GAMMA)	0.0	-1.0461E-04	-4.6575E-04	-5.7036E-04
FE (N,PROTON)	0.0	-4.4042E-06	-1.3131E-05	-1.7515E-05
FE (N,DEUTERIUM)	0.0	-4.2392E-07	-1.2687E-06	-1.6926E-06
FE (N,TRITIUM)	0.0	-9.0021E-09	-2.6895E-08	-3.5898E-08
FE (N,HELIUM 3)	0.0	-6.6210E-09	-1.9784E-08	-2.6405E-08
FE (N,ALPHA)	0.0	-1.1435E-06	-3.4329E-06	-4.5764E-06
FE INELASTIC	0.0	-2.1656E-04	-7.2651E-04	-9.4306E-04
LI7 ELASTIC	5.8452E-01	0.0	1.7923E-03	5.8631E-01
LI7 (N,2N)	6.9400E-03	0.0	4.6027E-07	6.9404E-03
LI7 (N,2N)ALPHA	9.8681E-03	0.0	6.3667E-07	9.8688E-03
LI7 (N,ALPHA N')T	1.6597E-01	0.0	1.3701E-05	1.6599E-01
LI7 (N,GAMMA)	-4.6624E-06	0.0	1.2627E-08	-4.6498E-06
LI7 (N,DEUTERIUM)	4.9150E-03	0.0	3.3453E-07	4.9154E-03
LI7 INELASTIC	-2.6705E-02	0.0	-3.3958E-06	-2.6709E-02

TABLE 8.

SENSITIVITY COEFFICIENTS OF STAINLESS STEEL DPA
TO LI6,PB,FE,L17 PARTIAL CROSS SECTIONS
NON FISSION REACTOR

	ZONE 4	ZONE 5	ZONE 6	SUM
LI6 ELASTIC	-4.2469E-02	0.0	-2.8234E-04	-4.2751E-02
LI6 (N,2N)ALPHA	-7.9830E-05	0.0	5.5752E-08	-7.9775E-05
LI6 (N,N')ALPHA	-1.9413E-03	0.0	3.5537E-08	-1.9412E-03
LI6 (N,GAMMA)	-2.1161E-06	0.0	-1.6237E-08	-2.1323E-06
LI6 (N,PROTON)	-8.3781E-05	0.0	-1.9267E-08	-8.3720E-05
LI6 (N,ALPHA)T	-8.7733E-02	0.0	-8.7505E-04	-8.8608E-02
LI6 INELASTIC	-2.0211E-03	0.0	9.1294E-08	-2.0210E-03

PB ELASTIC	-1.6717E+00	0.0	1.3378E-02	-1.6584E+00
PB (N,2N)	-1.9693E-01	0.0	1.1320E-04	-1.9682E-01
PB (N,3N)	-2.0672E-02	0.0	2.7274E-07	-2.0672E-02
PB (N,GAMMA)	-3.2775E-02	0.0	-2.4195E-04	-3.3017E-02
PB INELASTIC	-4.0623E-01	0.0	-3.8085E-04	-4.0653E-01

FE ELASTIC	0.0	1.7413E-04	5.9673E-02	5.9847E-02
FE (N,2N)	0.0	1.0113E-04	3.9532E-04	4.9646E-04
FE (N,N')ALPHA	0.0	-5.9714E-07	-5.9707E-07	-1.1942E-06
FE (N,N')PROTON	0.0	-4.9494E-06	-2.6522E-06	-7.6016E-06
FE (N,GAMMA)	0.0	-2.9103E-03	-9.9716E-03	-1.2882E-02
FE (N,PROTON)	0.0	-9.9757E-05	-2.1492E-04	-3.1467E-04
FE (N,DEUTERIUM)	0.0	-9.6894E-06	-1.9965E-05	-2.9575E-05
FE (N,TRITIUM)	0.0	-2.0482E-07	-4.2162E-07	-6.2645E-07
FE (N,HELIUM 3)	0.0	-1.5061E-07	-3.1020E-07	-4.6081E-07
FE (N,ALPHA)	0.0	-2.5879E-05	-5.4859E-05	-8.0737E-05
FE INELASTIC	0.0	-3.3470E-03	-8.6536E-03	-1.2003E-02

L17 ELASTIC	-4.2559E-01	0.0	-2.5592E-03	-4.2815E-01
L17 (N,2N)	-3.3053E-04	0.0	2.2007E-07	-3.3031E-04
L17 (N,2N)ALPHA	-1.0665E-03	0.0	1.9346E-07	-1.0663E-03
L17 (N,GAMMA)	-1.6075E-02	0.0	1.3530E-06	-1.6078E-02
L17 (N,ALPHA N')T	-2.6340E-05	0.0	-2.1718E-07	-2.6557E-05
L17 (N,DEUTERIUM)	-9.6918E-04	0.0	-9.0674E-08	-9.6927E-04
L17 INELASTIC	-4.9491E-02	0.0	-4.1814E-05	-4.9533E-02

TABLE 9

SENSITIVITY COEFFICIENTS OF FE(N,P) GAS PROD
TO LI6,PB,FE,L17 PARTIAL CROSS SECTIONS
NON FISSION REACTOR

	ZONE 4	ZONE 5	ZONE 6	SUM
LI6 ELASTIC	-2.2160E-02	0.0	-1.6086E-04	-2.2321E-02
LI6 (N,2N)ALPHA	-1.7307E-03	0.0	-1.2734E-05	-1.7434E-03
LI6 (N,N')ALPHA	-1.1055E-02	0.0	-8.1399E-05	-1.1137E-02
LI6 (N,GAMMA)	-4.5004E-07	0.0	-4.7183E-09	-4.5476E-07
LI6 (N,PROTON)	-2.2860E-04	0.0	-2.2136E-06	-2.3001E-04
LI6 (N,ALPHA)T	-4.3862E-03	0.0	-6.3326E-05	-4.4496E-03
LI6 INELASTIC	-1.2786E-02	0.0	-9.4133E-05	-1.2880E-02

PB ELASTIC	-3.5691E+00	0.0	-8.7865E-03	-3.5779E+00
PB (N,2N)	-3.5368E+00	0.0	-2.7622E-02	-3.5642E+00
PB (N,3N)	-0.6144E-02	0.0	-5.7467E-04	-9.6698E-02
PB (N,GAMMA)	-3.0392E-03	0.0	-5.1247E-05	-3.0905E-03
PB INELASTIC	-9.0007E-01	0.0	-9.5412E-03	-9.0961E-01

FE ELASTIC	0.0	-4.8281E-02	-1.4214E-01	-1.9042E-01
FE (N,2N)	0.0	-2.2638E-02	-1.0608E-01	-1.3324E-01
FE (N,N')ALPHA	0.0	-1.7037E-04	-8.2101E-04	-9.9138E-04
FE (N,N')PROTON	0.0	-2.0150E-03	-9.8106E-03	-1.1826E-02
FE (N,GAMMA)	0.0	-1.6876E-04	-1.3616E-03	-1.5303E-03
FE (N,PROTON)	0.0	1.5575E-01	8.0126E-01	9.5702E-01
FE (N,DEUTERIUM)	0.0	-1.0130E-03	-4.9892E-03	-6.0022E-03
FE (N,TRITIUM)	0.0	-2.2159E-05	-1.0668E-04	-1.2884E-04
FE (N,HELIUM 3)	0.0	-1.6275E-05	-7.8445E-05	-9.4700E-05
FE (N,ALPHA)	0.0	-2.4872E-03	-1.2735E-02	-1.5222E-02
FE INELASTIC	0.0	-6.6048E-02	-3.4744E-01	-4.1349E-01

L17 ELASTIC	-3.2359E-01	0.0	-2.2853E-03	-3.2507E-01
L17 (N,2N)	-6.9543E-03	0.0	-5.1352E-05	-7.0056E-03
L17 (N,2N)ALPHA	-1.0640E-02	0.0	-7.1256E-05	-1.0711E-02
L17 (N,ALPHA N')T	-1.0482E-01	0.0	-7.5924E-04	-1.0588E-01
L17 (N,GAMMA)	-5.3614E-06	0.0	-5.5275E-08	-5.4168E-06
L17 (N,DEUTERIUM)	-3.3268E-03	0.0	-2.2446E-05	-3.3492E-03
L17 INELASTIC	-1.4539E-01	0.0	-1.0899E-03	-1.4648E-01

TABLE 10

Average gas production in SS-316 APPM/yr

	H	He
Fissioning zone	94.8	24.2
Tritium zone	10.7	2.11
Reflector zone	0.15	0.024

Average DPA/FPY

	SS-316	Li ₂ O
Fissioning zone	4.53	-
Tritium zone	0.51	0.177
Reflector zone	0.014	-
Li6(n,alpha)T	1.5081	
Li7(n,n' alpha)T	8.5225E-2	
U238(n,fission)	3.9185E-1	
U235(n,fission)	4.8273E-2	
Li6 neutron heating	7.8901E+6	
Li7 neutron heating	1.1814E+6	
Fe neutron heating	4.1377E+5	
U238 neutron heating	7.2220E+7	
U235 neutron heating	7.5359E+6	
Ni neutron heating	1.6095E+5	
Cr neutron heating	1.0947E+5	
C neutron heating	1.2579E+6	
O neutron heating	4.5296E+5	
He neutron heating	1.5389E+5	
Ni gamma heating	2.1801E+5	
Cr gamma heating	2.8602E+5	
Fe gamma heating	1.1328E+6	
C gamma heating	3.0843E+5	
O gamma heating	2.8713E+5	
Li6 gamma heating	7.4519E+4	
Li7 gamma heating	2.2346E+5	
He gamma heating	1.0025E+4	
Mn gamma heating	3.3381E+4	

TABLE 11. Fissile reactor parameters.
(calculated with the P₃S₄ approximation).

SENSITIVITY COEFFICIENTS OF LI7 (N,ALPHA N') T
TO LI6,FE,LI7,U238,U235 PARTIAL CROSS SECTIONS
FISSILE CYLINDRICAL REACTOR

	ZONE 3	ZONE 4	ZONE 5	SUM
LI6 ELASTIC	0.0	-3.9960E-02	0.0	-3.9960E-02
LI6 (N,2N)ALPHA	0.0	-1.2524E-02	0.0	-1.2524E-02
LI6 (N,N')ALPHA	0.0	-8.1685E-02	0.0	-8.1685E-02
LI6 (N,GAMMA)	0.0	-2.5779E-06	0.0	-2.5779E-06
LI6 (N,PROTON)	0.0	-2.4807E-03	0.0	-2.4807E-03
LI6 (N,ALPHA) T	0.0	-9.4239E-03	0.0	-9.4239E-03
LI6 INELASTIC	0.0	-8.4403E-02	0.0	-8.4403E-02
FE ELASTIC	-5.1423E-02	-1.0454E-02	2.6376E-03	-5.9240E-02
FE (N,2N)	-4.7043E-02	-1.0913E-02	-6.5664E-05	-6.6021E-02
FE (N,N')ALPHA	-3.7305E-04	-1.3537E-04	-3.6843E-07	-5.0874E-04
FE (N,N')PROTON	-4.2477E-03	-1.6711E-03	-5.7004E-06	-5.9245E-03
FE (N,GAMMA)	-5.233E-05	-2.2625E-05	-1.9340E-07	-7.7051E-05
FE (N,PROTON)	-1.4192E-02	-8.4807E-03	-6.0787E-05	-2.2734E-02
FE (N,DEUTERIUM)	-2.0566E-03	-8.5994E-04	-3.3506E-06	-2.9199E-03
FE (N,TRITIUM)	-4.874E-05	-1.7462E-05	-4.6088E-08	-6.6292E-05
FE (N,HELIUM 3)	-3.5683E-05	-1.2069E-05	-3.4672E-08	-4.8507E-05
FE (N,ALPHA)	-4.480E-03	-2.4731E-03	-1.5931E-05	-6.9369E-03
FE INELASTIC	-8.4030E-02	-6.6508E-02	-3.5694E-04	-1.5090E-01
LI7 ELASTIC	0.0	-1.3846E-01	0.0	-1.3846E-01
LI7 (N,2N)	0.0	-1.1816E-02	0.0	-1.1816E-02
LI7 (N,2N)ALPHA	0.0	-1.5021E-02	0.0	-1.5021E-02
LI7 (N,N')ALPHA	0.0	1.7126E+00	0.0	1.7126E+00
LI7 (N,GAMMA)	0.0	-7.5336E-06	0.0	-7.5336E-06
LI7 (N,DEUTERIUM)	0.0	-4.7912E-03	0.0	-4.7912E-03
LI7 INELASTIC	0.0	-2.2132E-01	0.0	-2.2132E-01
U238 ELASTIC	-3.3728E-01	0.0	0.0	-3.3728E-01
U238 (N,2N)	-2.6016E-01	0.0	0.0	-2.6016E-01
U238 (N,3N)	-1.4161E-01	0.0	0.0	-1.4161E-01
U238 TOTAL FISSION	-2.9218E-01	0.0	0.0	-2.9218E-01
U238 (N,GAMMA)	-2.5375E-03	0.0	0.0	-2.5375E-03
U238 TOTAL INELASTIC	-1.1060E-01	0.0	0.0	-1.1060E-01
U235 ELASTIC	-2.5875E-03	0.0	0.0	-2.5875E-03
U235 (N,2N)	-6.6645E-04	0.0	0.0	-6.6645E-04
U235 (N,3N)	-7.2836E-05	0.0	0.0	-7.2836E-05
U235 TOTAL FISSION	8.3861E-04	0.0	0.0	8.3861E-04
U235 (N,GAMMA)	-3.3796E-05	0.0	0.0	-3.3796E-05
U235 TOTAL INELASTIC	-4.2784E-04	0.0	0.0	-4.2784E-04

TABLE 12

SENSITIVITY COEFFICIENTS OF LI6 (N,ALPHA) T
TO LI6,FE,LI7,U238,U235 PARTIAL CROSS SECTIONS
FISSILE CYLINDRICAL REACTOR

	ZONE 3	ZONE 4	ZONE 5	SUM
LI6 ELASTIC	0.0	-5.3508E-03	0.0	-5.3508E-03
LI6 (N,2N)ALPHA	0.0	1.8256E-03	0.0	1.8256E-03
LI6 (N,N')ALPHA	0.0	7.5801E-04	0.0	7.5801E-04
LI6 (N,GAMMA)	0.0	-2.7678E-05	0.0	-2.7678E-05
LI6 (N,PROTON)	0.0	-6.3301E-04	0.0	-6.3301E-04
LI6 (N,ALPHA) T	0.0	1.1446E-01	0.0	1.1446E-01
LI6 INELASTIC	0.0	7.6744E-04	0.0	7.6744E-04
FE ELASTIC	-1.7053E-02	-7.8870E-03	1.7004E-03	-2.3240E-02
FE (N,2N)	-2.6542E-03	2.6333E-03	1.8305E-04	-1.6214E-04
FE (N,N')ALPHA	-1.3497E-04	3.9349E-07	1.1340E-07	-1.3446E-04
FE (N,N')PROTON	-1.4457E-03	6.5056E-06	1.9871E-06	-1.4371E-03
FE (N,GAMMA)	-3.1451E-03	-2.9963E-03	-1.0339E-03	-7.1751E-03
FE (N,PROTON)	-9.4926E-03	-1.2914E-03	-1.0351E-04	-1.0807E-02
FE (N,DEUTERIUM)	-1.2800E-03	-1.1400E-04	-6.5745E-06	-1.4066E-03
FE (N,TRITIUM)	-3.0442E-05	-2.3102E-06	-1.0755E-07	-3.2800E-05
FE (N,HELIUM 3)	-2.2269E-05	-1.7025E-06	-8.0206E-08	-2.4022E-05
FE (N,ALPHA)	-3.8491E-03	-3.4232E-04	-2.5527E-05	-3.2168E-03
FE INELASTIC	-3.2801E-02	3.5123E-04	3.7064E-04	-3.2159E-02
LI7 ELASTIC	0.0	-5.8212E-03	0.0	-5.8212E-03
LI7 (N,2N)	0.0	1.7063E-03	0.0	1.7063E-03
LI7 (N,2N)ALPHA	0.0	2.1083E-03	0.0	2.1083E-03
LI7 (N,N')ALPHA	0.0	1.7191E-03	0.0	1.7191E-03
LI7 (N,GAMMA)	0.0	-1.0300E-04	0.0	-1.0300E-04
LI7 (N,DEUTERIUM)	0.0	-6.3768E-04	0.0	-6.3768E-04
LI7 INELASTIC	0.0	2.1611E-03	0.0	2.1611E-03
U238 ELASTIC	-4.1320E-02	0.0	0.0	-4.1320E-02
U238 (N,2N)	-1.4202E-02	0.0	0.0	-1.4202E-02
U238 (N,3N)	2.0790E-02	0.0	0.0	2.0790E-02
U238 TOTAL FISSION	4.0742E-01	0.0	0.0	4.0742E-01
U238 (N,GAMMA)	-2.3802E-01	0.0	0.0	-2.3802E-01
U238 TOTAL INELASTIC	-1.5708E-01	0.0	0.0	-1.5708E-01
U235 ELASTIC	-1.8094E-04	0.0	0.0	-1.8094E-04
U235 (N,2N)	-2.2840E-05	0.0	0.0	-2.2840E-05
U235 (N,3N)	1.0185E-05	0.0	0.0	1.0185E-05
U235 TOTAL FISSION	4.2032E-02	0.0	0.0	4.2032E-02
U235 (N,GAMMA)	-3.0613E-03	0.0	0.0	-3.0613E-03
U235 TOTAL INELASTIC	-7.6341E-04	0.0	0.0	-7.6341E-04

TABLE 13

SENSITIVITY COEFFICIENTS OF 238U (N,FISSION)
TO LI6,FE,LI7,U238,U235 PARTIAL CROSS SECTIONS
FISSILE CYLINDRICAL REACTOR

	ZONE 3	ZONE 4	ZONE 5	SUM		ZONE 3	ZONE 4	ZONE 5	SUM
LI6 ELASTIC	0.0	7.9905E-03	0.0	7.9905E-03	LI6 ELASTIC	0.0	2.1023E-02	0.0	2.1023E-02
LI6 (N,2N)ALPHA	0.0	-7.2910E-05	0.0	-7.2910E-05	LI6 (N,2N)ALPHA	0.0	5.5734E-04	0.0	5.5734E-04
LI6 (N,N')ALPHA	0.0	-6.6260E-04	0.0	-6.6260E-04	LI6 (N,N')ALPHA	0.0	9.6910E-04	0.0	9.6910E-04
LI6 (N,GAMMA)	0.0	-2.3115E-07	0.0	-2.3115E-07	LI6 (N,GAMMA)	0.0	-3.9244E-06	0.0	-3.9244E-06
LI6 (N,PROTON)	0.0	-1.3143E-04	0.0	-1.3143E-04	LI6 (N,PROTON)	0.0	-1.9902E-04	0.0	-1.9902E-04
LI6 (N,ALPHA)T	0.0	-4.6295E-03	0.0	-4.6295E-03	LI6 (N,ALPHA)T	0.0	-1.3809E-01	0.0	-1.3809E-01
LI6 INELASTIC	0.0	-6.5410E-04	0.0	-6.5410E-04	LI6 INELASTIC	0.0	9.8610E-04	0.0	9.8610E-04
FE ELASTIC	1.4969E-02	5.5017E-03	1.1305E-05	2.8562E-02	FE ELASTIC	3.9307E-02	2.0493E-02	3.1321E-05	5.9811E-02
FE (N,2N)	-1.8989E-02	-1.8265E-04	-2.1229E-07	-1.9172E-02	FE (N,2N)	-1.0712E-03	8.7788E-04	1.5364E-06	1.9506E-03
FE (N,N')ALPHA	-1.9614E-04	-2.4449E-06	-1.6180E-09	-1.9859E-04	FE (N,N')ALPHA	-1.0210E-04	5.9048E-07	-2.7022E-09	-1.0159E-04
FE (N,N')PROTON	-1.9870E-03	-2.3201E-05	-2.1919E-08	-2.0182E-03	FE (N,N')PROTON	-1.1119E-03	1.0070E-05	-2.1866E-08	-1.1038E-03
FE (N,GAMMA)	-1.9706E-04	-1.2529E-05	-3.5254E-08	-2.0622E-04	FE (N,GAMMA)	-4.9760E-03	-4.0944E-04	-1.3765E-06	-5.3870E-03
FE (N,PROTON)	-8.6652E-03	-2.6227E-04	-2.5571E-07	-8.9277E-03	FE (N,PROTON)	-8.5617E-03	-3.8059E-04	-1.6366E-06	-8.9439E-03
FE (N,DEUTERIUM)	-1.1721E-03	-1.9549E-05	-1.6154E-08	-1.1917E-03	FE (N,DEUTERIUM)	-1.1542E-03	-3.8105E-05	-9.8845E-08	-1.1845E-03
FE (N,TRITIUM)	-2.7672E-05	-3.7055E-07	-2.3929E-10	-2.8051E-05	FE (N,TRITIUM)	-2.7252E-05	-6.0050E-07	-1.5580E-09	-2.7854E-05
FE (N,HELIUM 3)	-2.8243E-05	-2.7917E-07	-1.7955E-10	-2.8523E-05	FE (N,HELIUM 3)	-1.9936E-05	-4.4272E-07	-1.1643E-09	-2.0380E-05
FE (N,ALPHA)	-2.6059E-03	-6.8223E-05	-6.7431E-08	-2.6742E-03	FE (N,ALPHA)	-2.5645E-03	-9.8301E-05	-4.0313E-07	-2.6633E-03
FE INELASTIC	-5.0719E-02	-1.5752E-03	-1.2174E-06	-5.2294E-02	FE INELASTIC	-2.1544E-02	1.2227E-04	-1.9460E-06	-2.1444E-02
LI7 ELASTIC	0.0	2.6035E-02	0.0	2.6035E-02	LI7 ELASTIC	0.0	5.9235E-02	0.0	5.9235E-02
LI7 (N,2N)	0.0	-7.0028E-05	0.0	-7.0028E-05	LI7 (N,2N)	0.0	5.2293E-04	0.0	5.2293E-04
LI7 (N,2N)ALPHA	0.0	-2.4184E-04	0.0	-2.4184E-04	LI7 (N,2N)ALPHA	0.0	5.9688E-04	0.0	5.9688E-04
LI7 (N,ALPHA N')T	0.0	3.8185E-04	0.0	3.8185E-04	LI7 (N,ALPHA N')T	0.0	2.6907E-03	0.0	2.6907E-03
LI7 (N,GAMMA)	0.0	-7.1943E-07	0.0	-7.1943E-07	LI7 (N,GAMMA)	0.0	-1.3805E-05	0.0	-1.3805E-05
LI7 (N,DEUTERIUM)	0.0	-1.1331E-04	0.0	-1.1331E-04	LI7 (N,DEUTERIUM)	0.0	-1.7174E-04	0.0	-1.7174E-04
LI7 INELASTIC	0.0	7.0662E-04	0.0	7.0662E-04	LI7 INELASTIC	0.0	4.4610E-03	0.0	4.4610E-03
U238 ELASTIC	1.1708E-01	0.0	0.0	1.1708E-01	U238 ELASTIC	2.2808E-01	0.0	0.0	2.2808E-01
U238 (N,2N)	-1.2631E-01	0.0	0.0	-1.2631E-01	U238 (N,2N)	7.5241E-03	0.0	0.0	7.5241E-03
U238 (N,3N)	-7.0967E-02	0.0	0.0	-7.0967E-02	U238 (N,3N)	4.1152E-02	0.0	0.0	4.1152E-02
U238 TOTAL FISSION	1.1498E+00	0.0	0.0	1.1498E+00	U238 TOTAL FISSION	4.3043E-01	0.0	0.0	4.3043E-01
U238 (N,GAMMA)	-1.4996E-02	0.0	0.0	-1.4996E-02	U238 (N,GAMMA)	-3.9521E-01	0.0	0.0	-3.9521E-01
U238 TOTAL INELASTIC	-2.4806E-01	0.0	0.0	-2.4806E-01	U238 TOTAL INELASTIC	-2.6054E-02	0.0	0.0	-2.6054E-02
U235 ELASTIC	0.8871E-04	0.0	0.0	0.8871E-04	U235 ELASTIC	1.5733E-03	0.0	0.0	1.5733E-03
U235 (N,2N)	-2.9807E-04	0.0	0.0	-2.9807E-04	U235 (N,2N)	2.8452E-05	0.0	0.0	2.8452E-05
U235 (N,3N)	-3.7419E-05	0.0	0.0	-3.7419E-05	U235 (N,3N)	2.0980E-05	0.0	0.0	2.0980E-05
U235 TOTAL FISSION	2.6475E-02	0.0	0.0	2.6475E-02	U235 TOTAL FISSION	1.0557E+00	0.0	0.0	1.0557E+00
U235 (N,GAMMA)	-1.7361E-04	0.0	0.0	-1.7361E-04	U235 (N,GAMMA)	-5.0108E-03	0.0	0.0	-5.0108E-03
U235 TOTAL INELASTIC	-9.6129E-04	0.0	0.0	-9.6129E-04	U235 TOTAL INELASTIC	8.7427E-06	0.0	0.0	8.7427E-06

TABLE 14.

TABLE 15.

SENSITIVITY COEFFICIENTS OF LI7 NUCLEAR HEATING
TO LI6,FE,LI7,U238,U235 PARTIAL CROSS SECTIONS
FISSILE CYLINDRICAL REACTOR

	ZONE 3	ZONE 4	ZONE 5	SUM
LI6 ELASTIC	0.0	-6.5895E-02	0.0	-6.5895E-02
LI6 (N,2N)ALPHA	0.0	-6.5638E-03	0.0	-6.5638E-03
LI6 (N,N')ALPHA	0.0	-5.5739E-02	0.0	-5.5739E-02
LI6 (N,GAHMA)	0.0	-3.3766E-06	0.0	-3.3766E-06
LI6 (N,PROTON)	0.0	-2.2023E-03	0.0	-2.2023E-03
LI6 (N,ALPHA)T	0.0	-5.3594E-02	0.0	-5.3594E-02
LI6 INELASTIC	0.0	-5.6209E-02	0.0	-5.6209E-02
FE ELASTIC	-4.6418E-02	-1.1996E-02	2.2163E-03	-5.6198E-02
FE (N,2N)	-3.5268E-02	-1.1411E-02	4.7395E-06	-4.6675E-02
FE (N,N')ALPHA	-3.1647E-04	-9.4544E-05	-2.8802E-07	-4.1139E-04
FE (N,N')PROTON	-3.4867E-03	-1.1147E-03	-3.3509E-06	-4.6048E-03
FE (N,GAHMA)	-3.1164E-04	-1.3954E-04	-1.5224E-05	-4.6641E-04
FE (N,PROTON)	-1.2908E-02	-6.1422E-03	-7.4792E-05	-1.9125E-02
FE (N,DEUTERIUM)	-1.8490E-03	-6.3349E-04	-4.2158E-06	-2.4867E-03
FE (N,TRITIUM)	-4.3913E-05	-1.3051E-05	-6.2973E-08	-5.7027E-05
FE (N,HELIUM 3)	-3.2118E-05	-9.6123E-06	-4.7208E-08	-4.1792E-05
FE (N,ALPHA)	-3.9871E-03	-1.7597E-03	-1.8372E-05	-5.7657E-03
FE INELASTIC	-7.0314E-02	-4.7405E-02	-3.4505E-04	-1.2608E-01
LI7 ELASTIC	0.0	3.7625E-01	0.0	3.7625E-01
LI7 (N,2N)	0.0	2.2828E-03	0.0	2.2828E-03
LI7 (N,2N)ALPHA	0.0	2.0166E-03	0.0	2.0166E-03
LI7 (N,ALPHA N')T	0.0	1.7745E-01	0.0	1.7745E-01
LI7 (N,GAHMA)	0.0	-5.1242E-06	0.0	-5.1242E-06
LI7 (N,DEUTERIUM)	0.0	2.8287E-03	0.0	2.8287E-03
LI7 INELASTIC	0.0	-1.3778E-01	0.0	-1.3778E-01
U238 ELASTIC	-2.7142E-01	0.0	0.0	-2.7142E-01
U238 (N,2N)	-1.9519E-01	0.0	0.0	-1.9519E-01
U238 (N,3N)	-1.0951E-01	0.0	0.0	-1.0951E-01
U238 TOTAL FISSION	-5.7612E-01	0.0	0.0	-5.7612E-01

SENSITIVITY COEFFICIENTS OF LI6 NUCLEAR HEATING
TO LI6,FE,LI7,U238,U235 PARTIAL CROSS SECTIONS
FISSILE CYLINDRICAL REACTOR

	ZONE 3	ZONE 4	ZONE 5	SUM
LI6 ELASTIC	0.0	1.5258E-02	0.0	1.5258E-02
LI6 (N,2N)ALPHA	0.0	4.0593E-03	0.0	4.0593E-03
LI6 (N,N')ALPHA	0.0	2.0541E-02	0.0	2.0541E-02
LI6 (N,GAHMA)	0.0	-2.5402E-05	0.0	-2.5402E-05
LI6 (N,PROTON)	0.0	6.6992E-05	0.0	6.6992E-05
LI6 (N,ALPHA)T	0.0	1.7906E-01	0.0	1.7906E-01
LI6 INELASTIC	0.0	-3.0646E-03	0.0	-3.0646E-03
FE ELASTIC	-1.9297E-02	-8.2999E-03	1.7215E-03	-2.5875E-02
FE (N,2N)	-4.8706E-03	1.7127E-03	1.7149E-04	-2.9864E-03
FE (N,N')ALPHA	-1.4758E-04	-5.9382E-06	8.6842E-08	-1.5341E-04
FE (N,N')PROTON	-1.5868E-03	-6.8319E-05	1.6254E-06	-1.6535E-03
FE (N,GAHMA)	-2.9775E-03	-2.7664E-03	-9.4931E-04	-6.6432E-03
FE (N,PROTON)	-9.7394E-03	-1.6265E-03	-1.0181E-04	-1.1468E-02
FE (N,DEUTERIUM)	-1.3277E-03	-1.4929E-04	-6.4267E-06	-1.4815E-03
FE (N,TRITIUM)	-3.1395E-05	-3.0371E-06	-1.0471E-07	-3.4537E-05
FE (N,HELIUM 3)	-2.2966E-05	-2.2308E-06	-7.8106E-08	-2.5284E-05
FE (N,ALPHA)	-2.9305E-03	-4.3967E-04	-2.5101E-05	-3.3953E-03
FE INELASTIC	-3.8121E-02	-2.8998E-03	3.2013E-04	-3.8700E-02
LI7 ELASTIC	0.0	-2.4335E-02	0.0	-2.4335E-02
LI7 (N,2N)	0.0	1.1826E-03	0.0	1.1826E-03
LI7 (N,2N)ALPHA	0.0	1.3439E-03	0.0	1.3439E-03
LI7 (N,ALPHA N')T	0.0	-5.4376E-03	0.0	-5.4376E-03
LI7 (N,GAHMA)	0.0	-9.5341E-05	0.0	-9.5341E-05
LI7 (N,DEUTERIUM)	0.0	-8.3242E-04	0.0	-8.3242E-04
LI7 INELASTIC	0.0	-7.7223E-03	0.0	-7.7223E-03
U238 ELASTIC	-5.7780E-02	0.0	0.0	-5.7780E-02
U238 (N,2N)	-2.6684E-02	0.0	0.0	-2.6684E-02
U238 (N,3N)	-1.7568E-02	0.0	0.0	-1.7568E-02
U238 TOTAL FISSION	-1.1756E-01	0.0	0.0	-1.1756E-01

SENSITIVITY COEFFICIENTS OF FE NUCLEAR HEATING
TO LI6,FE,L17,U238,U235 PARTIAL CROSS SECTIONS
FISSILE CYLINDRICAL REACTOR

	ZONE 3	ZONE 4	ZONE 5	SUM
LI6 ELASTIC	0.0	-2.0365E-02	0.0	-2.0365E-02
LI6 (N,2N)ALPHA	0.0	-3.2756E-03	0.0	-3.2756E-03
LI6 (N,N')ALPHA	0.0	-2.3367E-02	0.0	-2.3367E-02
LI6 (N,GAMMA)	0.0	-1.1651E-06	0.0	-1.1651E-06
LI6 (N,PROTON)	0.0	-6.4192E-04	0.0	-6.4192E-04
LI6 (N,ALPHA)T	0.0	-1.9587E-02	0.0	-1.9587E-02
LI6 INELASTIC	0.0	-2.3607E-02	0.0	-2.3607E-02
FE ELASTIC	1.2860E-01	4.7645E-02	5.4722E-03	1.8172E-01
FE (N,2N)	2.3217E-02	5.7730E-03	-9.5688E-04	2.8033E-02
FE (N,N')ALPHA	3.2992E-03	7.1170E-04	2.7893E-05	4.0380E-03
FE (N,N')PROTON	1.6620E-02	3.7910E-03	8.4746E-05	2.0462E-02
FE (N,GAMMA)	-1.4178E-04	-3.8823E-06	-2.4368E-04	-3.8934E-04
FE (N,PROTON)	2.5609E-01	8.1876E-02	6.1520E-03	3.4412E-01
FE (N,DEUTERIUM)	1.9471E-02	4.6772E-03	2.0934E-04	2.4358E-02
FE (N,TRITIUM)	1.4812E-04	3.2553E-05	3.1473E-07	1.8092E-04
FE (N,HELIUM 3)	1.0966E-04	4.1180E-05	1.0426E-06	2.3194E-04
FE (N,ALPHA)	1.0389E-01	3.2111E-02	2.4397E-03	1.3844E-01
FE INELASTIC	-5.1899E-02	-1.6699E-02	-9.2738E-03	-7.7872E-02
L17 ELASTIC	0.0	-7.0023E-02	0.0	-7.0023E-02
L17 (N,2N)	0.0	-3.1353E-03	0.0	-3.1353E-03
L17 (N,2N)ALPHA	0.0	-4.4147E-03	0.0	-4.4147E-03
L17 (N,ALPHA N')T	0.0	-5.1447E-02	0.0	-5.1447E-02
L17 (N,GAMMA)	0.0	-3.4333E-06	0.0	-3.4333E-06
L17 (N,DEUTERIUM)	0.0	-1.5044E-03	0.0	-1.5044E-03
L17 INELASTIC	0.0	-6.4594E-02	0.0	-6.4594E-02
U238 ELASTIC	-8.8710E-03	0.0	0.0	-8.8710E-03
U238 (N,2N)	-1.4556E-01	0.0	0.0	-1.4556E-01
U238 (N,3N)	-8.1646E-02	0.0	0.0	-8.1646E-02
U238 TOTAL FISSION	-8.5904E-02	0.0	0.0	-8.5904E-02
U238 (N,GAMMA)	-1.7183E-02	0.0	0.0	-1.7183E-02
U238 TOTAL INELASTIC	-1.1327E-01	0.0	0.0	-1.1327E-01
U235 ELASTIC	-7.0495E-05	0.0	0.0	-7.0495E-05
U235 (N,2N)	-3.6667E-04	0.0	0.0	-3.6667E-04
U235 (N,3N)	-4.2239E-05	0.0	0.0	-4.2239E-05
U235 TOTAL FISSION	-9.0283E-03	0.0	0.0	-9.0283E-03
U235 (N,GAMMA)	-2.1061E-04	0.0	0.0	-2.1061E-04
U235 TOTAL INELASTIC	-4.9441E-04	0.0	0.0	-4.9441E-04

TABLE 18

SENSITIVITY COEFFICIENTS OF 238U NUCLEAR HEATING
TO LI6,FE,L17,U238,U235 PARTIAL CROSS SECTIONS
FISSILE CYLINDRICAL REACTOR

	ZONE 3	ZONE 4	ZONE 5	SUM
LI6 ELASTIC	0.0	7.9266E-03	0.0	7.9266E-03
LI6 (N,2N)ALPHA	0.0	-8.2368E-05	0.0	-8.2368E-05
LI6 (N,N')ALPHA	0.0	-6.5675E-04	0.0	-6.5675E-04
LI6 (N,GAMMA)	0.0	-2.2284E-07	0.0	-2.2284E-07
LI6 (N,PROTON)	0.0	-1.2778E-04	0.0	-1.2778E-04
LI6 (N,ALPHA)T	0.0	-4.4364E-03	0.0	-4.4364E-03
LI6 INELASTIC	0.0	-6.4862E-04	0.0	-6.4862E-04
FE ELASTIC	1.4851E-02	5.4590E-03	1.0976E-05	2.0321E-02
FE (N,2N)	-1.9517E-02	-1.9041E-04	-2.0709E-07	-1.9707E-02
FE (N,N')ALPHA	-1.5644E-04	-2.4050E-06	-1.5700E-09	-1.9885E-04
FE (N,N')PROTON	-2.0112E-03	-2.3375E-05	-2.1358E-08	-2.0346E-03
FE (N,GAMMA)	-1.8918E-04	-1.2038E-05	-3.3902E-08	-2.0125E-04
FE (N,PROTON)	-0.6106E-03	-2.5594E-04	-2.4650E-07	-8.6672E-03
FE (N,DEUTERIUM)	-1.1670E-03	-1.9068E-05	-1.5634E-08	-1.1660E-03
FE (N,TRITIUM)	-2.7555E-05	-3.6092E-07	-2.3171E-10	-2.7924E-05
FE (N,HELIUM 3)	-2.0158E-05	-2.7228E-07	-1.7306E-10	-2.0430E-05
FE (N,ALPHA)	-2.5917E-03	-6.6629E-05	-6.5146E-08	-2.6583E-03
FE INELASTIC	-5.0126E-02	-1.5432E-03	-1.1711E-06	-5.1671E-02
L17 ELASTIC	0.0	2.5767E-02	0.0	2.5767E-02
L17 (N,2N)	0.0	-7.8862E-05	0.0	-7.8862E-05
L17 (N,2N)ALPHA	0.0	-2.4047E-04	0.0	-2.4047E-04
L17 (N,ALPHA N')T	0.0	2.0788E-04	0.0	2.0788E-04
L17 (N,GAMMA)	0.0	-6.9133E-07	0.0	-6.9133E-07
L17 (N,DEUTERIUM)	0.0	-1.1057E-04	0.0	-1.1057E-04
L17 INELASTIC	0.0	6.4000E-04	0.0	6.4000E-04
U238 ELASTIC	1.1654E-01	0.0	0.0	1.1654E-01
U238 (N,2N)	-1.1384E-01	0.0	0.0	-1.1384E-01
U238 (N,3N)	-7.0241E-02	0.0	0.0	-7.0241E-02
U238 TOTAL FISSION	1.0990E+00	0.0	0.0	1.0990E+00
U238 (N,GAMMA)	-1.4287E-02	0.0	0.0	-1.4287E-02
U238 TOTAL INELASTIC	-2.3668E-01	0.0	0.0	-2.3668E-01
U235 ELASTIC	0.7886E-04	0.0	0.0	0.7886E-04
U235 (N,2N)	-2.9581E-04	0.0	0.0	-2.9581E-04
U235 (N,3N)	-3.7493E-05	0.0	0.0	-3.7493E-05
U235 TOTAL FISSION	2.5077E-02	0.0	0.0	2.5077E-02
U235 (N,GAMMA)	-1.6631E-04	0.0	0.0	-1.6631E-04
U235 TOTAL INELASTIC	-9.1921E-04	0.0	0.0	-9.1921E-04

TABLE 19

SENSITIVITY COEFFICIENTS OF 235U NUCLEAR HEATING
TO LI6,FE,L17,U238,U235 PARTIAL CROSS SECTIONS
FISSILE CYLINDRICAL REACTOR

	ZONE 3	ZONE 4	ZONE 5	SUM
LI6 ELASTIC	0.0	2.0571E-02	0.0	2.0571E-02
LI6 (N,2N)ALPHA	0.0	5.4502E-04	0.0	5.4502E-04
LI6 (N,N')ALPHA	0.0	9.3924E-04	0.0	9.3924E-04
LI6 (N,GAMMA)	0.0	-3.8404E-06	0.0	-3.8404E-06
LI6 (N,PROTON)	0.0	-1.9752E-04	0.0	-1.9752E-04
LI6 (N,ALPHA)T	0.0	-1.3544E-01	0.0	-1.3544E-01
LI6 INELASTIC	0.0	9.5538E-04	0.0	9.5538E-04
FE ELASTIC	3.8767E-02	2.0108E-02	3.0825E-05	5.8906E-02
FE (N,2N)	7.9859E-04	8.5879E-04	1.4862E-06	1.6389E-03
FE (N,N')ALPHA	-1.0380E-04	5.2966E-07	-2.6794E-09	-1.0327E-04
FE (N,N')PROTON	-1.1304E-03	9.4155E-06	-2.1946E-08	-1.1210E-03
FE (N,GAMMA)	-4.8856E-03	-4.0085E-04	-1.3458E-06	-5.2878E-03
FE (N,PROTON)	-8.5695E-03	-3.7789E-04	-1.5993E-06	-8.9494E-03
FE (N,DEUTERIUM)	-1.1557E-03	-2.9944E-05	-9.6597E-08	-1.1857E-03
FE (N,TRITIUM)	-2.7287E-05	-5.9551E-07	-1.5220E-09	-2.7884E-05
FE (N,HELIUM 3)	-1.9962E-05	-4.3905E-07	-1.1374E-09	-2.0402E-05
FE (N,ALPHA)	-2.5674E-03	-9.7687E-05	-3.9404E-07	-2.6655E-03
FE INELASTIC	-2.2050E-02	8.4505E-05	-1.9340E-06	-2.1967E-02
LI7 ELASTIC	0.0	5.0377E-02	0.0	5.0377E-02
LI7 (N,2N)	0.0	5.1190E-04	0.0	5.1190E-04
LI7 (N,2N)ALPHA	0.0	5.7999E-04	0.0	5.7999E-04
LI7 (N,ALPHA N')T	0.0	2.6563E-03	0.0	2.6563E-03
LI7 (N,GAMMA)	0.0	-1.3519E-05	0.0	-1.3519E-05
LI7 (N,DEUTERIUM)	0.0	-1.7041E-04	0.0	-1.7041E-04
LI7 INELASTIC	0.0	4.3524E-03	0.0	4.3524E-03
U238 ELASTIC	2.2674E-01	0.0	0.0	2.2674E-01
U238 (N,2N)	5.5673E-03	0.0	0.0	5.5673E-03
U238 (N,3N)	3.9356E-02	0.0	0.0	3.9356E-02
U238 TOTAL FISSION	4.2559E-01	0.0	0.0	4.2559E-01
U238 (N,GAMMA)	-3.8639E-01	0.0	0.0	-3.8639E-01
U238 TOTAL INELASTIC	-3.2648E-02	0.0	0.0	-3.2648E-02
U235 ELASTIC	1.6097E-03	0.0	0.0	1.6097E-03
U235 (N,2N)	3.8548E-04	0.0	0.0	3.8548E-04
U235 (N,3N)	2.1423E-05	0.0	0.0	2.1423E-05
U235 TOTAL FISSION	1.1070E+00	0.0	0.0	1.1070E+00
U235 (N,GAMMA)	-4.8903E-03	0.0	0.0	-4.8903E-03
U235 TOTAL INELASTIC	-3.2590E-05	0.0	0.0	-3.2590E-05

TABLE 20

SENSITIVITY COEFFICIENTS OF FE(N,P) GAS PHOD
TO LI6,FE,L17,U238,U235 PARTIAL CROSS SECTIONS
FISSILE CYLINDRICAL REACTOR

	ZONE 3	ZONE 4	ZONE 5	SUM
LI6 ELASTIC	0.0	-1.3356E-02	0.0	-1.3356E-02
LI6 (N,2N)ALPHA	0.0	-3.6430E-03	0.0	-3.6430E-03
LI6 (N,N')ALPHA	0.0	-2.4187E-02	0.0	-2.4187E-02
LI6 (N,GAMMA)	0.0	-7.1604E-07	0.0	-7.1604E-07
LI6 (N,PROTON)	0.0	-6.1622E-04	0.0	-6.1622E-04
LI6 (N,ALPHA)T	0.0	-3.1574E-03	0.0	-3.1574E-03
LI6 INELASTIC	0.0	-2.4434E-02	0.0	-2.4434E-02
FE ELASTIC	-9.6277E-05	-3.3009E-03	-2.5726E-03	-5.9698E-03
FE (N,2N)	-3.0866E-02	-5.7120E-03	-2.2759E-03	-3.8844E-02
FE (N,N')ALPHA	-2.4673E-04	-4.0931E-05	-1.3894E-05	-3.0161E-04
FE (N,N')PROTON	-2.0015E-03	-5.0409E-04	-1.9664E-04	-3.5023E-03
FE (N,GAMMA)	-5.2656E-05	-7.9754E-06	-5.5430E-06	-6.6174E-05
FE (N,PROTON)	7.8931E-01	2.8235E-01	2.5689E-02	1.0973E+00
FE (N,DEUTERIUM)	-1.3716E-03	-2.6050E-04	-1.1158E-04	-1.7437E-03
FE (N,TRITIUM)	-3.2396E-05	-5.2996E-06	-1.7605E-06	-3.9456E-05
FE (N,HELIUM 3)	-2.3700E-05	-3.9061E-06	-1.3165E-06	-2.8923E-05
FE (N,ALPHA)	-2.9940E-03	-7.1153E-04	-4.0735E-04	-4.1128E-03
FE INELASTIC	-5.3365E-02	-1.7526E-02	-1.3057E-02	-8.3948E-02
LI7 ELASTIC	0.0	-4.7546E-02	0.0	-4.7546E-02
LI7 (N,2N)	0.0	-3.4661E-03	0.0	-3.4661E-03
LI7 (N,2N)ALPHA	0.0	-4.5279E-03	0.0	-4.5279E-03
LI7 (N,ALPHA N')T	0.0	-5.5017E-02	0.0	-5.5017E-02
LI7 (N,GAMMA)	0.0	-2.1033E-06	0.0	-2.1033E-06
LI7 (N,DEUTERIUM)	0.0	-1.4435E-03	0.0	-1.4435E-03
LI7 INELASTIC	0.0	-6.5283E-02	0.0	-6.5283E-02
U238 ELASTIC	3.0998E-03	0.0	0.0	3.0998E-03
U238 (N,2N)	-1.7456E-01	0.0	0.0	-1.7456E-01
U238 (N,3N)	-9.3511E-02	0.0	0.0	-9.3511E-02
U238 TOTAL FISSION	-1.6220E-01	0.0	0.0	-1.6220E-01
U238 (N,GAMMA)	-2.9878E-03	0.0	0.0	-2.9878E-03
U238 TOTAL INELASTIC	-8.2570E-02	0.0	0.0	-8.2570E-02
U235 ELASTIC	2.3007E-05	0.0	0.0	2.3007E-05
U235 (N,2N)	-4.4247E-04	0.0	0.0	-4.4247E-04
U235 (N,3N)	-4.8082E-05	0.0	0.0	-4.8082E-05
U235 TOTAL FISSION	3.2190E-03	0.0	0.0	3.2190E-03
U235 (N,GAMMA)	-3.7730E-05	0.0	0.0	-3.7730E-05
U235 TOTAL INELASTIC	-2.9479E-04	0.0	0.0	-2.9479E-04

TABLE 21

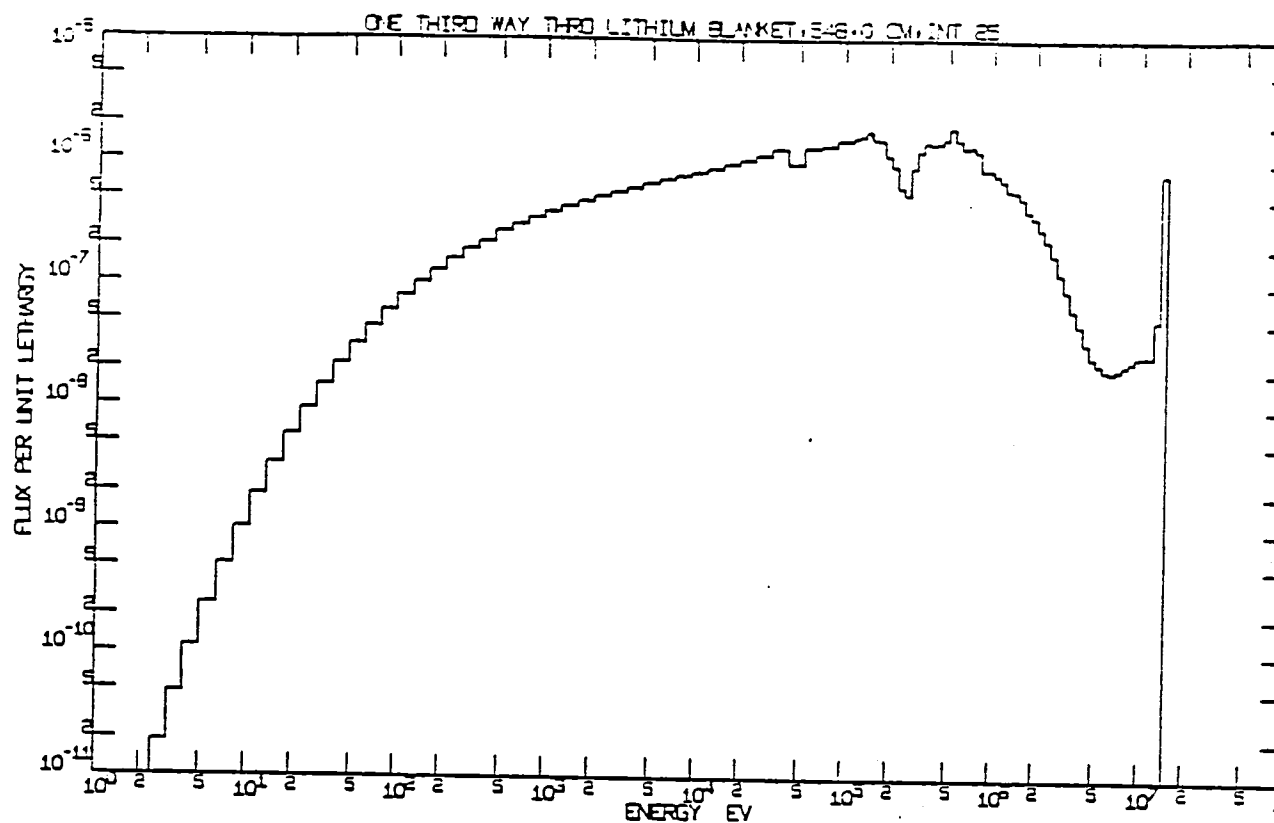


FIG. 2

scalar forward flux at a spatial position $1/3$ of the way into the Lithium blanket in the NFR.

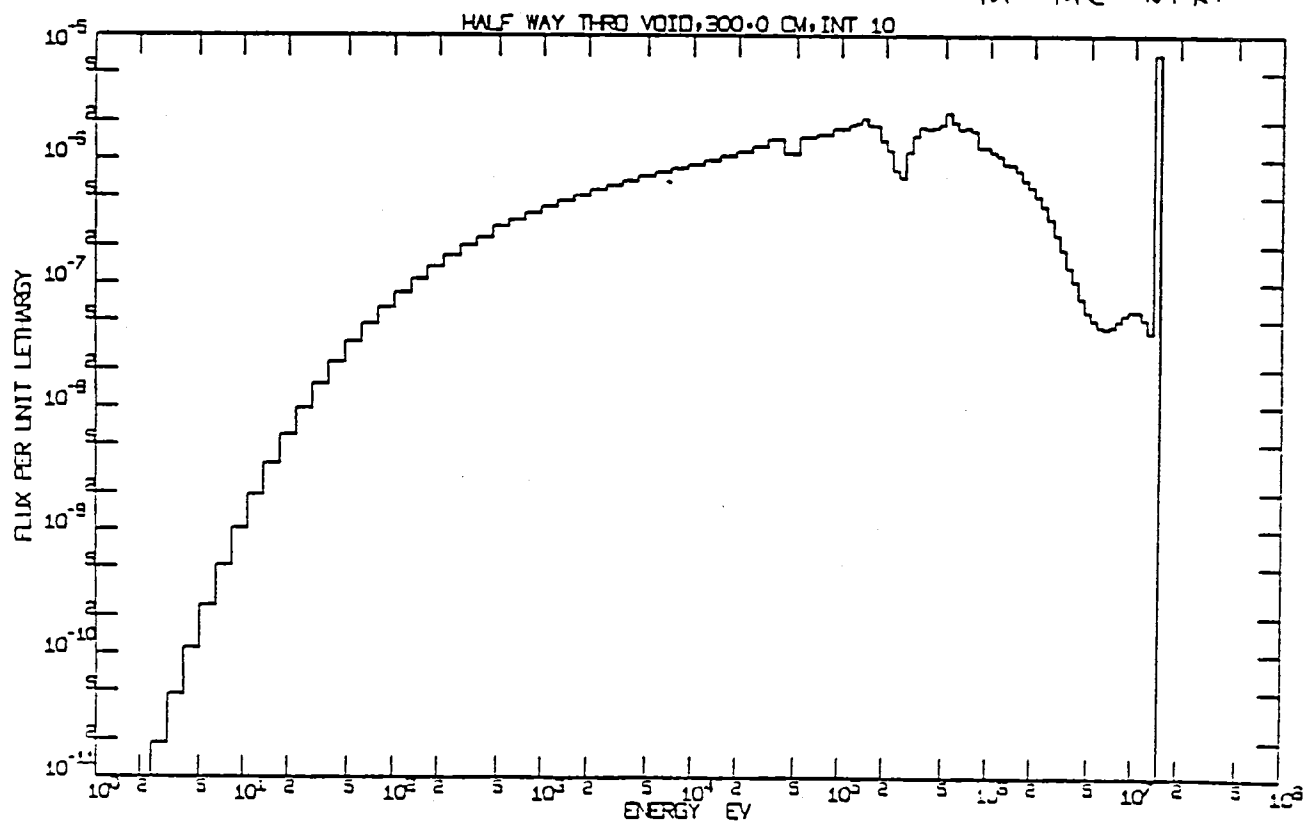


FIG. 1

scalar forward flux at a spatial position halfway through the void in the NFR.

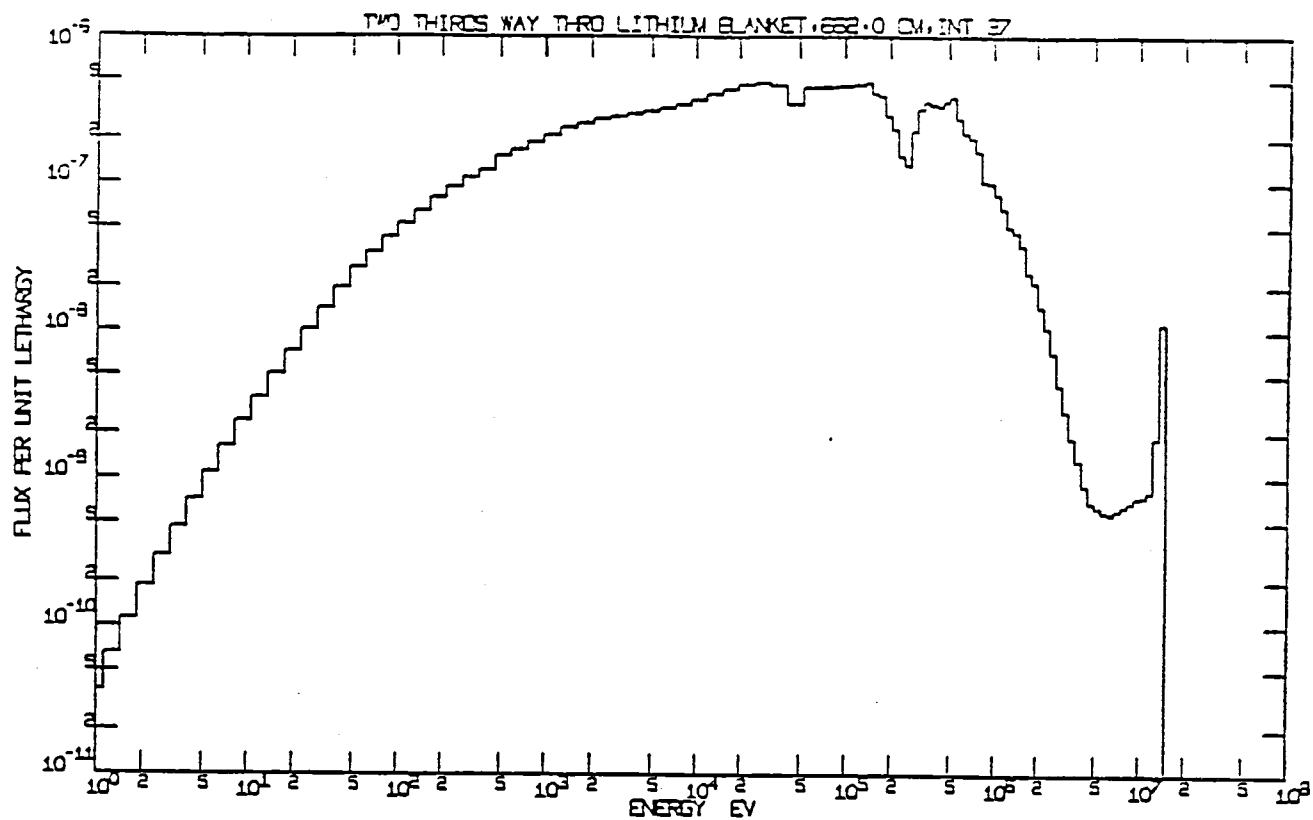


FIG. 3

Scalar flux at a position
2/3 into Lithium blanket.
in NFR.

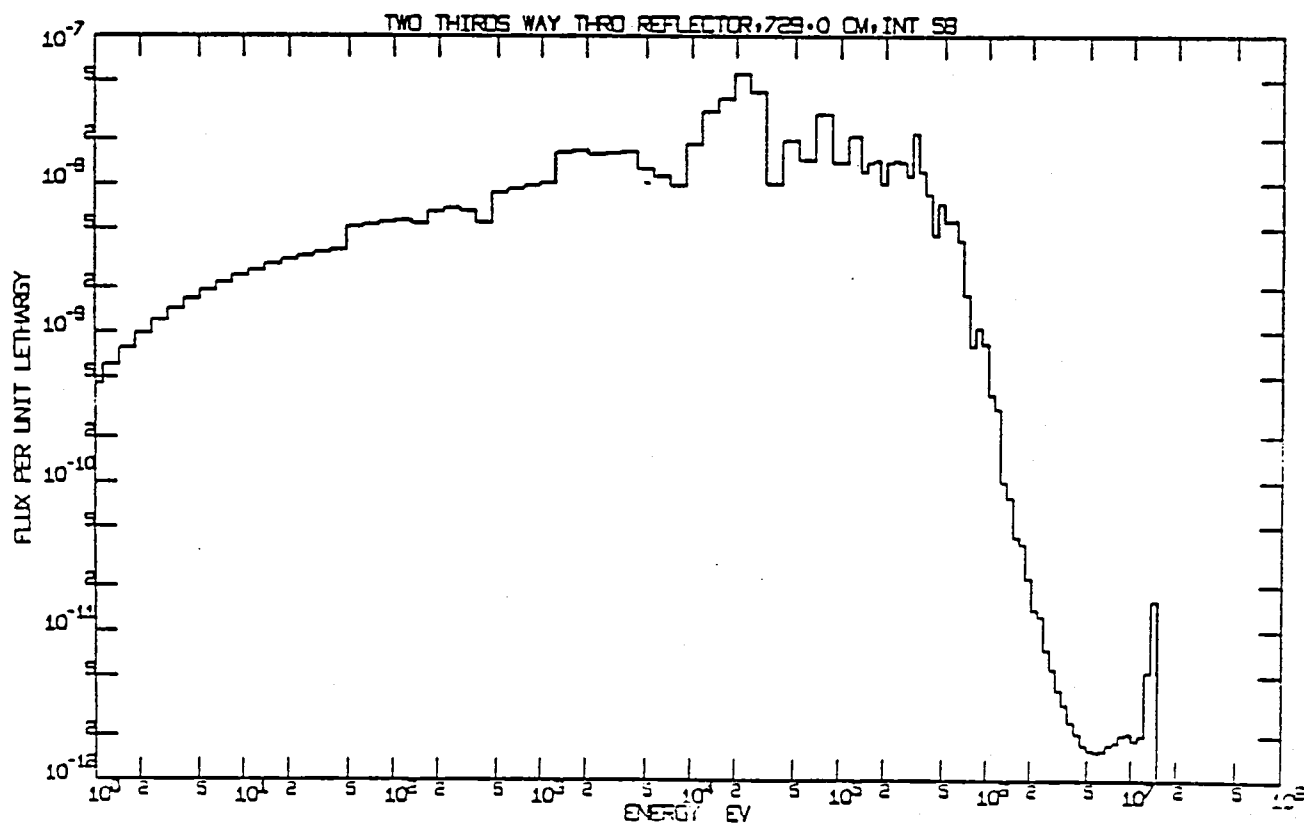


FIG. 4

Scalar flux at a position
2/3 into reflector region
in NFR.

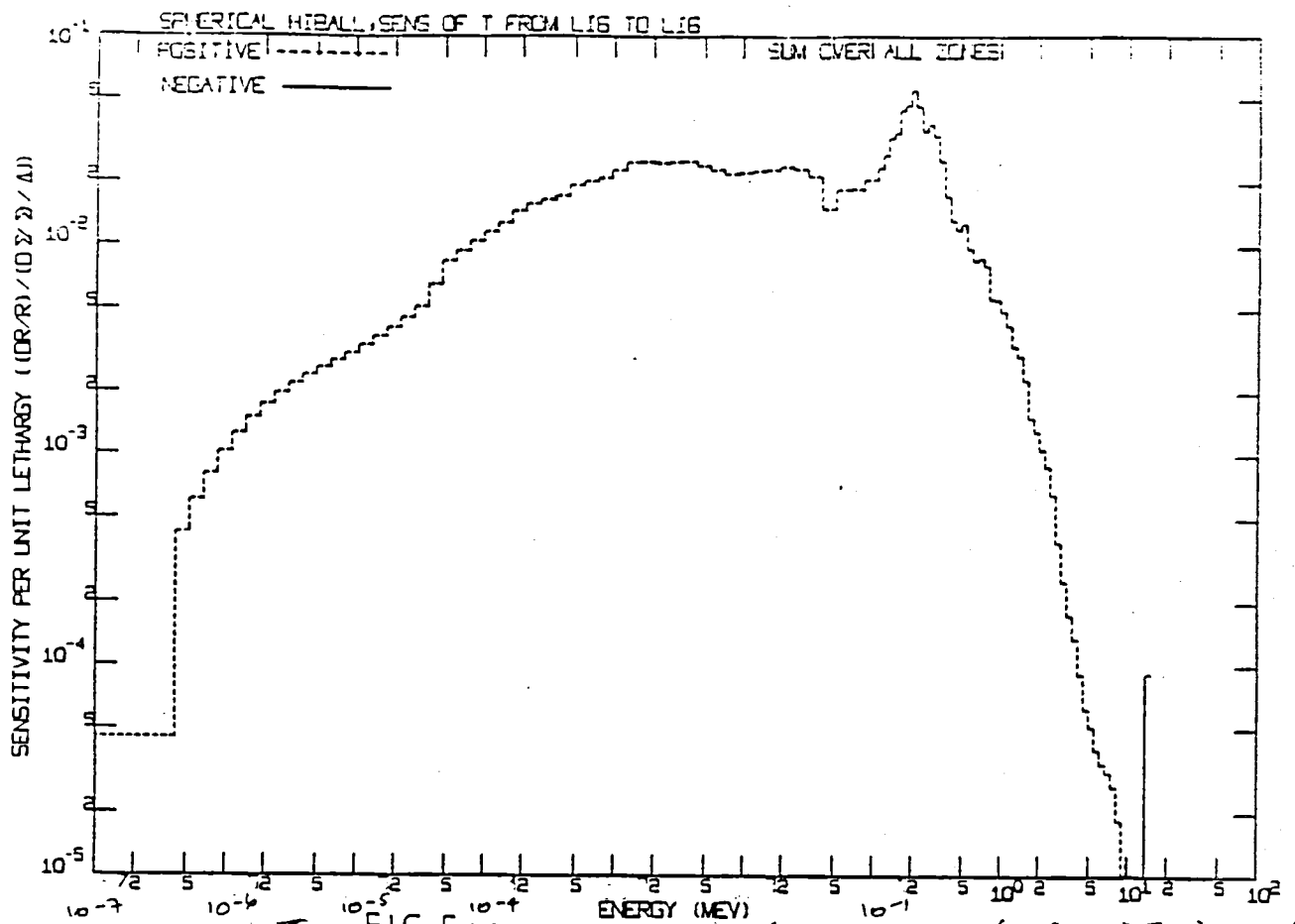


FIG. 5 sensitivity of ${}^6\text{Li}(n, \alpha)\text{T}$ to ${}^6\text{Li}(n, \alpha)\text{T}$ in NFR.

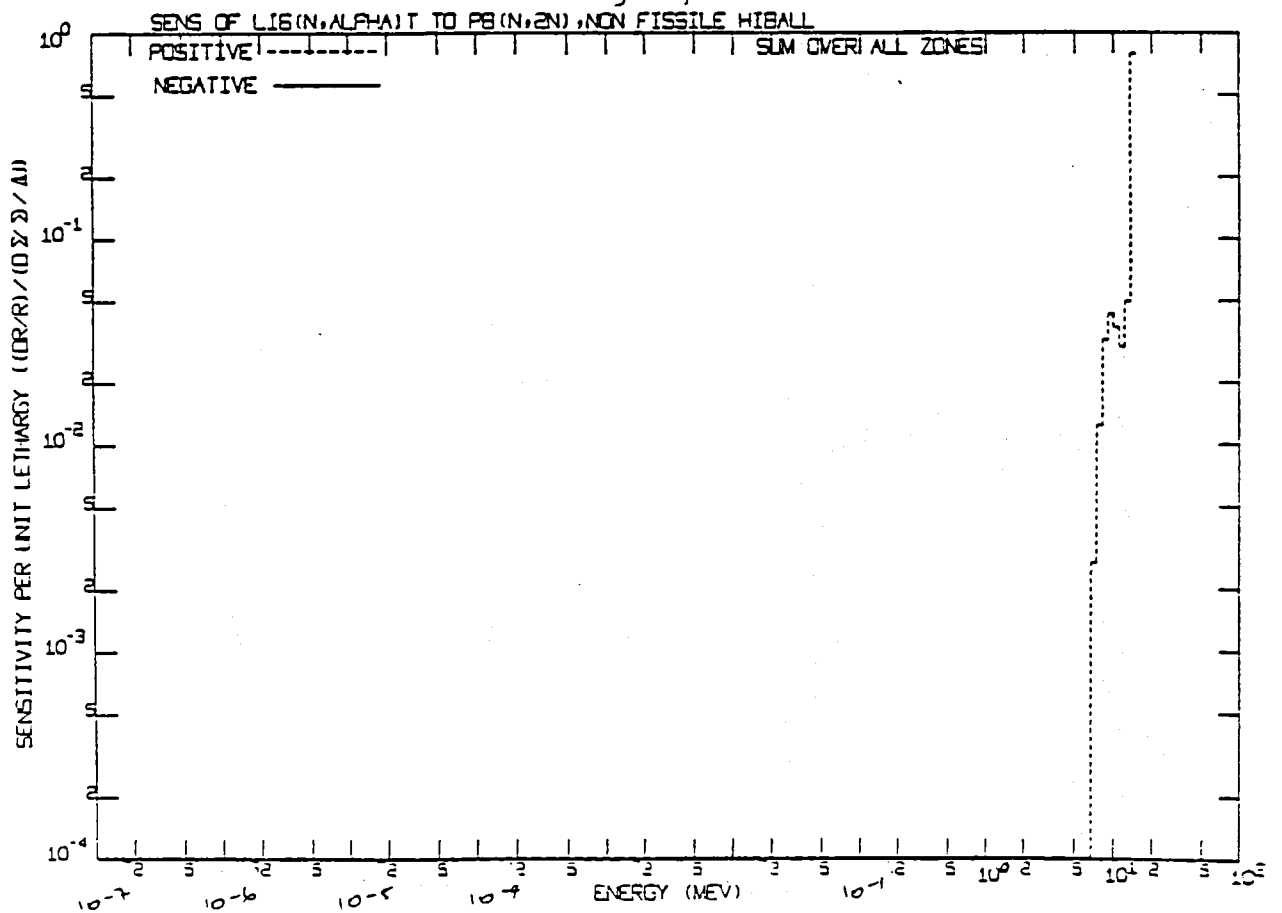


FIG. 6 sensitivity of ${}^6\text{Li}(n, \alpha)\text{T}$ to $\text{Pb}(n, 2n)$ in NFR.

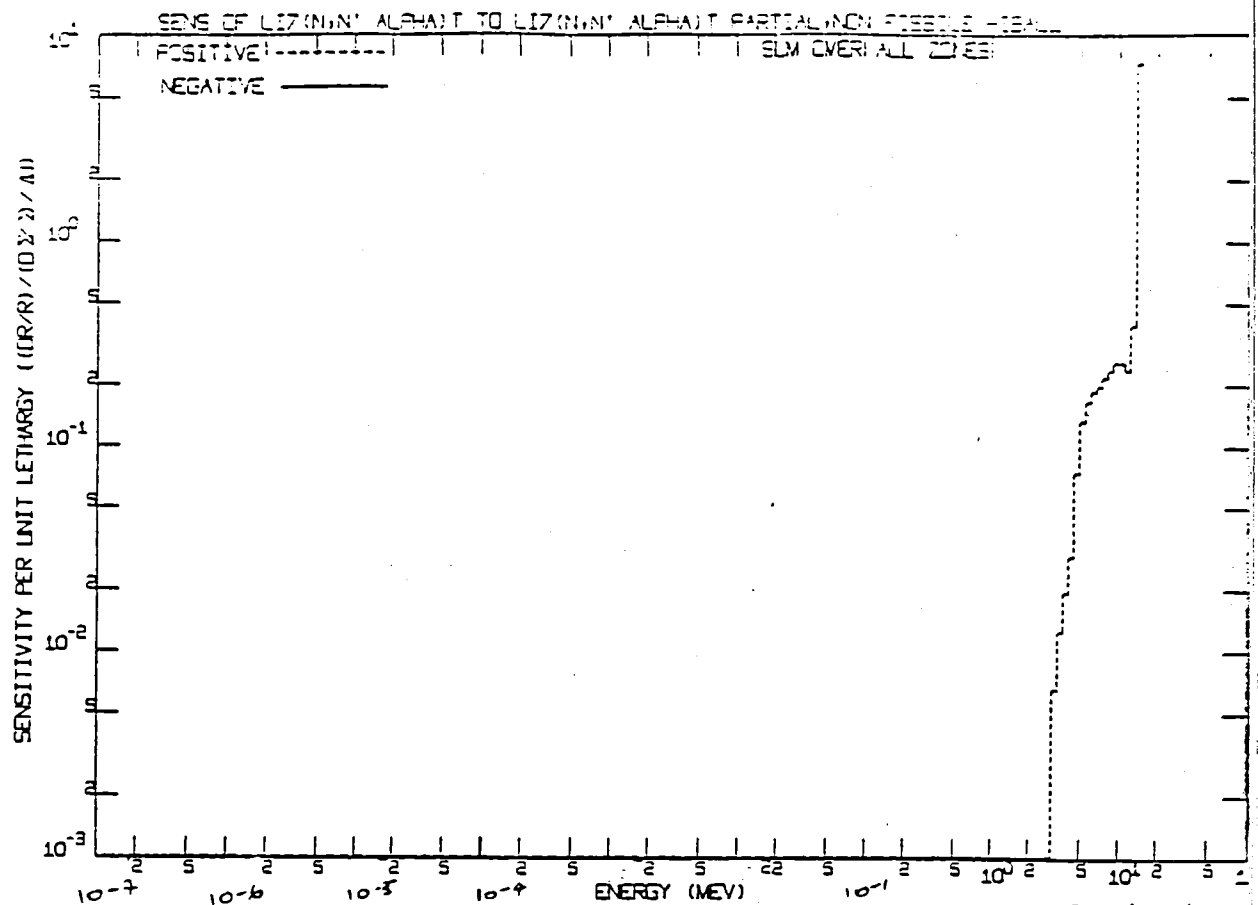


FIG. 7 sensitivity of ${}^7\text{Li}(n,n')\alpha T$ to ${}^7\text{Li}(n,n')\alpha$ in NFR.

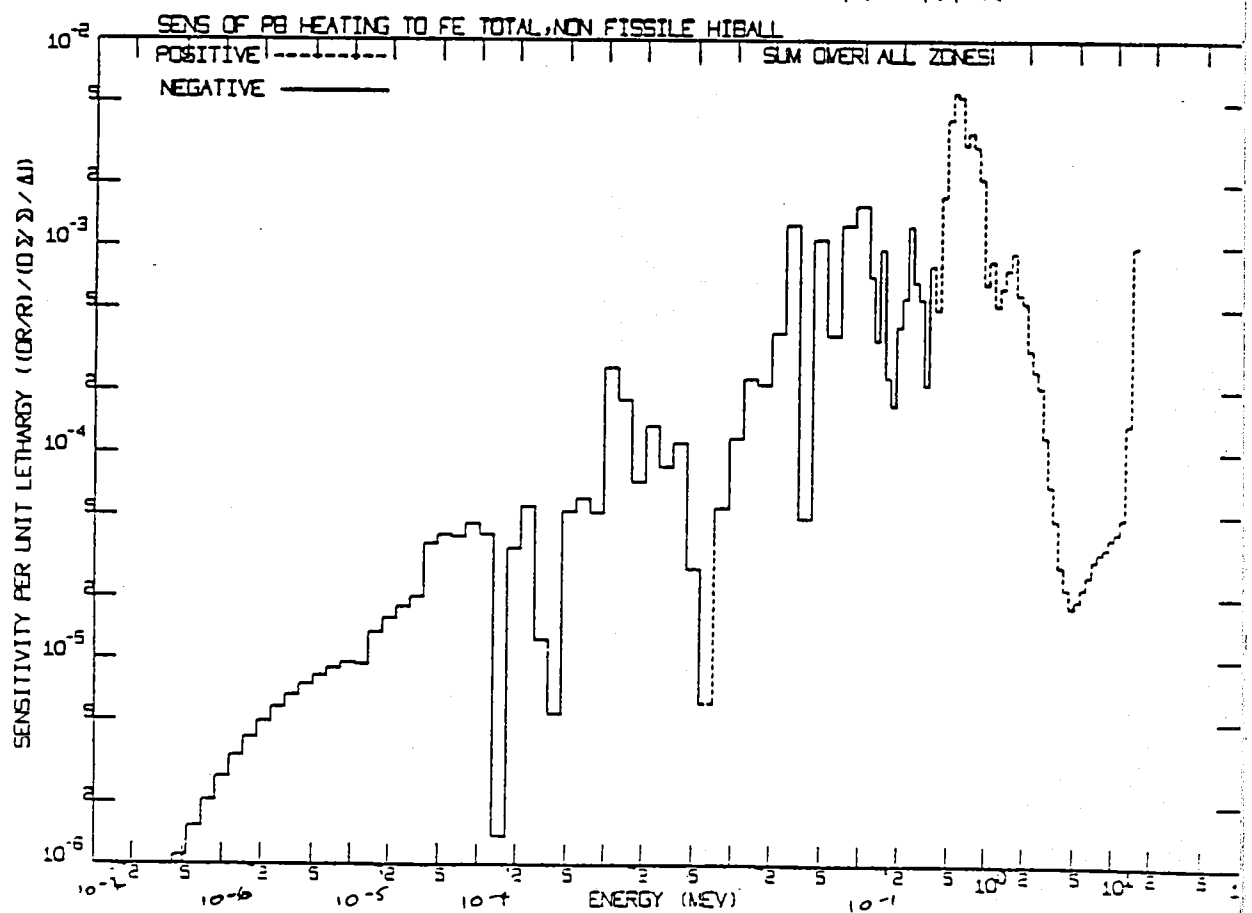


FIG. 8 sensitivity of total Pb heating to the total iron cross section in NFR.

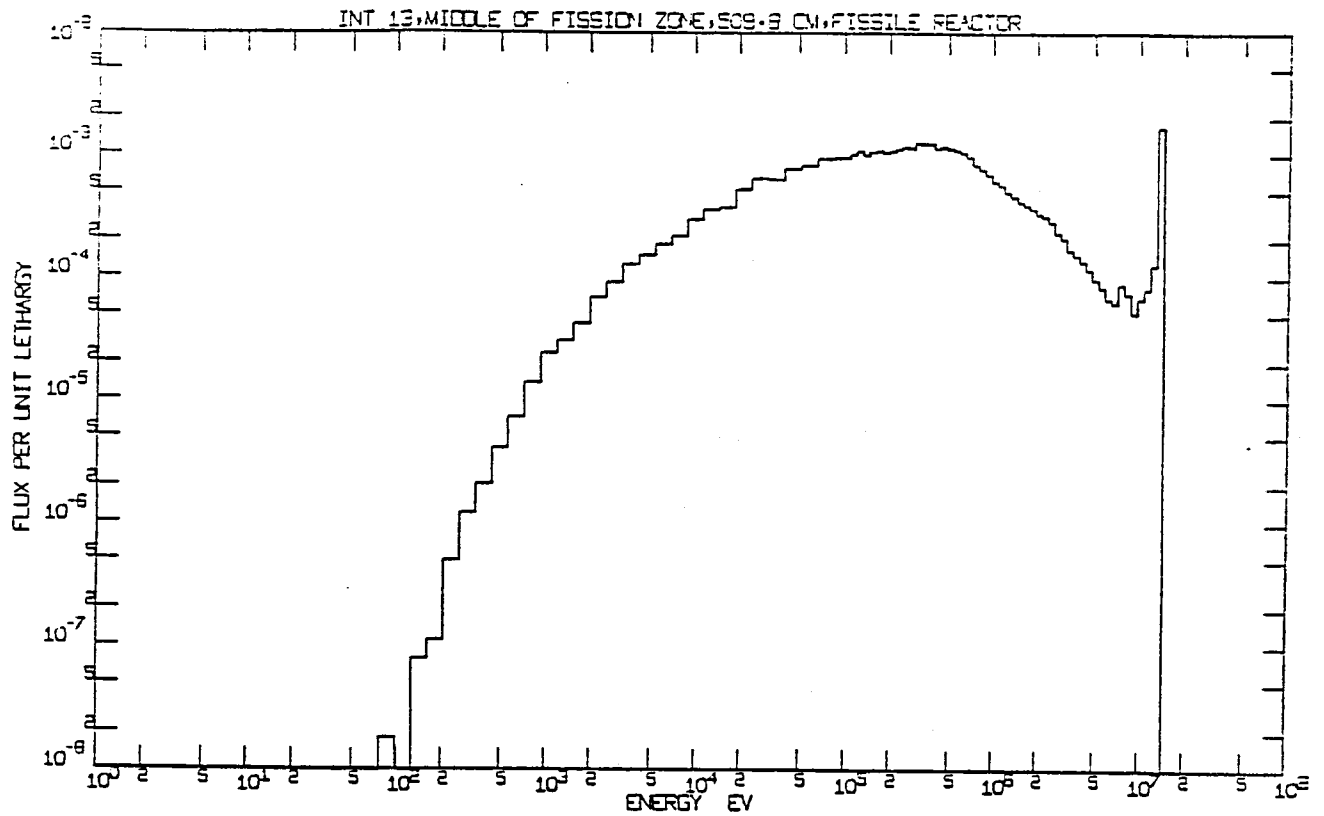


FIG. 9 forward scalar flux in the middle of the fission zone in FR.

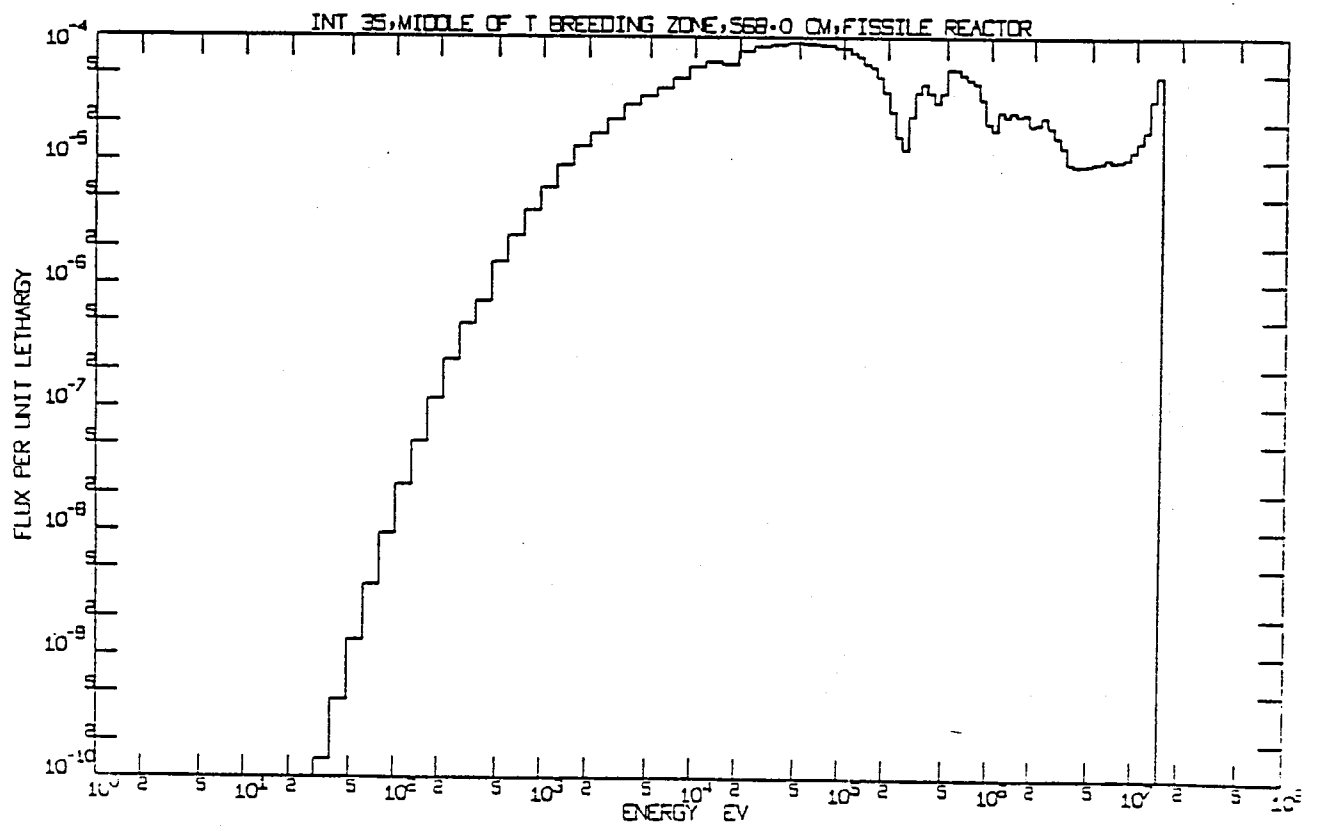


FIG. 10 forward scalar flux in the middle of the Lithium blanket in the FR.

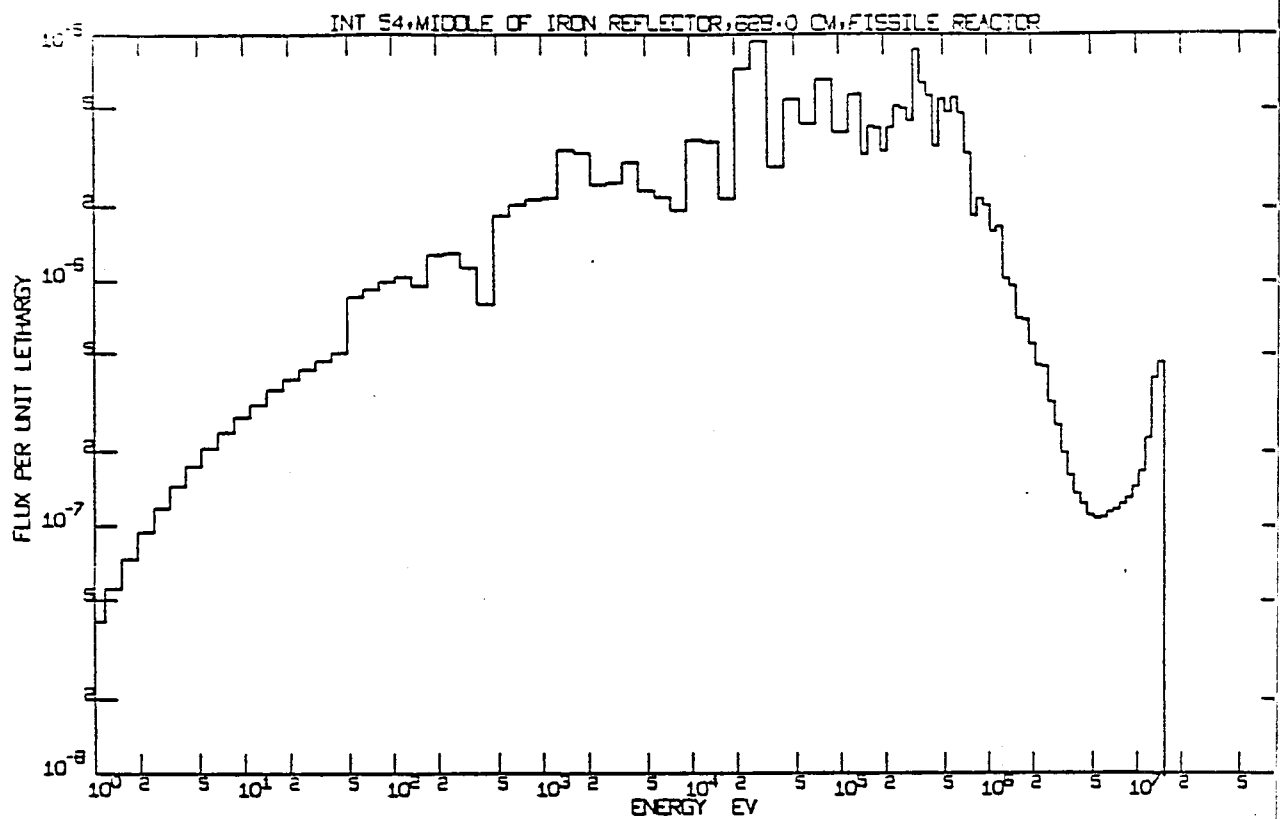


FIG. 11 forward scalar flux in the middle of the reflector zone in the FR.

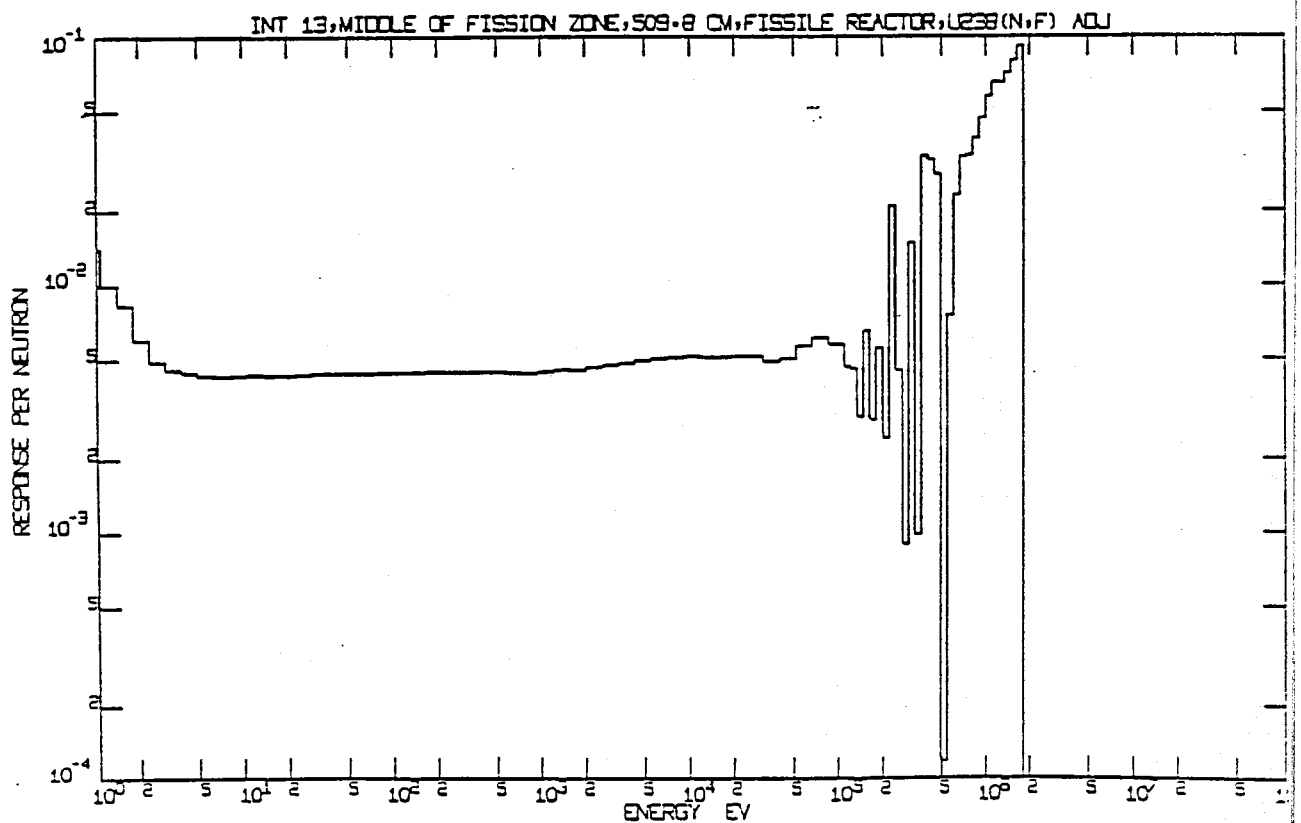


FIG. 12 adjoint scalar flux in the middle of the fission zone $^{238}\text{U}(n,f)$ is adjoint source.

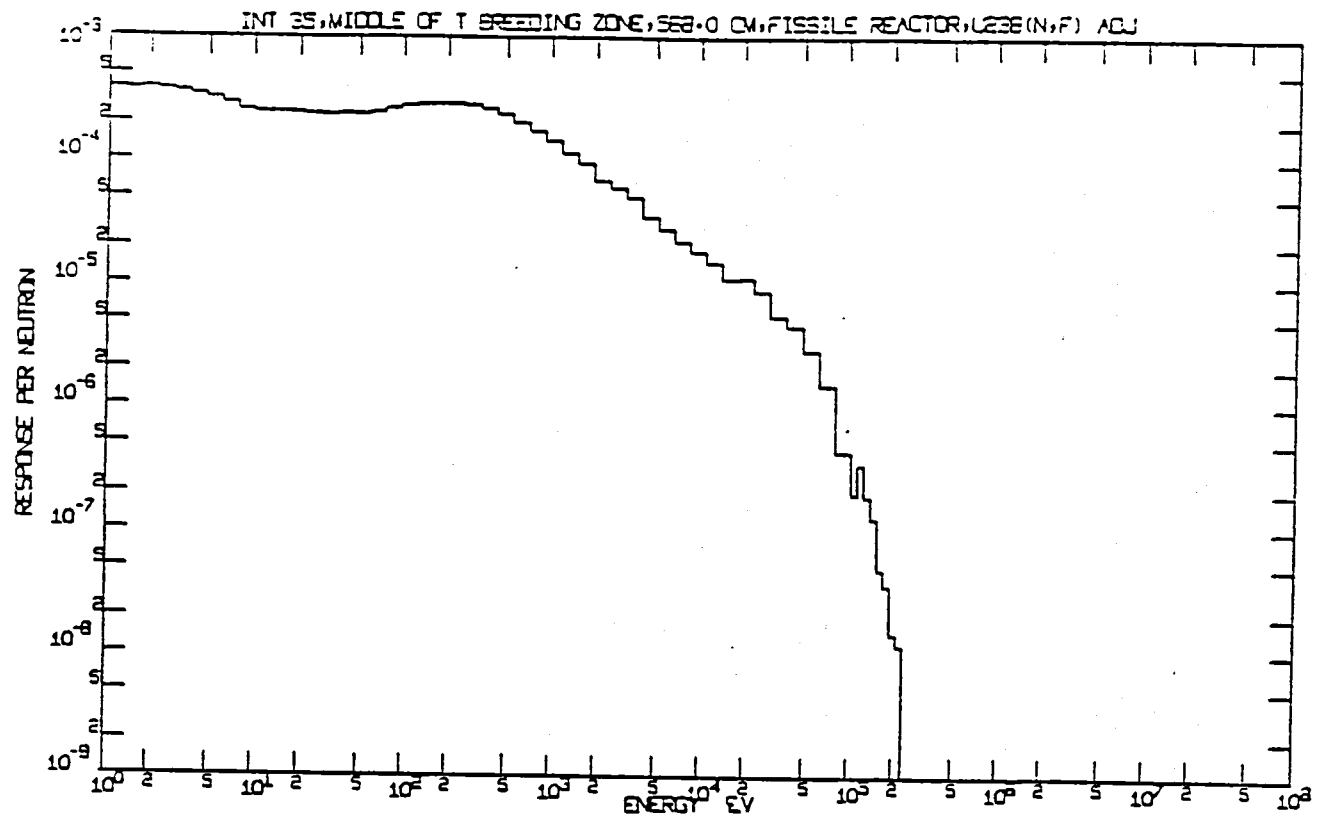


FIG.13 adjoint scalar flux in the middle of the Lithium zone $^{238}\text{U}(n,f)$ is adjoint source.

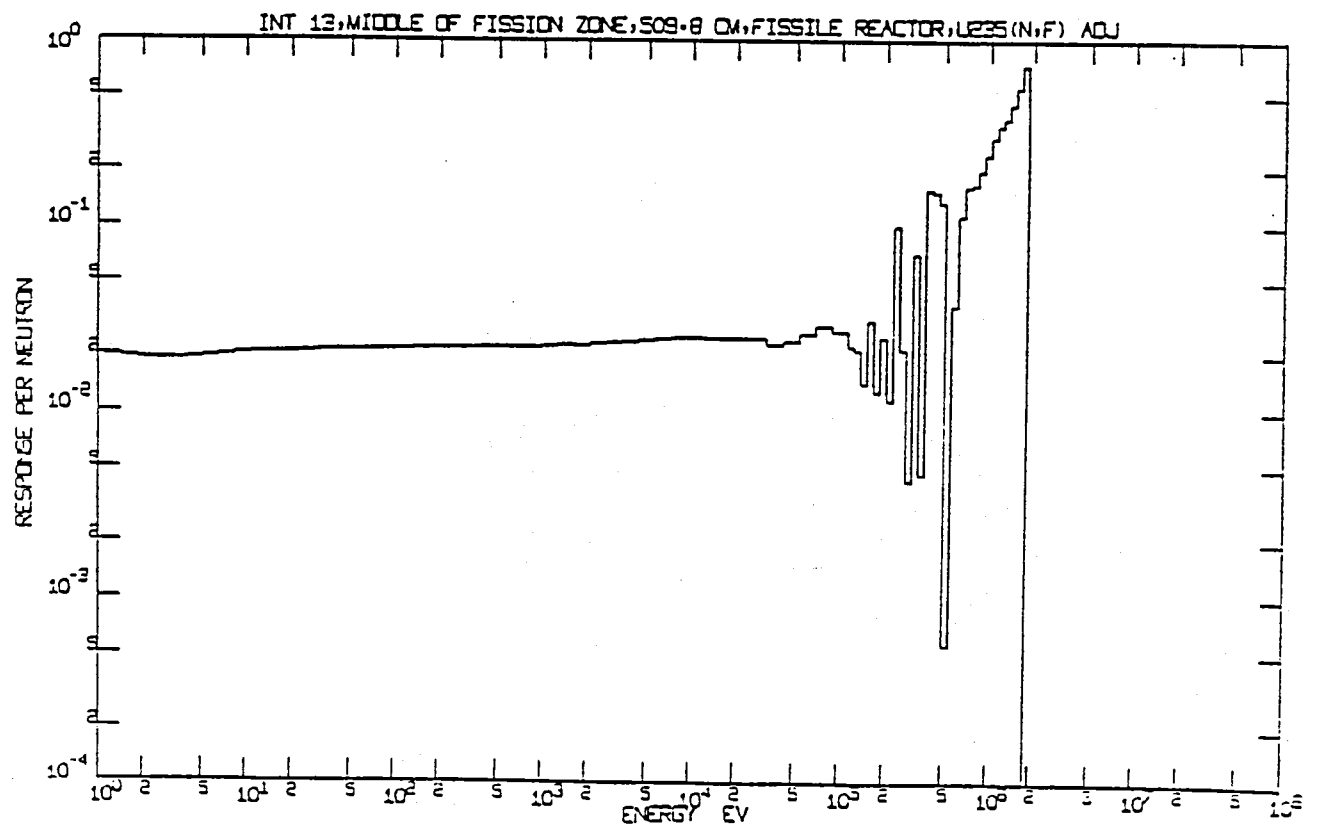


FIG.14 adjoint scalar flux in the middle of the fission zone $^{235}\text{U}(n,f)$ is adjoint source.

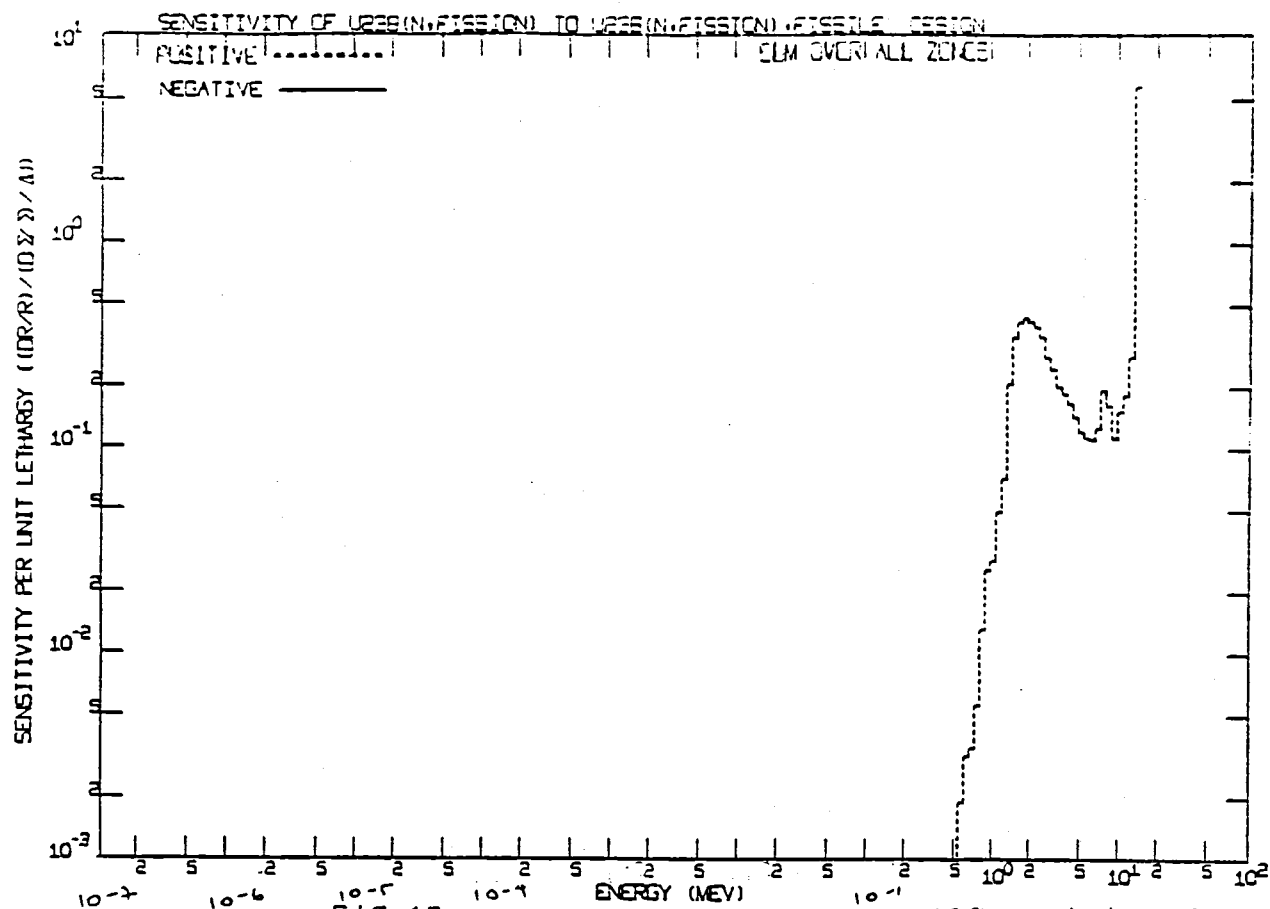


FIG. 15 sensitivity of $^{238}\text{U}(n,f)$ to $^{238}\text{U}(n,f)$ in FR.

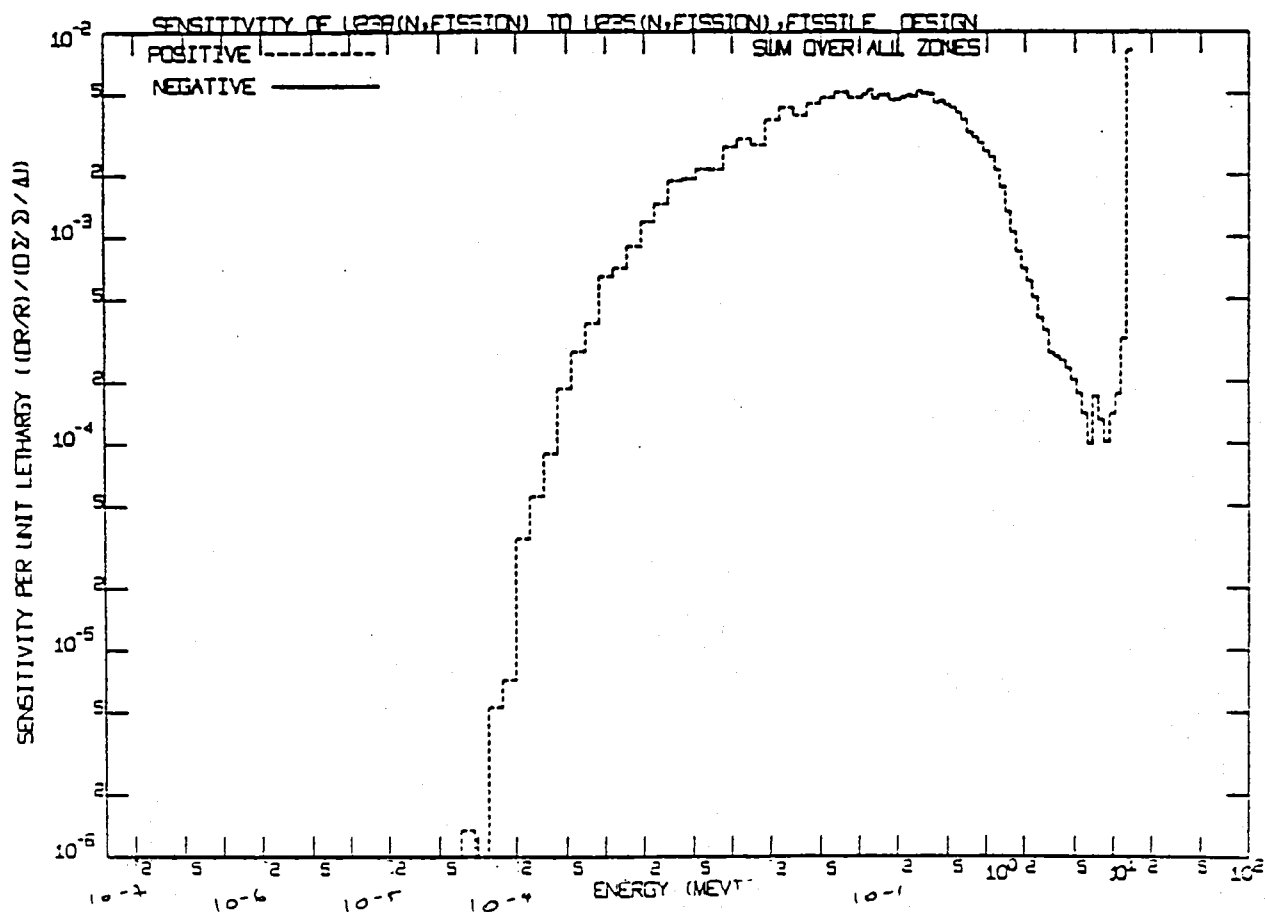


FIG. 16 sensitivity of $^{238}\text{U}(n,f)$ to $^{235}\text{U}(n,f)$ in FR.

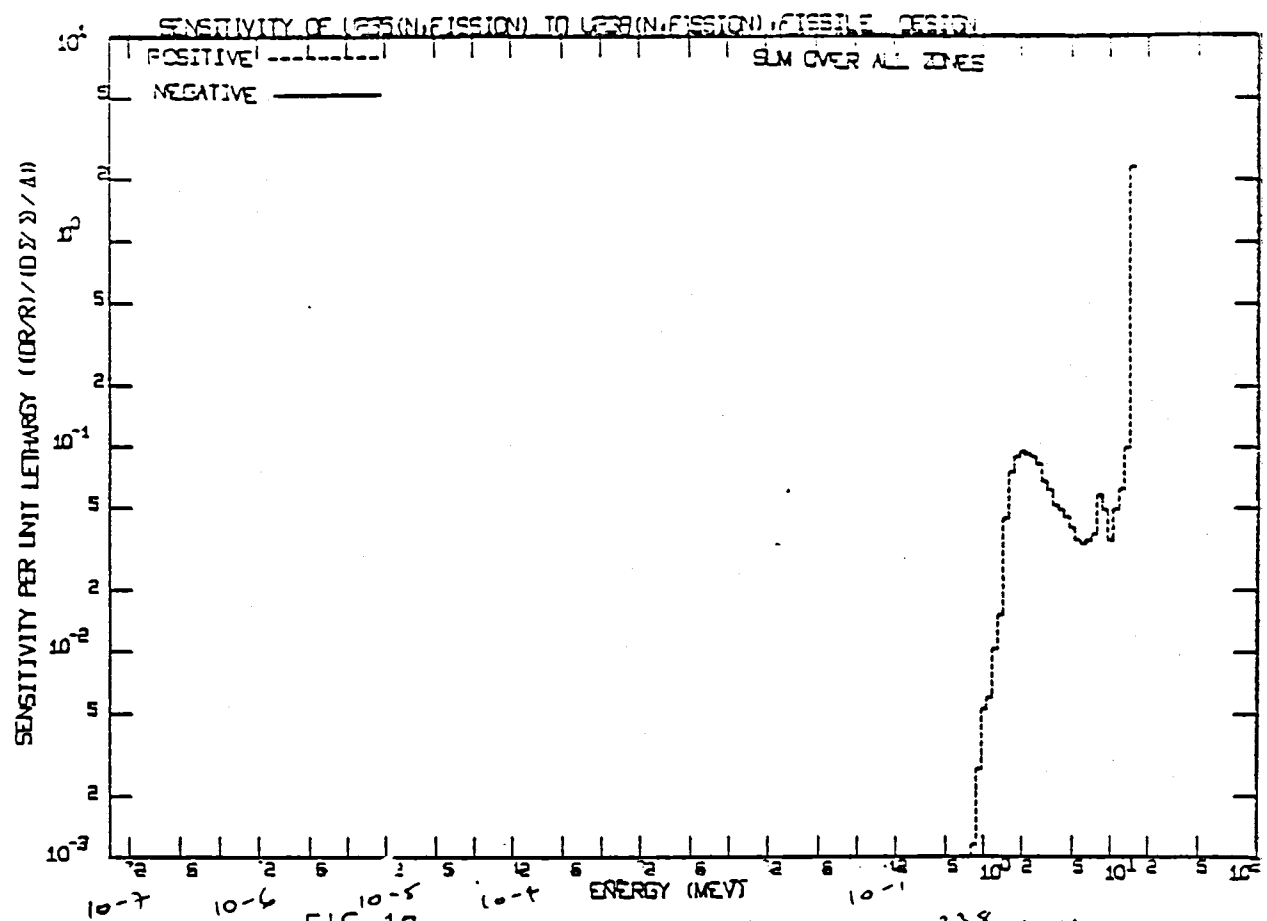


FIG. 17 sensitivity of $^{235}\text{U}(n,f)$ to $^{238}\text{U}(n,f)$ in FR.

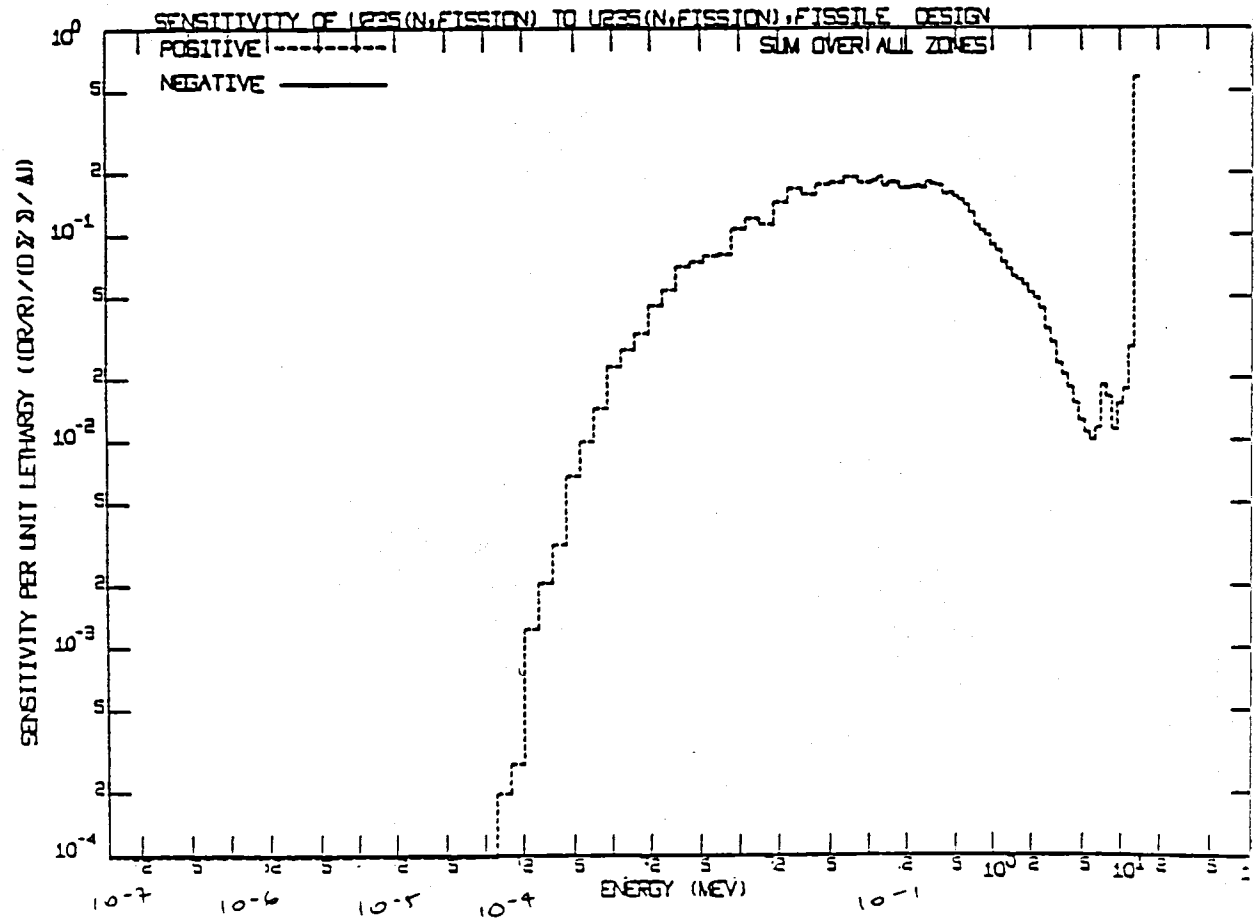


FIG. 18 sensitivity of $^{235}\text{U}(n,f)$ to $^{235}\text{U}(n,f)$ in FR.

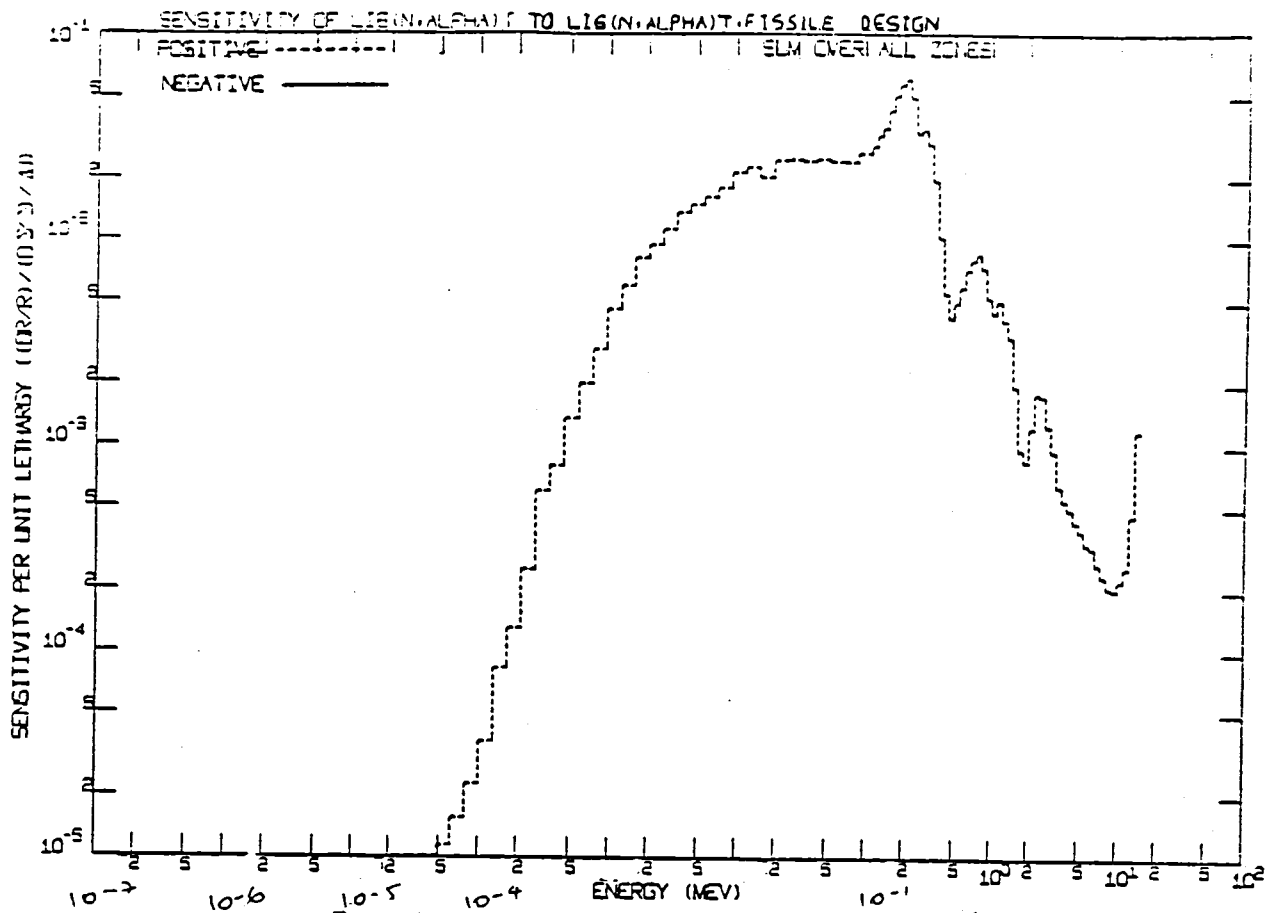


FIG.19 sensitivity of $Li_6(n, \alpha)T$ to $Li_6(n, \alpha)T$ in FR

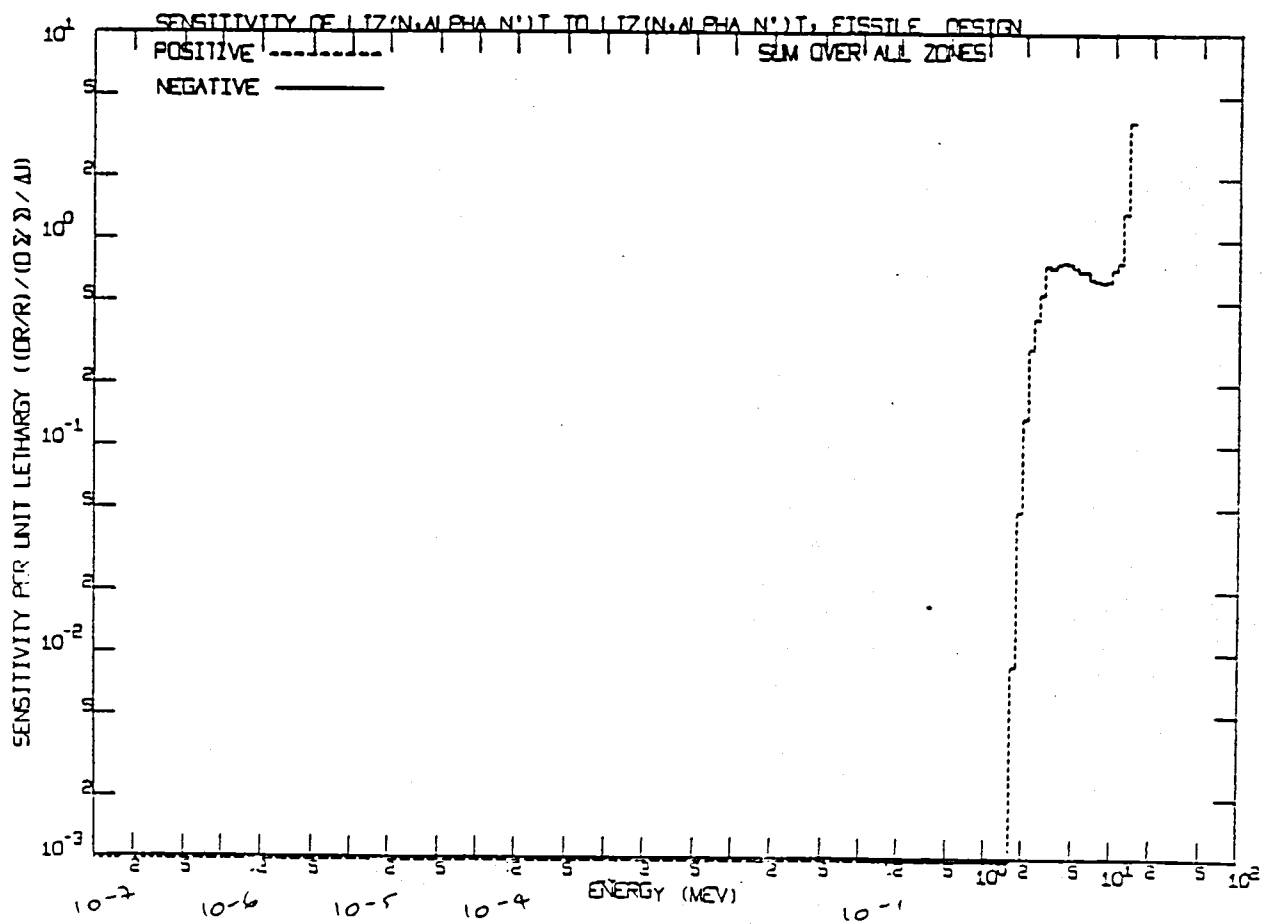


FIG.20 sensitivity of $Li_7(n, \alpha)T$ to $Li_7(n, \alpha)T$ in FR

— 110 —

CHAPTER 7

RESULTS AND DESIGN OF THE NON FISSILE INTEGRAL EXPERIMENT

7.1 INTRODUCTION

This chapter details the design of the non fissile integral experiment via the use of sensitivity analysis. Various intermediate designs, as well as the final working design, are analysed and discussed to show the changes in the neutronics. Finally, since only a few sensitivity profiles were used for the actual tuning, the matching of other profiles between the non fissile reactor and non fissile integral experiment are discussed and a summary of its overall suitability is given. Also shown here is the integral experiment system parameters and how they relate to the non fissile reactor.

7.2 OVERVIEW

The non fissile integral experiment is an attempt to construct a design which can easily be physically built and which models the radiation field and neutron transport properties of the non fissile fusion reactor. Although the final goal of all this work is to adjust group ENDF/B-4 cross sections for some selected isotopes pertinent to the non fissile reactor, the first stage must be to design an experiment which in some way copies the radiation properties of the reactor. The way to do this is to match sensitivity profiles of selected reactor parameters and cross sections. The radiation field properties of the experiment are altered until the similarity between the profiles is as close as possible. However, it is clearly impractical to use all the

profiles to design such an assembly, so some type of reduction was necessary. It was decided to mostly use the sensitivity of $\text{Li6}(n,\alpha)T$ to $\text{Li6}(n,\alpha)T$ as the tuning profile. There are two reasons for this. Firstly, the $\text{Li6}(n,\alpha)T$ cross section covers the whole energy range of interest and so gives information over this range. Secondly, the direct term in this sensitivity profile (as in most profiles) is largely dominant over any indirect collisional loss term and so this profile essentially shows the behaviour of the system—averaged, energy—dependent scalar flux. So in using this profile, we are really tuning the scalar fluxes between the experiment and the reactor. The profiles discussed and shown throughout this work have been summed over all spatial zones. As well as using the shape of the sensitivity of $\text{Li6}(n,\alpha)T$ to $\text{Li6}(n,\alpha)T$ for the tuning, the total sensitivity coefficient of this profile was used. It is a single number that gives the percentage change in a parameter for a unit percentage change in a cross section. It is the total area under a sensitivity curve, before division by lethargy width. The total sensitivity coefficient for the non fissile reactor is 0.21. All the integral designs are made with this number in mind and the attempt here, along with the sensitivity profile shape, is to get as close to this number as possible. Also included in the discussion are the values for the right and left leakages and the tritium breeding reaction rates. The non fissile intermediate designs are shown in figure 19. The notion of right and left hand leakage can be understood by observing these designs. The designs are drawn in the ANISN type format, and the left leakage is that which travels back past the source and is lost. The right leakage is likewise that leakage which escapes from the right hand side and is lost. For reference,

table 1 gives a summary of these quantities for each of the considered designs. As a final note, it should be remembered that this integral experiment design comes only from a 1-dimensional consideration, using the ANISN code. The neutronics of the experiment will contain all the approximations inherent in a 1-dimensional code. An attempt to correct for this by using a 3-dimensional consideration is given in chapter 10. Throughout this work, NFR means the non fissile reactor and NFIE means the non fissile integral experiment.

7.3 THE DESIGN OF THE NON FISSILE INTEGRAL EXPERIMENT

Design b

The first design change was to remove those materials from the reactor which will not be used for the integral experiment design. Hopefully too, such materials will be unimportant to the transport dynamics of the reactor system. After the removal of these elements, only Pb, Li and Fe remain. This leaves a spherical assembly exactly the same as the NFR, except that zone 1 contains 66cm of $Pb_{83}Li_{17}$, zone 2 1cm of Fe and zone 3 40cm of Fe. Their number densities remain that of the original reactor. Design b) gives a $Li6(n,\alpha)T$ reaction rate of 1.155 and a $Li7(n,n'\alpha)T$ of $3.02E-2$. The slight changes in the breeding ratios are due to a very slight hardening of the flux due to the removal of some elements which had high elastic and/or high inelastic scattering cross sections. Since the reaction rate changes were so small no further analysis and no sensitivity profiles were obtained. These slight changes demonstrate that the removed materials are indeed neutronically unimportant.

Design c

The second design change, design c, was to change the geometry from a sphere to a slab. The physical system envisaged is to have a 14.1 MeV point neutron source firing onto a series of material slabs making up the integral experiment. The neutron source will not be enclosed by the experiment but will be placed very close to the first slab surface. The solid angle subtended by the source on the slab should be as close as possible to 2π . Such a neutron source could be generated by firing a deuteron beam from a 300 KV Cockcroft Walton accelerator onto a thick titanium-tritium target (Seki 1977a). Further reasons for using a slab are that, unlike a sphere, it needs no additional underneath support, and also because such a simple geometry makes for a computationally cleaner model, in the sense that it reduces the uncertainties in the spatial modelling. The 5m void of design b) was removed and an anisotropic (using the P_3 approximation) 14.1 MeV neutron source was placed directly next to the first surface. No changes were made to the materials making up the assembly. In order to get a correct representation of this anisotropy, the angular quadrature was doubled from S_4 to S_8 . In addition because the slab geometry has a shorter execution time, the number of spatial intervals was increased from 63 to 100. It should be noted that as long as the fluxes and responses input to SWANLAKE are consistent, then the derived sensitivity profiles are independent of any source normalisation. Due to the geometry change, the boundary conditions were also altered. The NFR has a right boundary condition (outer surface) of albedo 50% to account for a biological shield, and because it is a sphere, it has a perfectly reflecting boundary condition in the centre (left boundary). A 3% albedo was set on the front and back surfaces of the slab for energies 1 MeV to thermal to

account for any likely room return neutrons from the walls of the building. The boundary condition change altered the neutronics drastically. The system is now very leaky, especially to low energy neutrons. The $\text{Li6}(n,\alpha)\text{T}$ reaction rate dropped by 80% from 1.155 to 0.231 and for $\text{Li7}(n,n'\alpha)\text{T}$ the drop was 6%, from $3.01\text{E}-2$ to $2.83\text{E}-2$. The differences in the reaction rate drops indicate the relatively small amount of fast flux in this design, irrespective of boundary condition. The right hand leakage only doubles, indicating that the flux here is mostly low energy and that the outer Fe zone is reflecting most of it back. The left hand leakage changes from 0.0 to 1.54, showing that a great deal of the source is reflected back out of the assembly. The leakage is greater than 1 because of multiplicative reactions, particularly $\text{Pb}(n,2n)$ and $\text{Pb}(n,3n)$. Figures 1 and 2 give the forward scalar fluxes of design a) and design c) at the same spatial positions. Small differences can be seen at higher energies but in the energy band 10-0.1 Kev the difference is quite pronounced. The lack of flux at this energy indicates just how leaky the slab geometry is. The difference in the flux magnitudes is due to the fact that the scalar flux from an ANISN run with slab geometry has dimensions of per unit area.

Figures 3 and 4 give the profiles of the sensitivity of $\text{Li6}(n,\alpha)\text{T}$ to $\text{Li6}(n,\alpha)\text{T}$. There is a huge increase in sensitivity at the high energies in figure 4. This has nothing to do with source normalisation, because the sensitivity profile is independent of normalisation, or the area factor on the sphere fluxes because the fluxes are volume integrated. The high energy sensitivity (fig 4) produces a total sensitivity coefficient of 0.515, which is over twice that given by figure 3, 0.21. The reason is due to geometry and albedo. The NFR is spherical with

1211

a total diameter of 7.41m and 50% albedos imposed on its outer surface. Design c) is a slab with total thickness 2.41m and a 3% albedo imposed on its front and back surfaces. In the sphere case the Fe reflector has a massive volume and is a great deal bigger than the Li zone. The solid angle subtended by the smaller spherical lithium zone from any point in the Fe reflector will be very much smaller than in the corresponding slab case, and so in the sphere it will be very much harder for a reflected high energy neutron to get back into the Li zone and contribute to the breeding. In essence, the difference in sensitivity between figures 3 and 4 is that the sphere geometry finds it much harder to score a reflected neutron back into the Li region because of the subtended angle effect. Also included here is the effect of anisotropy. At higher neutron energies there is a great deal more anisotropy after a collision, resulting in less chance of the neutron scoring in the lithium zone, which then exaggerates the effects in the sphere geometry. The energy of a neutron getting back into the Li zone in the sphere case is more likely to be lower in the slab because of the 3-dimensional volume effects, which effectively increases the mean free path. This effect can be seen in the energy range 30-900 eV where the amount of flux in the sphere is greater than in the slab. Below 30 eV the slab flux increases since the capture cross section of Fe becomes appreciable at these energies and the flux in the sphere, because of the volume effect, is attenuated much more.

Design d

The above design, although giving valuable information on the effect of boundary conditions, represents something which cannot easily and practically be built. So the next design, design d),

is a heterogeneous construction of the homogeneous design c) comprising slabs of natural Fe, Pb and lithium. In this way an impractical homogeneous design is transferred into a readily buildable heterogeneous one. The $Pb_{83}Li_{17}$ was broken up into its volume fractions (being a eutectic, not a compound) to yield separate amounts of Li and Pb, and the Fe in zones 2 and 3 was added together. For input into the ANISN code, the thickness of each material zone was adjusted so that the density of the element in the zone is that of the natural element. This gives 57.3cm of lead, 31.3cm of iron and 8.35cm of lithium.

In order for the NFIE neutronics to correspond as closely as possible to that of the NFR, and hence to get similar sensitivity profiles, the Li and Pb were stacked one behind the other. The idea here is an attempt to reproduce the neutronics properties of the homogeneously mixed $Pb_{83}Li_{17}$. The lead zone was located nearest the neutron source and the Fe was positioned at the furthest point from the source to act as a reflector. Each lead and lithium zone is 11.47cm and 1.67cm thick, respectively and contains the natural densities of the materials. In choosing lead as the first zone, rather than lithium, the essential difference is that the flux spectrum of the NFR is being recreated in the NFIE since Pb is also the dominant material in the reactor. With this set-up, the left and right leakages are 1.55 and 0.09, respectively, with $Li6(n,\alpha)T$ being 0.247 and $Li7(n,n'\alpha)T$ being $1.69E-2$. The corresponding figures with lithium in the first zone are 1.48 and 0.1, with $Li6(n,\alpha)T$ being 0.257 and $Li7(n,n'\alpha)T$ being $5.56E-2$. The $Li7(n,n'\alpha)T$ reaction is seen to increase by 230% with lithium in the first zone. This is simply due to there being more fast flux passing through the lithium zone.

Design e

Before attempting to optimise the design, there is one more modification to be made. The natural lithium of design d) is converted into natural lithium fluoride (LiF) for design e). The Pb and Fe slabs present no handling problems and are readily available in the form of blocks, but lithium is a very toxic element. At Birmingham it comes in the form of a very fine powder. When used in an integral experiment it will be surrounded by some type of cladding material, usually stainless steel (essentially iron) or aluminium. In converting from lithium to LiF the total amount of lithium was kept constant. Each lithium zone thickness decreases from 1.67cm to 1.3cm because LiF has a higher nuclide density than natural lithium. The main effect of the fluorine in this design is to act as a moderator at high energies via the high elastic and inelastic scattering cross sections and as an absorber at low energies via the (n,gamma) reaction. In particular the (n,gamma) reaction has strong absorption peaks in the range 10-100 keV. These can be seen on figure 5 where the sensitivity of $Li6(n,\alpha)T$ to $Li6(n,\alpha)T$ is plotted. The energy of the resonances, from left to right, is 25-32 keV, 41-52 keV and the deepest one is 87-111 keV. With the addition of fluorine the $Li6(n,\alpha)T$ rate increased by 35%, from 0.247 to 0.333 but $Li7(n,n'\alpha)T$ was reduced by 5% from $1.69E-2$ to $1.6E-2$. Both the right and left leakages decreased, from 0.09 to 0.043 (52%) and from 1.54 to 1.48 (4%), respectively. Both decreases come about because of the lower leakages in low energy neutrons due solely to absorption in fluorine. The effect of the fluorine on the sensitivity profile is to bring the neutronics behaviour of the NFIE closer to the NFR. The overall coefficient dropped by 31%

from 0.49 to 0.34, which is only 60% removed from the NFR value of 0.21. In particular, the low energy profile shows a slight flattening out at low energies, again similar to that of the NFR. Design e) in this form would be an acceptable integral experiment since it has neutronics resembling the NFR. Except for some fluorine absorption structure, the likeness is very good and data adjustments via foil response measurements made to this design would be relevant to the NFR. However, as discussed shortly, one of the constraints on the design is its weight. In anticipation of this the amounts of Fe and Pb were decreased.

For a total 50% decrease in Pb, the total sensitivity coefficient increased by 40%, from 0.344 to 0.48, and for a 75% decrease the same drop was 69% from 0.344 to 0.574. The profile for a 75% decrease in Pb is shown in figure 6. There is a substantial amount of higher energy flux, particularly in the saddle centred around 12 Mev. The decrease in Pb is biasing the flux towards higher energies and away from the NFR profile. With the removal of the Fe reflector, the overall coefficient increased by 29%, from 0.344 to 0.445, because less of the high energy flux was absorbed in the Fe allowing more high energy neutrons to be reflected by the albedo. The reduction in low energy (100 ev to thermal) is very sharp indeed and indicates the necessity of an Fe reflector, as can be seen in figure 7.

As a final test in this design, with no Pb or Fe reductions, 10cm of Fe was removed from the reflector and 10 lots of 1cm thick Fe zones were placed on either side of each LiF zone. There were very slight decreases, 10% (0.333 to 0.3) and 23% ($1.67\text{E}-2$ to $1.28\text{E}-2$) in $\text{Li6}(n,\alpha)\text{T}$ and $\text{Li7}(n,n'\alpha)\text{T}$. There were also slight decreases in the leakages. The overall

125

sensitivity coefficient did increase very slightly, from 0.344 to 0.349, though this increase is due to there being more low energy flux. The high energy profile was left virtually untouched, as can be seen by comparing figures 5 and 8. The low energy sensitivity increase is evident and due solely to the enhanced downscattering produced by the Fe, with the extra bonus of Fe being a good reflector at low energies. From these results, as will be seen next, it was decided to use intermediate zones of natural Fe as additional neutron softening agents.

Design f

The final design stage, design f), is to optimise design e) and obtain a working integral experiment. The optimisation is between four quantities,

- 1) keeping the total weight below 3 metric tonnes.
- 2) making the y, z dimensions as large as possible.
- 3) keeping the overall sensitivity coefficient as low as possible.
- 4) attempting to increase the sensitivity at lower energies.

The integral experiment is most likely to be built in a room with a low albedo floor. To avoid the floor needing extra structural support, the weight must be kept as low as possible. A weight of about 3 metric tonnes is considered to be a maximum (Scott 1984). Equally important is that of size. In the 1-dimensional ANISN representation of a slab, one direction is completely defined (x, say) whilst the other two are left undefined (y,z) and considered to be of unit length. To reduce leakage, and particularly from the sides of the slabs which the ANISN code cannot calculate, the (y,z) dimensions should be made as large as possible within the confines of the weight constraint. The value of the tritium breeding and the cost of the integral experiment are not

important design considerations. The integral assembly will be mostly used for measuring fluxes and foil responses (for subsequent cross section adjustment) and the value of the tritium activity will not be of prime interest. The cost of the assembly materials, though certainly not negligible, should not be too appreciable. It is the relatively low cost of the integral experiments that makes them so attractive to build. The experiment in its present form contains 5 zones of 11.47cm natural lead, 5 zones of 1.3cm thick natural LiF and a 31.34cm thickness of natural iron, and with (y,z) dimensions of 100cm X 100cm the assembly weighs 9.14 metric tonnes. It is much too heavy.

It is seen previously that reducing the amount of Fe and Pb increases the overall sensitivity coefficient and sharpens the lower energy sensitivity profile - two effects which are not desired. In order to reduce these effects extra zones of materials have had to be added. Since the LiF regions will be surrounded by cladding in the assembly, it was decided to include them in the neutronics. At first they were only about 1mm thick and were just included for completeness but because of their neutron softening properties, they were later made much thicker and introduced into the experiment as separate, distinct zones. Various thicknesses of cladding were tried and analysed. However, there must first be a weight reduction. Because the assembly is more sensitive to the Pb than the Fe, it was decided to reduce the amount of Fe in the reflector by 70% and the amount of Pb by 30%. The integral design was then continued from this point. This leaves 5 zones of 8cm thick Pb and an Fe zone of thickness 9.4cm. The LiF remains unchanged. This assembly, with no Fe cladding, gave a sensitivity coefficient of 0.476 and a

much reduced profile at lower energies. The $\text{Li7}(n,n'\alpha)\text{T}$ are now 0.253 and $3.19\text{E-}2$, profile is shown in figure 9. The design was cladding included. Firstly, 1mm of Fe cladding on sides of each LiF zone. The overall sensitivity decreased very slightly by 2% , from 0.476 to 0.466 of sensitivity at high energies. There was a decrease in sensitivity at lower energies due to the enhancement of the Fe. Subsequently, the 1mm cladding was increased to 2mm of cladding. The total sensitivity coefficient decreased from 0.476 to 0.432 and the increase in sensitivity at low energies is quite noticeable. Figure 10 gives $\text{Li6}(n,\alpha)\text{T}$ to $\text{Li6}(n,\alpha)\text{T}$ with 10mm of cladding surrounding all the LiF zones. Comparing figure 10, the effect of the Fe in figure 10 can be seen as an increase in the lower energy sensitivity is due to the effect of the $\text{Fe}(n,\gamma)$ absorption resonance at $0.96\text{--}1.23\text{ keV}$. The sensitivity is reduced though not by any appreciable amount. For calculation, but with 20mm of cladding, the sensitivity coefficient decrease of 13% , from 0.476 to 0.413 , and a large increase in low energy sensitivity. The effect of the Fe at low energies can be seen by looking at the sensitivities. For instance, at group 80, $49\text{--}61\text{ eV}$, the sensitivity coefficient increases with cladding. The corresponding increases for 10mm and 20mm of cladding is 134% and 260% , respectively. The decrease in total sensitivity coefficient, at low energy sensitivity is increasing rapidly, is due to the fact that the sensitivity is decreasing slightly and the

these higher energies are much larger than. The decrease in sensitivity coefficient linear with the thickness of Fe cladding change was to replace the Fe cladding by an amount of aluminium was tested at cladding 10mm and 20mm. The results are essentially obtained with the iron. The sensitivity $\text{Li6}(n,\alpha)\text{T}$ with 10mm of aluminium cladding zone is shown in figure 11. This can be compared with figure 10 which contains Fe cladding. In fact the $\text{Fe}(n,\gamma)$ resonance at 0.96-1.23 keV is a larger amount of low energy flux. The aluminium has very little structure of its own into the present sensitivity 'spikes' can be seen which correspond to valleys. The high energy flux is virtually the same. The sensitivity coefficient for Al is slightly lower due to the lack of low energy flux. The reason aluminium is fairly transparent to low energy neutrons is that its (n,γ) cross section being typically ten times smaller than $\text{Fe}(n,\gamma)$. Because of this lack of low level flux it was decided to stay with Fe as the cladding material.

The final change is to further reduce the weight and bring the weight down to 3 tonnes. This reduction was achieved by removing the Fe cladding and without the Fe cladding. The Pb in the zone source was trebled, without any Fe cladding, the other zones were decreased to keep the weight constant. The sensitivity coefficient decreased very slightly due to the further removal of high energy neutrons. The $\text{Li6}(n,\alpha)\text{T}$ rate increased from 0.25 to 0.27. The amount of Pb by 90%, in the first zone only.

coefficient from 0.48 to 0.59 and decreases the $\text{Li6}(n,\alpha)\text{T}$ rate from 0.25 to 0.2. In particular, the flux dip at about 12 MeV fills up as the amount of Pb is decreased.

Including the Fe cladding has the same effect as increasing the Pb, and it is just over 30% lighter. So as not to increase the weight, the Fe cladding was taken from the reflector. The overall effect of the Fe cladding is to downscatter neutrons. The Fe cladding was chosen to be 5mm thick with the reflector having a thickness 10cm. It was decided to choose the (y,z) dimensions as 80cm each. With these dimensions, and a total weight of 3 metric tonnes, the thickness of each Pb zone, including the zone nearest the source, becomes 5.9cm. The LiF zones remain unchanged at a thickness of 1.3cm. These are the final dimensions of the non fissile integral experiment (see design f), figure 19). The total sensitivity coefficient is 0.458, as opposed to 0.344 for the uncladded unreduced design. Its value is over double that of the NFR, at 0.21. The reason for this difference is due to the sphere geometry effect as previously explained. However, the final sensitivity profile does show that the low energy flux is behaving much the same in the NFIE as the NFR, as can be seen in figure 12.

7.4 FURTHER TUNING BETWEEN PROFILES

This section details the tuning between the reactor and integral experiment for profiles which were not used in the experimental design. The potential number of profiles is very large and so only a few important ones will be presented here. The comparisons made here are both analytic (using sensitivity profiles) and predictive (using sensitivity coefficients). This

further tuning is for comparative purposes only, not for any changes to be made to the integral experiment. An attempt will be made to explain the differences in profiles in terms of differences in the radiation transport media and to give an account of the particle properties in the assembly. The integral experiment is a lot easier to analyse and resolve than the non fissile reactor because the elements making it up are physically separated from each other and because there are fewer such elements.

The NFIE zone sensitivities are given in tables 3-10. These can be directly compared to tables 3-10 in chapter 6. The total sensitivity coefficient (under the heading 'sum' in the tables) gives a single number which is an overall measure of how a system behaves. It should be noted that the Fe occurred in too many zones to be included across a single page and so has zero as its entry. However, the total sensitivity coefficient is given correctly. Fluorine has been included for completeness. In Pb neutron nuclear heating there are no direct terms on the $Pb(n,3n)$ or Pb total inelastic, hence the negative terms. Some interesting spatial effects are occurring in the assembly due to it being heterogeneous. Some zones show a positive sensitivity, while others are negative. However, in the reactor a lot of the effects have been masked because the zones are so big and spatial information has been lost. An example is the sensitivity of Li7 nuclear heating to $Li7(n,\gamma)$. In the NFIE it has a positive sensitivity in the first two zones since the flux still has a hard component and so the $Li7(n,\gamma)$ rate remains fairly large. It should be mentioned that this sensitivity result only comes about because there are more (n,γ) events occurring ; the gammas do not contribute to the Li7 neutron heating in any way.

Zone 4 of the NFR for this sensitivity is negative, showing simply that an increase in (n,γ) reduces the number of neutrons available for Li^7 neutron heating. However, a detailed interval calculation in this zone would yield a positive sensitivity near the first surface edge, for the same reason given above.

The sensitivity of $\text{Li}^6(n,\alpha)\text{T}$ to $\text{Pb}(n,2n)$ shows quite clearly (table 3) the difference between sphere and slab geometry. The overall coefficient of the above sensitivity is negative due to a very high negative sensitivity in group 1 of the first zone only. A detailed spatial breakdown shows that the negative sensitivity comes from the first spatial interval. The negative sensitivity is really saying that the collisional loss is greater than the downscattering gain. A 14.1 MeV neutron having an interaction in the first interval, particularly an $(n,2n)$ or $(n,3n)$ reaction, has a high chance of scattering the neutron back out towards the source. In the slab case this neutron is then lost. The reason for the sensitivity being so high is that the flux is mostly 14.1 MeV at this spatial position (with a lower energy reflected contribution) and so the number of these scattering events is high, with the corresponding high chance of neutron leakage. The detailed energy dependent leakage shows that all the neutron backscatter towards the source is in the first lead zone, and in particular in the energy range 1-2 Mev, the mean energy at which the neutrons from an $(n,2n)$ reaction appear at. The sensitivity of Li^6 nuclear heating to the $\text{Pb}(n,2n)$ partial in NFIE shows the same effect, mainly because the Li^6 heating response is dominated by the $\text{Li}^6(n,\alpha)\text{T}$ partial at energies below about 0.5 MeV.

— 155 —

little absorption occurring, which tends to cancel out the volume absorption effect of the lithium. Figure 13 shows a low energy tail. This is due to a combination of the volume effect and the high $\text{Li6}(n,\alpha)\text{T}$ cross section producing a large absorption effect coupled with a rapidly decreasing forward flux, increasing the loss of iron heating at low energies in the sphere case.

Figures 15 and 16 show the sensitivity of total Pb nuclear heating to the $(n,2n)$ partial. The $\text{Pb}(n,2n)$ partial is one of the most dominant cross sections in both the reactor and experiment. Both profiles show the same shape - a positive sensitivity in the highest four or five energy groups and a negative sensitivity from that group to the $\text{Pb}(n,2n)$ threshold energy in the 6.7-7.4 MeV range. There is very little difference between the two profiles, except that the spherical reactor has a slightly lower sensitivity. In figure 16 the negative sensitivity occurs just below the 10^{-3} cut-off and so is not shown. In this profile, the indirect loss, downscattering gain and direct gain terms are invoked. The $\text{Pb}(n,2n)$ partial plays an important role at high energies in contributing to the Pb nuclear heating. This is represented by a high direct gain term in the top energy group. The large amount of flux at this energy means that the event reaction rate is high and hence gives a large direct gain term. The two neutrons resulting from the reaction cause additional heating and this heating sensitivity is stored in the energy group of the interaction energy. This additional energy is the reason for the positive sensitivity in groups 2 to 5 and is due to further reactions by the neutrons in the Pb inelastic. The amount of flux drops off rapidly from group 1 and the direct gain will become negligible. However, the Pb inelastic cross section has a threshold at group 33, about

0.55-0.6 MeV, and it is only the smaller (n,gamma) and elastic kerma factors that contribute to the heating. Neutrons cascading below the Pb inelastic threshold will thus contribute little to the Pb heating. So the reason for the sign change is that the neutrons from an (n,2n) reaction, at interaction energies below about 9 MeV, tend to spill into energies lower than the Pb inelastic cross section and so the amount of downscattering gain is sharply reduced.

Although these are not shown graphically, the sensitivities of $\text{Li7}(n,n'\alpha)\text{T}$ to $\text{Pb}(n,2n)$ are almost exactly the same for both reactor and experiment. This indicates that the dominant $\text{Pb}(n,2n)$ partial is acting in the same manner in the two designs at the highest energies. In this respect, the tuning at high energies is very good.

Figures 17 and 18 give the sensitivities of $\text{Li6}(n,\alpha)\text{T}$ to the Fe total inelastic scattering cross section. Only the collisional loss and the downscattering gain term are invoked in producing these profiles. The spherical reactor has a positive profile, except at 14.1 MeV, with a small total coefficient of $3.1\text{E-}4$ whilst the experiment is positive except at the highest 6 energy groups, 8.1-14.1 MeV. The negative effects outweigh the positive effects and the sensitivity coefficient is negative with a value of $-1.66\text{E-}2$. The Fe inelastic scattering cross section has a fairly flat shape between 14.1 MeV and 3 MeV, but falls rapidly between 3 MeV and its threshold value, in the energy range 0.82-0.9 MeV. The quantitative difference between the profiles can be explained by the volume effect of the sphere. For the sphere case the profile is saying that the $\text{Li6}(n,\alpha)\text{T}$ rate is changing very little indeed for a change in the Fe

inelastic cross section. This insensitivity is due to the difficulty that flux in these Fe zones experiences in getting back to the lithium zone. Qualitatively the profile shapes are similar except at the highest energies. For both profiles, the sharp increase in sensitivity towards the inelastic threshold is due to the rapidly increasing amount of flux, with the corresponding high sensitivity gains from having neutrons downscatter at these energies. The slab negative sensitivities, in comparison with the sphere, is due to the lack of albedo. Inelastic scattering anisotropy increases at higher energies, and towards the edge of the slab surface the increase in the Fe inelastic cross section at high energies would tend to push neutrons out of the system with the subsequent loss in breeding. The change in sign comes about because the anisotropy decreases to become less forward peaked as the neutron interaction energy decreases.

The NFR and NFIE design parameters are shown in table 2. These quantities are given as volume integrated averages. Because of the leaky nature of the slab geometry, its parameters are smaller than those of the reactor. The degree of smallness depends on the energy dependence of the particular response. Because, for the slab, the first zone is a 5.9cm thick block of lead, then the system is essentially composed of flux in the energy range 50 keV - 1 Mev. So those responses which are biased towards lower energies show the biggest decrease in value. The $\text{Li6}(n,\alpha)\text{T}$ reaction rate decreases by 92%, but $\text{Li7}(n,n'\alpha)\text{T}$ only decreases by 66%. These values are meaningful since the mean free path thickness of lithium that a neutron sees is the same for both systems. The Li6 neutron kerma response is dominated at low energies by the $\text{Li6}(n,\alpha)\text{T}$ and so this shows

150

the same decrease. The Li6 and Li7 gamma heating responses show a very small decrease. This is due to the fact that both responses are biased towards high gamma energies, and there are relatively few gammas in the system at these energies.

Difficulty occurs in trying to compare the DPA rates between the reactor and the experiment. The NFIE source strength was taken to be 10^9 neutrons sec^{-1} , whilst the NFR was $1.56\text{E}+20$ neutrons sec^{-1} . The rate of DPA scales proportionally with the source strength. Since the reactor is homogeneously mixed and the NFIE is heterogenous, there is difficulty in choosing spatial positions at which to compare the DPAs. Thus the DPA's were evaluated for the whole geometry.

	Design a)	Design b)	Design c)	Design d)	Design e)	Design f)
Li6(n,alpha)T	1.1703	1.1555	1.2307	0.247	1.333	1.251
Li7(n,n' alpha)T	0.0296	0.0302	0.0283	0.0169	0.0161	0.031
sensitivity coefficient	0.21	0.24	0.515	0.497	0.344	1.458
left leakage	0.0	0.0	1.54	1.55	1.487	1.30
right leakage	0.0391	0.0321	0.0787	0.090	0.043	0.197

TABLE 1. SOME RESULTS FROM INTERMEDIATE STEPS IN
DESIGNING THE NON FISSILE INTEGRAL EXPERIMENT

	NFR $P_3 S_4$	NFIE $P_3 S_g$	REDUCTION
Li6(n,alpha)T	1.1703	9.461E-2	66%
Li7(n,n' alpha)T	2.963E-2	9.871E-3	92%
Fe heating	8.636E+5	1.207E+5	86%
Pb heating	6.033E+5	1.006E+5	83%
Fl9 heating	-	1.791E+5	-
Li6 neutron heating	5.741E+6	4.769E+5	92%
Li7 neutron heating	8.311E+5	1.662E+5	80%
Fe gamma heating	2.529E+6	2.700E+5	89%
Pb gamma heating	8.096E+6	2.286E+6	71%
Li6 gamma heating	1.938E+3	1.330E+3	31%
Li7 gamma heating	2.454E+4	1.531E+4	38%
Fl9 gamma heating	-	5.288E+4	-
<u>DPA/year</u>			
Fe total DPA	0.3	2.679E-8	
Li6 total DPA	7.3	4.391E-9	
Li7 total DPA	5.4	3.110E-9	
Pb total DPA	24.0	3.453E-8	
<u>GAS APPM/year</u>			
Fe(n,proton)	0.3	4.468E-7	
Fe(n,alpha)	0.1	2.688E-7	

TABLE 2. NFR AND NFIE PERFORMANCE PARAMETERS.

TABLE 4

SENSITIVITY COEFFICIENTS OF L17 (N, ALPHA N⁺)
TO L16, P0, P1, L17, F19 PARTIAL CROSS SECTIONS
NON FISSION INTEGRAL EXPERIMENT

	ZONE 1	ZONE 7	ZONE 11	ZONE 15	ZONE 19	SUM
L16 ELASTIC	1.41E-04	2.33E-04	1.29E-04	7.16E-05	5.06E-05	6.27E-04
L16 (N,2N)ALPHA	-3.90E-04	-1.75E-04	-6.73E-05	-2.27E-05	-5.56E-06	-6.61E-04
L16 (N,N ⁺)ALPHA	-1.25E-03	-6.10E-04	-2.52E-04	-8.68E-05	-1.70E-05	-2.22E-03
L16 (N, GAMMA)	-6.89E-08	-3.28E-08	-1.31E-08	-4.64E-09	-1.26E-09	-1.21E-07
L16 (N, PROTON)	-5.93E-05	-2.97E-05	-1.24E-05	-4.55E-06	-1.32E-06	-1.07E-04
L16 (N, DEUTERIUM)	-2.08E-04	-1.04E-04	-4.29E-05	-1.57E-05	-4.52E-06	-3.75E-04
L16 INELASTIC	-1.25E-03	-6.11E-04	-2.52E-04	-8.69E-05	-1.69E-05	-2.22E-03
L17 ELASTIC	1.69E-03	2.81E-03	1.57E-03	9.07E-04	6.37E-04	7.64E-03
L17 (N,2N)	-1.44E-03	-6.49E-04	-2.49E-04	-8.37E-05	-2.05E-05	-2.44E-03
L17 (N,2N)ALPHA	-2.14E-03	-9.31E-04	-3.48E-04	-1.14E-04	-2.68E-05	-3.56E-03
L17 (N, ALPHA)	5.64E-01	2.56E-01	1.04E-01	4.07E-02	1.55E-02	9.80E-01
L17 (N, GAMMA)	-7.84E-07	-3.72E-07	-1.09E-07	-5.26E-08	-1.42E-08	-1.37E-06
L17 (N, PROTON)	-6.52E-04	-2.87E-04	-1.08E-04	-3.58E-05	-8.52E-06	-1.09E-03
L17 (N, DEUTERIUM)	-1.19E-02	-5.70E-03	-2.33E-03	-7.79E-04	-1.25E-04	-2.08E-02
L17 INELASTIC	2.85E-03	3.56E-03	1.90E-03	1.02E-03	6.32E-04	9.96E-03
F19 ELASTIC	-3.38E-03	-1.46E-03	-5.00E-04	-1.75E-04	-4.03E-05	-5.59E-03
F19 (N,2N)	-2.81E-03	-1.29E-03	-5.01E-04	-1.70E-04	-3.92E-05	-4.82E-03
F19 (N,N ⁺)ALPHA	-7.80E-04	-4.43E-04	-1.29E-04	-4.26E-05	-1.01E-05	-1.30E-03
F19 (N,N ⁺)PROTON	-9.76E-07	-4.83E-07	-1.99E-07	-7.26E-08	-2.07E-08	-1.75E-06
F19 (N, GAMMA)	-1.65E-03	-8.12E-04	-3.33E-04	-1.21E-04	-3.40E-05	-2.95E-03
F19 (N, PROTON)	-8.92E-04	-4.07E-04	-1.58E-04	-5.19E-05	-1.16E-05	-1.52E-03
F19 (N, DEUTERIUM)	-5.49E-04	-2.48E-04	-9.53E-05	-3.22E-05	-7.99E-06	-9.12E-04
F19 (N, TRITON)	-3.44E-03	-1.81E-03	-7.73E-04	-2.94E-04	-8.90E-05	-6.40E-03
F19 (N, ALPHA)	-3.47E-02	-1.61E-02	-6.36E-03	-2.14E-03	-4.47E-04	-5.98E-02
F19 INELASTIC	-2.87E-03	-1.46E-03	-5.00E-04	-1.75E-04	-4.03E-05	-5.59E-03

	ZONE 1	ZONE 5	ZONE 9	ZONE 13	ZONE 17	SUM
P0 ELASTIC	-2.87E-01	-4.84E-02	-6.59E-03	1.34E-03	3.05E-04	-3.38E-01
P0 (N,2N)	-1.10E-01	-2.16E-01	-8.73E-02	-3.11E-02	-9.00E-03	-7.54E-01
P0 (N,3N)	-1.20E-02	-5.70E-03	-2.15E-03	-7.20E-04	-1.94E-04	-2.07E-02
P0 (N, GAMMA)	-4.05E-05	-2.56E-05	-1.11E-05	-4.24E-06	-1.14E-06	-8.20E-05
P0 INELASTIC	-5.13E-02	-3.44E-02	-1.67E-02	-6.90E-03	-2.25E-03	-1.12E-01
P1 ELASTIC	0.0	0.0	0.0	0.0	0.0	5.24E-03
P1 (N,2N)	0.0	0.0	0.0	0.0	0.0	5.18E-02
P1 (N,N ⁺)ALPHA	0.0	0.0	0.0	0.0	0.0	4.43E-04
P1 (N,N ⁺)PROTON	0.0	0.0	0.0	0.0	0.0	-4.64E-01
P1 (N, GAMMA)	0.0	0.0	0.0	0.0	0.0	-3.42E-05
P1 (N, PROTON)	0.0	0.0	0.0	0.0	0.0	-1.65E-02
P1 (N, DEUTERIUM)	0.0	0.0	0.0	0.0	0.0	-2.26E-03
P1 (N, TRITON)	0.0	0.0	0.0	0.0	0.0	-5.25E-05
P1 (N, ALPHA)	0.0	0.0	0.0	0.0	0.0	-3.85E-05
P1 (N, HELIUM 3)	0.0	0.0	0.0	0.0	0.0	-5.08E-03
P1 INELASTIC	0.0	0.0	0.0	0.0	0.0	-8.20E-02

TABLE 3

SENSITIVITY COEFFICIENTS OF L16 (N, ALPHA)
TO L16, P0, P1, L17, F19 PARTIAL CROSS SECTIONS
NON FISSION INTEGRAL EXPERIMENT

	ZONE 1	ZONE 7	ZONE 11	ZONE 15	ZONE 19	SUM
L16 ELASTIC	2.19E-04	1.99E-04	2.51E-04	2.11E-04	1.37E-04	8.21E-04
L16 (N,2N)ALPHA	-3.11E-05	-3.62E-05	-2.90E-05	-1.45E-05	-6.06E-06	-5.46E-05
L16 (N,N ⁺)ALPHA	-1.15E-03	-4.19E-04	-1.00E-04	-4.03E-06	-1.32E-05	-1.66E-03
L16 (N, GAMMA)	-2.06E-06	-4.64E-06	-5.63E-06	-4.98E-06	-3.11E-06	-2.05E-05
L16 (N, PROTON)	-5.01E-05	-1.10E-05	-1.37E-05	-4.70E-06	-1.13E-06	-1.01E-04
L16 (N, ALPHA)	1.10E-01	1.12E-01	8.86E-02	7.21E-02	6.44E-02	4.56E-01
L16 INELASTIC	-1.16E-03	-4.25E-04	-1.02E-04	-4.35E-06	-1.31E-05	-1.68E-03
L17 ELASTIC	3.21E-03	2.50E-02	3.18E-02	2.73E-02	1.87E-02	1.06E-01
L17 (N,2N)	-1.10E-04	-1.32E-04	-1.06E-04	-5.29E-05	-2.22E-05	-1.95E-04
L17 (N,2N)ALPHA	-1.61E-05	-2.95E-04	-1.98E-04	-8.99E-05	-3.41E-05	-5.81E-04
L17 (N, ALPHA)	-1.11E-02	-4.41E-03	-1.29E-03	-2.18E-04	-6.17E-05	-1.69E-02
L17 (N, GAMMA)	-2.74E-05	-6.23E-05	-7.56E-05	-6.48E-05	-4.20E-05	-2.72E-04
L17 (N, PROTON)	-6.04E-04	-3.01E-04	-1.13E-04	-3.30E-05	-5.97E-06	-1.06E-03
L17 (N, DEUTERIUM)	-1.15E-02	-3.19E-03	-5.61E-05	-6.27E-04	-4.09E-04	-1.16E-02
F19 ELASTIC	1.04E-02	1.85E-02	4.61E-02	4.30E-02	3.33E-02	1.92E-01
F19 (N,2N)	-1.96E-04	-2.49E-04	-2.20E-04	-1.13E-04	-4.68E-05	-2.30E-04
F19 (N,N ⁺)ALPHA	-4.93E-03	-7.87E-04	-2.43E-04	-4.75E-05	-6.23E-06	-2.99E-03
F19 (N,N ⁺)PROTON	-1.02E-07	-1.50E-07	-4.16E-07	-6.40E-08	-2.04E-08	-5.99E-07
F19 (N, GAMMA)	-8.08E-04	-1.76E-03	-1.88E-03	-1.30E-03	-5.62E-04	-6.31E-03
F19 (N, PROTON)	-1.26E-03	-7.09E-04	-2.94E-04	-9.39E-05	-1.97E-05	-2.38E-03
F19 (N, DEUTERIUM)	-7.87E-04	-4.07E-04	-1.58E-04	-4.74E-05	-8.92E-06	-1.41E-03
F19 (N, TRITON)	-4.94E-04	-2.53E-04	-9.73E-05	-2.80E-05	-5.16E-06	-8.70E-04
F19 (N, ALPHA)	-2.40E-03	-1.59E-03	-7.19E-04	-2.50E-04	-6.16E-05	-5.10E-03
F19 INELASTIC	-2.64E-03	-1.94E-02	-2.31E-02	-1.59E-02	-7.29E-03	-6.30E-02

	ZONE 1	ZONE 5	ZONE 9	ZONE 13	ZONE 17	SUM
P0 ELASTIC	2.24E-01	-7.60E-02	2.55E-03	4.60E-02	8.77E-02	3.53E-01
P0 (N,2N)	-1.23E-01	-1.50E-02	-3.16E-02	-1.92E-02	-8.70E-03	-4.88E-02
P0 (N,3N)	1.93E-03	7.10E-03	4.20E-03	1.70E-03	5.83E-04	1.56E-02
P0 (N, GAMMA)	-1.33E-03	-5.51E-03	-7.38E-03	-6.31E-03	-3.72E-03	-2.42E-02
P0 INELASTIC	-2.03E-02	-4.63E-03	-5.65E-03	-5.86E-03	-3.24E-03	-1.02E-02
P1 ELASTIC	0.0	0.0	0.0	0.0	0.0	3.12E-01
P1 (N,2N)	0.0	0.0	0.0	0.0	0.0	4.79E-03
P1 (N,N ⁺)ALPHA	0.0	0.0	0.0	0.0	0.0	-1.66E-04
P1 (N,N ⁺)PROTON	0.0	0.0	0.0	0.0	0.0	-2.05E-03
P1 (N, GAMMA)	0.0	0.0	0.0	0.0	0.0	-1.04E-01
P1 (N, PROTON)	0.0	0.0	0.0	0.0	0.0	-1.62E-02
P1 (N, DEUTERIUM)	0.0	0.0	0.0	0.0	0.0	-2.26E-03
P1 (N, TRITON)	0.0	0.0	0.0	0.0	0.0	-5.26E-05
P1 (N, ALPHA)	0.0	0.0	0.0	0.0	0.0	-3.85E-05
P1 (N, HELIUM 3)	0.0	0.0	0.0	0.0	0.0	-5.08E-03
P1 INELASTIC	0.0	0.0	0.0	0.0	0.0	-1.66E-02

TABLE 5.

SENSITIVITY COEFFICIENTS OF P0 NEUTRON HEATING
TO L16, P0, FE, L17, F19 PARTIAL CROSS SECTIONS
NON FISSION INTEGRAL EXPERIMENT

	ZONE 3	ZONE 7	ZONE 11	ZONE 15	ZONE 19	SUM
L16 ELASTIC	-1.74E-03	-1.31E-03	-7.81E-04	-3.45E-04	-3.86E-05	-4.22E-03
L16(N,2N)ALPHA	-1.10E-04	-3.10E-05	-6.60E-06	1.42E-07	1.46E-06	-1.46E-04
L16(N,N')ALPHA	-1.03E-03	-4.42E-04	-1.56E-04	-3.92E-05	2.73E-06	-1.67E-03
L16(N,GAMMA)	-1.22E-07	-1.23E-07	-9.19E-08	-5.31E-08	-1.76E-08	-4.08E-07
L16(N,PROTON)	-4.16E-05	-2.19E-05	-8.92E-06	-2.88E-06	-4.74E-07	-7.58E-05
L16(N,ALPHA)T	-2.85E-03	-3.79E-03	-3.06E-03	-1.88E-03	-6.97E-04	-1.21E-02
L16 INELASTIC	-1.04E-03	-4.47E-04	-1.57E-04	-3.92E-05	-2.79E-06	-1.68E-03
L17 ELASTIC	-2.20E-02	-1.54E-02	-8.63E-03	-3.52E-03	-1.20E-04	-4.97E-02
L17(N,2N)	-4.13E-04	-1.17E-04	-2.57E-05	5.72E-08	3.30E-06	-5.50E-04
L17(N,2N)ALPHA	-9.08E-04	-3.16E-04	-9.25E-05	-1.76E-05	3.61E-06	-1.33E-03
L17(N,ALPHA)T	-8.26E-03	-3.13E-03	-9.71E-04	-1.89E-04	4.80E-05	-1.25E-02
L17(N,GAMMA)	-1.38E-06	-1.45E-06	-1.13E-06	-6.73E-07	-2.56E-07	-4.87E-06
L17(N,DEUTERIUM)	-4.75E-04	-2.05E-04	-7.23E-05	-1.99E-05	-2.18E-06	-7.75E-04
L17 INELASTIC	-1.17E-02	-5.60E-03	-2.28E-03	-6.98E-04	-1.46E-05	-2.03E-02
F19 ELASTIC	-1.62E-02	-1.16E-02	-8.42E-03	-3.39E-03	6.11E-04	-4.11E-02
F19(N,2N)	-1.04E-03	-2.92E-04	-6.43E-05	-1.23E-06	1.14E-05	-1.30E-03
F19(N,N')ALPHA	-1.41E-03	-5.25E-04	-1.62E-04	-3.28E-05	5.71E-06	-2.12E-03
F19(N,N')PROTON	-4.01E-04	-1.56E-04	-5.05E-05	-1.28E-05	7.14E-06	-6.19E-04
F19(N,GAMMA)	-3.89E-05	-6.41E-05	-5.84E-05	-3.61E-05	-1.24E-05	-2.10E-04
F19(N,PROTON)	-1.02E-03	-4.90E-04	-1.88E-04	-5.71E-05	-7.98E-06	-1.76E-03
F19(N,DEUTERIUM)	-6.23E-04	-2.79E-04	-1.01E-04	-2.87E-05	-3.39E-06	-1.03E-03
F19(N,TRITON)	-3.91E-04	-1.73E-04	-6.22E-05	-1.75E-05	-2.02E-06	-6.45E-04
F19(N,ALPHA)	-2.08E-03	-1.12E-03	-4.67E-04	-1.54E-04	-2.61E-05	-3.85E-03
F19 INELASTIC	-3.72E-02	-2.81E-02	-1.63E-02	-7.24E-03	-1.51E-03	-9.03E-02
P0 ELASTIC	2.40E-01	1.45E-01	1.10E-01	7.30E-02	4.69E-02	5.75E-01
P0(N,2N)	6.93E-02	4.72E-02	2.62E-02	1.31E-02	6.64E-03	1.63E-01
P0(N,N)	-7.23E-03	-3.15E-03	-1.09E-03	-3.08E-04	-4.52E-05	-1.18E-02
P0(N,GAMMA)	-2.88E-04	-6.50E-04	-5.91E-04	-3.79E-04	-1.70E-04	-2.08E-03
P0 INELASTIC	-4.29E-02	-4.19E-02	-2.27E-02	-9.61E-03	-2.74E-03	-1.20E-01
FE ELASTIC	0.0	0.0	0.0	0.0	0.0	-1.02E-02
FE(N,2N)	0.0	0.0	0.0	0.0	0.0	-1.49E-02
FE(N,N')ALPHA	0.0	0.0	0.0	0.0	0.0	-2.23E-04
FE(N,N')PROTON	0.0	0.0	0.0	0.0	0.0	-2.34E-03
FE(N,GAMMA)	0.0	0.0	0.0	0.0	0.0	-1.65E-03
FE(N,PROTON)	0.0	0.0	0.0	0.0	0.0	-1.19E-02
FE(N,DEUTERIUM)	0.0	0.0	0.0	0.0	0.0	-1.64E-03
FE(N,TRITON)	0.0	0.0	0.0	0.0	0.0	-3.82E-05
FE(N,HELIUM 3)	0.0	0.0	0.0	0.0	0.0	-2.79E-05
FE(N,ALPHA)	0.0	0.0	0.0	0.0	0.0	-3.64E-03
FE INELASTIC	0.0	0.0	0.0	0.0	0.0	-6.25E-02

TABLE 6.

SENSITIVITY COEFFICIENTS OF L16 NEUTRON HEATING
TO L16, P0, FE, L17, F19 PARTIAL CROSS SECTIONS
NON FISSION INTEGRAL EXPERIMENT

	ZONE 3	ZONE 7	ZONE 11	ZONE 15	ZONE 19	SUM
L16 ELASTIC	7.16E-03	6.57E-03	5.10E-03	3.44E-03	2.00E-03	2.43E-02
L16(N,2N)ALPHA	1.52E-03	6.89E-04	2.81E-04	1.08E-04	3.99E-05	2.63E-03
L16(N,N')ALPHA	8.14E-03	3.76E-03	1.59E-03	6.50E-04	2.54E-04	1.44E-02
L16(N,GAMMA)	-1.88E-06	-4.37E-06	-5.33E-06	-4.65E-06	-3.14E-06	-1.94E-05
L16(N,PROTON)	2.52E-04	1.06E-04	4.20E-05	1.69E-05	7.01E-06	4.24E-04
L16(N,ALPHA)T	1.22E-01	1.16E-01	9.09E-02	7.31E-02	6.36E-02	4.66E-01
L16 INELASTIC	-1.18E-03	-4.36E-04	-1.09E-04	-7.67E-06	1.23E-05	-1.72E-03
L17 ELASTIC	1.68E-03	2.27E-02	2.96E-02	2.57E-02	1.77E-02	9.74E-02
L17(N,2N)	-1.44E-04	1.15E-04	9.81E-05	4.99E-05	2.13E-05	1.41E-04
L17(N,2N)ALPHA	-9.78E-05	2.58E-04	1.81E-04	8.38E-05	3.24E-05	4.57E-04
L17(N,ALPHA)T	-1.11E-02	-4.42E-03	-1.30E-03	-2.19E-04	6.04E-05	-1.70E-02
L17(N,GAMMA)	-2.62E-05	-5.95E-05	-7.21E-05	-6.18E-05	-4.00E-05	-2.60E-04
L17(N,DEUTERIUM)	-6.99E-04	-3.02E-04	-1.14E-04	-3.32E-05	-6.06E-06	-1.06E-03
L17 INELASTIC	-1.17E-02	-3.40E-03	-1.01E-04	5.45E-04	3.82E-04	-1.43E-02
F19 ELASTIC	2.80E-02	3.59E-02	4.34E-02	4.15E-02	3.17E-02	1.80E-01
F19(N,2N)	-4.57E-04	2.12E-04	2.02E-04	1.05E-04	4.49E-05	1.07E-04
F19(N,N')ALPHA	-1.93E-03	-7.89E-04	-2.44E-04	-4.87E-05	5.70E-06	-3.01E-03
F19(N,N')PROTON	-4.13E-04	-1.55E-04	-4.38E-05	-7.29E-06	1.76E-06	-6.17E-04
F19(N,GAMMA)	-7.72E-04	-1.68E-03	-1.80E-03	-1.24E-03	-5.36E-04	-6.81E-03
F19(N,PROTON)	-1.28E-03	-7.14E-04	-2.95E-04	-9.47E-05	-2.08E-05	-2.40E-03
F19(N,DEUTERIUM)	-7.94E-04	-4.09E-04	-1.59E-04	-4.77E-05	-9.06E-06	-1.42E-03
F19(N,TRITON)	-4.98E-04	-2.54E-04	-9.76E-05	-2.91E-05	-5.45E-06	-8.85E-04
F19(N,ALPHA)	-2.52E-03	-1.60E-03	-7.24E-04	-2.53E-04	-6.25E-05	-5.16E-03
F19 INELASTIC	-5.33E-03	-1.61E-02	-2.05E-02	1.43E-02	6.67E-03	5.23E-02
P0 ELASTIC	2.95E-01	-1.09E-02	1.54E-03	4.47E-02	8.41E-02	1.24E-01
P0(N,2N)	-1.31E-01	9.49E-03	2.87E-02	1.79E-02	8.27E-03	-6.69E-02
P0(N,N)	1.22E-03	6.53E-03	3.99E-03	1.67E-03	5.50E-04	2.13E-02
P0(N,GAMMA)	-1.28E-03	-5.29E-03	-7.07E-03	-6.03E-03	-1.56E-03	-2.43E-02
P0 INELASTIC	-2.48E-02	-6.36E-03	4.05E-03	2.91E-03	-1.67E-03	-1.67E-02
FE ELASTIC	0.0	0.0	0.0	0.0	0.0	2.70E-01
FE(N,2N)	0.0	0.0	0.0	0.0	0.0	1.12E-01
FE(N,N')ALPHA	0.0	0.0	0.0	0.0	0.0	-1.75E-04
FE(N,N')PROTON	0.0	0.0	0.0	0.0	0.0	-2.13E-01
FE(N,GAMMA)	0.0	0.0	0.0	0.0	0.0	-9.90E-02
FE(N,PROTON)	0.0	0.0	0.0	0.0	0.0	-1.63E-02
FE(N,DEUTERIUM)	0.0	0.0	0.0	0.0	0.0	-2.27E-03
FE(N,TRITON)	0.0	0.0	0.0	0.0	0.0	-5.29E-05
FE(N,HELIUM 3)	0.0	0.0	0.0	0.0	0.0	-3.80E-05
FE(N,ALPHA)	0.0	0.0	0.0	0.0	0.0	-5.01E-01
FE INELASTIC	0.0	0.0	0.0	0.0	0.0	-2.01E-02

TABLE 8

SENSITIVITY COEFFICIENTS OF FE NEUTRON HEATING
TO L16, P0, P1, L17, F19 PARTIAL CROSS SECTIONS
NON FISSION INTEGRAL EXPERIMENT

	ZONE 3	ZONE 7	ZONE 11	ZONE 15	ZONE 19	SUM
L16 ELASTIC	-1.42E-03	-9.47E-04	-7.29E-04	-5.68E-04	-4.21E-04	-4.49E-03
L16(N,2N)ALPHA	-3.85E-04	-1.39E-04	-6.44E-05	-3.48E-05	-2.41E-05	-5.67E-04
L16(N,N')ALPHA	-1.82E-03	-8.93E-04	-4.37E-04	-2.42E-04	-1.66E-04	-3.56E-03
L16(N,GAMMA)	-8.61E-08	-6.92E-08	-5.14E-08	-3.90E-08	-3.11E-08	-2.77E-07
L16(N,PROTON)	-4.68E-05	-2.45E-05	-1.21E-05	-6.40E-06	-4.17E-06	-9.31E-05
L16(N,ALPHA)	-1.14E-03	-1.57E-03	-1.47E-03	-1.24E-03	-1.02E-03	-6.44E-03
L16 INELASTIC	-1.04E-03	-9.03E-04	-4.41E-04	-2.44E-04	-1.68E-04	-3.60E-03
L17 ELASTIC	-1.93E-02	-1.23E-02	-9.40E-03	-6.88E-03	-5.12E-03	-5.26E-02
L17(N,2N)	-1.13E-03	-5.17E-04	-2.39E-04	-1.29E-04	-8.96E-05	-2.10E-03
L17(N,N')ALPHA	-1.02E-03	-8.13E-04	-3.83E-04	-2.02E-04	-1.35E-04	-3.37E-03
L17(N,ALPHA)	-1.64E-02	-7.84E-03	-3.76E-03	-2.06E-03	-1.41E-03	-3.15E-02
L17(N,GAMMA)	-9.71E-07	-7.89E-07	-5.99E-07	-4.65E-07	-3.80E-07	-3.21E-06
L17(N,DEUTERIUM)	-6.32E-04	-3.04E-04	-1.42E-04	-7.35E-05	-4.71E-05	-1.20E-03
L17 INELASTIC	-2.01E-02	-1.02E-02	-5.19E-03	-2.95E-03	-2.00E-03	-4.04E-02
F19 ELASTIC	-1.30E-02	-9.73E-03	-8.60E-03	-7.59E-03	-6.50E-03	-4.55E-02
F19(N,2N)	-2.72E-03	-1.21E-03	-6.46E-04	-2.97E-04	-2.02E-04	-4.99E-03
F19(N,N')ALPHA	-2.09E-03	-1.37E-03	-6.44E-04	-3.46E-04	-2.36E-04	-5.49E-03
F19(N,N')PROTON	-6.98E-04	-3.23E-04	-1.50E-04	-7.87E-05	-5.21E-05	-1.29E-03
F19(N,GAMMA)	-1.62E-05	-2.79E-05	-2.93E-05	-2.73E-05	-2.99E-05	-2.31E-04
F19(N,PROTON)	-1.28E-03	-6.59E-04	-3.18E-04	-1.60E-04	-1.11E-04	-2.54E-03
F19(N,DEUTERIUM)	-8.22E-04	-4.04E-04	-1.91E-04	-9.96E-05	-6.50E-05	-1.50E-03
F19(N,TRITON)	-5.18E-04	-2.54E-04	-1.19E-04	-6.23E-05	-4.05E-05	-9.94E-04
F19(N,ALPHA)	-2.26E-03	-1.24E-03	-6.17E-04	-3.29E-04	-2.18E-04	-4.66E-03
F19 INELASTIC	-4.61E-02	-2.87E-02	-1.77E-02	-1.15E-02	-8.03E-03	-1.12E-01
P0 ELASTIC	-2.54E-01	-7.45E-02	-3.75E-02	-2.01E-02	-2.45E-02	-4.19E-01
P0(N,2N)	-3.59E-01	-1.81E-01	-8.26E-02	-4.13E-02	-2.56E-02	-6.91E-01
P0(N,3N)	-1.12E-02	-5.64E-03	-2.50E-03	-1.20E-03	-6.89E-04	-2.13E-02
P0(N,GAMMA)	-1.25E-04	-2.80E-04	-2.87E-04	-2.41E-04	-1.94E-04	-1.14E-03
P0 INELASTIC	-5.82E-02	-4.02E-02	-2.25E-02	-1.27E-02	-7.91E-03	-1.41E-01
P2 ELASTIC	0.0	0.0	0.0	0.0	0.0	1.75E-01
FE(N,2N)	0.0	0.0	0.0	0.0	0.0	1.06E-02
FE(N,N')ALPHA	0.0	0.0	0.0	0.0	0.0	4.01E-03
FE(N,N')PROTON	0.0	0.0	0.0	0.0	0.0	1.92E-02
FE(N,GAMMA)	0.0	0.0	0.0	0.0	0.0	-1.56E-03
FE(N,PROTON)	0.0	0.0	0.0	0.0	0.0	3.28E-01
FE(N,DEUTERIUM)	0.0	0.0	0.0	0.0	0.0	2.40E-02
FE(N,TRITON)	0.0	0.0	0.0	0.0	0.0	1.63E-04
FE(N,HELIUM 3)	0.0	0.0	0.0	0.0	0.0	2.22E-04
FE(N,ALPHA)	0.0	0.0	0.0	0.0	0.0	1.34E-01
FE INELASTIC	0.0	0.0	0.0	0.0	0.0	-1.18E-01

TABLE 7

SENSITIVITY COEFFICIENTS OF L17 NEUTRON HEATING
TO L16, P0, P1, L17, F19 PARTIAL CROSS SECTIONS
NON FISSION INTEGRAL EXPERIMENT

	ZONE 3	ZONE 7	ZONE 11	ZONE 15	ZONE 19	SUM
L16 ELASTIC	-1.22E-03	-1.29E-03	-7.87E-04	-3.88E-04	-1.03E-04	-4.29E-03
L16(N,2N)ALPHA	-1.94E-04	-7.30E-05	-2.42E-05	-6.99E-06	-4.86E-07	-2.98E-04
L16(N,N')ALPHA	-1.31E-03	-6.23E-04	-2.45E-04	-7.95E-05	-1.22E-05	-2.29E-03
L16(N,GAMMA)	-1.16E-07	-9.16E-08	-6.15E-08	-3.52E-08	-1.59E-08	-3.22E-07
L16(N,PROTON)	-5.80E-05	-3.11E-05	-1.32E-05	-6.74E-06	-1.28E-06	-1.08E-04
L16(N,ALPHA)	-1.00E-03	-2.11E-03	-1.63E-03	-1.01E-03	-4.77E-04	-7.03E-03
L16 INELASTIC	-1.34E-03	-6.28E-04	-2.47E-04	-7.90E-05	-1.22E-05	-2.31E-03
L17 ELASTIC	2.34E-01	1.49E-01	8.18E-02	4.23E-02	2.06E-02	5.27E-01
L17(N,2N)	6.81E-03	2.89E-03	1.12E-03	4.22E-04	1.59E-04	1.14E-02
L17(N,N')ALPHA	9.71E-03	4.02E-03	1.54E-03	5.76E-04	2.17E-04	1.61E-02
L17(N,ALPHA)	1.71E-01	7.62E-02	3.05E-02	1.18E-02	4.49E-03	2.96E-01
L17(N,GAMMA)	1.12E-06	2.55E-07	4.45E-08	8.49E-08	3.52E-08	1.21E-06
L17(N,DEUTERIUM)	5.20E-03	2.15E-03	8.22E-04	3.07E-04	1.14E-04	8.59E-03
L17 INELASTIC	-1.47E-02	-7.56E-03	-3.35E-03	-1.21E-03	-2.35E-04	-2.70E-02
F19 ELASTIC	-1.24E-02	-8.77E-03	-4.59E-03	-1.94E-03	2.39E-04	-2.79E-02
F19(N,2N)	-1.80E-03	-6.59E-04	-2.15E-04	-5.61E-05	-4.61E-06	-2.73E-03
F19(N,N')ALPHA	-1.01E-03	-7.35E-04	-2.56E-04	-7.14E-05	-4.75E-06	-2.08E-03
F19(N,N')PROTON	-5.75E-04	-2.42E-04	-8.67E-05	-2.60E-05	-4.75E-06	-9.34E-04
F19(N,GAMMA)	-1.69E-05	-2.40E-05	-2.13E-05	-1.30E-05	-6.01E-06	-8.11E-05
F19(N,PROTON)	-1.39E-03	-6.92E-04	-2.78E-04	-9.55E-05	-2.40E-05	-2.40E-03
F19(N,DEUTERIUM)	-8.26E-04	-3.82E-04	-1.46E-04	-4.73E-05	-1.08E-05	-1.41E-03
F19(N,TRITON)	-5.16E-04	-2.37E-04	-8.96E-05	-2.88E-05	-6.46E-06	-8.77E-04
F19(N,ALPHA)	-2.97E-03	-1.61E-03	-7.44E-04	-2.58E-04	-7.16E-05	-5.63E-03
F19 INELASTIC	-4.74E-02	-3.45E-02	-2.00E-02	-9.58E-03	-3.17E-03	-1.15E-01
P0 ELASTIC	-1.97E-01	-6.21E-02	-1.29E-02	1.71E-03	6.64E-03	-2.64E-01
P0(N,2N)	-2.87E-01	-1.17E-01	-4.06E-02	-1.22E-02	-2.39E-03	-4.59E-01
P0(N,3N)	-1.00E-02	-5.24E-03	-1.98E-03	-6.41E-04	-1.60E-04	-1.80E-02
P0(N,GAMMA)	-1.85E-04	-4.02E-04	-3.48E-04	-2.11E-04	-9.97E-05	-1.24E-03
P0 INELASTIC	-5.68E-02	-5.34E-02	-2.96E-02	-1.32E-02	-4.43E-03	-1.57E-01
P2 ELASTIC	0.0	0.0	0.0	0.0	0.0	-1.43E-02
FE(N,2N)	0.0	0.0	0.0	0.0	0.0	-1.02E-02
FE(N,N')ALPHA	0.0	0.0	0.0	0.0	0.0	-3.39E-04
FE(N,N')PROTON	0.0	0.0	0.0	0.0	0.0	-3.60E-03
FE(N,GAMMA)	0.0	0.0	0.0	0.0	0.0	-9.47E-04
FE(N,PROTON)	0.0	0.0	0.0	0.0	0.0	-1.58E-02
FE(N,DEUTERIUM)	0.0	0.0	0.0	0.0	0.0	-2.20E-03
FE(N,TRITON)	0.0	0.0	0.0	0.0	0.0	-5.13E-05
FE(N,HELIUM 3)	0.0	0.0	0.0	0.0	0.0	-3.75E-05
FE(N,ALPHA)	0.0	0.0	0.0	0.0	0.0	-4.84E-03
FE INELASTIC	0.0	0.0	0.0	0.0	0.0	-9.12E-02

TABLE 9

SENSITIVITY COEFFICIENTS OF F19 NEUTRON HEATING
TO L16, P0, FE, L17, F19 PARTIAL CROSS SECTIONS
NON FISSION INTEGRAL EXPERIMENT

	ZONE 3	ZONE 7	ZONE 11	ZONE 15	ZONE 19	SUM
L16 ELASTIC	-1.11E-03	-1.01E-03	-6.62E-04	-3.27E-04	-8.66E-05	-3.41E-03
L16(N,2N)ALPHA	-2.34E-04	-9.07E-05	-3.11E-05	-8.95E-06	-1.29E-06	-3.66E-04
L16(N,N')ALPHA	-1.53E-03	-6.84E-04	-2.60E-04	-8.19E-05	-1.40E-05	-2.57E-03
L16(N,GAMMA)	-1.11E-07	-8.92E-08	-5.61E-08	-2.94E-08	-1.10E-08	-2.96E-07
L16(N,PROTON)	-5.10E-05	-2.64E-05	-1.09E-05	-3.86E-06	-1.01E-06	-9.32E-05
L16(N,ALPHA)T	-1.96E-03	-2.16E-03	-1.77E-03	-1.01E-03	-4.19E-04	-7.55E-03
L16 INELASTIC	-1.54E-03	-6.91E-04	-2.62E-04	-8.27E-05	-1.41E-05	-2.59E-03
L17 ELASTIC	-1.74E-02	-1.21E-02	-7.37E-03	-3.39E-03	-6.81E-04	-4.10E-02
L17(N,2N)	-8.70E-04	-3.30E-04	-1.16E-04	-3.36E-05	-4.93E-06	-3.36E-03
L17(N,2N)ALPHA	-1.52E-03	-6.09E-04	-2.11E-04	-6.31E-05	-1.08E-05	-2.41E-03
L17(N,ALPHA)T	-1.34E-02	-5.68E-03	-2.07E-03	-6.24E-04	-9.89E-05	-3.10E-02
L17(N,GAMMA)	-1.22E-06	-9.84E-07	-6.27E-07	-3.34E-07	-1.26E-07	-3.29E-06
L17(N,DEUTERIUM)	-6.27E-04	-2.80E-04	-1.04E-04	-3.30E-05	-7.25E-06	-1.05E-03
L17 INELASTIC	-1.70E-02	-8.20E-03	-3.42E-03	-1.20E-03	-2.33E-04	-3.01E-02
F19 ELASTIC	9.80E-02	6.91E-02	4.20E-02	2.41E-02	1.33E-02	2.47E-01
F19(N,2N)	1.97E-03	8.86E-04	3.69E-04	1.56E-04	7.15E-05	3.45E-03
F19(N,N')ALPHA	2.15E-02	9.22E-03	3.62E-03	1.38E-03	5.25E-04	3.62E-02
F19(N,N')PROTON	3.91E-03	1.63E-03	6.28E-04	2.16E-04	8.90E-05	6.53E-03
F19(N,GAMMA)	-1.41E-05	-2.41E-05	-2.15E-05	-1.38E-05	-5.79E-06	-7.93E-05
F19(N,PROTON)	1.57E-02	6.90E-03	2.77E-03	1.07E-03	4.11E-04	2.69E-02
F19(N,DEUTERIUM)	8.01E-03	3.39E-03	1.32E-03	4.98E-04	1.88E-04	1.31E-02
F19(N,TRITON)	1.87E-03	1.62E-03	6.27E-04	2.36E-04	8.96E-05	6.41E-03
F19(N,ALPHA)	1.84E-02	1.00E-02	7.52E-03	3.01E-03	1.18E-03	6.81E-02
F19 INELASTIC	-4.94E-02	-3.47E-02	-2.00E-02	-9.50E-03	-3.21E-03	-1.17E-01
P0 ELASTIC	-2.11E-01	-6.09E-02	-1.35E-02	-1.00E-03	6.27E-03	-2.70E-01
P0(N,2N)	-3.10E-01	-1.32E-01	-4.65E-02	-1.44E-02	-3.17E-03	-5.06E-01
P0(N,N')	-1.06E-02	-4.08E-03	-1.01E-03	-5.81E-04	-1.41E-04	-1.80E-02
P0(N,GAMMA)	-1.96E-04	-4.52E-04	-3.99E-04	-2.53E-04	-1.21E-04	-1.42E-03
P0 INELASTIC	-5.41E-02	-4.09E-02	-2.11E-02	-9.02E-03	-2.89E-03	-1.28E-01
FE ELASTIC	0.0	0.0	0.0	0.0	0.0	-1.12E-02
FE(N,2N)	0.0	0.0	0.0	0.0	0.0	-1.26E-02
FE(N,N')ALPHA	0.0	0.0	0.0	0.0	0.0	-3.42E-04
FE(N,N')PROTON	0.0	0.0	0.0	0.0	0.0	-3.76E-03
FE(N,GAMMA)	0.0	0.0	0.0	0.0	0.0	-9.41E-04
FE(N,PROTON)	0.0	0.0	0.0	0.0	0.0	-1.55E-02
FE(N,DEUTERIUM)	0.0	0.0	0.0	0.0	0.0	-2.21E-03
FE(N,TRITON)	0.0	0.0	0.0	0.0	0.0	-5.18E-05
FE(N,HELIUM 3)	0.0	0.0	0.0	0.0	0.0	-3.79E-05
FE(N,ALPHA)	0.0	0.0	0.0	0.0	0.0	-4.81E-03
FE INELASTIC	0.0	0.0	0.0	0.0	0.0	-8.44E-02

TABLE 10

SENSITIVITY COEFFICIENTS OF FE(N,P) GAS PRODUCTION
TO L16, P0, FE, L17, F19 PARTIAL CROSS SECTIONS
NON FISSION INTEGRAL EXPERIMENT

	ZONE 3	ZONE 7	ZONE 11	ZONE 15	ZONE 19	SUM
L16 ELASTIC	-8.40E-04	-3.35E-04	-2.62E-04	-2.37E-04	-1.97E-04	-1.84E-03
L16(N,2N)ALPHA	-3.75E-04	-1.80E-04	-8.53E-05	-4.60E-05	-3.31E-05	-7.21E-04
L16(N,N')ALPHA	-1.79E-03	-8.98E-04	-4.56E-04	-2.70E-04	-2.01E-04	-1.62E-03
L16(N,GAMMA)	-6.12E-08	-3.19E-08	-1.55E-08	-8.59E-09	-6.22E-09	-1.25E-07
L16(N,PROTON)	-4.90E-05	-2.52E-05	-1.22E-05	-6.74E-06	-5.00E-06	-9.82E-05
L16(N,ALPHA)T	-1.99E-04	-1.13E-04	-5.95E-05	-3.45E-05	-2.56E-05	-4.12E-04
L16 INELASTIC	-1.01E-03	-9.07E-04	-4.61E-04	-2.71E-04	-2.05E-04	-3.65E-03
L17 ELASTIC	-1.13E-02	-4.90E-03	-3.60E-03	-3.23E-03	-2.66E-03	-2.58E-02
L17(N,2N)	-1.39E-03	-6.66E-04	-3.16E-04	-1.74E-04	-1.23E-04	-2.67E-03
L17(N,2N)ALPHA	-2.11E-03	-9.90E-04	-4.63E-04	-2.52E-04	-1.74E-04	-3.99E-03
L17(N,ALPHA)T	-1.64E-02	-8.09E-03	-4.06E-03	-2.38E-03	-1.77E-03	-3.27E-02
L17(N,GAMMA)	-7.20E-07	-3.62E-07	-1.76E-07	-9.73E-08	-7.03E-08	-1.43E-06
L17(N,DEUTERIUM)	-6.45E-04	-3.05E-04	-1.41E-04	-7.80E-05	-5.43E-05	-1.23E-03
L17 INELASTIC	-1.86E-02	-9.31E-03	-4.77E-03	-2.87E-03	-2.16E-03	-3.70E-02
F19 ELASTIC	-7.34E-03	-2.59E-03	-2.39E-03	-2.42E-03	-2.40E-03	-1.60E-02
F19(N,2N)	-3.31E-03	-1.54E-03	-7.19E-04	-3.90E-04	-2.68E-04	-6.23E-03
F19(N,N')ALPHA	-3.17E-03	-1.54E-03	-7.41E-04	-4.15E-04	-2.99E-04	-6.16E-03
F19(N,N')PROTON	-7.64E-04	-3.61E-04	-1.70E-04	-9.26E-05	-6.45E-05	-1.45E-03
F19(N,GAMMA)	-9.31E-07	-5.24E-07	-2.75E-07	-1.61E-07	-1.22E-07	-2.41E-06
F19(N,PROTON)	-1.40E-03	-7.09E-04	-3.43E-04	-1.89E-04	-1.40E-04	-2.78E-03
F19(N,DEUTERIUM)	-8.55E-04	-4.14E-04	-1.97E-04	-1.08E-04	-7.67E-05	-1.65E-03
F19(N,TRITON)	-5.35E-04	-2.58E-04	-1.22E-04	-6.69E-05	-4.74E-05	-1.03E-03
F19(N,ALPHA)	-2.52E-03	-1.33E-03	-6.55E-04	-3.63E-04	-2.75E-04	-5.15E-03
F19 INELASTIC	-4.16E-02	-2.07E-02	-1.04E-02	-6.11E-03	-4.53E-03	-8.14E-02
P0 ELASTIC	-2.93E-01	-6.40E-02	-2.91E-02	-2.27E-02	-2.00E-02	-4.10E-01
P0(N,2N)	-4.10E-01	-2.26E-01	-1.06E-01	-5.46E-02	-3.48E-02	-8.12E-01
P0(N,N')	-1.21E-02	-6.15E-03	-2.75E-03	-1.16E-03	-8.21E-04	-2.11E-02
P0(N,GAMMA)	-4.33E-05	-3.25E-05	-1.82E-05	-1.05E-05	-7.12E-06	-1.12E-04
P0 INELASTIC	-5.77E-02	-3.70E-02	-1.97E-02	-1.12E-02	-7.81E-03	-1.14E-01
FE ELASTIC	0.0	0.0	0.0	0.0	0.0	-1.70E-02
FE(N,2N)	0.0	0.0	0.0	0.0	0.0	-6.69E-02
FE(N,N')ALPHA	0.0	0.0	0.0	0.0	0.0	-5.19E-04
FE(N,N')PROTON	0.0	0.0	0.0	0.0	0.0	-5.99E-03
FE(N,GAMMA)	0.0	0.0	0.0	0.0	0.0	-5.94E-05
FE(N,PROTON)	0.0	0.0	0.0	0.0	0.0	-9.04E-01
FE(N,DEUTERIUM)	0.0	0.0	0.0	0.0	0.0	-2.96E-03
FE(N,TRITON)	0.0	0.0	0.0	0.0	0.0	-6.78E-05
FE(N,HELIUM 3)	0.0	0.0	0.0	0.0	0.0	-4.97E-05
FE(N,ALPHA)	0.0	0.0	0.0	0.0	0.0	-6.67E-03
FE INELASTIC	0.0	0.0	0.0	0.0	0.0	-1.23E-01

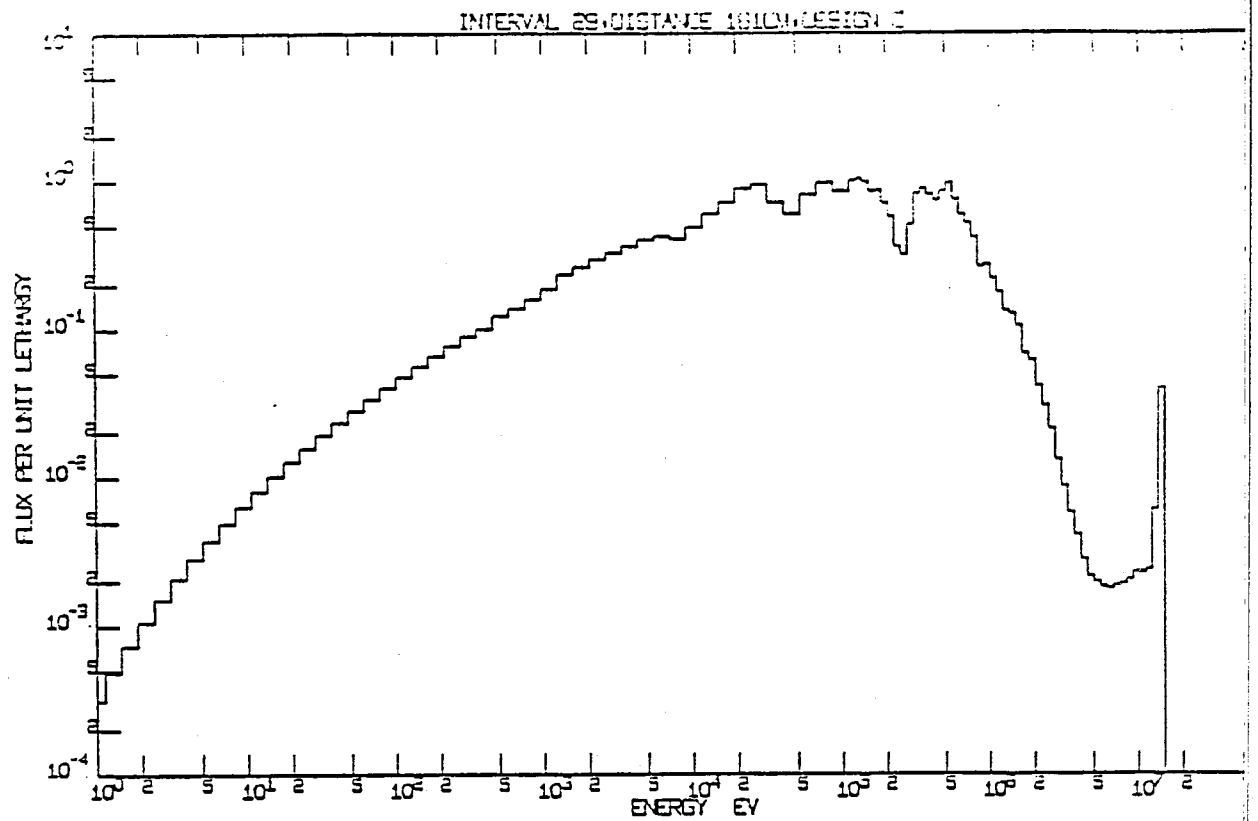


FIG. 2 scalar flux at a position
181 cm in design c), NFR.

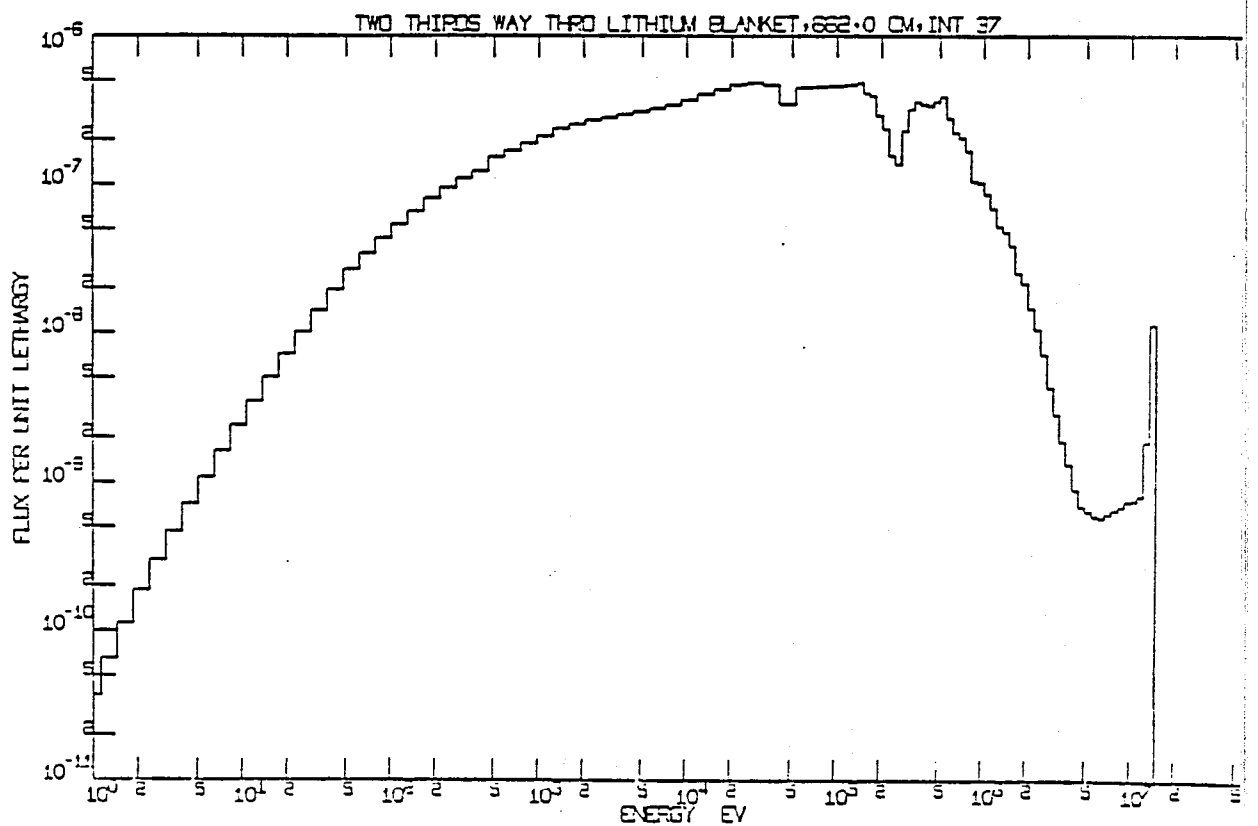
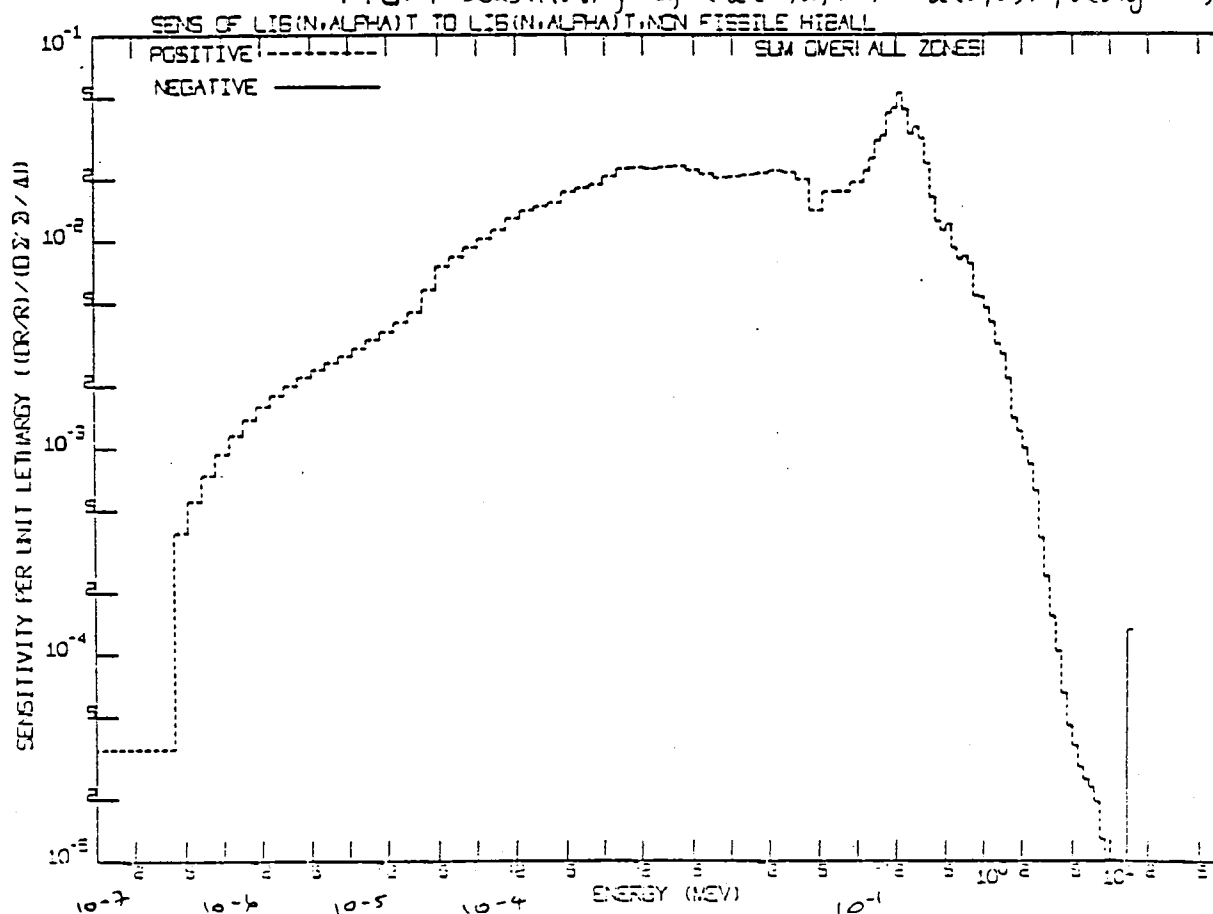
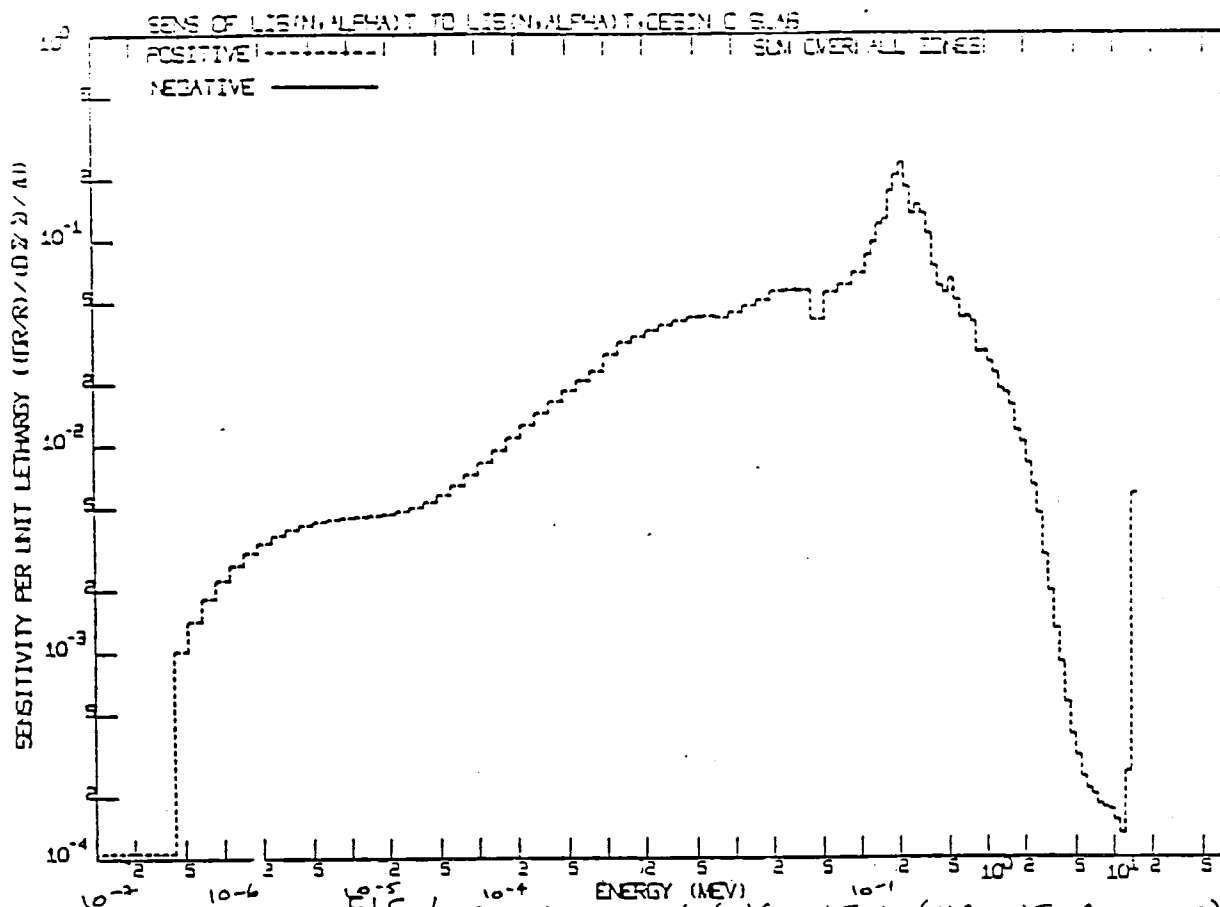


FIG. 1 scalar flux at a position
2/3 into Lithium blanket
in NFR.



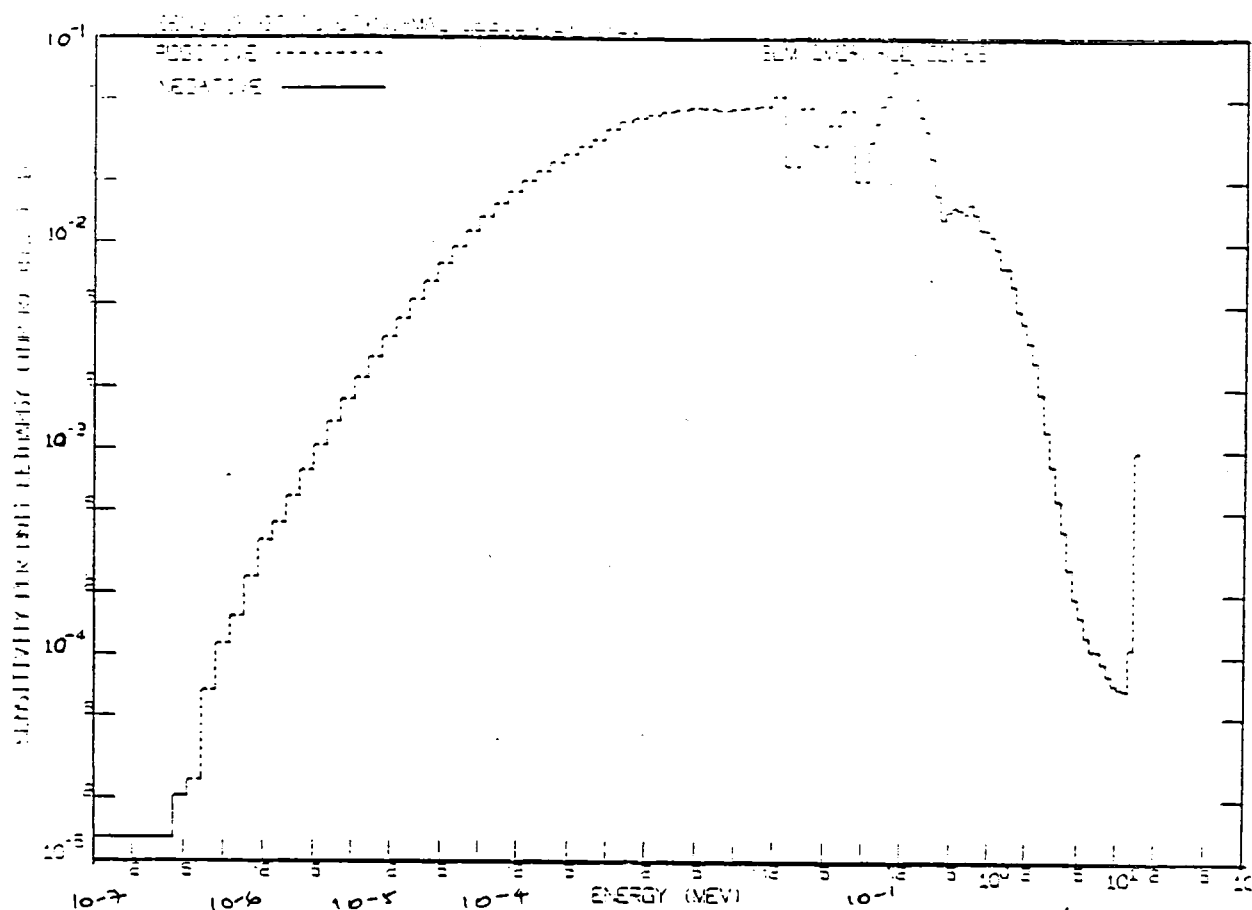


FIG. 5 sensitivity of $\phi_{Li}(u, x)T$ to $\phi_{Li}(u, x)T$ of an unchanged design e

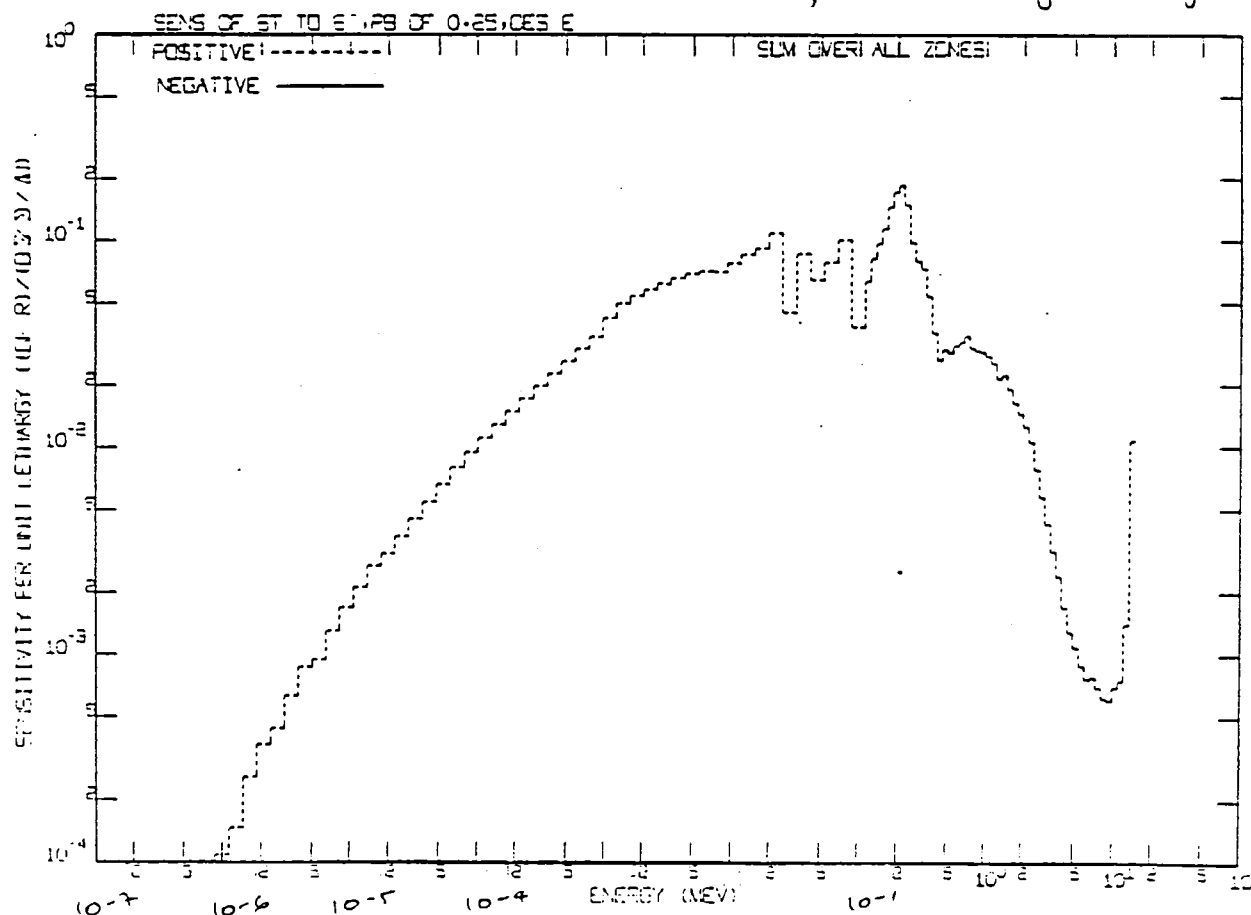


FIG. 6 sensitivity of $\phi_{Li}(u, x)T$ to $\phi_{Li}(u, x)$ of design e), with 75% of the Lead removed.

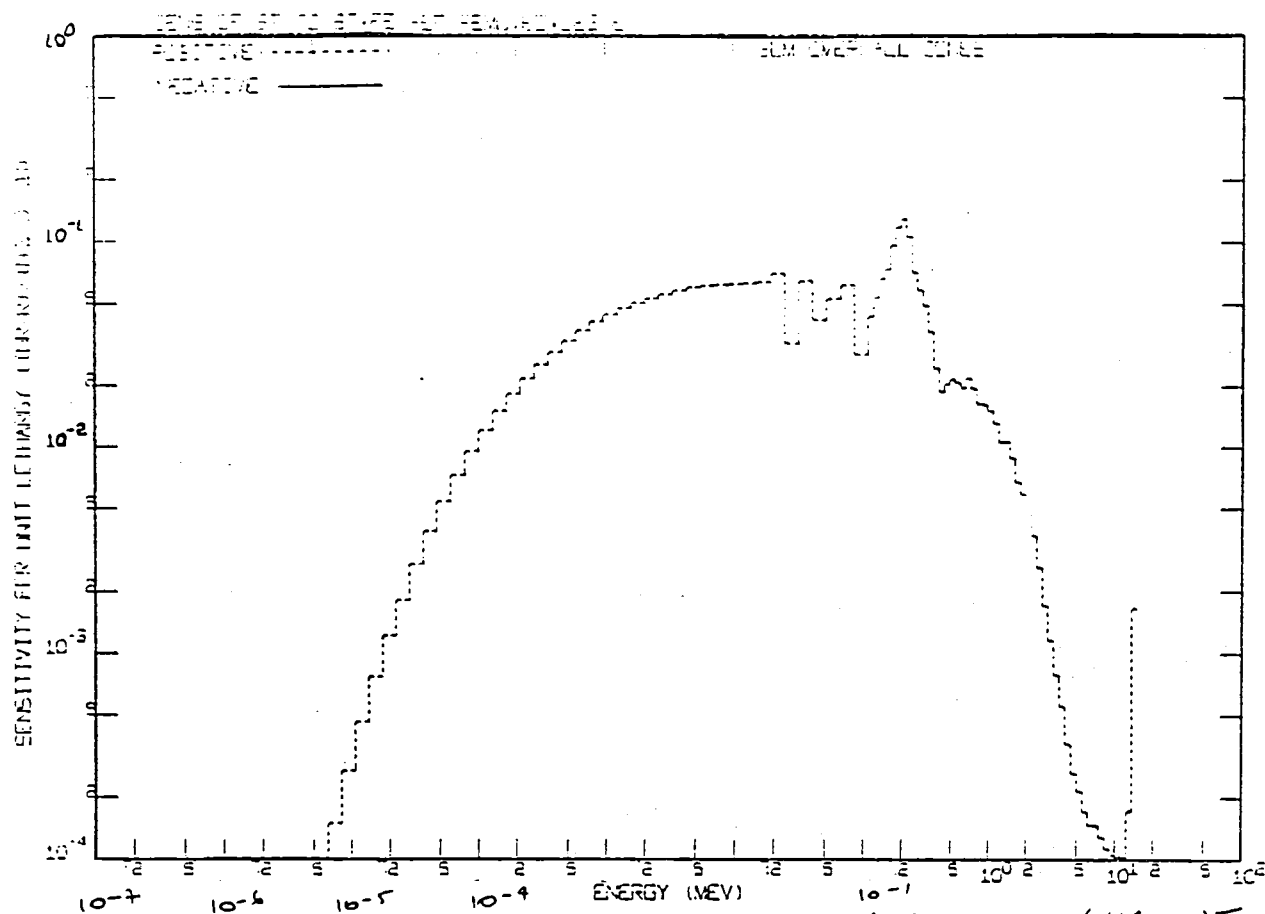


FIG. 7 sensitivity of ${}^6\text{Li}(\alpha, \alpha)\text{T}$ to ${}^6\text{Li}(\alpha, \alpha)\text{T}$ of design e), with iron reflector totally removed.

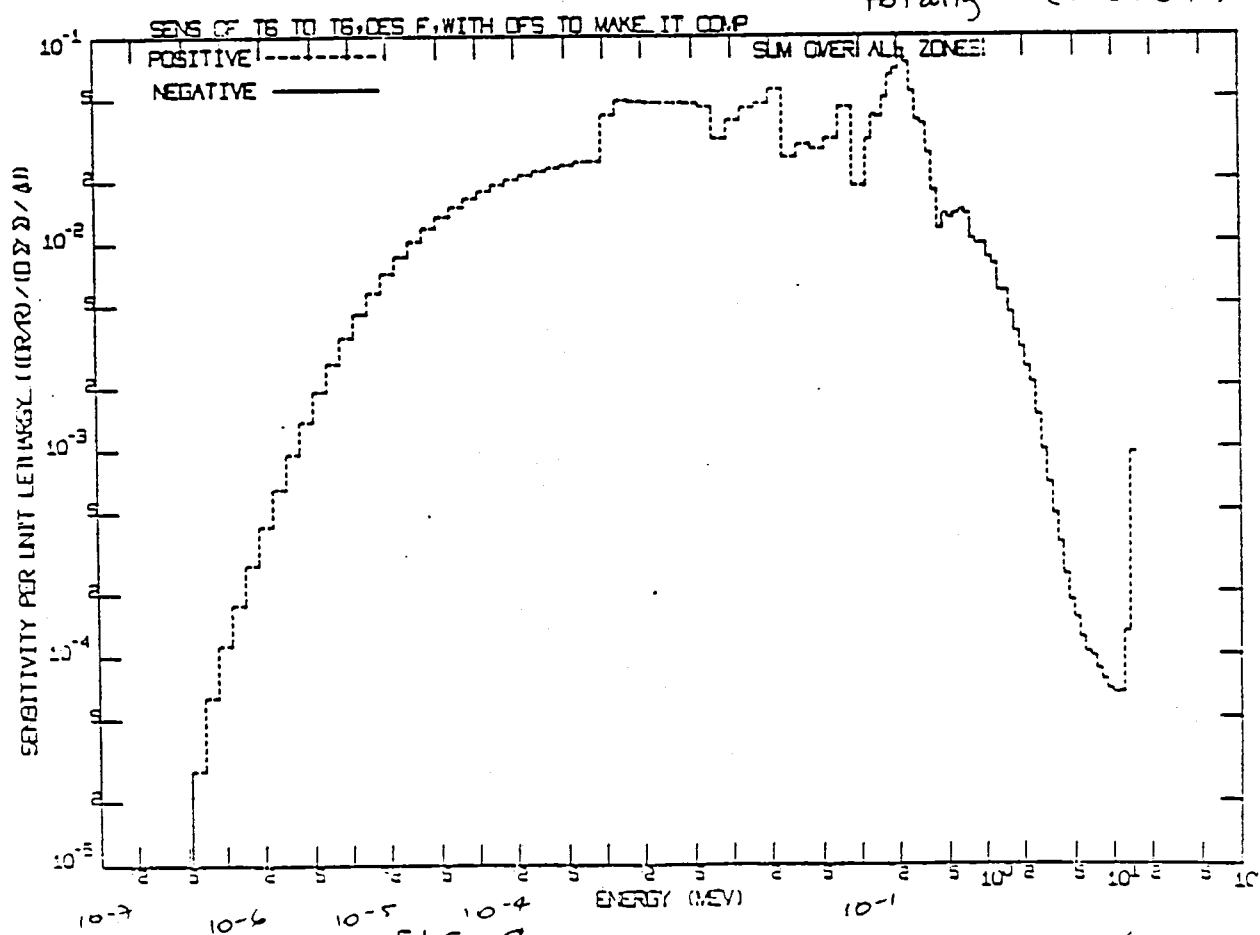
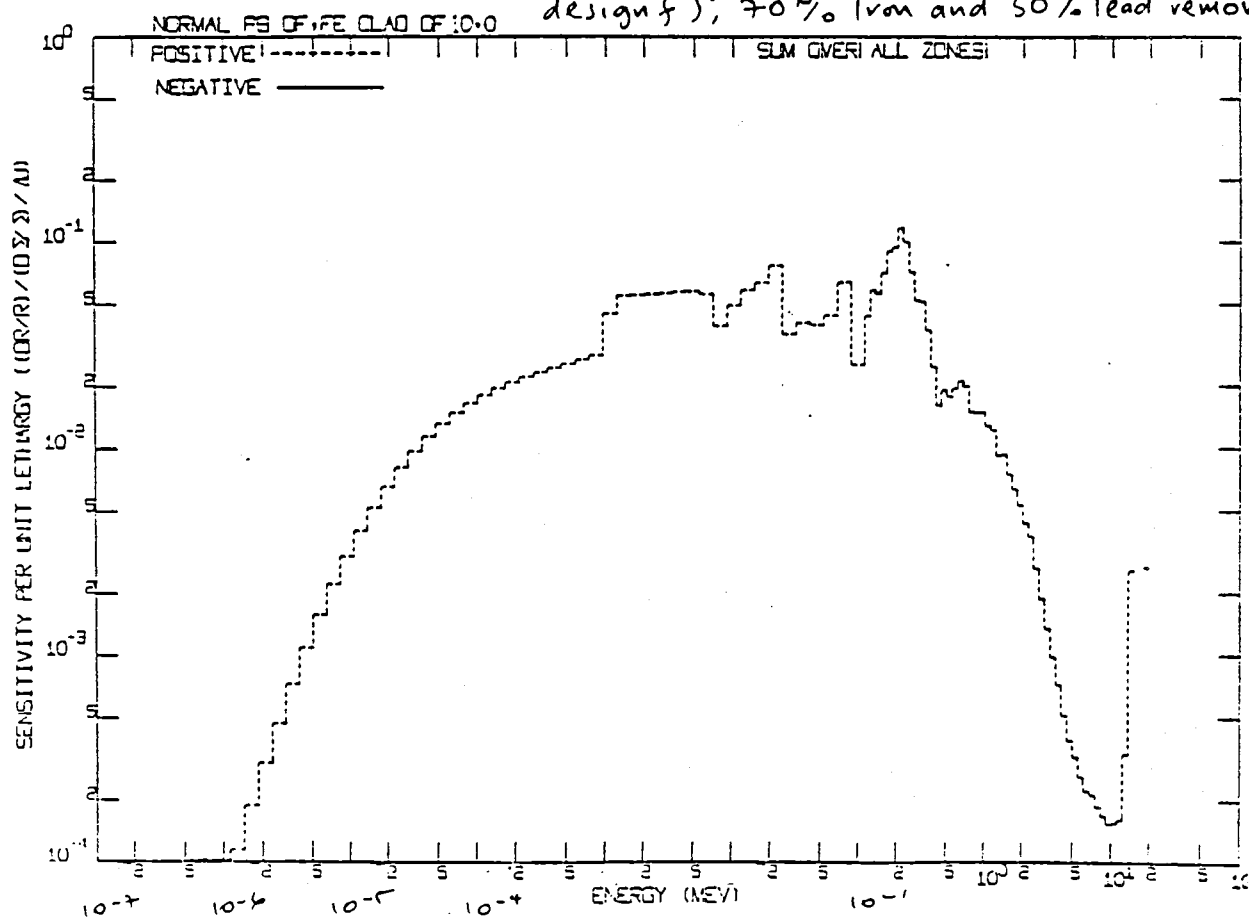
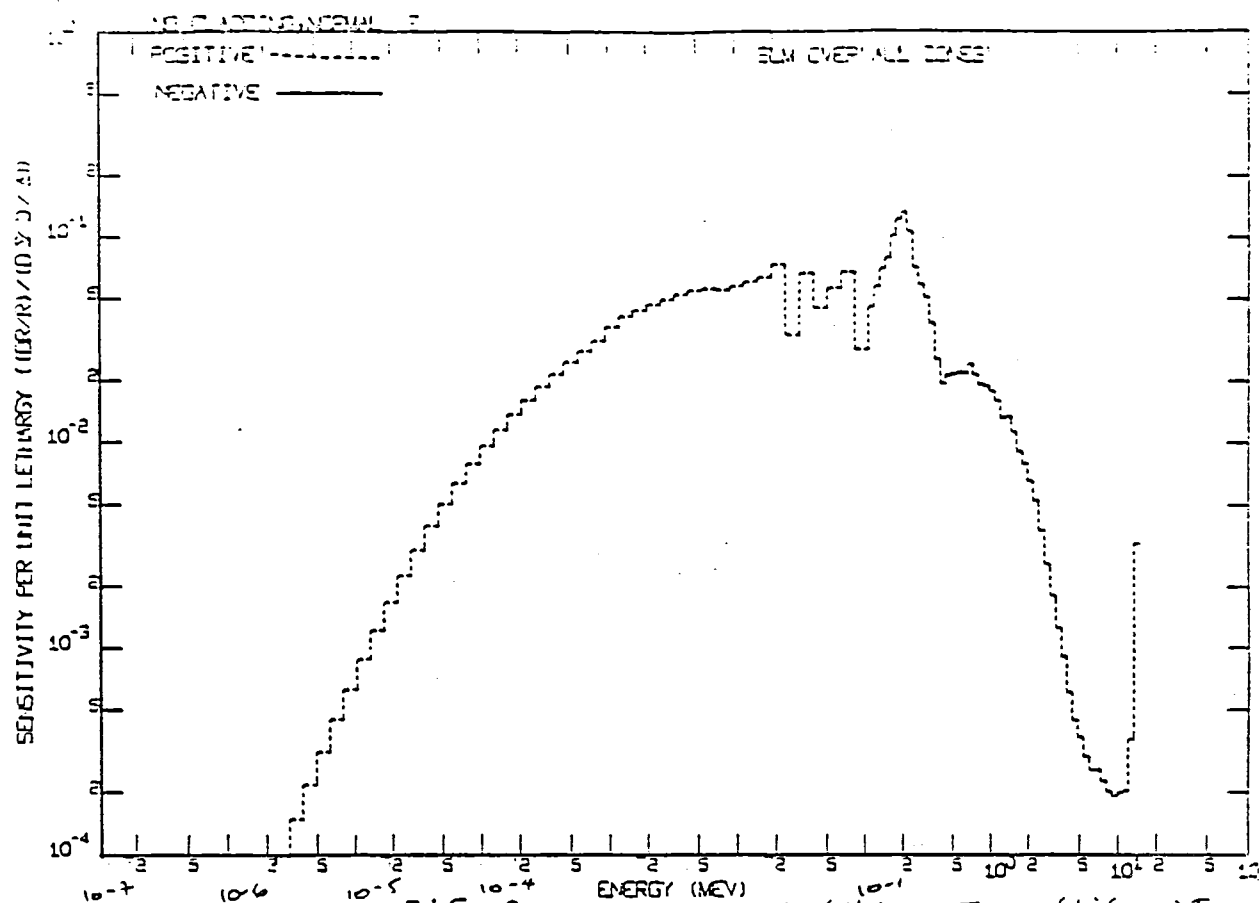


FIG. 8 sensitivity of ${}^6\text{Li}(\alpha, \alpha)\text{T}$ to ${}^6\text{Li}(\alpha, \alpha)\text{T}$ of design e), with 1cm thick iron cladding around each LiF zone.



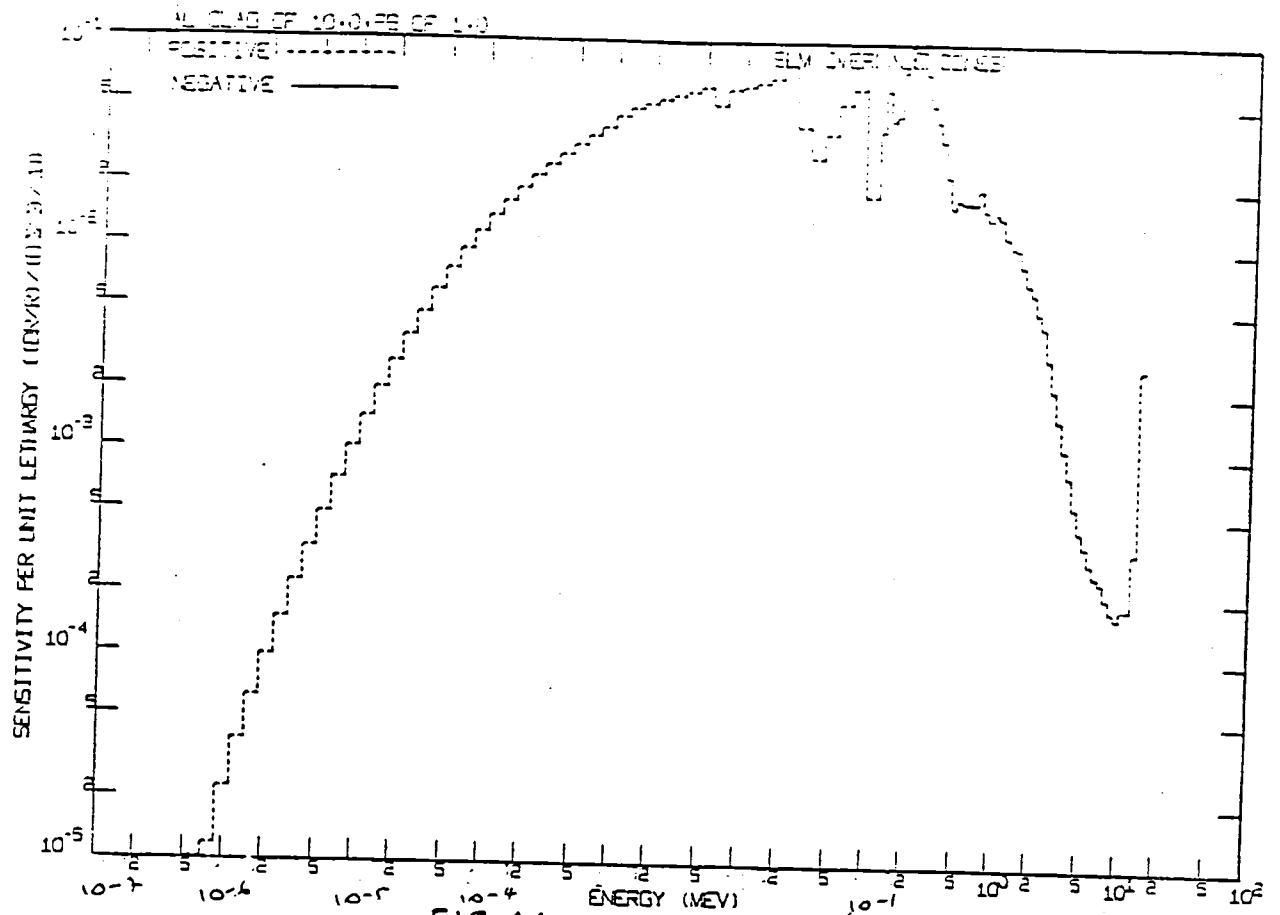


FIG. 11 Sensitivity of ${}^6\text{Li}(n, \alpha)\text{T}$ to ${}^6\text{Li}(n, \alpha)\text{T}$ design f), 1cm thick Aluminium cladding around each LIF zone.

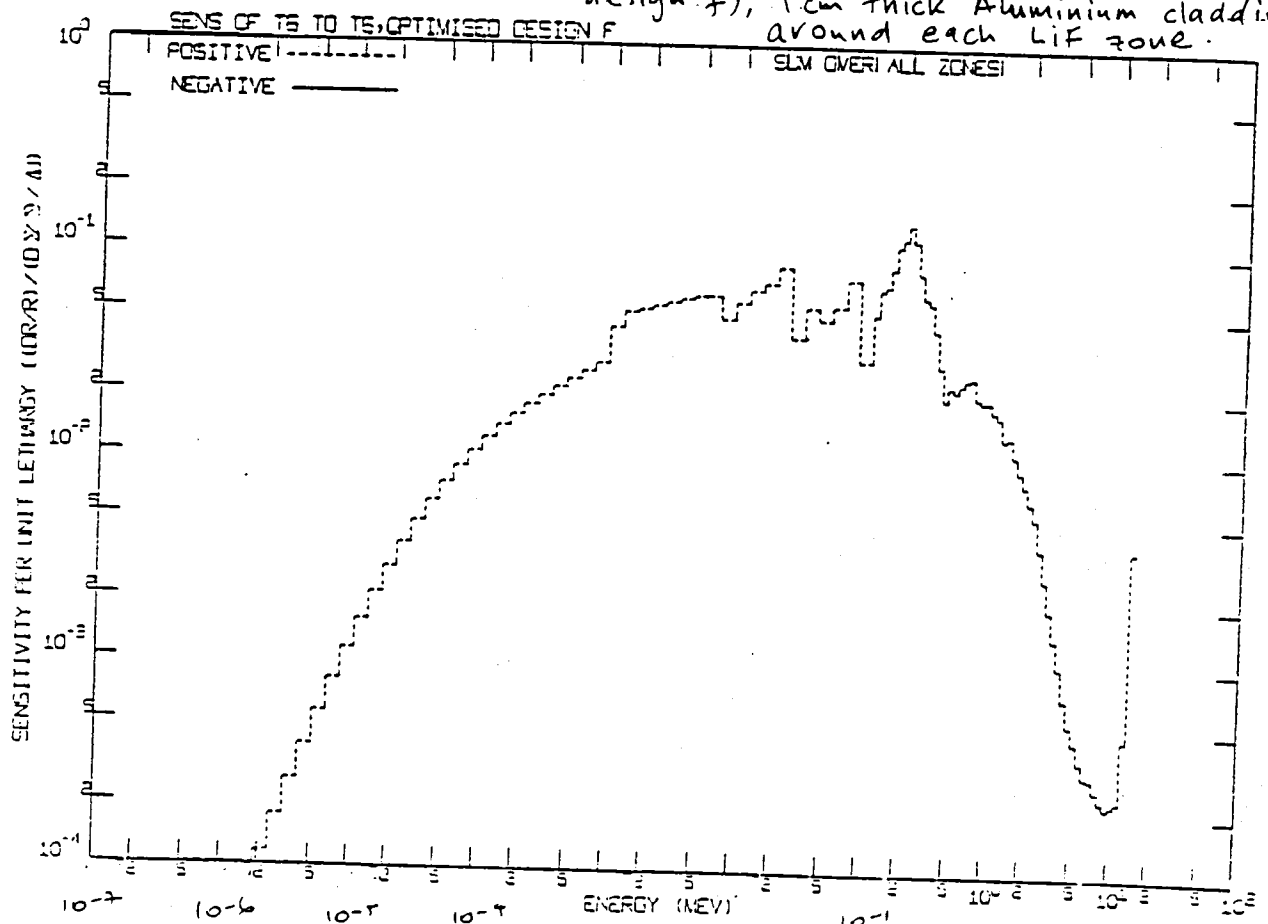


FIG. 12 Sensitivity of ${}^6\text{Li}(n, \alpha)\text{T}$ to ${}^6\text{Li}(n, \alpha)\text{T}$ Final optimised design f), NFIE.

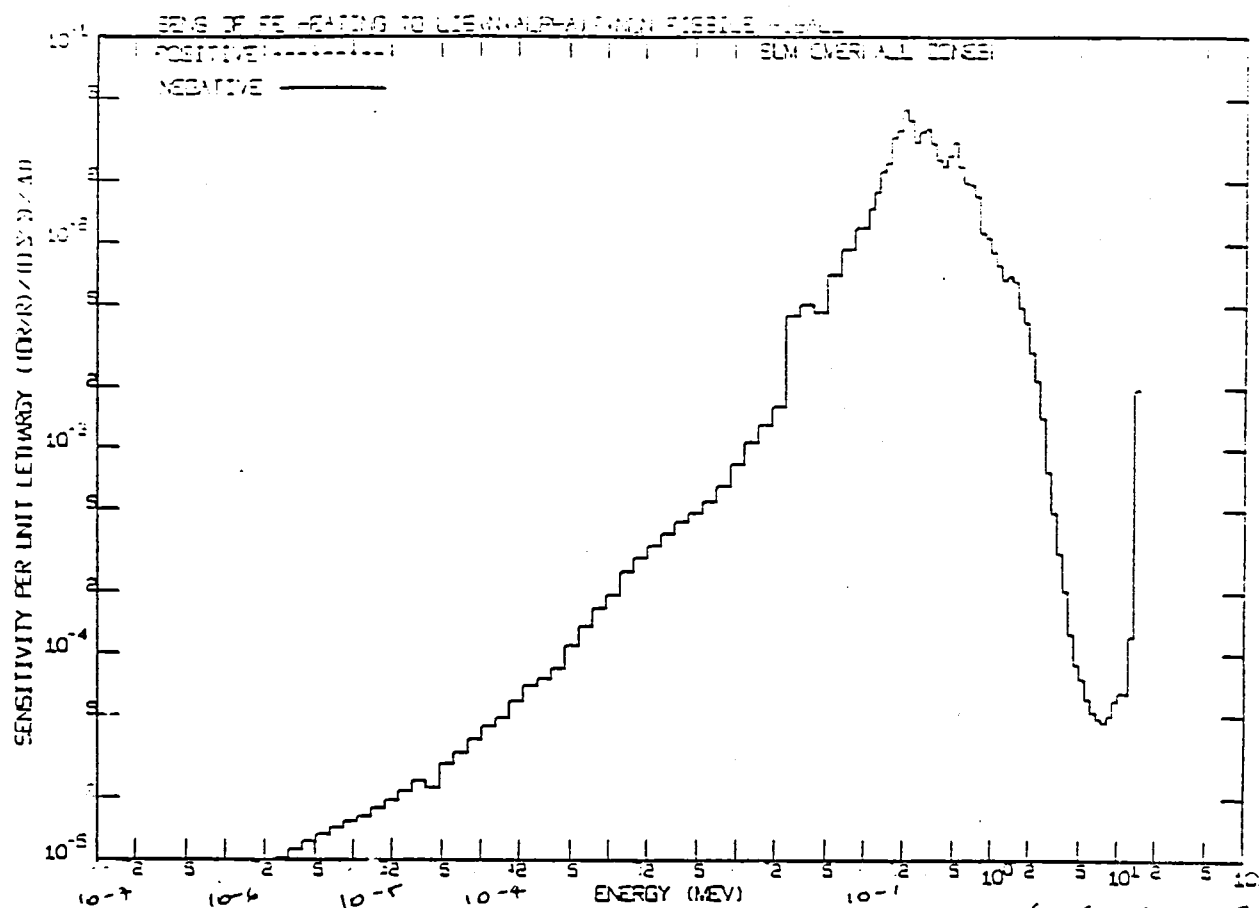


FIG. 13 sensitivity of Fe heating to $\phi L(N, \alpha)T$, NF

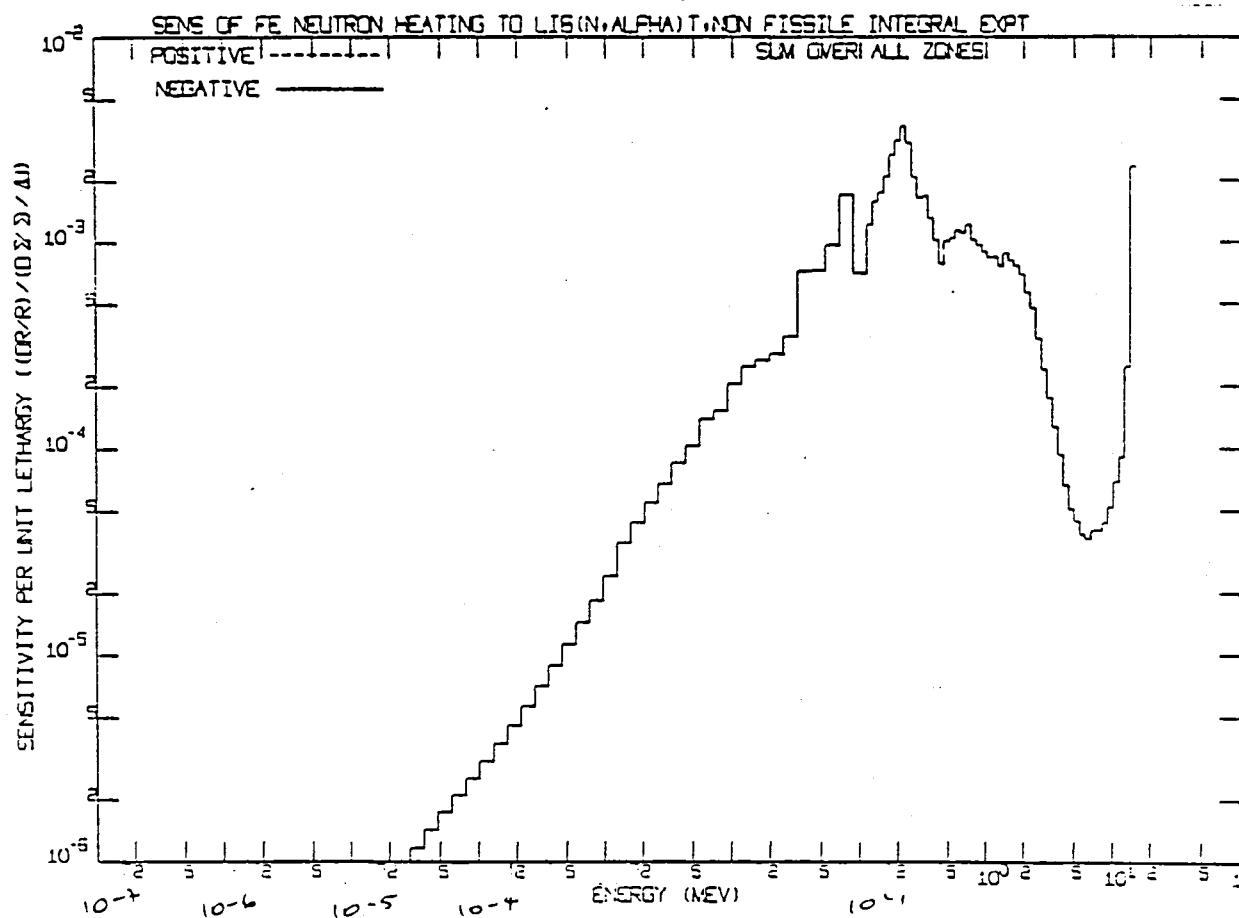


FIG. 14 sensitivity of Fe heating to $\phi L(N, \alpha)T$, in NFIE.

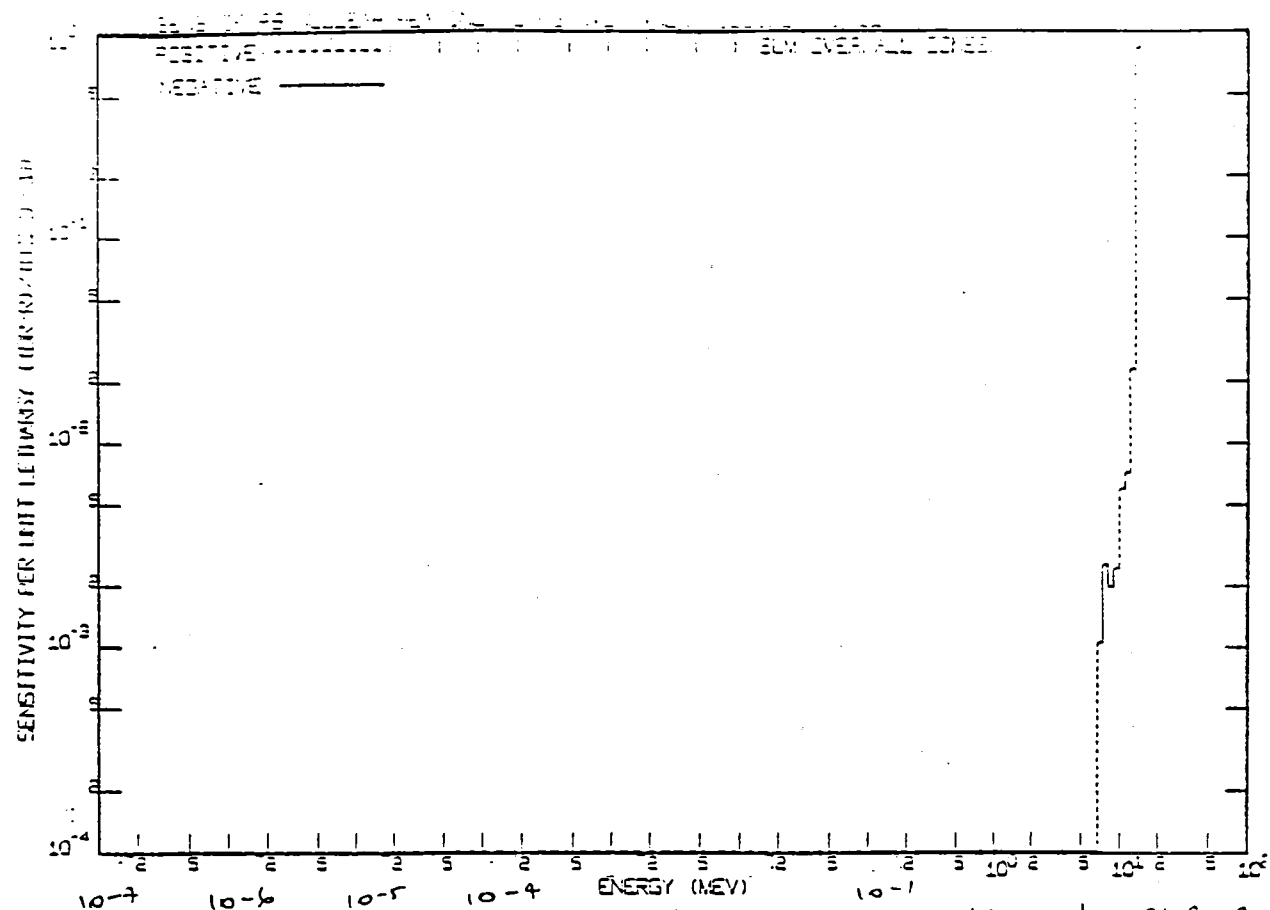


FIG. 15 Sensitivity of Pb heating to Pb(n, 2n) in NFR.

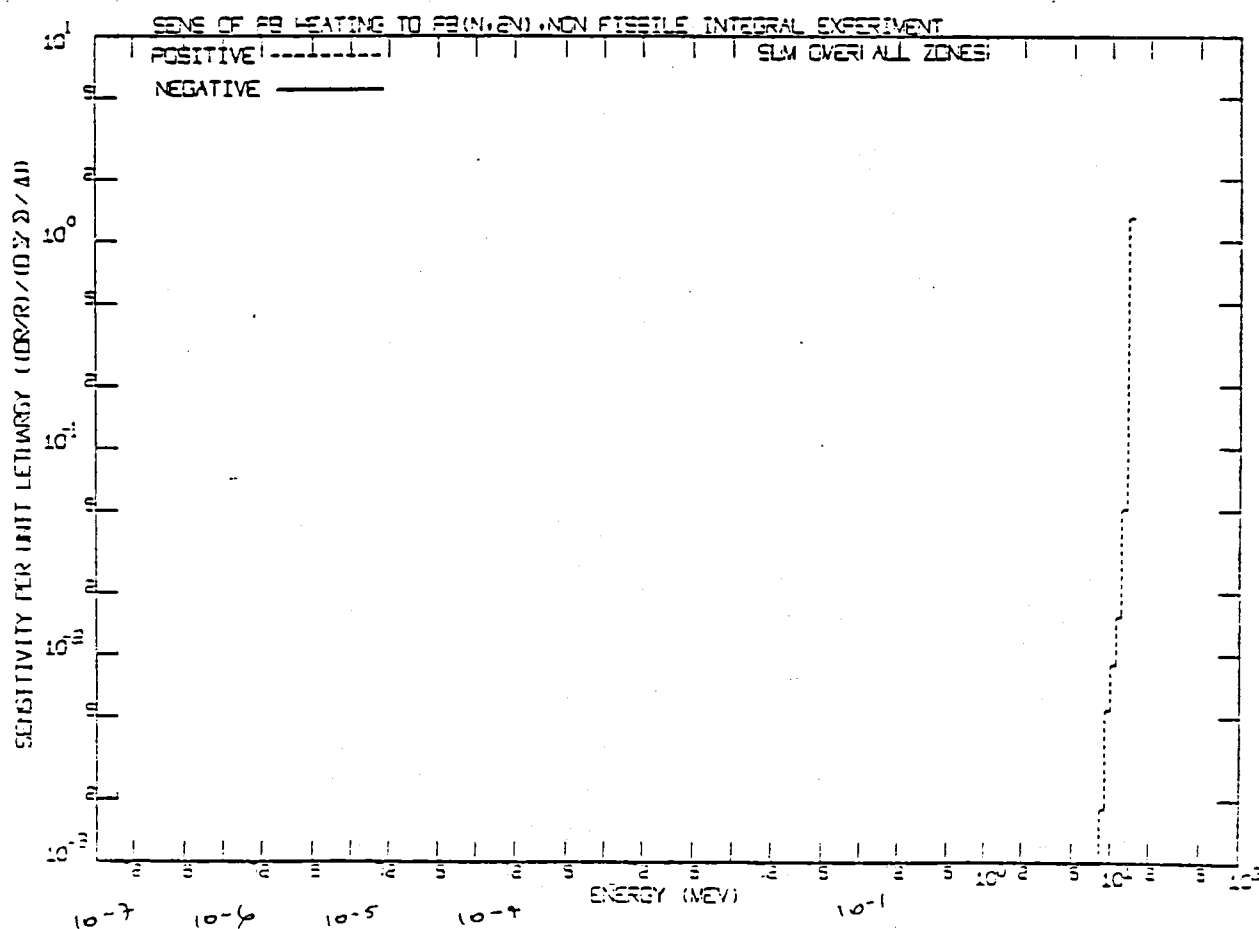


FIG. 16 Sensitivity of Pb heating to Pb(n, 2n), in NFIE.

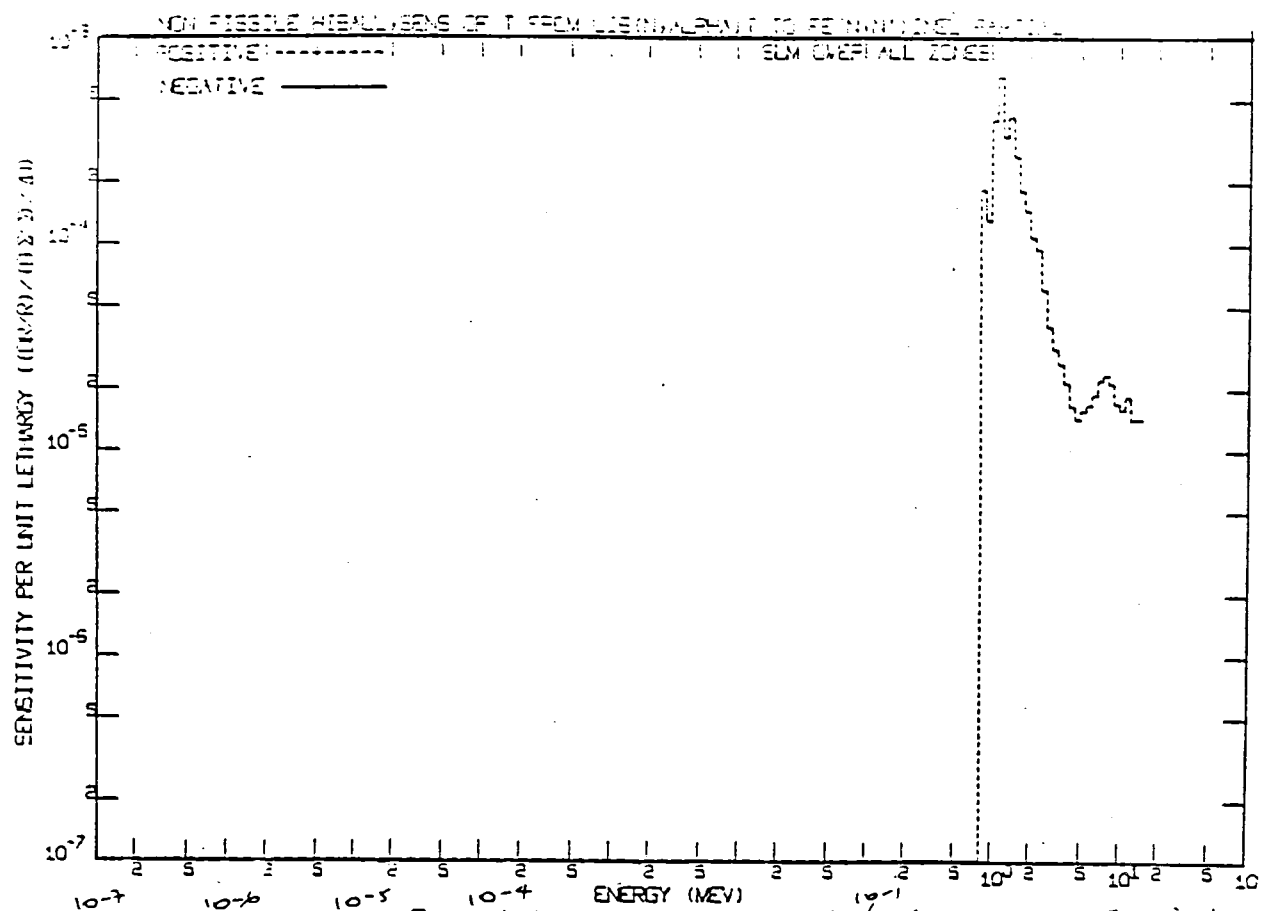


FIG. 17 sensitivity of $\rho_{\text{Li}}(n, \alpha)T$ to Fe inelas in NFR

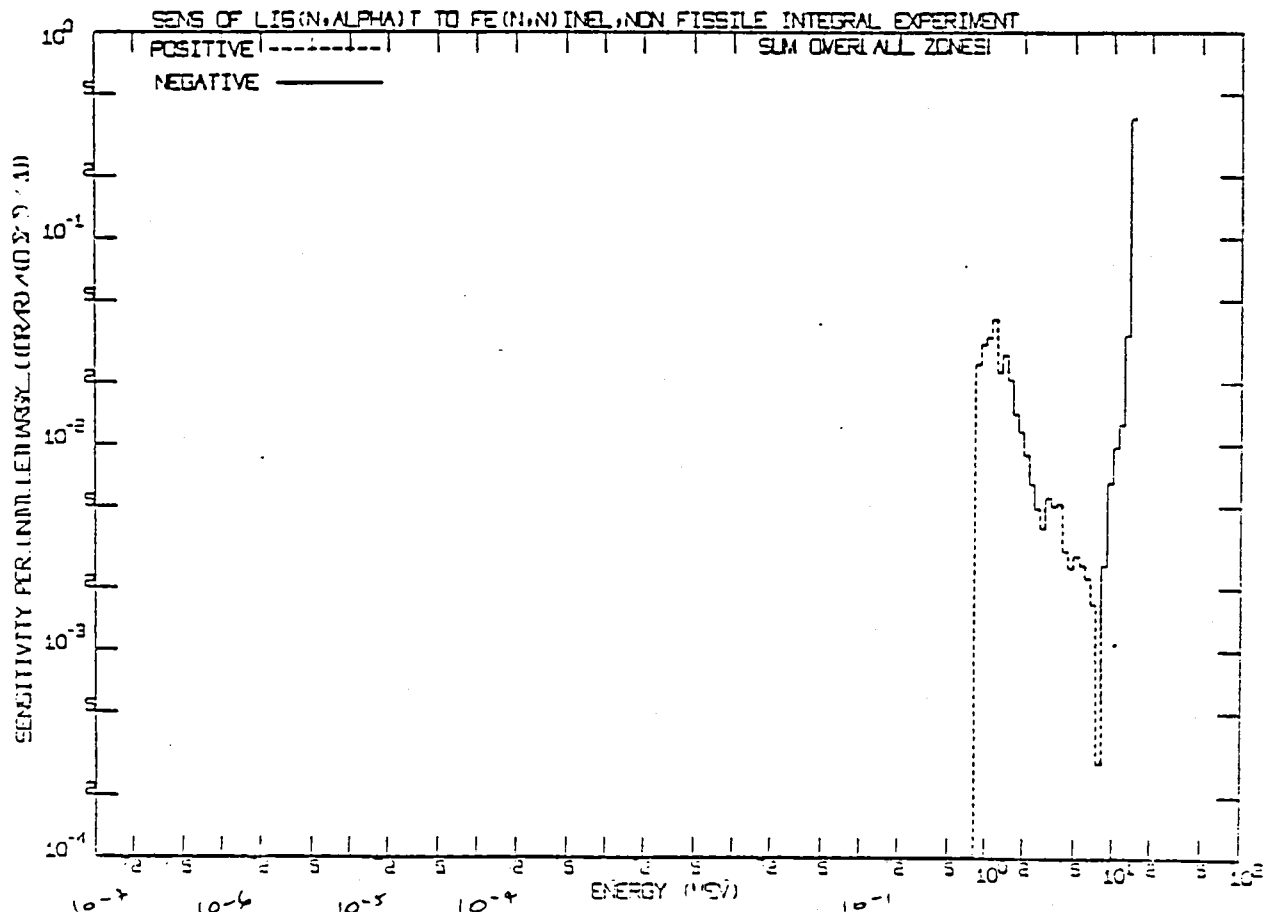


FIG. 18

Sensitivity of $\rho_{\text{Li}}(n, \alpha)T$
to Fe inelastic in NFIE.

ZONE	1	2	3	4	5
	SOURCE	VOIO	Pb, bLi, γ Li at 33% density	Fe	Pb, bLi, γ Li, Fe
	0.0	8.0	500	700	701
					741 cm.

DESIGN B)

SPHERE,

HOMOGENEOUS.

ZONE	1	2	3	4
	S	Pb, 6Li, 7Li	Fe	Pb, 6Li, 7Li, Fe
(SOURCE)		at 33% density		
0.0	1.0	2.0	2.0	2.2 cm

DESIGN (C)

SLAB, HOMOGENEOUS

DESIGN D)												
SLAB, HETEROGENEOUS												
1	2	3	4	5	6	7	8	9	10	11	12	
S	Pb	Li	Pb	Li	Pb	Li	Pb	Li	Pb	Li	Fe	
0.0	1.0	12.47	14.14	25.61	27.28	38.75	40.42	51.89	53.56	65.03	66.7	98.04 cm

	1	2	3	4	5	6	7	8	9	10	11	12
S	Pb	Pb	LiF	Pb	LiF	Pb	LiF	Pb	LiF	Pb	LiF	Fe
O.D	1.0	12.47	13.73	14.99	16.25	23.72	28.98	40.45	41.71	53.18	54.44	85.78 cm

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22
11 S	Pb	Fe	LiF	Fe	Pb	Fe	LiF	Fe	Pb	Fe	LiF	Fe	Pb	Fe	LiF	Fe	Pb	Fe	LiF	Fe	Pb	Fe
0-0 10	6.9	7.7	8.7	9.2	15.1	15.6	16.9	17.4	23.3	23.8	25.1	25.6	31.5	32	33.3	33.8	37.7	40.2	41.5	42	52	6m

CHAPTER 8

RESULTS AND DESIGN OF FISSILE INTEGRAL EXPERIMENT

8.1 INTRODUCTION

This chapter details the design of the fissile integral experiment (FR). It was analysed and constructed in the same manner as that presented in chapter 7, for the non fissile integral experiment (NFIE). Because the non fissile assembly was designed first, the ideas and lessons learnt in its analysis have been used directly in designing the fissile experiment. For this reason, and because these methods and ideas are given in such depth in chapter 7, much less detail has been presented here. The reading of chapter 7, before chapter 8, is a necessary prerequisite to understanding this chapter. The four design modifications, needed to change the cylindrical reactor into the fissile assembly in slab geometry, have been neutronically analysed and are presented. The optimised experiment is discussed and justified. Finally, as in the non fissile case, some sensitivity profiles, not used in the integral construction, are compared between the reactor and assembly.

8.2 OVERVIEW

The design of the fissile integral experiment is quite similar to that of the non fissile case, except that uranium now plays the role of the multiplying medium and the change in reactor geometry is from a cylinder to a slab rather than that of a sphere to a slab. Different sensitivity profiles have been used for tuning the fissile reactor to the assembly. Since the

neutronics is dominated by the uranium, the fission reaction rates $^{238}\text{U}(n,\text{fission})$ and $^{235}\text{U}(n,\text{fission})$ were chosen as the profiles for the neutronic tuning. Both sensitivity profiles were dominated by the direct term and fissile gain terms. Additionally, the sensitivity of $\text{Li6}(n,\alpha)\text{T}$ to $\text{Li6}(n,\alpha)\text{T}$ was used, as in the non fissile case, because it gives a measure of the system scalar flux due to its direct term dominance. This profile was included because the cylindrical reactor contains 25% enriched lithium in the Li6 isotope, whilst the integral experiment contains lithium fluoride in its natural state.

The justification of the fissile assembly, in the neutronics sense, is given in somewhat more detail than in the non fissile case. This is due to the fact that the cross sections of uranium are poorly known at high energies (3-14 MeV) since there is no great demand for reactor analysis at these energies. The use of uranium, and the fissile nuclides in general, in the role as energy multipliers for fusion reactors has only recieved serious attention in the last 10 or so years. Basically the uranium serves as an additional energy multiplication medium and correspondingly reduces the fusion gain needed on the DT pellet for the same energy output. The cross sections of uranium at energies below about 3 MeV, roughly corresponding to the highest neutron energies encountered in a fission reactor, are very well known. Since fusion reactors act at considerably higher energies it is very important that (in this case at least) the uranium cross sections be accurately known. Therefore the main purpose of the fissile integral experiment is to improve the accuracy of these cross sections. Similarly to the non fissile design, the discussion in this chapter is via sensitivities and the important radiation processes occuring in the assemblies.

A summary of the performance parameters for the five fissile assemblies (including the cylindrical reactor), is given in table 1. The following four integral designs, labelled design b), c), d) and e) contain the successive modifications needed to obtain the final fissile integral experiment. Design a) is the fissile reactor and has been discussed in chapter 6. As in the non fissile case, each separate design has been analysed neutronically and sensitivity profiles used to determine particle behaviour. In the nomenclature used here, FR and FIE mean the fissile reactor and the fissile integral experiment, respectively. The fissile designs are shown diagrammatically in figure 15.

8.3 DESIGN OF FISSILE INTEGRAL EXPERIMENT

Design b)

Those elements which will not be used to build the FIE with, are removed from the fissile reactor to produce design b). This leaves a cylindrical geometry containing uranium, iron and lithium as the reactor materials. The number densities of these materials remain unchanged and the zones containing the elements are homogeneously mixed. All other parameters, such as boundary conditions and interval distances remain unchanged. Design b) did not converge satisfactorily in the ANISN iterative scheme, even up to 100 outer iterations, showing the importance of the discarded elements in producing damping and stability via extra absorption in the reactor. Since a properly converged forward run was not obtained, the reactor parameters for this design have been obtained from the adjoint solution mode. The reason the adjoint mode converges and the forward mode does not, is not because the system is near criticality since K_{eff} is only about

0.26, but because of the difference in energy dependence of the fission spectra used for the forward and adjoint solutions, and the high energy nature of the fixed source. In the forward run, the number of outer iterations is dictated by the quantity $\nu(\epsilon) \xi_f(\epsilon)$, which increases fairly sharply at lower energies whilst the adjoint uses the quantity $\chi(\epsilon)$ which becomes very small at either end of the energy range. ANISN has the ability to converge very quickly in a system that contains only fissile material with no external source and the fluxes would normally settle down very quickly with the outer iteration sweeps. However, the high energy source is causing a great deal of fast fission in ^{238}U , the dominant isotope. One way of overcoming this convergence problem is to add an additional low energy source to act as a damping factor. Running the system with the low energy source only, and then the high energy source together with the low energy source, we can subtract the fluxes and responses of the former from the latter to yield the results for the high energy source. This solution comes about because of the linear nature of the Boltzmann equation.

To obtain information on the importance of the discarded elements, a sensitivity analysis was made on them using the forward and adjoint fluxes from design a). It was found to be the oxygen content which caused nearly all the spectrum softening. In the following discussions the sensitivities are given for a 1% increase in the cross section, whereas in going from design a) to design b) the materials have been totally removed. This cross section increase was preserved because the sensitivity profiles are given in this form. Table 2 gives the total sensitivity coefficients, as a percentage, for a 1% decrease in the specified oxygen partial. Figure 1 gives the

sensitivity of $^{238}\text{U}(\text{n},\text{fission})$ to ^{16}O inelastic. Although not easily seen in this graph, the highest energy group is positive with all the other groups being negative down to its threshold energy centred at 6.0-6.7 MeV. However, these sensitivities are very small, typically 10^{-5} . The positive sensitivity indicates that some of the neutrons being downscattered from this energy appear at energies above the $^{238}\text{U}(\text{n},\text{f})$ threshold of 2.8 MeV, thus being able to contribute to the fission reaction rate. For inelastic scattering events occurring below the highest group, most neutrons appear at energies below the $^{238}\text{U}(\text{n},\text{f})$ threshold. Figure 2 gives the sensitivity of $^{235}\text{U}(\text{n},\text{fission})$ to ^{16}O elastic. The deep valley at 0.2 MeV is the 285 KeV $\text{Li6}(\text{n},\alpha)\text{T}$ absorption resonance. Changes in partial cross sections at this energy have small effects because of the reduction in flux due to this resonance. The peak at about 0.5 MeV is an absorption peak in the ^{16}O elastic partial and indicates the large amount of flux at these energies. The negative sensitivities, seen at lower energies, indicate energies at which the elastic cross section tends to feed neutrons into energy regions where the non fissile absorption resonances (mostly in ^{238}U) are located. The sensitivity of $\text{Li6}(\text{n},\alpha)\text{T}$ to the ^{16}O elastic partial (not shown graphically) is positive from 14.1 MeV to about 2.5 MeV and thereafter negative to thermal energies, with a few isolated positive sensitivity bands. The change in sign occurs at the $^{238}\text{U}(\text{n},\text{f})$ threshold and indicates the dominance of the ^{238}U isotope. Neutrons downscattered at energies above 2.8 MeV via the ^{16}O elastic cross section have a chance that they will appear above the $^{238}\text{U}(\text{n},\text{f})$ threshold, thus causing fission and ultimately contributing to the $\text{Li6}(\text{n},\alpha)\text{T}$ reaction. The wholly negative region between 2.5 MeV and 20 KeV indicates the

energy band where most of the flux resides and where the $\text{Li6}(n,\alpha)\text{T}$ reaction rate is at its highest, particularly at the resonance. Increasing the elastic partial removes the flux from these energies - the sensitivity profile is more strongly negative at the 285 KeV $\text{Li6}(n,\alpha)\text{T}$ resonance. The positive sensitivities indicate regions where the enhanced downscattering is taking neutrons past absorption resonances in materials other than Li6. The highest positive sensitivity at these energies occurs at the $\text{Fe}(n,\gamma)$ 1 KeV absorption resonance.

Design c)

The second design, design c), is a slab geometry version of design b) with physically separate zones of uranium, lithium and iron. These zones are an attempt to simulate the previous homogeneous design. As with the non fissile case, the number of spatial intervals was increased from 60 to 100 and the quadrature set was increased from S_4 to S_8 , and the change in boundary conditions brought about the usual large change in the integral parameters. Boundary conditions of zero reflection have been used. The lithium remains in its enriched form. In going from design b) to design c) the total thickness of material was kept constant, as it must be to keep the same radiation environment. The natural uranium zone was placed nearest the anisotropic source, with 12 zones of enriched lithium and natural iron in an attempt to recreate the radiation environment of the original cylindrical fissile reactor. A 20.8 cm thick block of natural Fe was placed at the back of the design to play the role of a reflector.

In going from design b) to design c), the $\text{Li6}(n,\alpha)\text{T}$ response dropped by 48%, from 1.51 to 0.789 whilst the $\text{Li7}(n,n'\alpha)\text{T}$ reaction increased by 56%, from 0.085 to 0.133. Both fission rates dropped, the $^{238}\text{U}(n,f)$ by 19% from 0.392 to 0.318 and 72% for the $^{235}\text{U}(n,f)$ rate from $4.83\text{E}-2$ to $1.32\text{E}-2$. The large drop for the $^{235}\text{U}(n,f)$ fission rate indicates just how important the reflective boundary conditions were in producing this response. All the reaction rate drops are indicative of the increased neutron leakage. However, the Li7 and Fe responses show an increase. The Li7 has increased because the competitive softening reactions of oxygen are now absent and the Fe has increased because of the reduction of the competitive reactions and because more of it is physically closer to the source.

Design d)

As in the non fissile case, the final modification to be made before optimisation is to convert the lithium to lithium fluoride. The lithium used above is 25% enriched in the Li6 isotope though the lithium fluoride available to be used in an eventual experiment is in its natural unenriched state. This is the reason why the sensitivity of $\text{Li6}(n,\alpha)\text{T}$ to $\text{Li6}(n,\alpha)\text{T}$ was not used as one of the main profiles for designing the fissile experiment. It was decided to use natural lithium fluoride because of its availability and cheapness. In a later section, the sensitivity profiles for the fissile assembly are also derived with enriched lithium fluoride. In converting from the enriched lithium of design c) to the natural lithium fluoride of design d), the total amount of lithium (sum of both isotopes) was kept constant. Thus the isotopic concentrations are changed

from 25% Li6, 75% Li7 to 7% Li6 and 93% Li7. This is in contrast to keeping the total amount of either the Li7 or Li6 isotope constant so that a neutron travelling through the assembly sees the same mean free path thickness. However, either of the above criteria would be acceptable since the only difference between them is the final thickness of lithium fluoride that results. The amount of natural lithium fluoride can so easily be adjusted at a later date depending on the closeness or otherwise of the relevant sensitivity profiles. With the above in mind, the 50.2 cm of 25% enriched Li6 is converted to the same thickness of natural lithium. This 50.2 cm of natural lithium is next converted to 37.9 cm of natural lithium fluoride. This is the amount of lithium fluoride that is subsequently used in the optimisation of the integral experiment. The effect of the fluorine can be seen clearly in table 1.

The change in the reaction rates is also due to the change of the isotopic concentration of the lithium. The lithium response, for heating and breeding, decreased. Both fissile responses increased, but not by very much. This is probably due to the enhanced reflection of neutrons back towards the source with the fluorine present. In addition, for the $^{235}\text{U}(n,f)$ response, the amount of competition for neutrons from the $\text{Li6}(n,\alpha)\text{T}$ reaction is much reduced. The K_{eff} quantity, closely related to the fissile responses, also increased slightly. The sensitivity profiles for the sensitivity of $^{238}\text{U}(n,f)$ to $^{238}\text{U}(n,f)$, $^{235}\text{U}(n,f)$ to $^{235}\text{U}(n,f)$, and $\text{Li6}(n,\alpha)\text{T}$ to $\text{Li6}(n,\alpha)\text{T}$ are shown in figures 3, 4 and 5. No further system changes have been made to this design, and the optimisation is now presented.

Design e)

As for the non fissile case the optimisation of design d), to produce the working fissile assembly design e), is subject to four main constraints.

- 1) keeping the total weight at about 3 metric tonnes.
- 2) making the y,z dimensions as large as possible.
- 3) trying to keep the overall sensitivity profiles of the assembly as close as possible to that of the reactor.
- 4) attempting to decrease the amount of flux at lower energies.

The integral experiment, in its present form, before any optimisation, has the following dimensions ; a natural uranium zone of thickness 4.85cm has been placed directly nearest the source and a 20.8cm thick block of Fe is located at the back end of the assembly to act as a reflector. In between these two lie alternating zones of natural LiF and Fe. There are 6 zones of LiF each of thickness 6.3cm and 6 zones of Fe each of thickness 2cm. Because fission is so important in this system the y and z dimensions were both chosen to be each 100cm. With the choice of these dimensions, the fissile assembly weighs about 4.5 metric tonnes. Thus the weight constraint is much less severe than in the non fissile case, which means that the change needed in the neutronic environment will be correspondingly much less. The profiles in figures 3, 4 and 5 form the baseline to this optimisation. Any system changes to the assembly will have its profiles compared to the three mentioned above.

In response to a necessary weight reduction, the amount of all materials in the assembly was halved. The $^{238}\text{U}(n,f)$ fission rate very nearly halved, showing the linear dependence of this reaction on the uranium thickness. The $^{235}\text{U}(n,f)$ reaction rate

decreased by 62%, from $1.69\text{E-}2$ to $6.5\text{E-}3$. The reason for this large decrease is not only because of the loss in the uranium but also because virtually all the $^{235}\text{U}(\text{n},\text{f})$ response occurs by low energy neutrons being reflected back towards the source region. The amount of right leakage increased by 2600%, from $5.4\text{E-}3$ to 0.146 which accounts for the decrease in the $^{235}\text{U}(\text{n},\text{f})$ fission rate. The reason for the reaction not decreasing further is because there is also now less absorption in the lithium. The $\text{Li6}(\text{n},\alpha)\text{T}$ reaction decreased by 35%, from 0.58 to 0.38 but the $\text{Li7}(\text{n},\text{n}'\alpha)\text{T}$ actually increased by 51%, from $9.31\text{E-}2$ to 0.141 . This illustrates that for this reaction, the higher proportion of fast neutrons, due to the reduction of fast fissions in ^{238}U , is more important than the reduction in the amount of lithium. The sensitivity profiles for this design (not shown graphically) correspond very closely to those given in figures 3, 4 and 5. There is only one noticeable difference. The sensitivity of $\text{Li6}(\text{n},\alpha)\text{T}$ to $\text{Li6}(\text{n},\alpha)\text{T}$ is higher at all energies for this design, and particularly at the highest energies, than that given in figure 5. This reflects the fact that, due to the reduction in uranium and lithium, there is now more fast flux, and less absorption at lower energies. The higher sensitivity occurs because the sensitivity profile is related to the number of mean free paths in the system. The amount of lithium has been halved, so the $\text{Li6}(\text{n},\alpha)\text{T}$ reaction is much lower and any change in the amount of lithium necessarily brings about an increased $\text{Li6}(\text{n},\alpha)\text{T}$ response value. Apart from this, the sensitivity profiles of $^{238}\text{U}(\text{n},\text{f})$ to $^{238}\text{U}(\text{n},\text{f})$ and $^{235}\text{U}(\text{n},\text{f})$ to $^{235}\text{U}(\text{n},\text{f})$, seem fairly insensitive to any change in the radiation media.

everything else constant, the amount reduced by 90% in each zone. The amount increased substantially, particularly where located, resulting in very poor neutronic hand leakage (back towards the source) increased to 1.77 and the right leakage by 5418%, from the $\text{Li6}(n,\alpha)\text{T}$ reaction rate dropped by 0.146. The sensitivity of $\text{Li6}(n,\alpha)\text{T}$ is shown in figure 6. Comparing figures 5 and 6 that although the profiles have roughly the same down to energies of about 1 KeV, figure 6 has higher sensitivity. The total sensitivity coefficient 5 is 0.18 whilst for figure 6 it is 0.86. The remembering that the amount of lithium fluoride by 90%, is again because of the connection of mean free path thickness. It is now a lot harder $\text{Li6}(n,\alpha)\text{T}$ event. The greater the mean free path from the source to the detector, then the higher the difference in profiles can be viewed as just a change of the detector will be to system changes. Alter flux, since it is mainly the direct term that produces these profiles. With a reduction in fluoride, particularly for the lithium-6 and fluoride now a lot more flux in the integral experiment. The flux is hardly affected though, as can be seen by comparing profiles. Because of the increased leakage and loss breeding it was decided to maintain the full amount of fluoride and to use this amount in the final fissile design.

The next design modification was to move the fissile material around the assembly. The total amount of uranium was kept fixed. The $^{238}\text{U}(n,f)$ reaction scales directly with the thickness of the uranium zone, showing that fast fissions are occurring fairly uniformly throughout the uranium zone and seems fairly insensitive to any other material changes. Because of this insensitivity, half of the uranium zone was moved to the middle of the assembly. It was placed in the same spatial position as one of the Fe cladding zones. The displaced Fe was added to the reflector. It would have made little sense to place the uranium towards the end of the assembly, furthest from the source. At that position a great deal of the thermal flux will have been absorbed by other reactions and there will be virtually no fast flux component. Coupled with this, the amount of right leakage would be very great indeed. As expected, the $^{238}\text{U}(n,f)$ reaction rate dropped by only 33%, from 0.318 to 0.213, showing that a significant amount of fast fission is occurring in the middle uranium zone. The $^{235}\text{U}(n,f)$ activation rate increased by 35%, from 1.69E-2 to 2.29E-2. The amount of left leakage decreased by 20%, from 1.68 to 1.35, and the amount of right hand leakage increased by 26%, from 5.4E-3 to 6.81E-3. Though the right leakage has increased slightly, it is still a small figure when compared to the loss at the left hand edge. The optimisation here is of one between keeping the neutron population as high as possible via having many $^{238}\text{U}(n,f)$ reactions and of reducing the left hand leakage. Since the K_{eff} of these fissile slab systems is about 0.15, then the repeat fission process may still be important. In breaking up the fissile zone in this way, an attempt has been made to catch stray high energy neutrons via $^{238}\text{U}(n,f)$ and to catch low energy neutrons via the repeat fission

process of $^{235}\text{U}(n,f) \longrightarrow ^{238}\text{U}(n,f)$. The 35% increase in the $^{235}\text{U}(n,f)$ activation rate shows just how important the uranium zone located at the centre of the assembly is in helping to increase the neutron population by mopping up thermal neutrons. For this reason it was decided to keep half the uranium at this position in the final design.

The final design step is to reduce the amount of Fe in the assembly. For each of the cladding zones the thickness of Fe was halved, as was the amount of Fe reflector. In addition, 75% of the uranium was placed in the first zone and the rest of it in the middle zone. The $^{238}\text{U}(n,f)$ rate decreased by 18%, from 0.31 to 0.261 and the $^{235}\text{U}(n,f)$ rate decreased by 14%, from $1.69\text{E}-2$ to $1.46\text{E}-2$. This demonstrates that the loss of Fe has a more important effect on the $^{235}\text{U}(n,f)$ reaction than putting the uranium in a more thermal flux. The left hand leakage decreased by 10%, from 1.68 to 1.51, as expected since the amount of fast fission in the first zone is less, whilst the amount of right leakage increased by 201%, from $5.4\text{E}-3$ to $1.63\text{E}-2$, because of the removal of some of the Fe, particularly at the reflector region. The $\text{Li6}(n,\alpha)\text{T}$ reaction rate decreased and the $\text{Li7}(n,n'\alpha)\text{T}$ rate increased, as expected. The changes in activation rates were 10%, from 0.57 to 0.52 and 49%, from 0.093 to 0.139, respectively. The sensitivity profile is shown in figure 7 for the sensitivity of $^{235}\text{U}(n,f)$ to $^{235}\text{U}(n,f)$. The profiles are virtually the same for the sensitivity of $^{238}\text{U}(n,f)$ to $^{238}\text{U}(n,f)$. Figures 4 and 7 are virtually the same except for figure 7 having some additional low energy sensitivity below about 0.1 MeV and losing some high energy sensitivity between 0.1 MeV and 2 MeV. The reason for this gain in sensitivity is because of the loss of thermal flux from the removal of the Fe softening properties.

The loss of sensitivity at higher energies indicates the greater amount of high energy flux, therefore we get more fissions occurring in $^{235}\text{U}(n,f)$ at higher energies (although this is very low in value) and so it becomes less sensitive.

To achieve the necessary weight reduction, the Fe cladding was kept fixed at 2cm and the Fe reflector was reduced to a thickness of 5cm. From the lessons learned on designing the non fissile assembly, an Fe reflector is always located at the position furthest from the source. With the y and z dimensions set at 100cm, the fissile integral experiment in this form weighs 3.1 metric tonnes. The total thickness is 57.6cm. The final assembly has the following dimensions, a 2.4cm thick uranium zone nearest the source, then 3 sets of Fe cladding and LiF zones. The Fe cladding is 2cm and the LiF is 6.3cm thick, respectively. The next zone is 2.4cm of natural uranium. Next comes 3 zones of LiF separated by 2 zones of Fe. Finally, 5cm of Fe is located at the furthest point from the source to act as a reflector.

With the final fissile integral experiment designed using natural lithium fluoride, a last analysis will be made of this design in using 25% Li6 enriched lithium fluoride instead. These results are included because enriched lithium fluoride may become available at a later date for use as a material. The following neutronic comparison is made between the final FIE and the enriched form. The amount of enriched lithium in this design is not the same as that in the reactor (because of the reduction method used in converting design c) to design d)) and so no direct comparison can be made between the enriched FIE and the FR. The change brought about a 12% increase in $\text{Li6}(n,\alpha)T$ and a 12% decrease in $\text{Li7}(n,n'\alpha)T$. The $^{238}\text{U}(n,f)$ rate hardly

changed but the $^{235}\text{U}(n,f)$ response decreased by 48% due to the extra absorption in Li6. For the same reason, the right and left leakages both decreased. Only the sensitivity profiles of $^{235}\text{U}(n,f)$ to $^{235}\text{U}(n,f)$ and Li6(n,alpha)T to Li6(n,alpha)T are shown in figures 8 and 9. The sensitivity profile of $^{238}\text{U}(n,f)$ to $^{238}\text{U}(n,f)$ shows exactly the same shape but is reduced for the enriched case. This is due to the loss of the repeat fission process via $^{235}\text{U}(n,f) \longrightarrow ^{238}\text{U}(n,f)$ particularly in the 2nd zone. By comparing figures 4 and 8 it can be seen that the profile for the enriched case has dropped considerably. The overall drop in sensitivity is about a factor 10. The drop in profile is considerable below about 1 keV. This is due to the sharp reduction in the amount of flux below this energy. The sensitivity of Li6(n,alpha)T to Li6(n,alpha)T is shown in figure 9. Comparing figures 5 and 9, the profiles are roughly the same, except at energies below about 1 keV, where the neutron flux is very low in the enriched case.

8.4 FURTHER TUNING BETWEEN PROFILES

This section details how some of the other sensitivity profiles, those not used in the FIE construction, of the FR and FIE correspond to one another. The method is the same used for the non fissile case. The attempt here is to explain the differences in terms of neutronic environments by the sensitivity profiles. Also included here is a summary, tables 4-13, of the zone sensitivities for the fissile integral experiment. These are given for some important reactor type parameters. Due to lack of space, the sensitivities of zones 12 and 14 for the Fe partials have been added together and placed in the zone 12

column. Likewise, the sensitivities of zones 11 and 13 have been added and placed in zone 11 for the Li6, Li7 and 19F isotopes.

Figures 10 and 11 give the FR and FIE sensitivities of Li6(n,alpha)T to $^{238}\text{U}(n,f)$. These profiles have been included because of the importance of these two reactions. The shape and magnitude of the profiles is very similar; this is because in both cases, and particularly the FR, the 14.1 MeV neutron source is incident on a fairly substantial thickness of uranium. Therefore the flux spectra of the FR and FIE can be considered to be roughly the same. The essential differences between the FR and FIE are that the FR contains oxygen and enriched lithium but does not contain fluorine. Also there is the geometry and albedo change from a cylindrical shape to that of a slab. The two differences between figures 10 and 11 can be accounted for by these changes. The magnitude of figure 10 is roughly double that of figure 11 and also contains a prominent spike centred at about 7.4-8.2 MeV. The FIE decrease can be explained by the fact that half of the uranium has been removed from the first zone, thereby decreasing the effect of the high energy biased $^{238}\text{U}(n,f)$ by roughly a factor half. The 7.4-8.2 MeV spike does exist in the FIE case, but it is barely noticeable. The reason for the lack of spike in the FIE case is probably due to the iron. Before any neutrons can react with the first LiF slab, they must pass through 2cm of natural Fe. For an interaction in the 2nd LiF zone, yet another 2cm of Fe must be penetrated. The Fe elastic scattering cross section is high at these energies and in addition the Fe inelastic cross section has a resonance centred in the energy range 7.4-8.2 MeV. Both these reactions very effectively downscatter neutrons to lower energies. The effect of the Fe must also be coupled with the fact that the number of

fast fissions in ^{238}U has been effectively halved, thereby decreasing the amount of second and higher order chance fissions. Contrast this with the FR case where the full fission zone, with no uranium lost, has only an effective Fe thickness of 1cm before the lithium zone is encountered. The effective thickness of Fe in the lithium zone of the FR is only itself about 0.5cm. Therefore the amount of 7.4-8.2 MeV flux attenuation will only be half as much, whilst the number of fast fissions will be roughly twice as large as compared to the FIE case. In both cases the importance of the high energy flux can be seen. This peak at the 14.1 MeV energy range is due solely to the fissile gain term in the sensitivity equation.

Figures 12 and 13 give the sensitivity of ^{238}U neutron nuclear heating to the $\text{Li6}(n,\alpha)\text{T}$ partial. Mathematically, only the indirect loss term is invoked in the sensitivity equation to produce these profiles, which explains why they are negative. Physically the reason is because any increase in the purely absorptive $\text{Li6}(n,\alpha)\text{T}$ reaction reduces the number of neutrons available for the ^{238}U heating response. This response is very high energy biased and is completely dominated by the fission cross section down to the threshold at about 2.8 MeV. The rapid attenuation of the flux can be seen clearly for the FR in figure 12. It is cut-off (goes below a sensitivity of 10^{-6}) at about an energy of 200 eV whilst for the FIE the cut-off is about 8 eV. The 14.1 MeV sensitivity is very low, reflecting the fact that the $\text{Li6}(n,\alpha)\text{T}$ partial has a low value at high energies. The FR has a slightly higher negative sensitivity at 14.1 MeV because the lithium is enriched in the Li6 isotope. In both graphs, the greatest loss in ^{238}U heating occurs at two definite peaks. The one at about 0.2 MeV is the 285 keV $\text{Li6}(n,\alpha)\text{T}$ resonance. The

peak to its right, centred at about 2 MeV, is the maximum of the product of the fission spectrum and the $^{238}\text{U}(n,f)$ partial cross section. This is the energy region where most of the neutrons from fission are appearing at. Thus any increase in $\text{Li6}(n,\alpha)\text{T}$ has a big effect on the flux in this energy region. The trough between the two peaks is a minima of the product of the $\text{Li6}(n,\alpha)\text{T}$ cross section and the $\phi\phi^*$ quantity (see equation 2.22, chapter 2). The fluorine (n,γ) absorption structure can be clearly seen in figure 13. Neutrons at these energies are preferentially absorbed by $^{19}\text{F}(n,\gamma)$ rather than by $\text{Li6}(n,\alpha)\text{T}$ and so the profile at these energies is less negative.

Figure 14 has been included because it shows clearly the effect of the fission process in $^{235}\text{U}(n,f)$ on the high energy partial $\text{Li7}(n,n'\alpha)\text{T}$. The negative sensitivity on the right hand side is due to the direct interaction between $^{235}\text{U}(n,f)$ and $\text{Li7}(n,n'\alpha)\text{T}$. Any increase in $^{235}\text{U}(n,f)$ at the highest energies, above the $\text{Li7}(n,n'\alpha)\text{T}$ threshold at about 3.8 MeV, will absorb neutrons and decrease the chance of a $\text{Li7}(n,n'\alpha)\text{T}$ event occurring. The fact that the sensitivity is negative indicates that the loss of high energy neutrons to the $^{235}\text{U}(n,f)$ fission cross section is more important than the subsequent neutrons appearing at these high energies from the fission process. It also indicates that the probability of neutrons appearing at high energies from a high energy interaction in $^{235}\text{U}(n,f)$ is quite small. The change in sign corresponds to the $\text{Li7}(n,n'\alpha)\text{T}$ threshold, at about 3.8 MeV. At these high energies the scale on figure 14 is not very accurate. The positive sensitivity below the $\text{Li7}(n,n'\alpha)\text{T}$ threshold indicates that there is now no longer any competition

between the two cross sections and that there is a fair chance that neutrons coming from a fission in $^{235}\text{U}(n,f)$ appear at energies above 3.8 MeV and so contribute to the $\text{Li7}(n,n'\alpha)\text{T}$ response. The highest positive sensitivity, at about 50 KeV, indicates the maximum of the product of flux and the $^{235}\text{U}(n,f)$ cross section.

The FR and FIE design parameters are given in table 3. The greatest decrease, for $\text{Li6}(n,\alpha)\text{T}$ and $^{235}\text{U}(n,f)$, is due to their low energy dependence. The system is more leaky to low energy neutrons, since the interaction mean free path thickness is less at low energies. The small decreases in the Fe neutron and gamma heating indicates the high energy nature of both these responses. It is interesting to note that K_{eff} only decreased by about a third, yet the $^{238}\text{U}(n,f)$ partial, which is responsible for virtually all the reactivity, decreased by nearly two thirds. The source strength used for the DPA calculation on the FIE was $1.0\text{E}+9$ neutrons sec^{-1} and for the FR it was $1.39\text{E}+19$ neutrons sec^{-1} . Both sources are isotropic. The DPA cross sections for fluorine and both uranium isotopes were not available, so no DPA values are given for these. The DPA values for Li6 and Li7 are quite low, much lower than those given for the non fissile reactor. However, the gas production from Fe is much increased for the fissile reactor. This is due solely to the higher average energy of the flux from the fission process.

	FR P.S.	FIE P.S.	REDUCTION
L16(n,alpha)T	1.5081	0.2615	83%
L17(n,n' alpha)T	0.00522	0.01184	60%
238U(n,fission)	0.19185	0.1145	65%
235U(n,fission)	4.827E-2	7.459E-3	85%
L16 neutron heating	7.890E+6	1.318E+6	83%
L17 neutron heating	1.181E+6	4.717E+5	60%
Fe neutron heating	4.117E+5	2.553E+5	38%
238U neutron heating	7.222E+7	2.502E+7	65%
235U neutron heating	7.535E+6	1.191E+6	84%
L16 gamma heating	7.452E+4	8.427E+3	89%
L17 gamma heating	2.215E+5	1.119E+5	50%
Fe gamma heating	1.134E+6	7.816E+5	31%
P19 neutron heating	-	5.066E+5	-
P19 gamma heating	-	1.847E+5	-
K effective	0.24	0.16	33%

DPA/year

Fe total DPA	1.0	1.965E-8
L16 total DPA	0.1	1.128E-9
L17 total DPA	0.1	2.242E-9

GAS APPM/year

Fe(n,proton)	18.3	1.591E-7
Fe(n,alpha)	10.7	1.996E-7

TABLE 1. FISSILE REACTOR AND FISSILE INTEGRAL EXPERIMENT PERFORMANCE PARAMETERS.

	<u>Des a</u>	<u>Des b</u>	<u>Des c</u>	<u>Des d</u>	<u>Des e</u>
<u>Li6(n,alpha)T</u>	1.5081	1.4121	0.7896	0.5816	0.4901
<u>Li7(n,n' alpha)T</u>	0.522E-2	0.4912	0.1112	9.439E-2	0.1104
<u>238U(n,fission)</u>	0.1918	0.25689	0.11891	0.12157	0.1826
<u>235U(n,fission)</u>	4.8273E-2	2.1494E-2	1.3289E-2	1.7046E-2	1.110E-2
<u>Li6 neutron heating</u>	7.891E+6	7.492E+6	4.648E+6	2.927E+6	2.517E+6
<u>Li7 neutron heating</u>	1.181E+6	6.296E+6	1.674E+6	1.173E+6	1.549E+6
<u>Fe neutron heating</u>	4.117E+5	1.441E+6	7.775E+5	5.053E+5	7.005E+5
<u>238U neutron heating</u>	7.222E+7	4.751E+7	3.415E+7	5.959E+7	3.114E+7
<u>235U neutron heating</u>	7.535E+6	1.697E+6	1.247E+6	2.719E+6	1.731E+6
<u>F19 neutron heating</u>	-	-	-	1.374E+6	1.502E+6
<u>Li6 gamma heating</u>	7.452E+4	4.319E+4	9.101E+4	1.835E+4	2.106E+4
<u>Li7 gamma heating</u>	2.214E+5	1.397E+5	2.731E+5	1.174E+6	1.549E+6
<u>Fe gamma heating</u>	1.112E+6	9.126E+5	2.123E+6	1.554E+6	1.809E+6
<u>F19 gamma heating</u>	-	-	-	8.803E+5	9.586E+5
<u>K effective</u>	0.24	0.26	0.1916	0.1958	0.159

TABLE 1. SOME INTERMEDIATE RESULTS FROM DESIGNING
THE FISSION INTEGRAL EXPERIMENT

<u>238U(n,fission)</u>			
160 elastic	-1.67E-2	160(n,deuterium)	7.59E-5
160(n,gamma)	8.16E-10	160(n,alpha)	1.55E-1
160(n,proton)	2.50E-4	160 inelastic	-4.34E-4
<u>235U(n,fission)</u>			
160 elastic	-6.61E-2	160(n,deuterium)	1.16E-4
160(n,gamma)	2.46E-8	160(n,alpha)	2.22E-3
160(n,proton)	3.77E-4	160 inelastic	-1.66E-3
<u>Li6(n,alpha)T</u>			
160 elastic	2.22E-2	160 inelastic	-7.46E-4

TABLE 2. TOTAL SENSITIVITY COEFFICIENTS (%) FOR A

1% DECREASE IN THE SPECIFIED OXYGEN PARTIAL.

SENSITIVITY COEFFICIENTS OF L17(N,ALPHA)N TO L16,U23H,U21S
FE,L17,F19 PARTIAL CROSS SECTIONS,FISSILE INTEGRAL EXPERTMENT

	ZONE 1	ZONE 5	ZONE 7	ZONE 9	ZONE 11	SUM
L16 ELASTIC	-4.11E-03	-8.19E-04	-1.04E-04	-8.51E-05	-6.62E-06	-5.19E-01
L16(N,2N)ALPHA	-1.56E-03	-5.94E-04	-1.07E-04	-5.34E-05	-1.48E-05	-2.41E-03
L16(N,N')ALPHA	-7.68E-03	-3.29E-03	-1.15E-03	-3.70E-04	-1.09E-04	-1.25E-02
L16(N,GAMMA)	-2.79E-07	-1.19E-07	-4.39E-08	-1.18E-08	-3.82E-09	-4.59E-07
L16(N,PROTON)	-2.30E-04	-1.08E-04	-3.84E-05	-1.20E-05	-3.66E-06	-4.01E-04
L16(N,ALPHA)T	-9.10E-04	-4.53E-04	-2.68E-04	-9.54E-05	-2.20E-05	-1.74E-03
L16 INELASTIC	-7.65E-03	-3.31E-03	-1.16E-03	-3.73E-04	-1.10E-04	-1.26E-02
L17 ELASTIC	-6.20E-02	-1.10E-02	-1.16E-03	-1.41E-03	-1.46E-04	-7.96E-02
L17(N,2N)	-6.59E-03	-2.51E-03	-7.06E-04	-2.24E-04	-6.18E-05	-1.02E-02
L17(N,N')ALPHA	-9.28E-03	-3.23E-03	-9.54E-04	-2.60E-04	-6.72E-05	-1.38E-02
L17(N,ALPHA)T	5.46E-01	2.21E-01	7.66E-02	2.37E-02	7.71E-03	8.79E-01
L17(N,GAMMA)	-3.63E-06	-1.54E-06	-5.03E-07	-1.85E-07	-5.06E-08	-6.00E-06
L17(N,DEUTERIUM)	-2.87E-03	-1.01E-03	-3.10E-04	-8.57E-05	-2.26E-05	-4.32E-03
L17 INELASTIC	-8.89E-02	-3.78E-02	-1.32E-02	-4.24E-03	-1.24E-03	-1.45E-01
F19 ELASTIC	-4.60E-02	-5.54E-03	-3.15E-04	-1.32E-04	1.56E-04	-5.26E-02
F19(N,2N)	-1.45E-02	-4.06E-03	-1.40E-03	-3.74E-04	-9.45E-05	-2.12E-02
F19(N,N')ALPHA	-1.51E-02	-6.04E-03	-1.97E-03	-5.79E-04	-1.65E-04	-2.19E-02
F19(N,N')PROTON	-3.41E-03	-1.22E-03	-3.68E-04	-1.01E-04	-2.67E-05	-5.16E-03
F19(N,GAMMA)	-6.32E-06	-3.91E-06	-2.54E-06	-1.00E-06	-2.33E-07	-1.40E-05
F19(N,PROTON)	-7.95E-03	-3.52E-03	-1.24E-03	-3.83E-04	-1.17E-04	-1.31E-02
F19(N,DEUTERIUM)	-4.08E-03	-1.60E-03	-5.16E-04	-1.50E-04	-2.23E-05	-6.40E-03
F19(N,TRITON)	-2.50E-03	-9.59E-04	-3.03E-04	-8.67E-05	-2.49E-05	-3.08E-03
F19(N,ALPHA)	-1.60E-02	-7.01E-03	-2.89E-03	-9.31E-04	-2.92E-04	-2.00E-02
F19 INELASTIC	-3.00E-01	-8.13E-02	-2.73E-02	-8.36E-03	-2.40E-03	-3.19E-01

	ZONE 2	ZONE 4	ZONE 6	ZONE 10	ZONE 12	SUM
FE ELASTIC	-1.04E-01	-1.50E-02	-2.36E-03	-6.36E-05	3.96E-04	-1.21E-01
FE(N,2N)	-8.18E-02	-3.02E-02	-9.05E-03	-6.36E-04	-1.41E-04	-1.22E-01
FE(N,N')ALPHA	-6.52E-04	-2.22E-04	-6.35E-05	-4.17E-06	-8.00E-07	-9.43E-04
FE(N,N')PROTON	-7.37E-03	-2.67E-03	-7.94E-04	-5.55E-05	1.33E-05	-1.09E-02
FE(N,GAMMA)	-4.75E-05	-2.88E-05	-1.67E-05	-2.83E-06	5.47E-07	-9.64E-05
FE(N,PROTON)	-2.28E-02	-1.13E-02	-3.94E-03	-3.52E-04	-9.18E-05	-3.85E-02
FE(N,DEUTERIUM)	-3.53E-03	-1.35E-03	-4.13E-04	-3.01E-05	-6.03E-06	-5.13E-03
FE(N,TRITON)	-8.52E-05	-2.87E-05	-8.18E-06	-3.32E-07	-1.12E-07	-1.23E-04
FE(N,HELIUM 3)	-6.23E-05	-2.11E-05	-6.04E-06	-9.5E-07	-8.32E-08	-8.99E-05
FE(N,ALPHA)	-7.28E-03	-3.41E-03	-1.17E-03	-1.01E-04	-2.56E-05	-1.20E-02
FE INELASTIC	-1.23E-01	-7.26E-02	-2.81E-02	-2.94E-03	-7.03E-04	-2.28E-01

TABLE 4

	ZONE 1	ZONE 8	SUM
U230 ELASTIC	-2.23E-01	-1.25E-03	-2.24E-01
U238(N,2N)	-9.25E-02	-7.00E-03	-9.95E-02
U238(N,3N)	-6.10E-02	-1.90E-03	-6.29E-02
U238 TOTAL FISSION	-1.23E-01	-5.07E-03	-1.28E-01
U238(N,GAMMA)	-2.03E-04	-2.16E-04	-4.19E-04
U238 TOTAL INELASTIC	-3.37E-02	-3.55E-03	-3.73E-02
U235 ELASTIC	-1.62E-03	-9.27E-06	-1.63E-03
U235(N,2N)	-2.31E-04	-1.64E-05	-2.48E-04
U235(N,3N)	-3.03E-05	-8.71E-07	-3.12E-05
U235 TOTAL FISSION	-1.32E-03	-4.53E-04	-1.77E-03
U235(N,GAMMA)	-3.06E-06	-2.64E-06	-5.70E-06
U235 TOTAL INELASTIC	-1.44E-04	-1.17E-05	-1.56E-04

SENSITIVITY COEFFICIENTS OF L16(N,ALPHA)N TO L16,U23H,U21S
FE,L17,F19 PARTIAL CROSS SECTIONS,FISSILE INTEGRAL EXPERTMENT

	ZONE 1	ZONE 5	ZONE 7	ZONE 9	ZONE 11	SUM
L16 ELASTIC	-2.09E-03	-1.19E-04	-1.40E-04	2.18E-04	1.12E-04	1.11E-03
L16(N,2N)ALPHA	5.43E-04	3.36E-04	8.45E-05	4.28E-05	1.40E-05	1.02E-03
L16(N,N')ALPHA	-1.60E-03	-9.34E-04	-4.22E-04	3.81E-05	2.95E-05	-2.43E-03
L16(N,GAMMA)	-7.84E-06	-9.32E-06	-4.90E-06	-2.11E-06	-1.74E-06	-2.67E-05
L16(N,PROTON)	-1.97E-04	-1.15E-04	-5.57E-05	-1.52E-05	-1.76E-06	-3.08E-04
L16(N,ALPHA)T	-1.36E-01	-6.84E-02	-6.89E-02	2.60E-02	9.50E-03	-3.07E-01
L16 INELASTIC	-1.62E-03	-5.01E-04	-4.27E-04	3.84E-05	2.97E-05	-2.47E-03
L17 ELASTIC	-2.27E-02	-4.18E-03	-1.76E-04	1.65E-03	5.15E-03	-5.49E-03
L17(N,2N)	-2.27E-03	-1.40E-03	3.48E-04	1.78E-04	5.84E-05	-4.27E-03
L17(N,N')ALPHA	-3.31E-03	-1.83E-03	-3.93E-04	2.06E-04	6.77E-05	-5.82E-03
L17(N,ALPHA)T	-1.77E-02	-5.78E-03	-4.22E-03	3.29E-04	2.21E-04	-2.71E-02
L17(N,GAMMA)	-1.22E-04	-1.46E-04	-7.79E-05	-3.37E-05	-2.67E-05	-4.19E-04
L17(N,DEUTERIUM)	-1.83E-03	-8.08E-04	-3.38E-04	-5.80E-05	-1.24E-05	-3.05E-03
L17 INELASTIC	-2.13E-02	-6.84E-03	-5.25E-03	5.55E-04	3.03E-04	-3.23E-02
F19 ELASTIC	-1.15E-02	-8.53E-03	-1.60E-02	2.45E-03	8.12E-03	1.17E-02
F19(N,2N)	-4.42E-03	-2.54E-03	5.60E-04	3.00E-04	9.26E-05	-7.93E-03
F19(N,N')ALPHA	-2.89E-03	-8.98E-04	-6.51E-04	3.20E-05	2.76E-05	-4.17E-03
F19(N,N')PROTON	-5.51E-04	-1.55E-04	-1.25E-04	5.40E-06	5.48E-06	-8.18E-04
F19(N,GAMMA)	-3.60E-03	-3.34E-04	-1.70E-03	-7.69E-04	-3.49E-04	-9.94E-03
F19(N,PROTON)	-4.78E-03	-2.60E-03	-1.25E-03	-2.07E-04	-6.86E-05	-9.00E-03
F19(N,DEUTERIUM)	-2.55E-03	-1.21E-03	-5.40E-04	-1.04E-04	-2.30E-05	-4.43E-03
F19(N,TRITON)	-1.57E-03	-7.33E-04	-3.22E-04	-5.96E-05	-1.30E-05	-2.70E-03
F19(N,ALPHA)	-1.20E-02	-7.33E-03	-3.62E-03	-9.98E-04	-2.50E-04	-2.42E-02
F19 INELASTIC	-3.79E-02	-7.62E-03	-9.89E-03	8.59E-04	1.63E-03	-5.20E-02

	ZONE 2	ZONE 4	ZONE 6	ZONE 10	ZONE 12	SUM
FE ELASTIC	5.32E-03	-1.75E-02	1.82E-03	1.03E-03	1.40E-02	4.72E-03
FE(N,2N)	7.73E-03	1.53E-02	4.06E-03	6.14E-04	2.08E-04	2.07E-02
FE(N,N')ALPHA	-1.61E-04	-2.73E-05	-1.53E-05	6.56E-07	4.09E-07	-2.03E-04
FE(N,N')PROTON	-1.84E-03	-3.43E-04	-1.04E-04	0.45E-06	5.40E-06	-2.16E-03
FE(N,GAMMA)	-7.25E-03	-1.75E-02	-1.40E-02	-3.91E-03	-3.13E-03	-4.50E-02
FE(N,PROTON)	-1.48E-02	-0.78E-03	-3.96E-03	-2.64E-04	-6.26E-05	-2.79E-02
FE(N,DEUTERIUM)	-2.14E-03	-1.03E-03	-4.11E-04	-1.99E-05	-4.10E-06	-3.60E-03
FE(N,TRITON)	-5.11E-05	-2.22E-05	-8.32E-06	-3.54E-07	-6.81E-08	-8.20E-05
FE(N,HELIUM 3)	-3.74E-05	-1.63E-05	-6.13E-06	-2.63E-07	-5.08E-08	-6.01E-05
FE(N,ALPHA)	-4.56E-03	-2.57E-03	-1.13E-03	-6.81E-05	-1.55E-05	-8.14E-03
FE INELASTIC	-3.30E-02	-5.24E-03	-3.93E-03	6.99E-04	5.64E-04	-4.09E-02

TABLE 5

	ZONE 1	ZONE 8	SUM
U230 ELASTIC	7.56E-02	8.43E-03	8.37E-02
U238(N,2N)	-1.05E-02	3.44E-03	-7.02E-03
U238(N,3N)	9.78E-03	2.37E-03	1.21E-02
U238 TOTAL FISSION	1.43E-01	2.85E-02	1.72E-01
U238(N,GAMMA)	-4.77E-02	-9.26E-02	-1.40E-01
U238 TOTAL INELASTIC	-2.02E-03	-3.55E-03	-5.57E-03
U235 ELASTIC	3.62E-04	4.01E-05	4.02E-04
U235(N,2N)	-2.45E-05	8.60E-06	-1.59E-05
U235(N,3N)	4.74E-06	1.00E-06	5.82E-06
U235 TOTAL FISSION	5.57E-03	6.58E-03	1.21E-02
U235(N,GAMMA)	-5.52E-04	-1.46E-03	-1.61E-03
U235 TOTAL INELASTIC	2.10E-06	-1.34E-05	-1.13E-05

SENSITIVITY COEFFICIENTS OF 245(N,FISSIOM), TO L10, U238, U235,
FE, L17, P19 PARTIAL CROSS SECTIONS, FISSILE INTERNAL EXPERIMENT

	ZONE 1	ZONE 5	ZONE 7	ZONE 9	ZONE 11	SUM
L16 ELASTIC	-6.16E-04	-2.14E-03	-1.17E-03	5.27E-04	2.94E-05	-3.38E-03
L16(N,2N)ALPHA	3.11E-04	1.16E-04	2.90E-05	2.68E-05	4.88E-06	5.00E-04
L16(N,N')ALPHA	-0.57E-04	-5.01E-04	-4.10E-04	5.09E-05	1.18E-05	-1.79E-03
L16(N,GAMMA)	-1.14E-06	-2.02E-06	-2.50E-06	-1.06E-06	-1.86E-07	-9.01E-06
L16(N,PROTON)	-1.27E-04	-6.81E-05	3.78E-05	-8.50E-06	1.40E-06	-2.43E-04
L16(N,ALPHA)T	-0.70E-02	-5.44E-02	-6.66E-02	2.74E-02	-4.81E-03	-2.41E-01
L16 INELASTIC	-0.60E-04	-5.09E-04	-4.22E-04	5.14E-05	1.19E-05	-1.01E-03
L17 ELASTIC	-1.63E-02	-1.41E-02	-2.03E-02	7.47E-03	4.42E-04	-6.27E-02
L17(N,2N)	1.14E-03	4.70E-04	1.17E-04	1.12E-04	2.03E-05	2.11E-03
L17(N,2N)ALPHA	1.89E-03	5.17E-04	1.04E-04	1.30E-04	2.18E-05	2.67E-03
L17(N,ALPHA)T	-9.01E-03	-5.65E-03	-3.96E-03	4.56E-04	1.09E-04	-1.80E-02
L17(N,GAMMA)	-5.00E-05	-1.15E-05	-4.16E-05	-1.71E-05	-3.08E-06	-1.46E-04
L17(N,DEUTERIUM)	-1.13E-03	-5.04E-04	-2.41E-04	-2.95E-05	-4.23E-06	-1.91E-03
L17 INELASTIC	-1.05E-02	-7.54E-03	-5.19E-03	8.12E-04	1.70E-04	-2.22E-02
P19 ELASTIC	-2.17E-02	-4.44E-02	-4.10E-02	1.62E-02	3.61E-03	-9.13E-02
P19(N,2N)	2.77E-03	7.89E-04	1.55E-04	1.94E-04	3.24E-05	3.95E-03
P19(N,N')ALPHA	-1.55E-03	-8.80E-04	-6.04E-04	5.04E-05	1.34E-05	-2.97E-03
P19(N,N')PROTON	-3.16E-04	-1.94E-04	-1.20E-04	8.10E-06	1.98E-06	-6.40E-04
P19(N,GAMMA)	-1.96E-03	-1.16E-03	-1.04E-03	-4.19E-04	-8.77E-05	-4.89E-03
P19(N,PROTON)	-3.07E-03	-1.50E-03	-8.72E-04	-1.58E-04	-2.52E-05	-5.71E-03
P19(N,DEUTERIUM)	-1.59E-03	-7.51E-04	-3.83E-04	-5.03E-05	-8.12E-06	-2.79E-03
P19(N,TRITON)	-9.74E-04	-4.55E-04	-2.80E-04	-3.07E-05	-4.53E-06	-1.69E-03
P19(N,ALPHA)	-7.82E-03	-4.32E-03	-2.46E-03	-5.00E-04	-9.31E-05	-1.53E-02
P19 INELASTIC	-9.70E-03	-1.84E-02	-1.07E-02	3.05E-03	5.79E-04	-3.43E-02
FE ELASTIC	-1.41E-02	-1.43E-02	-2.25E-02	5.21E-03	1.75E-03	-4.40E-02
FE(N,2N)	1.69E-02	5.64E-03	1.27E-03	2.73E-04	4.85E-05	2.41E-02
FE(N,N')ALPHA	-5.88E-05	-3.20E-05	-1.79E-05	3.42E-07	7.41E-08	-1.08E-04
FE(N,N')PROTON	-6.76E-04	-3.66E-04	-2.11E-04	4.68E-06	1.08E-06	-1.25E-03
FE(N,GAMMA)	-7.05E-03	-4.65E-03	-4.40E-03	-9.08E-04	-1.53E-04	-1.72E-02
FE(N,PROTON)	-1.05E-02	-5.23E-03	-2.59E-03	-1.10E-04	-1.80E-05	-1.85E-02
FE(N,DEUTERIUM)	-1.43E-03	-6.17E-04	-2.74E-04	-7.89E-06	-1.14E-06	-2.34E-03
FE(N,TRITON)	-3.19E-05	-1.34E-05	-5.58E-06	-1.38E-07	-1.87E-08	-5.30E-05
FE(N,HELIUM 3)	-2.48E-05	-9.82E-06	-4.12E-06	-1.02E-07	-1.39E-08	-3.88E-05
FE(N,ALPHA)	-3.10E-03	-1.53E-03	-7.43E-04	-2.80E-05	-4.42E-06	-5.49E-03
FE INELASTIC	-1.28E-02	-8.06E-03	-6.38E-03	3.62E-04	1.05E-04	-2.68E-02

TABLE 6

	ZONE 1	ZONE 5	ZONE 7	ZONE 9	ZONE 11	SUM
U238 ELASTIC	1.14E-01	1.90E-03	1.16E-01			
U238(N,2N)	9.66E-03	2.69E-03	1.24E-02			
U238(N,3N)	2.82E-02	1.88E-03	3.01E-02			
U238 TOTAL FISSION	2.01E-01	2.10E-02	2.22E-01			
U238(N,GAMMA)	-9.55E-02	-1.03E-01	-1.98E-01			
U238 TOTAL INELASTIC	2.64E-03	-9.58E-04	1.68E-03			
U235 ELASTIC	7.22E-04	2.05E-05	7.43E-04			
U235(N,2N)	2.61E-05	6.70E-06	3.28E-05			
U235(N,3N)	1.39E-05	8.63E-07	1.48E-05			
U235 TOTAL FISSION	7.30E-04	2.79E-04	1.01E-03			
U235(N,GAMMA)	-1.21E-03	-1.23E-03	-2.44E-03			
U235 TOTAL INELASTIC	5.21E-05	-1.01E-06	5.11E-05			

TABLE 7

	ZONE 1	ZONE 5	ZONE 7	ZONE 9	ZONE 11	SUM
FE ELASTIC	5.97E-02	-1.39E-05	-6.09E-03	5.05E-04	5.98E-05	5.42E-02
FE(N,2N)	-3.32E-03	-3.81E-03	-2.03E-03	-1.37E-05	-9.09E-07	-9.90E-03
FE(N,N')ALPHA	-7.88E-05	-3.11E-05	-2.14E-05	-1.55E-07	-9.15E-09	-1.31E-04
FE(N,N')PROTON	-6.24E-04	-3.53E-04	-2.54E-04	-1.52E-06	-1.16E-07	-1.21E-03
FE(N,GAMMA)	-1.03E-04	-4.28E-05	-3.96E-05	-8.18E-06	-1.25E-06	-7.95E-04
FE(N,PROTON)	-5.06E-03	-1.52E-03	-1.15E-03	-1.94E-05	-1.55E-06	-7.75E-03
FE(N,DEUTERIUM)	-5.64E-04	-1.90E-04	-1.35E-04	-1.25E-06	-9.40E-08	-8.21E-04
FE(N,TRITON)	-1.26E-05	-4.19E-06	-2.82E-06	-2.06E-08	-1.46E-09	-1.96E-05
FE(N,HELIUM 3)	-9.23E-06	-3.88E-06	-2.88E-06	-1.53E-08	-1.09E-09	-1.41E-05
FE(N,ALPHA)	-1.45E-03	-4.59E-04	-4.46E-04	-4.96E-06	-3.96E-07	-2.26E-03
FE INELASTIC	-2.00E-02	-8.72E-03	-8.41E-03	-1.15E-04	-7.65E-06	-1.71E-02
U238 ELASTIC	1.19E-01	3.43E-03	1.43E-01			
U238(N,2N)	-4.03E-02	-3.38E-03	-4.37E-02			
U238(N,3N)	-2.72E-02	-8.68E-04	-2.81E-02			
U238 TOTAL FISSION	1.06E-01	5.42E-02	1.11E-01			
U238(N,GAMMA)	-2.51E-03	-9.75E-04	-3.48E-03			
U238 TOTAL INELASTIC	-8.66E-02	-8.00E-03	-9.55E-02			
U235 ELASTIC	1.02E-03	2.55E-05	1.05E-03			
U235(N,2N)	-8.07E-05	-7.22E-06	-8.88E-05			
U235(N,3N)	-1.41E-05	-4.02E-07	-1.45E-05			
U235 TOTAL FISSION	5.05E-03	2.15E-03	7.20E-03			
U235(N,GAMMA)	-2.47E-05	-1.12E-05	-1.59E-05			
U235 TOTAL INELASTIC	-3.02E-04	-3.34E-05	-3.35E-04			

SENSITIVITY COEFFICIENTS OF L17 HEAVYION TO L16, L22, L23, L24
PE, L17, L19 PARTIAL CROSS SECTIONS, FISSILE MATERIAL EXPENDITURE

	ZONE 1	ZONE 5	ZONE 7	ZONE 9	ZONE 11	SUM
L16 ELASTIC	-6.17E-03	-2.01E-03	-7.11E-04	-2.45E-04	-5.97E-05	-9.41E-03
L16(N,2N)ALPHA	-1.02E-03	-3.67E-04	-1.18E-04	-3.11E-05	-7.92E-06	-1.55E-03
L16(N,N')ALPHA	-6.51E-03	-2.70E-03	-9.04E-04	-3.01E-04	-8.31E-05	-1.06E-02
L16(N,N')ALPHA	-1.57E-07	-1.02E-07	-8.84E-08	-1.20E-08	-1.06E-08	-6.71E-07
L16(N,PROTON)	-2.95E-04	-1.16E-04	-4.35E-05	-1.37E-05	-4.07E-06	-4.23E-04
L16(N,ALPHA)T	-1.67E-03	-2.36E-03	-1.40E-03	-6.09E-04	-2.09E-04	-8.40E-03
L16 INELASTIC	-6.57E-03	-2.73E-03	-9.92E-04	-3.03E-04	-8.37E-05	-1.07E-02
L17 ELASTIC	1.97E-01	9.67E-02	4.02E-02	1.45E-02	4.97E-03	1.55E-01
L17(N,2N)	4.60E-01	1.50E-01	4.51E-01	1.25E-01	4.17E-02	6.73E-01
L17(N,2N)ALPHA	5.46E-01	1.64E-01	4.78E-01	1.32E-01	4.42E-02	7.77E-01
L17(N,ALPHA)T	1.02E-01	6.95E-02	2.14E-02	6.87E-03	2.26E-03	2.05E-01
L17(N,GAMMA)	1.60E-06	-1.21E-06	-7.72E-07	-3.01E-07	-9.34E-08	-4.09E-06
L17(N,DEUTERIUM)	4.16E-03	1.13E-03	4.02E-04	1.09E-04	3.39E-05	6.04E-03
L17 INELASTIC	-7.57E-02	-3.03E-02	-1.09E-02	-3.27E-03	-8.91E-04	-1.21E-01
L19 ELASTIC	-6.09E-02	-1.76E-02	-5.00E-03	-1.79E-03	-3.32E-04	-9.43E-02
L19(N,2N)	-1.02E-02	-3.12E-03	-9.98E-04	-2.48E-04	-5.90E-05	-1.48E-02
L19(N,N')ALPHA	-1.01E-02	-3.76E-03	-1.25E-03	-3.19E-04	-8.94E-05	-1.56E-02
L19(N,N')PROTON	-2.75E-03	-9.68E-04	-3.02E-04	-7.77E-05	-1.66E-05	-4.12E-03
L19(N,GAMMA)	-4.52E-05	-3.47E-05	-2.12E-05	-8.98E-06	-3.00E-06	-1.14E-04
L19(N,PROTON)	-6.69E-03	-3.00E-03	-1.08E-03	-3.18E-04	-9.41E-05	-1.12E-02
L19(N,DEUTERIUM)	-3.66E-03	-1.44E-03	-4.76E-04	-1.38E-04	-3.55E-05	-5.75E-03
L19(N,TRITON)	-2.26E-03	-8.71E-04	-2.84E-04	-7.63E-05	-2.95E-05	-3.52E-03
L19(N,ALPHA)	-1.51E-02	-7.57E-03	-2.91E-03	-9.21E-04	-2.81E-04	-2.60E-02
L19 INELASTIC	-1.85E-01	-7.58E-02	-2.00E-02	-8.91E-03	-2.51E-03	-3.01E-01
FE ELASTIC	-9.96E-02	-1.74E-02	-3.91E-03	-1.14E-04	4.02E-04	-1.21E-01
FE(N,2N)	-6.60E-02	-2.25E-02	-6.86E-03	-4.50E-04	-8.65E-05	-9.59E-02
FE(N,N')ALPHA	-5.78E-04	-1.96E-04	-5.77E-05	-3.59E-06	-7.28E-07	-8.36E-04
FE(N,N')PROTON	-6.37E-03	-2.24E-03	-6.81E-04	-4.49E-05	-9.30E-06	-9.34E-03
FE(N,N')ALPHA	-1.84E-04	-1.65E-04	-1.11E-04	-2.14E-05	-9.75E-06	-4.92E-04
FE(N,PROTON)	-2.13E-02	-1.04E-02	-3.70E-03	-3.16E-04	-8.19E-05	-3.50E-02
FE(N,DEUTERIUM)	-3.27E-03	-1.27E-03	-3.98E-04	-2.76E-05	-6.25E-06	-4.97E-03
FE(N,TRITON)	-7.89E-05	-2.74E-05	-8.95E-06	-5.01E-07	-1.06E-07	-1.15E-04
FE(N,HELIUM 3)	-3.86E-05	-1.65E-05	-6.11E-06	-2.69E-07	-5.24E-08	-6.11E-05
FE(N,ALPHA)	-6.74E-03	-3.13E-03	-1.08E-03	-8.01E-05	-2.21E-05	-1.11E-02
FE INELASTIC	-1.14E-01	-6.21E-02	-2.50E-02	-2.61E-03	-6.45E-04	-2.04E-01
U238 ELASTIC	-1.04E-01	-7.10E-02	-1.94E-02	-1.84E-03	-1.84E-04	-1.04E-01
U238(N,2N)	-8.06E-02	-5.16E-03	-5.54E-02	-5.54E-03	-5.54E-04	-8.06E-02
U238(N,N')ALPHA	-5.38E-02	-1.57E-03	-5.54E-02	-5.54E-03	-5.54E-04	-5.38E-02
U238 TOTAL FISSION	-6.90E-02	-1.08E-03	-6.71E-02	-6.71E-03	-6.71E-04	-6.90E-02
U238(N,GAMMA)	-1.72E-03	-1.08E-03	-3.19E-03	-3.19E-04	-3.19E-05	-1.72E-03
U238 TOTAL INELASTIC	-4.53E-02	-8.75E-03	-5.41E-02	-5.41E-03	-5.41E-04	-4.53E-02
U235 ELASTIC	-1.35E-03	-5.15E-06	-1.35E-03	-1.35E-04	-1.35E-05	-1.35E-03
U235(N,2N)	-1.98E-04	-1.17E-05	-2.09E-04	-2.09E-05	-2.09E-06	-1.98E-04
U235(N,N')ALPHA	-2.69E-05	-7.31E-07	-2.76E-05	-2.76E-06	-2.76E-07	-2.69E-05
U235 TOTAL FISSION	4.39E-04	2.22E-03	2.66E-04	2.66E-05	2.66E-06	4.39E-04
U235(N,GAMMA)	-1.82E-05	-1.60E-05	-3.51E-05	-3.51E-06	-3.51E-07	-1.82E-05
U235 TOTAL INELASTIC	-1.96E-04	-3.53E-05	-2.32E-04	-2.32E-05	-2.32E-06	-1.96E-04

TABLE 8

	ZONE 1	ZONE 8	SUM
U238 ELASTIC	5.92E-02	7.44E-03	6.67E-02
U238(N,2N)	-1.48E-02	2.90E-03	-1.19E-02
U238(N,N')ALPHA	5.85E-03	2.12E-03	7.97E-03
U238 TOTAL FISSION	1.30E-02	2.69E-02	1.57E-01
U238(N,GAMMA)	-4.46E-02	8.63E-02	1.31E-01
U238 TOTAL INELASTIC	-4.80E-03	-3.92E-03	-8.71E-03
U235 ELASTIC	2.55E-04	3.78E-05	2.92E-04
U235(N,2N)	-3.51E-05	7.31E-06	-2.78E-05
U235(N,N')ALPHA	2.78E-06	9.69E-07	3.75E-06
U235 TOTAL FISSION	5.26E-03	6.32E-03	1.16E-02
U235(N,GAMMA)	-5.16E-04	-9.88E-04	-1.50E-03
U235 TOTAL INELASTIC	-1.07E-05	-1.50E-05	-2.58E-05

TABLE 9

	ZONE 1	ZONE 5	ZONE 7	ZONE 9	ZONE 11	SUM
L16 ELASTIC	-6.17E-03	-2.01E-03	-7.11E-04	-2.45E-04	-5.97E-05	-9.41E-03
L16(N,2N)ALPHA	-1.02E-03	-3.67E-04	-1.18E-04	-3.11E-05	-7.92E-06	-1.55E-03
L16(N,N')ALPHA	-6.51E-03	-2.70E-03	-9.04E-04	-3.01E-04	-8.31E-05	-1.06E-02
L16(N,N')ALPHA	-1.57E-07	-1.02E-07	-8.84E-08	-1.20E-08	-1.06E-08	-6.71E-07
L16(N,PROTON)	-2.95E-04	-1.16E-04	-4.35E-05	-1.37E-05	-4.07E-06	-4.23E-04
L16(N,ALPHA)T	-1.67E-03	-2.36E-03	-1.40E-03	-6.09E-04	-2.09E-04	-8.40E-03
L16 INELASTIC	-6.57E-03	-2.73E-03	-9.92E-04	-3.03E-04	-8.37E-05	-1.07E-02
L17 ELASTIC	1.97E-01	9.67E-02	4.02E-02	1.45E-02	4.97E-03	1.55E-01
L17(N,2N)	4.60E-01	1.50E-01	4.51E-01	1.25E-01	4.17E-02	6.73E-01
L17(N,2N)ALPHA	5.46E-01	1.64E-01	4.78E-01	1.32E-01	4.42E-02	7.77E-01
L17(N,ALPHA)T	1.02E-01	6.95E-02	2.14E-02	6.87E-03	2.26E-03	2.05E-01
L17(N,GAMMA)	1.60E-06	-1.21E-06	-7.72E-07	-3.01E-07	-9.34E-08	-4.09E-06
L17(N,DEUTERIUM)	4.16E-03	1.13E-03	4.02E-04	1.09E-04	3.39E-05	6.04E-03
L17 INELASTIC	-7.57E-02	-3.03E-02	-1.09E-02	-3.27E-03	-8.91E-04	-1.21E-01
L19 ELASTIC	-6.09E-02	-1.76E-02	-5.00E-03	-1.79E-03	-3.32E-04	-9.43E-02
L19(N,2N)	-1.02E-02	-3.12E-03	-9.98E-04	-2.48E-04	-5.90E-05	-1.48E-02
L19(N,N')ALPHA	-1.01E-02	-3.76E-03	-1.25E-03	-3.19E-04	-8.94E-05	-1.56E-02
L19(N,N')PROTON	-2.75E-03	-9.68E-04	-3.02E-04	-7.77E-05	-1.66E-05	-4.12E-03
L19(N,GAMMA)	-4.52E-05	-3.47E-05	-2.12E-05	-8.98E-06	-3.00E-06	-1.14E-04
L19(N,PROTON)	-6.69E-03	-3.00E-03	-1.08E-03	-3.18E-04	-9.41E-05	-1.12E-02
L19(N,DEUTERIUM)	-3.66E-03	-1.44E-03	-4.76E-04	-1.38E-04	-3.55E-05	-5.75E-03
L19(N,TRITON)	-2.26E-03	-8.71E-04	-2.84E-04	-7.63E-05	-2.95E-05	-3.52E-03
L19(N,ALPHA)	-1.51E-02	-7.57E-03	-2.91E-03	-9.21E-04	-2.81E-04	-2.60E-02
L19 INELASTIC	-1.85E-01	-7.58E-02	-2.00E-02	-8.91E-03	-2.51E-03	-3.01E-01
FE ELASTIC	-9.96E-02	-1.74E-02	-3.91E-03	-1.14E-04	4.02E-04	-1.21E-01
FE(N,2N)	-6.60E-02	-2.25E-02	-6.86E-03	-4.50E-04	-8.65E-05	-9.59E-02
FE(N,N')ALPHA	-5.78E-04	-1.96E-04	-5.77E-05	-3.59E-06	-7.28E-07	-8.36E-04
FE(N,N')PROTON	-6.37E-03	-2.24E-03	-6.81E-04	-4.49E-05	-9.30E-06	-9.34E-03
FE(N,N')ALPHA	-1.84E-04	-1.65E-04	-1.11E-04	-2.14E-05	-9.75E-06	-4.92E-04
FE(N,PROTON)	-2.13E-02	-1.04E-02	-3.70E-03	-3.16E-04	-8.19E-05	-3.50E-02
FE(N,DEUTERIUM)	-3.27E-03	-1.27E-03	-3.98E-04	-2.76E-05	-6.25E-06	-4.97E-03
FE(N,TRITON)	-7.89E-05	-2.74E-05	-8.95E-06	-5.01E-07	-1.06E-07	-1.15E-04
FE(N,HELIUM 3)	-3.86E-05	-1.65E-05	-6.11E-06	-2.69E-07	-5.24E-08	-6.11E-05
FE(N,ALPHA)	-6.74E-03	-3.13E-03	-1.08E-03	-8.01E-05	-2.21E-05	-1.11E-02
FE INELASTIC	-1.14E-01	-6.21E-02	-2.50E-02	-2.61E-03	-6.45E-04	-2.04E-01
U238 ELASTIC	-1.04E-01	-7.10E-02	-1.94E-02	-1.84E-03	-1.84E-04	-1.04E-01
U238(N,2N)	-8.06E-02	-5.16E-03	-5.54E-02	-5.54E-03	-5.54E-04	-8.06E-02
U238(N,N')ALPHA	-5.38E-02	-1.57E-03	-5.54E-02	-5.54E-03	-5.54E-04	-5.38E-02
U238 TOTAL FISSION	-6.90E-02	-1.08E-03	-6.71E-02	-6.71E-03	-6.71E-04	-6.90E-02
U238(N,GAMMA)	-1.72E-03	-1.08E-03	-3.19E-03	-3.19E-04	-3.19E-05	-1.72E-03
U238 TOTAL INELASTIC	-4.53E-02	-8.75E-03	-5.41E-02	-5.41E-03	-5.41E-04	-4.53E-02
U235 ELASTIC	-1.35E-03	-5.15E-06	-1.35E-03	-1.35E-04	-1.35E-05	-1.35E-03
U235(N,2N)	-1.98E-04	-1.17E-05	-2.09E-04	-2.09E-05	-2.09E-06	-1.98E-04
U235(N,N')ALPHA	-2.69E-05	-7.31E-07	-2.76E-05	-2.76E-06	-2.76E-07	-2.69E-05
U235 TOTAL FISSION	4.39E-04	2.22E-03	2.66E-04	2.66E-05	2.66E-06	4.39E-04
U235(N,GAMMA)	-1.82E-05	-1.60E-05	-3.51E-05	-3.51E-06	-3.51E-07	-1.82E-05
U235 TOTAL INELASTIC	-1.96E-04	-3.53E-05	-2.32E-04	-2.32E-05	-2.32E-06	-1.96E-04

SENSITIVITY COEFFICIENTS OF FE HEATING, TO L16, U219, U235
FE, L17, F19 PARTIAL CROSS SECTIONS, FISSIONLE INTEGRAL EXPERIMENT

	ZONE 1	ZONE 5	ZONE 7	ZONE 9	ZONE 11	SUM
L16 ELASTIC	1.21E-03	-6.91E-04	-8.18E-04	1.64E-04	1.29E-05	-1.11E-04
L16(N,2N)ALPHA	-1.11E-04	-9.69E-05	-7.77E-05	-1.06E-06	-1.49E-07	-2.87E-04
L16(N,N')ALPHA	-5.94E-04	-6.17E-04	-6.32E-04	-1.41E-05	-8.46E-07	-1.88E-03
L16(N,GAMMA)	-7.86E-06	-3.44E-08	-4.40E-08	-1.20E-08	-1.72E-09	-1.75E-07
L16(N,PROTON)	-4.66E-05	-1.90E-05	-2.01E-05	-3.14E-06	-1.92E-07	-8.90E-05
L16(N,ALPHA)T	-4.66E-03	-5.50E-04	-6.91E-04	-2.59E-04	-4.15E-05	-2.61E-03
L16 INELASTIC	-5.96E-04	-6.43E-04	-6.38E-04	-1.18E-05	-8.37E-07	-1.89E-03
L17 ELASTIC	1.61E-02	-1.00E-02	-1.31E-02	2.35E-03	1.85E-04	-5.24E-03
L17(N,2N)	-4.74E-04	-4.11E-04	-3.31E-04	-4.97E-06	-6.57E-07	-1.22E-03
L17(N,2N)ALPHA	-1.01E-03	-6.16E-04	-4.77E-04	-1.86E-05	-2.12E-06	-2.12E-03
L17(N,ALPHA)T	-1.93E-03	-5.52E-03	-5.39E-03	9.25E-05	2.51E-06	-1.52E-02
L17(N,GAMMA)	-1.00E-06	-5.56E-07	-6.33E-07	-1.82E-07	-2.79E-08	-2.49E-06
L17(N,DEUTERIUM)	-1.76E-04	-2.01E-04	-1.60E-04	9.22E-06	-5.83E-07	-7.47E-04
L17 INELASTIC	-4.87E-03	-7.41E-03	-7.30E-03	1.15E-04	8.35E-06	-1.95E-02
F19 ELASTIC	2.27E-02	-8.21E-03	-1.00E-02	2.92E-03	2.27E-04	7.52E-03
F19(N,2N)	-1.00E-03	-9.10E-04	-6.61E-04	-7.02E-06	-1.01E-06	-2.66E-03
F19(N,N')ALPHA	-8.69E-04	-9.70E-04	-8.05E-04	3.09E-06	-7.68E-07	-2.64E-03
F19(N,N')PROTON	-5.12E-04	-2.88E-04	-1.78E-04	-6.63E-06	-4.86E-07	-7.55E-04
F19(N,GAMMA)	-1.79E-05	-1.17E-05	-9.16E-06	3.55E-05	-7.26E-07	-4.32E-05
F19(N,PROTON)	-1.19E-03	-5.42E-04	-5.34E-04	-6.22E-05	-4.04E-06	-2.34E-03
F19(N,DEUTERIUM)	-5.61E-04	-2.88E-04	-2.49E-04	1.90E-05	-2.28E-06	-1.12E-03
F19(N,TRITUM)	-3.37E-04	-1.77E-04	-1.50E-04	1.03E-05	-6.61E-07	-6.75E-04
F19(N,ALPHA)	-3.02E-03	-1.19E-03	-1.32E-03	2.16E-04	-3.68E-05	-5.76E-03
F19 INELASTIC	-1.59E-02	-1.55E-02	-1.55E-02	-2.49E-04	-8.81E-06	-4.71E-02
FE ELASTIC	6.48E-02	-3.80E-05	-6.50E-03	5.32E-04	6.27E-05	5.89E-02
FE(N,2N)	-4.21E-03	-4.08E-03	-3.04E-03	-1.49E-05	-1.05E-06	-1.13E-02
FE(N,N')ALPHA	-8.44E-05	-1.32E-05	-2.30E-05	-1.42E-07	-9.57E-09	-1.41E-04
FE(N,N')PROTON	-6.95E-04	-3.77E-04	-2.73E-04	-1.62E-06	-1.23E-07	-1.35E-03
FE(N,GAMMA)	-1.07E-04	-4.68E-05	-4.13E-05	-8.55E-06	-1.30E-06	-2.03E-04
FE(N,PROTON)	-5.35E-03	-1.61E-03	-1.23E-03	-2.03E-05	-1.62E-06	-8.21E-03
FE(N,DEUTERIUM)	-5.96E-04	-2.02E-04	-1.45E-04	1.31E-06	-9.81E-08	-9.45E-04
FE(N,TRITUM)	-1.31E-05	-4.46E-06	-3.03E-06	-2.15E-08	-1.52E-09	-2.08E-05
FE(N,HELIUM 3)	-9.74E-06	-3.28E-06	-2.23E-06	-1.60E-08	-1.13E-09	-1.53E-05
FE(N,ALPHA)	-1.53E-03	-4.87E-04	-3.70E-04	-5.19E-06	-4.13E-07	-2.39E-03
FE INELASTIC	-2.12E-02	-9.25E-03	-8.98E-03	-1.20E-04	-7.95E-06	-3.95E-02
U238 ELASTIC	1.53E-01	3.74E-03	1.74E-03	1.56E-01		
U238(N,2N)	-2.00E-02	-3.05E-03	-3.10E-02			
U238(N,3N)	-2.84E-02	-9.17E-04	-2.93E-02			
U238 TOTAL FISSION	1.10E+00	5.54E-02	1.16E+00			
U238(N,GAMMA)	-2.54E-03	-1.01E-03	-3.55E-03			
U238 TOTAL INELASTIC	-8.94E-02	-9.08E-03	-9.85E-02			
U235 ELASTIC	1.11E-03	2.75E-05	1.14E-03			
U235(N,2N)	-9.19E-05	-7.94E-06	-9.99E-05			
U235(N,3N)	-1.53E-05	-4.37E-07	-1.57E-05			
U235 TOTAL FISSION	5.15E-03	2.20E-03	7.35E-03			
U235(N,GAMMA)	-2.53E-05	-1.16E-05	-3.69E-05			
U235 TOTAL INELASTIC	-3.12E-04	-3.45E-05	-3.46E-04			

TABLE 10

TABLE 11

	ZONE 1	ZONE 8	SUM
U238 ELASTIC	-1.31E-01	-1.57E-01	-1.32E-01
U238(N,2N)	-8.24E-02	-1.70E-01	-8.41E-02
U238(N,3N)	-5.48E-02	-6.56E-04	-5.47E-02
U238 TOTAL FISSION	-9.19E-02	-9.29E-02	-9.29E-02
U238(N,GAMMA)	-1.57E-03	-2.55E-04	-1.83E-03
U238 TOTAL INELASTIC	-3.71E-02	-1.31E-03	-3.85E-02
U235 ELASTIC	-9.56E-04	-1.15E-05	-9.68E-04
U235(N,2N)	-2.04E-04	-4.00E-06	-2.08E-04
U235(N,3N)	-2.69E-05	-3.09E-07	-2.72E-05
U235 TOTAL FISSION	-3.98E-04	2.63E-04	-1.35E-04
U235(N,GAMMA)	-1.78E-05	-3.02E-06	-2.08E-05
U235 TOTAL INELASTIC	-1.59E-04	-5.16E-06	-1.64E-04

SENSITIVITY COEFFICIENTS OF F19 HEATING, TO L16, U21B, U235
FE, L17, F19 PARTIAL CROSS SECTIONS, FISSION INTEGRAL EXPERIMENT

	ZONE 1	ZONE 5	ZONE 7	ZONE 9	ZONE 11	SUM
L16 ELASTIC	-5.31E-04	-2.00E-03	-1.13E-03	5.43E-04	1.17E-05	-1.16E-03
L16(N,2N)ALPHA	1.24E-04	1.10E-04	2.61E-05	2.64E-05	4.79E-06	4.97E-04
L16(N,N')ALPHA	-8.57E-04	-5.92E-04	4.10E-04	4.97E-05	1.16E-05	-1.82E-04
L16(N,GAMMA)	-1.15E-06	-2.04E-06	-2.60E-06	-1.46E-06	-1.88E-07	-9.06E-06
L16(N,PROTON)	-1.27E-04	-6.75E-05	3.77E-05	-8.45E-06	-1.38E-06	-2.42E-04
L16(N,DEUTERIUM)	-8.70E-02	-5.47E-02	-6.70E-02	-2.76E-02	-4.87E-03	-2.42E-01
L16 INELASTIC	-8.67E-04	-5.99E-04	-4.34E-04	5.02E-05	1.17E-05	-1.84E-03
L17 ELASTIC	-1.42E-02	-3.33E-02	1.94E-02	7.70E-03	4.81E-04	-5.80E-02
L17(N,2N)	1.35E-03	4.57E-04	1.95E-04	1.10E-04	1.99E-05	2.85E-03
L17(N,2N)ALPHA	1.83E-03	4.86E-04	8.70E-05	1.27E-04	2.14E-05	2.56E-03
L17(N,N')ALPHA	-8.95E-03	-5.73E-03	-4.86E-03	4.51E-04	1.07E-04	-1.82E-02
L17(N,GAMMA)	-5.01E-05	-3.37E-05	-4.19E-05	-1.72E-05	-3.12E-06	-1.46E-04
L17(N,PROTON)	-1.12E-03	-5.01E-04	-2.41E-04	-2.93E-05	4.17E-06	-1.90E-03
L17 INELASTIC	-1.05E-02	-7.65E-03	-5.34E-03	7.97E-04	1.66E-04	-2.25E-02
F19 ELASTIC	-2.07E-02	-4.29E-02	-4.19E-02	1.63E-02	1.63E-03	-8.46E-02
F19(N,2N)	2.71E-03	7.47E-04	1.32E-04	1.90E-04	3.18E-05	3.87E-03
F19(N,N')ALPHA	-1.54E-03	-8.94E-04	-6.19E-04	4.96E-05	1.32E-05	-2.99E-03
F19(N,N')PROTON	-3.39E-04	-1.98E-04	-1.24E-04	7.80E-06	1.93E-06	-6.51E-04
F19(N,GAMMA)	-1.93E-03	-1.35E-03	-1.02E-03	-4.09E-04	-8.69E-05	-4.80E-03
F19(N,PROTON)	-3.86E-03	-1.57E-03	-8.73E-04	1.57E-04	-2.49E-05	-6.59E-03
F19(N,DEUTERIUM)	-1.56E-03	-7.47E-04	-3.84E-04	-5.39E-05	-8.03E-06	-2.77E-03
F19(N,TRITON)	-5.68E-04	-4.53E-04	-2.29E-04	3.05E-05	4.47E-06	-6.92E-03
F19(N,ALPHA)	-7.70E-03	-4.20E-03	-2.45E-03	-5.57E-04	-9.20E-05	-1.52E-02
F19 INELASTIC	-1.08E-02	-1.84E-02	-1.13E-02	3.68E-03	5.71E-04	-1.61E-02

	ZONE 2	ZONE 4	ZONE 6	ZONE 10	ZONE 12	SUM
FE ELASTIC	-1.05E-02	-1.39E-02	-2.21E-02	5.20E-03	1.75E-03	-1.95E-02
FE(N,2N)	1.68E-02	5.42E-03	1.15E-03	2.67E-04	4.77E-05	2.37E-02
FE(N,N')ALPHA	-5.93E-05	-3.25E-05	-1.81E-05	3.29E-07	7.30E-08	-1.10E-04
FE(N,N')PROTON	-6.77E-04	-3.72E-04	-2.16E-04	4.52E-06	1.06E-06	-1.26E-03
FE(N,GAMMA)	-7.10E-03	-4.69E-03	-4.46E-03	-1.01E-03	-1.56E-04	-1.74E-02
FE(N,PROTON)	-1.05E-02	-5.19E-03	-2.58E-03	-1.09E-04	-1.77E-05	-1.84E-02
FE(N,DEUTERIUM)	-3.37E-05	-6.13E-04	-2.74E-04	-7.79E-06	-1.2E-06	-2.11E-05
FE(N,TRITON)	-3.37E-05	-1.33E-05	-5.58E-06	-1.36E-07	-1.84E-08	-5.27E-05
FE(N,HELIUM 3)	-2.47E-05	-9.75E-06	4.11E-06	-1.01E-07	-1.37E-08	-3.07E-05
FE(N,ALPHA)	-3.10E-03	-1.52E-03	-7.41E-04	-2.76E-05	-4.36E-06	-5.47E-03
FE INELASTIC	-1.30E-02	-8.21E-03	-6.55E-03	3.50E-04	1.04E-04	-2.71E-02

	ZONE 1	ZONE 8	SUM
U238 ELASTIC	1.17E-01	2.11E-03	1.19E-01
U238(N,2N)	8.86E-03	2.57E-03	1.14E-02
U238(N,N')ALPHA	2.76E-02	1.83E-03	2.95E-02
U238 TOTAL FISSION	2.01E-01	2.08E-02	2.22E-01
U238(N,GAMMA)	-9.48E-02	-1.03E-01	-1.97E-01
U238 TOTAL INELASTIC	4.55E-04	-1.22E-03	-7.67E-04
U235 ELASTIC	8.01E-04	2.59E-05	8.26E-04
U235(N,2N)	8.18E-04	3.63E-05	8.55E-04
U235(N,N')ALPHA	1.70E-05	9.02E-07	1.79E-05
U235 TOTAL FISSION	7.89E-01	2.90E-01	1.08E-00
U235(N,GAMMA)	-1.21E-03	-1.23E-03	-2.44E-03
U235 TOTAL INELASTIC	4.24E-05	-2.23E-06	4.01E-05

TABLE 13

SENSITIVITY COEFFICIENTS OF F19 HEATING, TO L16, U21B, U235
FE, L17, F19 PARTIAL CROSS SECTIONS, FISSION INTEGRAL EXPERIMENT

	ZONE 1	ZONE 5	ZONE 7	ZONE 9	ZONE 11	SUM
L16 ELASTIC	-6.06E-03	-2.00E-03	-7.12E-04	-2.40E-04	-5.85E-05	-9.97E-03
L16(N,2N)ALPHA	1.26E-04	1.42E-04	1.40E-04	3.68E-05	-9.41E-06	1.89E-03
L16(N,N')ALPHA	-7.97E-03	-1.13E-03	-1.00E-03	-1.15E-04	-8.61E-05	-1.26E-02
L16(N,GAMMA)	-1.76E-07	-1.79E-07	-8.35E-08	-2.90E-08	-8.45E-09	-6.79E-07
L16(N,PROTON)	-2.18E-04	1.07E-04	3.90E-05	-1.19E-05	-3.47E-06	-4.00E-04
L16(N,DEUTERIUM)	-4.21E-03	-2.46E-03	-1.40E-03	-6.10E-04	-1.75E-04	-8.97E-03
L16 INELASTIC	-8.06E-03	-1.16E-03	-1.09E-03	-3.18E-04	-8.69E-05	-1.37E-02
L17 ELASTIC	-1.04E-01	-1.42E-02	-1.16E-02	-3.76E-03	-9.20E-04	-1.56E-01
L17(N,2N)	-5.35E-03	-1.88E-03	-5.96E-04	-1.50E-04	-8.44E-05	-8.94E-03
L17(N,2N)ALPHA	-8.48E-03	-2.85E-03	-8.64E-04	-2.20E-04	-5.47E-05	-1.25E-02
L17(N,N')ALPHA	-7.91E-02	-1.00E-02	-1.00E-02	-2.81E-03	-7.60E-04	-1.23E-01
L17(N,GAMMA)	-4.80E-06	-2.36E-06	-1.14E-06	-4.13E-07	-1.92E-07	-8.94E-06
L17(N,PROTON)	-3.03E-03	-1.07E-03	-3.32E-04	-8.65E-05	-2.23E-05	-4.54E-03
L17 INELASTIC	-9.74E-02	-1.76E-02	-1.29E-02	-3.73E-03	-1.02E-03	-1.31E-01
F19 ELASTIC	5.92E-02	4.15E-02	1.91E-02	7.47E-03	2.73E-03	1.31E-01
F19(N,2N)	-6.97E-03	-2.24E-03	-6.64E-04	-1.62E-04	-3.35E-05	-1.01E-02
F19(N,N')ALPHA	2.12E-02	7.67E-03	2.51E-03	7.19E-04	2.36E-04	3.42E-02
F19(N,N')PROTON	2.92E-03	9.35E-04	2.84E-04	7.79E-05	2.55E-05	4.35E-03
F19(N,GAMMA)	-4.66E-05	-3.72E-05	-2.20E-05	-9.39E-06	-3.16E-06	-1.22E-04
F19(N,PROTON)	1.97E-02	7.49E-03	2.56E-03	7.59E-04	2.50E-04	3.09E-02
F19(N,DEUTERIUM)	8.57E-03	2.97E-03	9.49E-04	2.67E-04	8.45E-05	1.29E-02
F19(N,TRITON)	3.61E-03	1.20E-03	3.77E-04	1.05E-04	3.35E-05	5.44E-03
F19(N,ALPHA)	5.77E-02	2.40E-02	8.00E-03	2.77E-03	9.24E-04	9.45E-02
F19 INELASTIC	-2.25E-01	-8.87E-02	-3.17E-02	-9.89E-03	-2.77E-03	-3.58E-01

	ZONE 2	ZONE 4	ZONE 6	ZONE 10	ZONE 12	SUM
FE ELASTIC	-1.10E-01	-1.93E-02	-4.37E-03	-1.57E-04	4.60E-04	-1.42E-01
FE(N,2N)	-7.79E-02	-2.61E-02	-7.86E-03	-5.19E-04	-1.01E-04	-1.13E-01
FE(N,N')ALPHA	-6.33E-04	-2.24E-04	-6.54E-05	-1.11E-05	-8.30E-07	-9.68E-04
FE(N,N')PROTON	-7.19E-03	-2.59E-03	-7.80E-04	-5.17E-05	-1.08E-06	-1.09E-02
FE(N,GAMMA)	-2.18E-04	-1.77E-04	-1.07E-04	-2.03E-05	-6.10E-06	-5.20E-04
FE(N,PROTON)	-2.45E-02	-1.13E-02	-3.94E-03	-3.20E-04	-8.16E-05	-4.01E-02
FE(N,DEUTERIUM)	-3.80E-03	-1.44E-03	-4.40E-04	-3.11E-05	-7.66E-06	-5.73E-03
FE(N,TRITON)	-9.19E-05	-3.13E-05	-9.16E-06	-5.76E-07	-1.22E-07	-1.33E-04
FE(N,HELIUM 3)	-6.72E-05	-2.30E-05	-6.75E-06	-4.27E-07	-9.07E-08	-9.75E-05
FE(N,ALPHA)	-7.80E-03	-3.46E-03	-1.18E-03	-9.41E-05	-2.35E-05	-1.26E-02
FE INELASTIC	-1.32E-01	-6.63E-02	-2.52E-02	-2.45E-03	-5.07E-04	-2.26E-01

	ZONE 1	ZONE 8	SUM
U238 ELASTIC	-2.21E-01	-9.54E-04	-2.21E-01
U238(N,2N)	-9.26E-02	-5.24E-03	-9.78E-02
U238(N,N')ALPHA	-6.12E-02	-1.72E-03	-6.30E-02
U238 TOTAL FISSION	-9.46E-02	4.41E-04	-9.41E-02
U238(N,GAMMA)	-2.86E-03	-1.41E-04	-3.49E-03
U238 TOTAL INELASTIC	-4.57E-02	-6.91E-03	-5.26E-02
U235 ELASTIC	-1.62E-03	-6.69E-06	-1.63E-03
U235(N,2N)	-2.29E-04	-1.20E-05	-2.41E-04
U235(N,N')ALPHA	-3.05E-05	-8.01E-07	-3.13E-05
U235 TOTAL FISSION	-6.71E-05	1.89E-03	1.82E-01
U235(N,GAMMA)	-2.24E-05	-1.65E-05	-3.89E-05
U235 TOTAL INELASTIC	-2.05E-04	-2.82E-05	-2.33E-04

TABLE 12

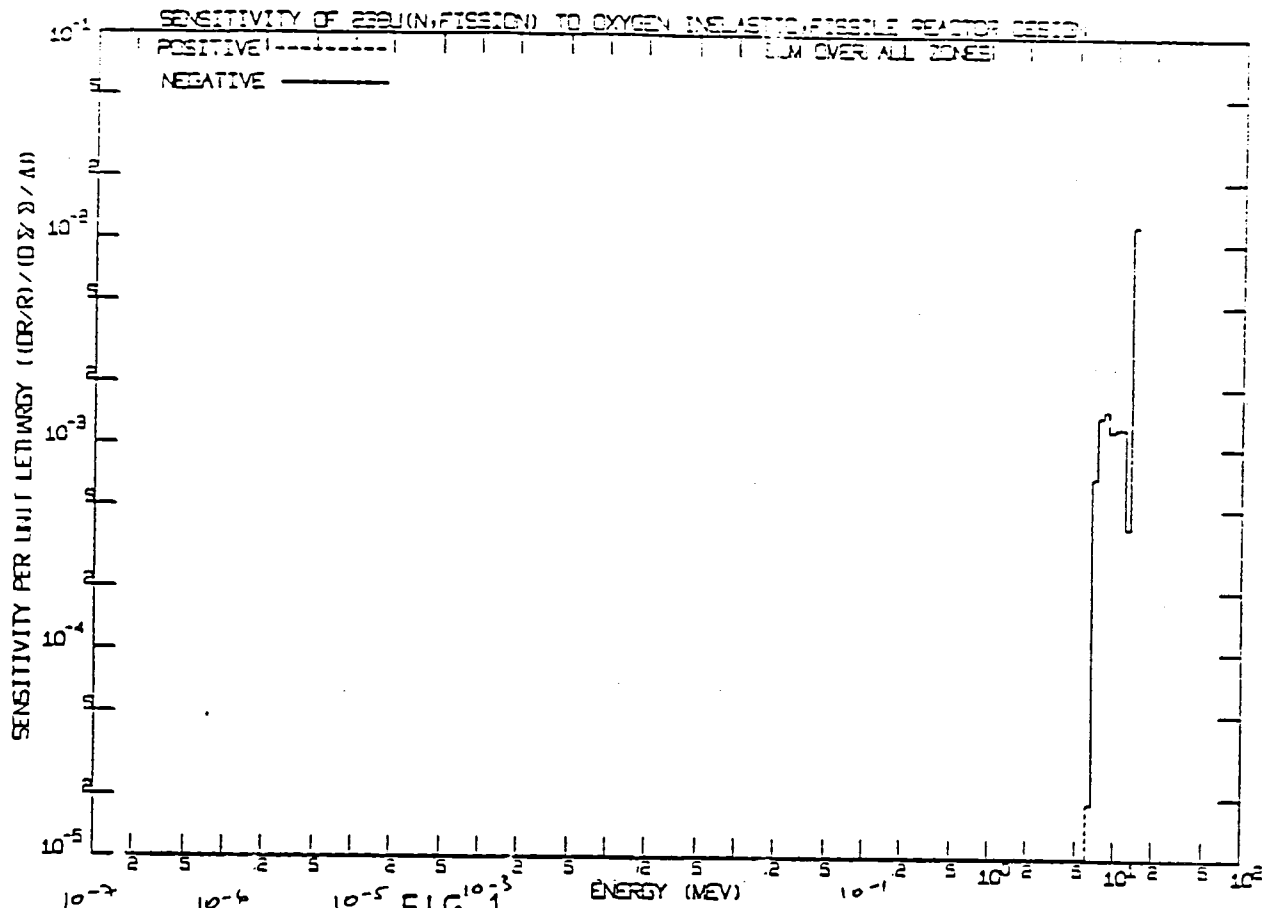


FIG. 1 sensitivity of $^{239}\text{U}(n, f)$ to ^{16}O inelastic in FR.

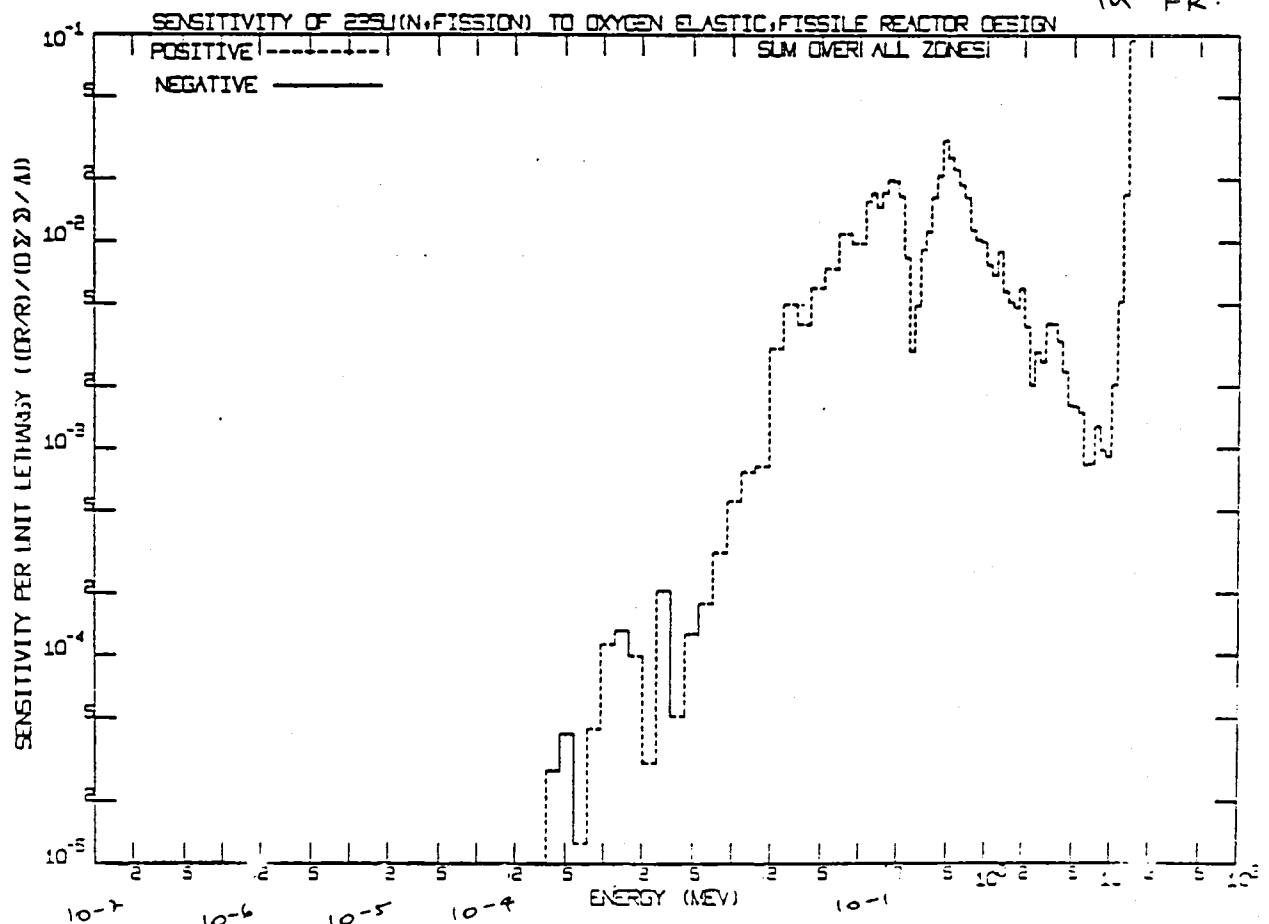


FIG. 2

sensitivity of $^{235}\text{U}(n, f)$ to ^{16}O elastic in FR.

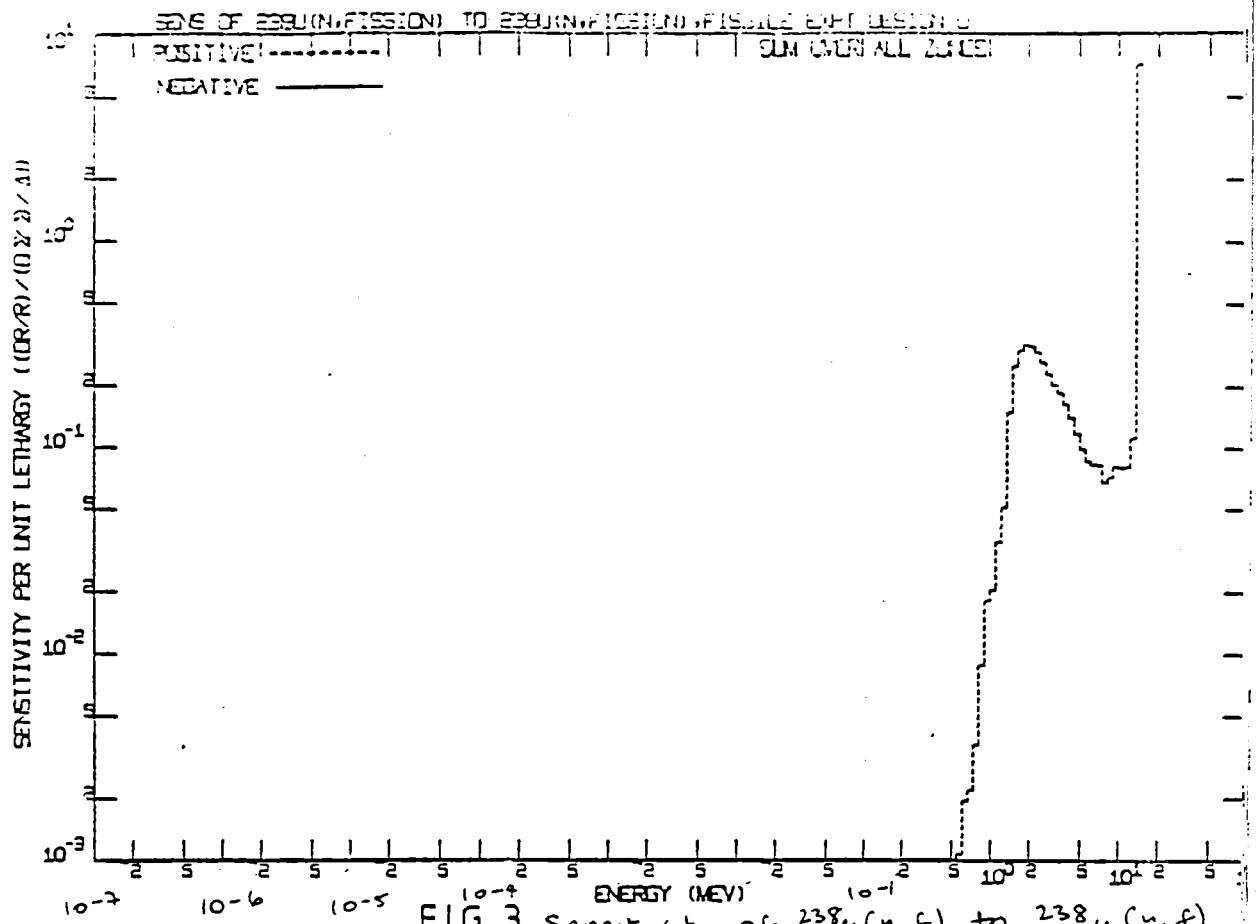


FIG. 3 Sensitivity of $^{238}\text{U}(n,f)$ to $^{238}\text{U}(n,f)$ in design d), FR.

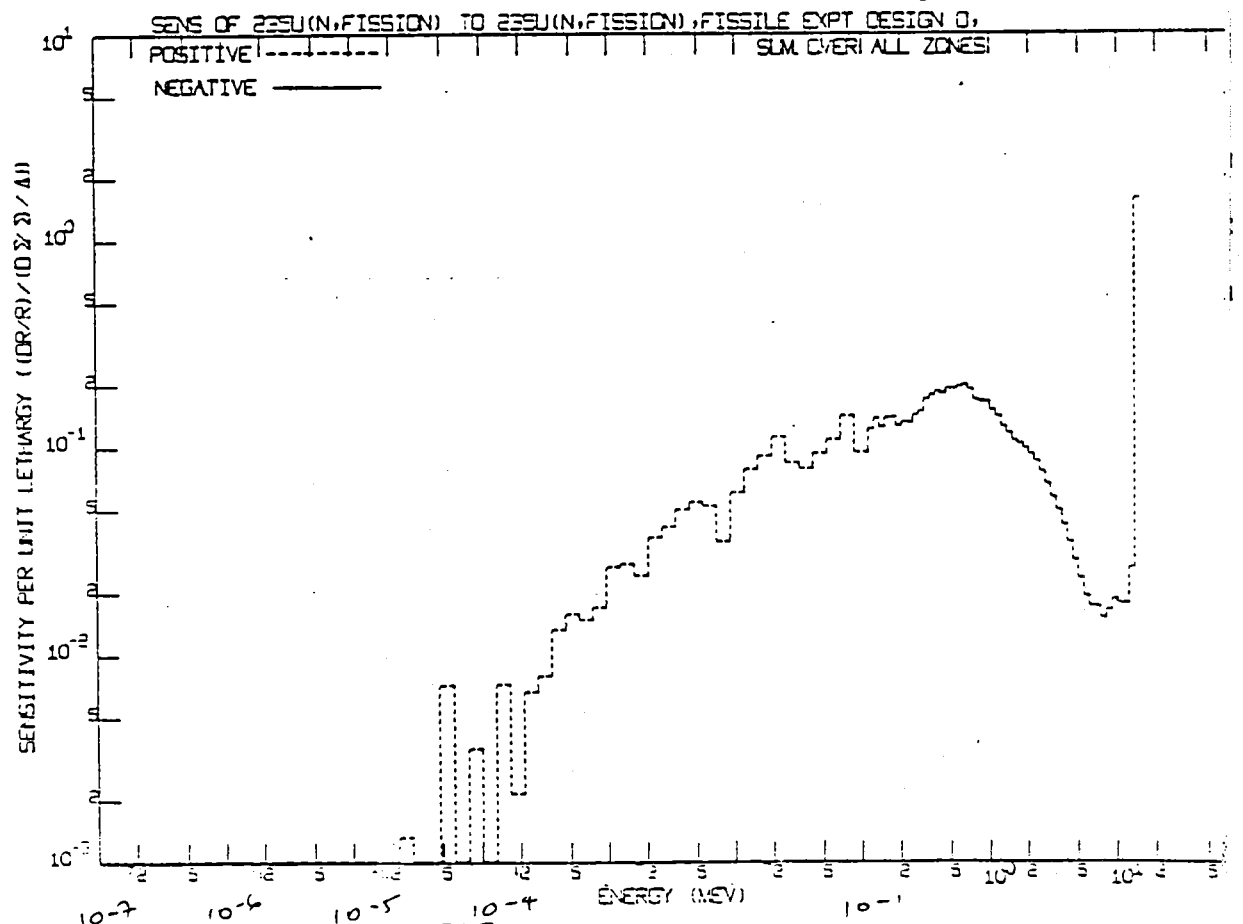


FIG. 4

Sensitivity of $^{235}\text{U}(n,f)$ to $^{235}\text{U}(n,f)$ in design d), FR.

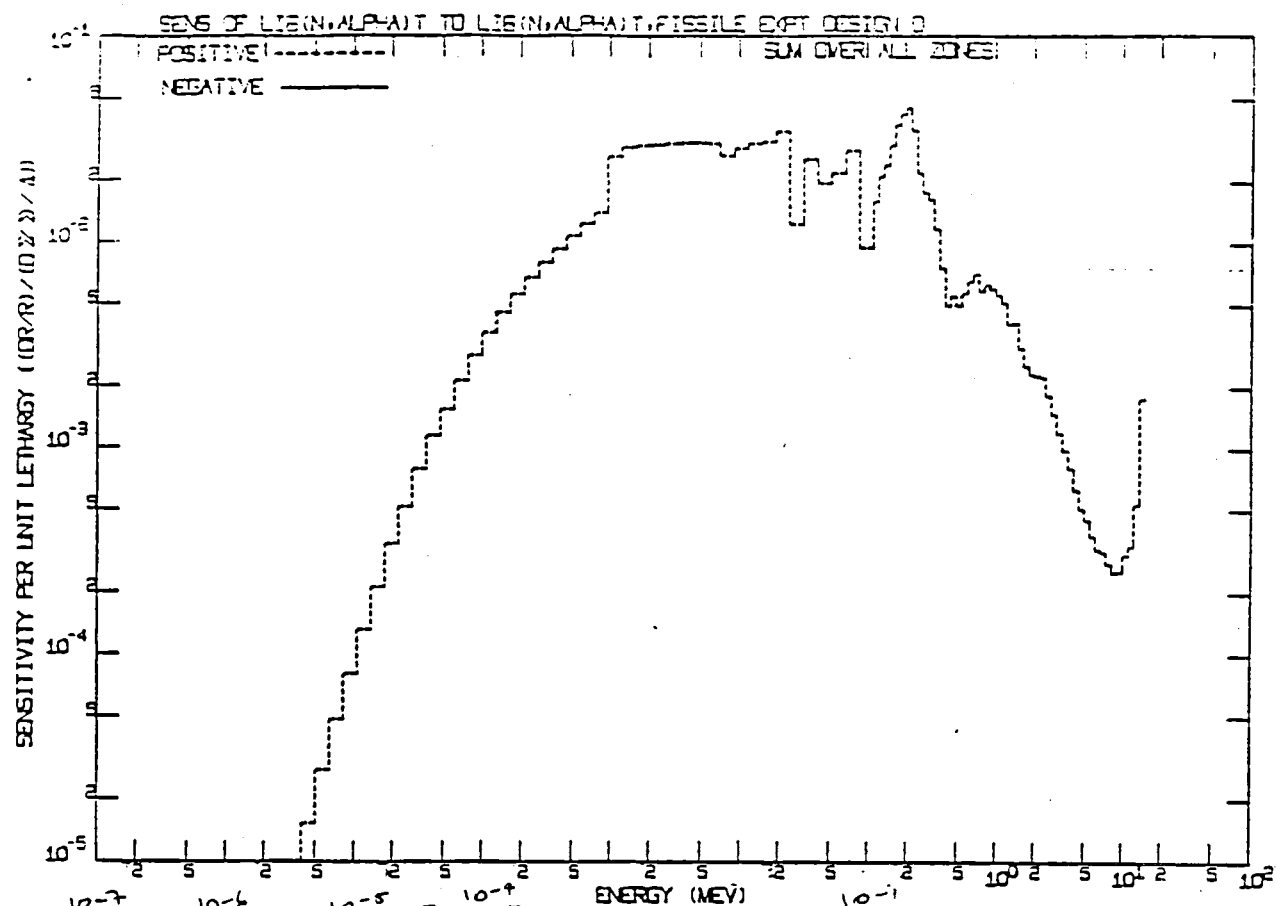


FIG. 5 sensitivity of $\phi Li(n,\alpha)T$ to $\phi Li(n,\alpha)T$ in design d), FR.

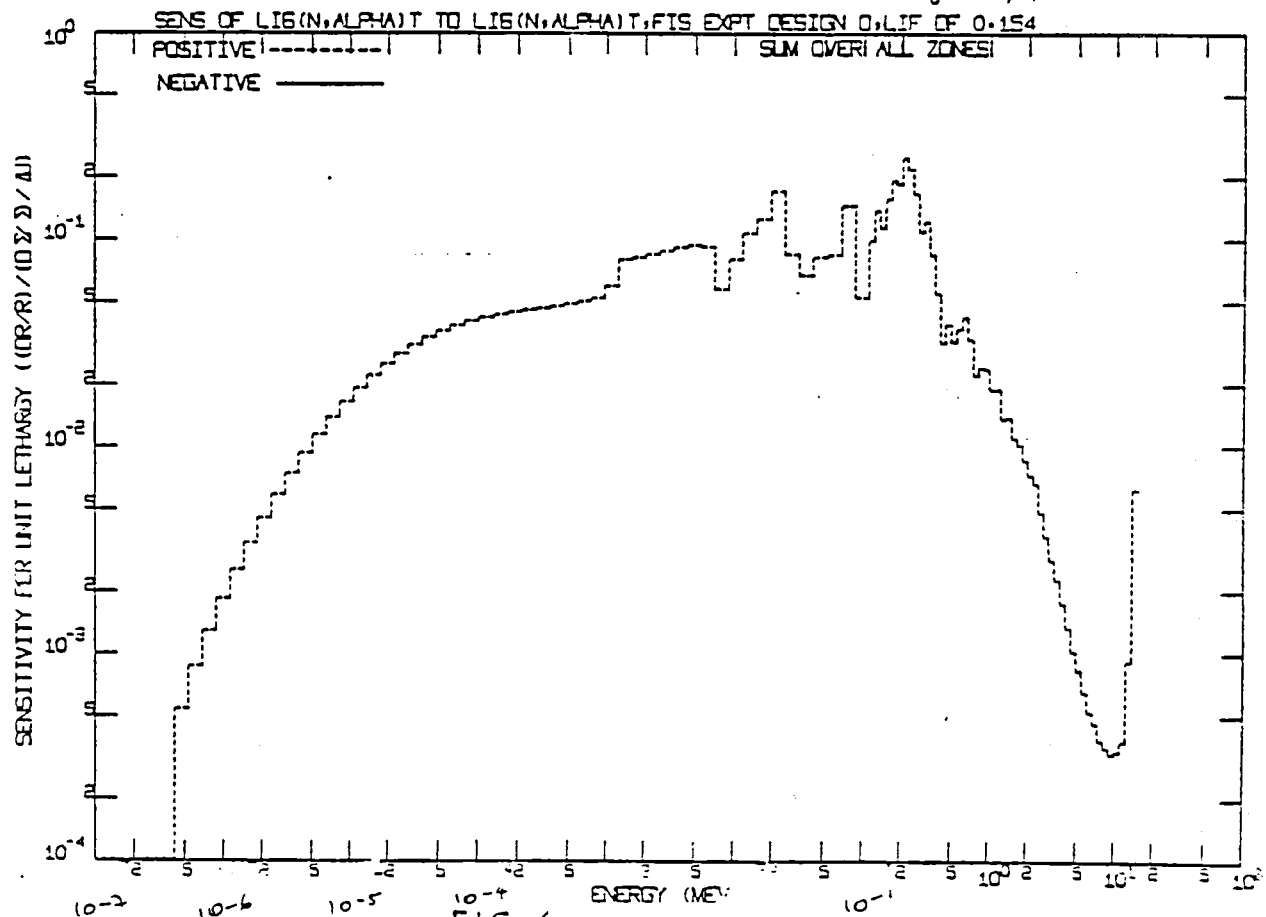


FIG. 6 sensitivity of $\phi Li(n,\alpha)T$ to $\phi Li(n,\alpha)T$, design d), 90% of the LIF is removed.

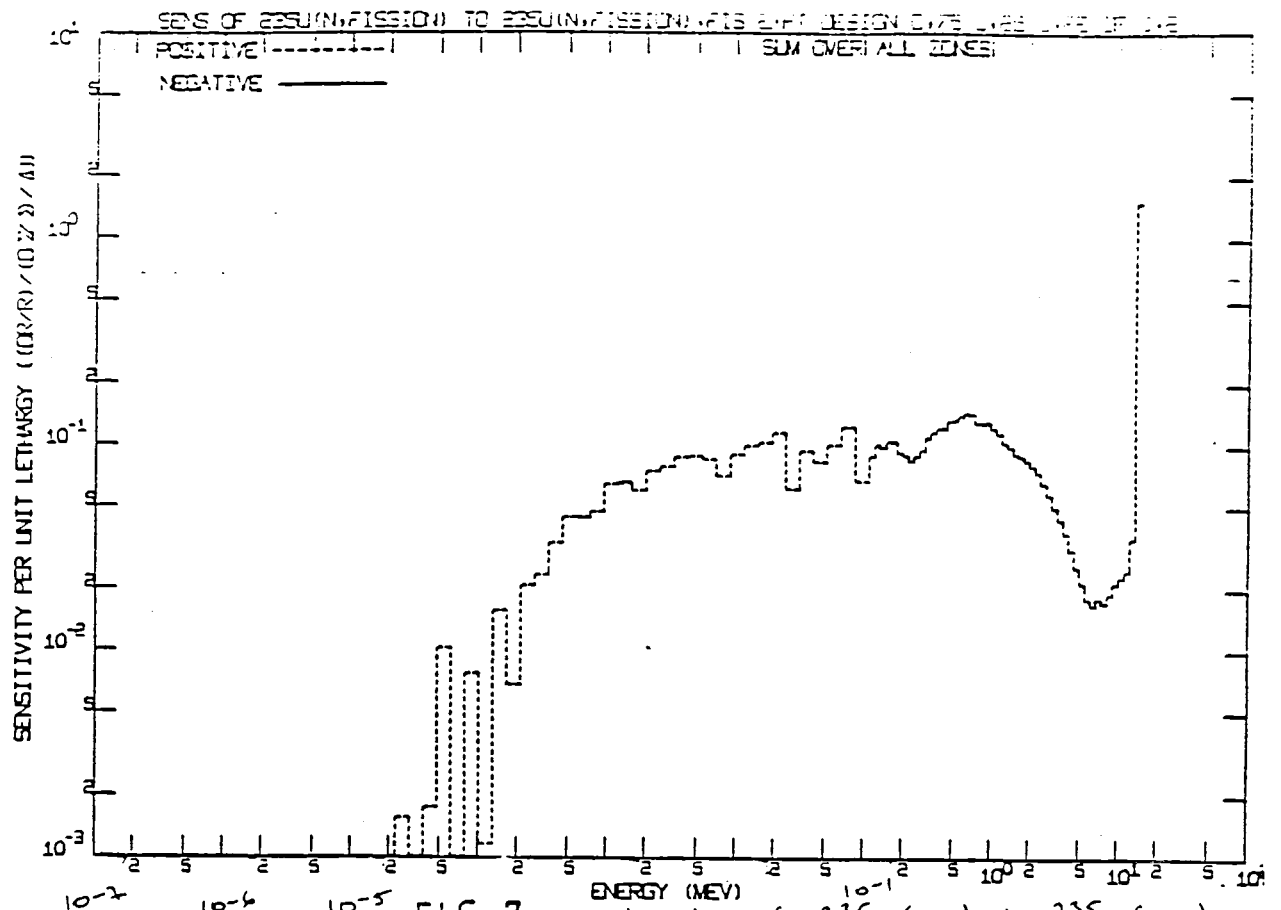


FIG. 7 sensitivity of $^{235}\text{U}(n,f)$ to $^{235}\text{U}(n,f)$,
design d), 80% of reflector removed,
25% of uranium in centre of design

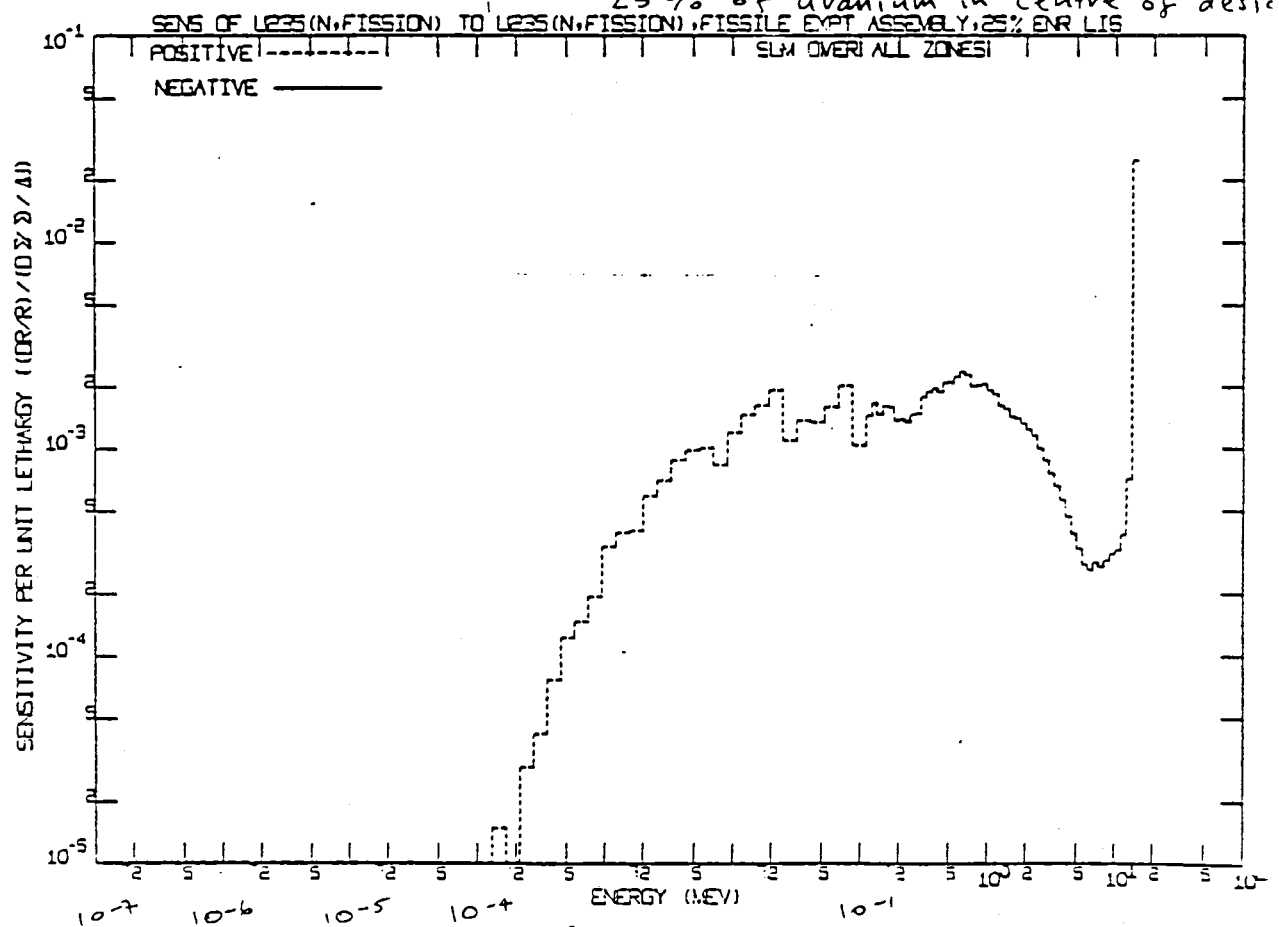


FIG. 8 sensitivity of $^{235}\text{U}(n,f)$ to $^{235}\text{U}(n,f)$
FIE, ^{235}U isotope is 25%
enriched.

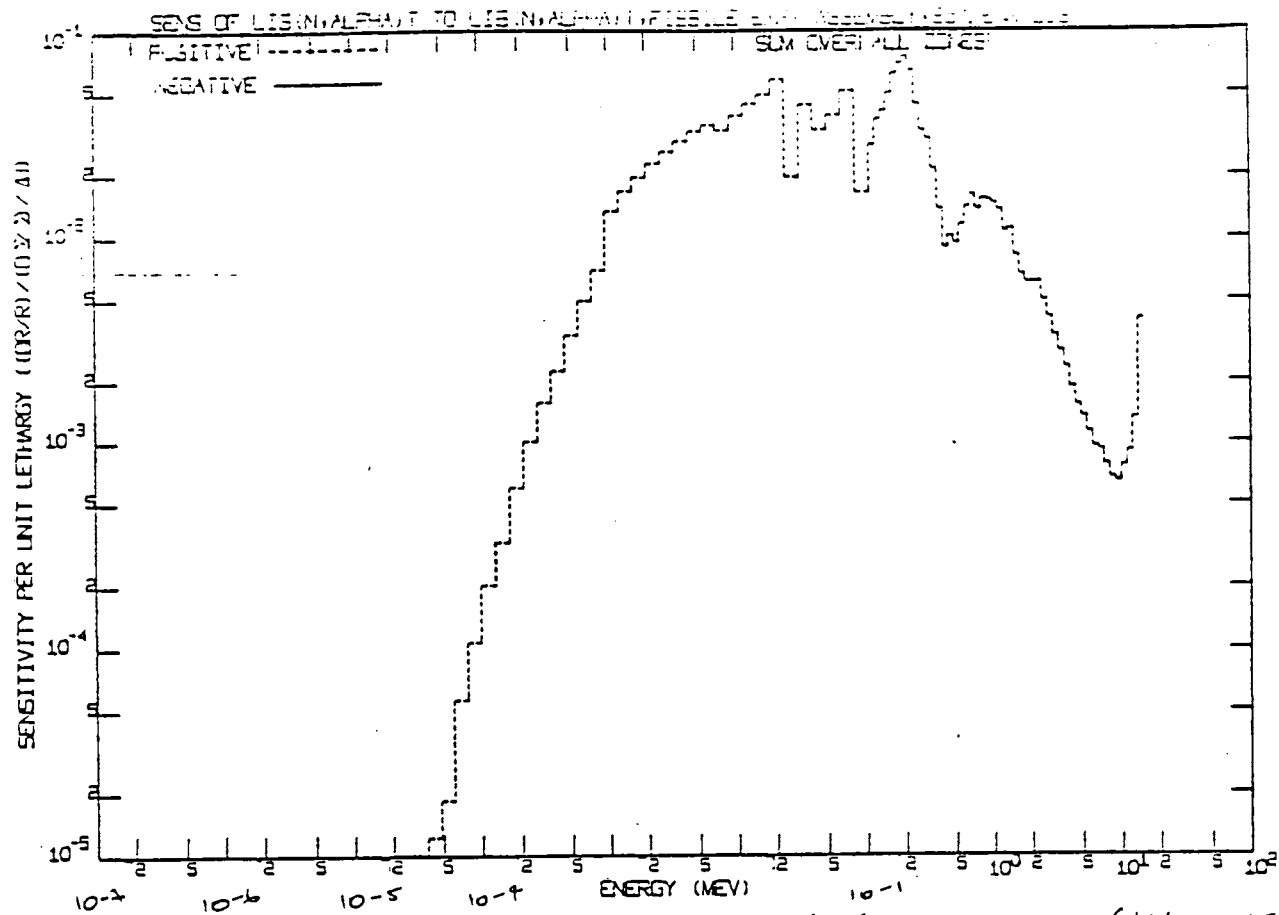


FIG. 9 sensitivity of ${}^6\text{Li}(\alpha, n)\text{T}$ to ${}^6\text{Li}(\alpha, n)\text{T}$ FIE, ${}^6\text{Li}$ isotope is 25% enriched.

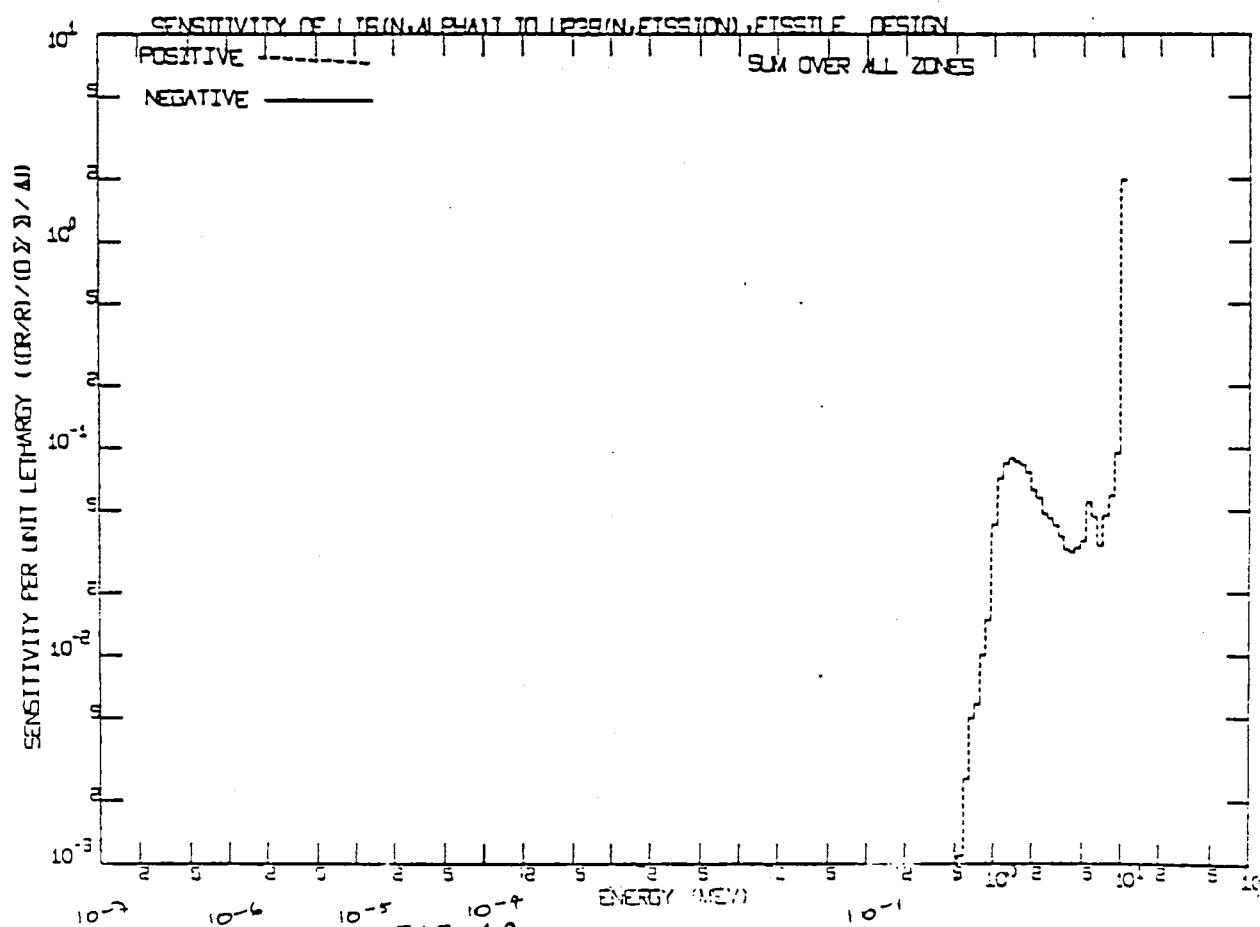


FIG. 10 sensitivity of ${}^6\text{Li}(\alpha, n)\text{T}$ to ${}^{238}\text{U}(\text{n}, \text{f})$, in FR.

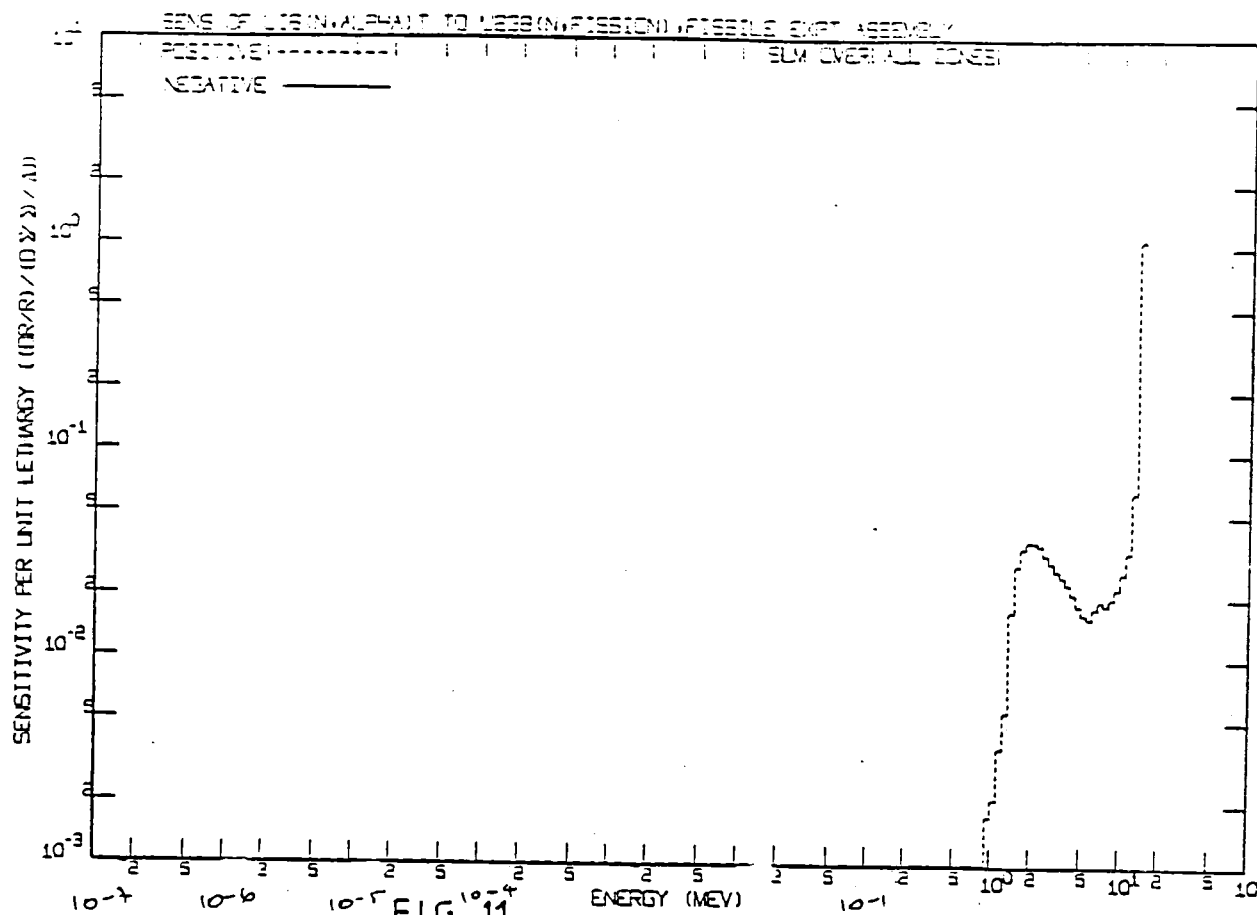


FIG. 11 Sensitivity of ${}^6\text{Li}(n, \alpha)T$ to ${}^{238}\text{U}(n, f)$ in FIE

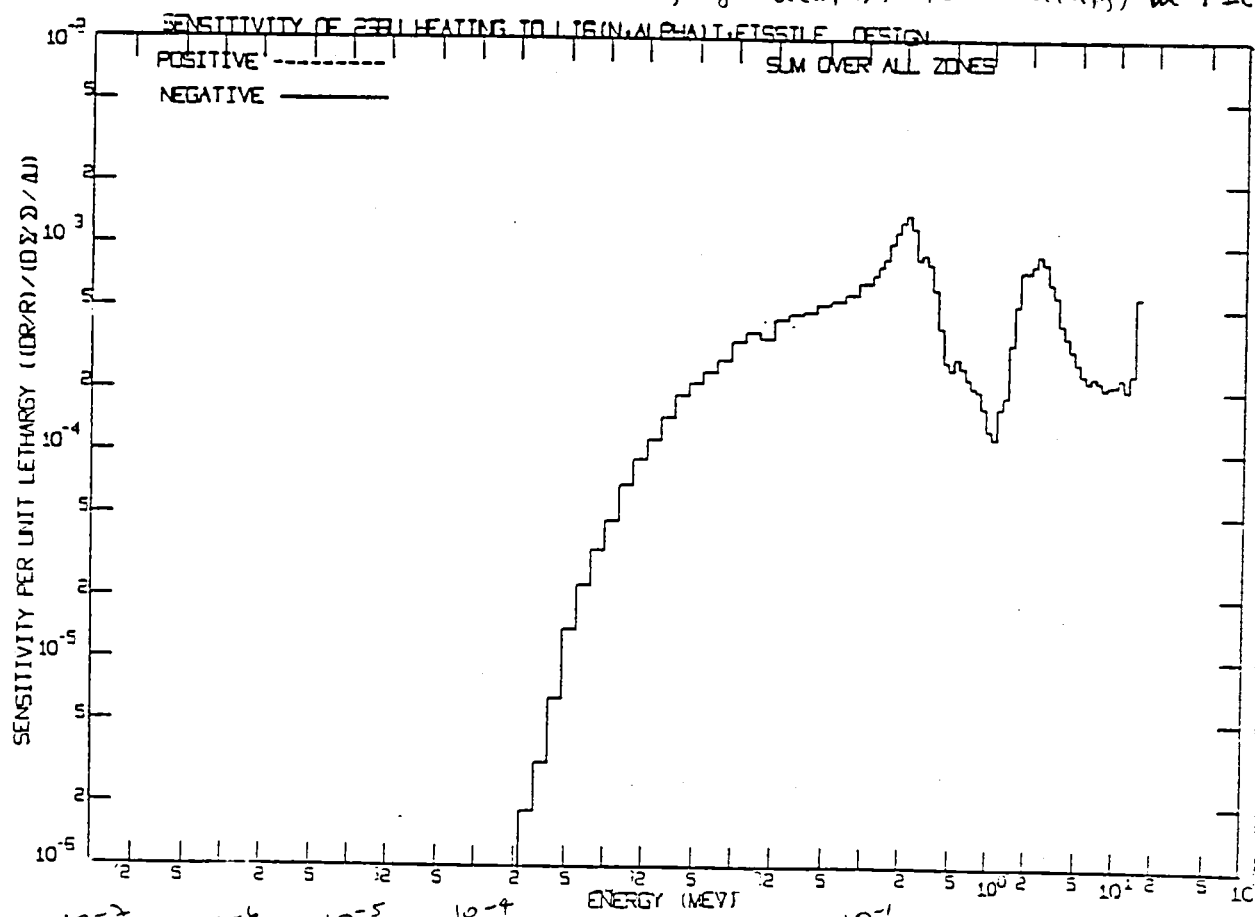
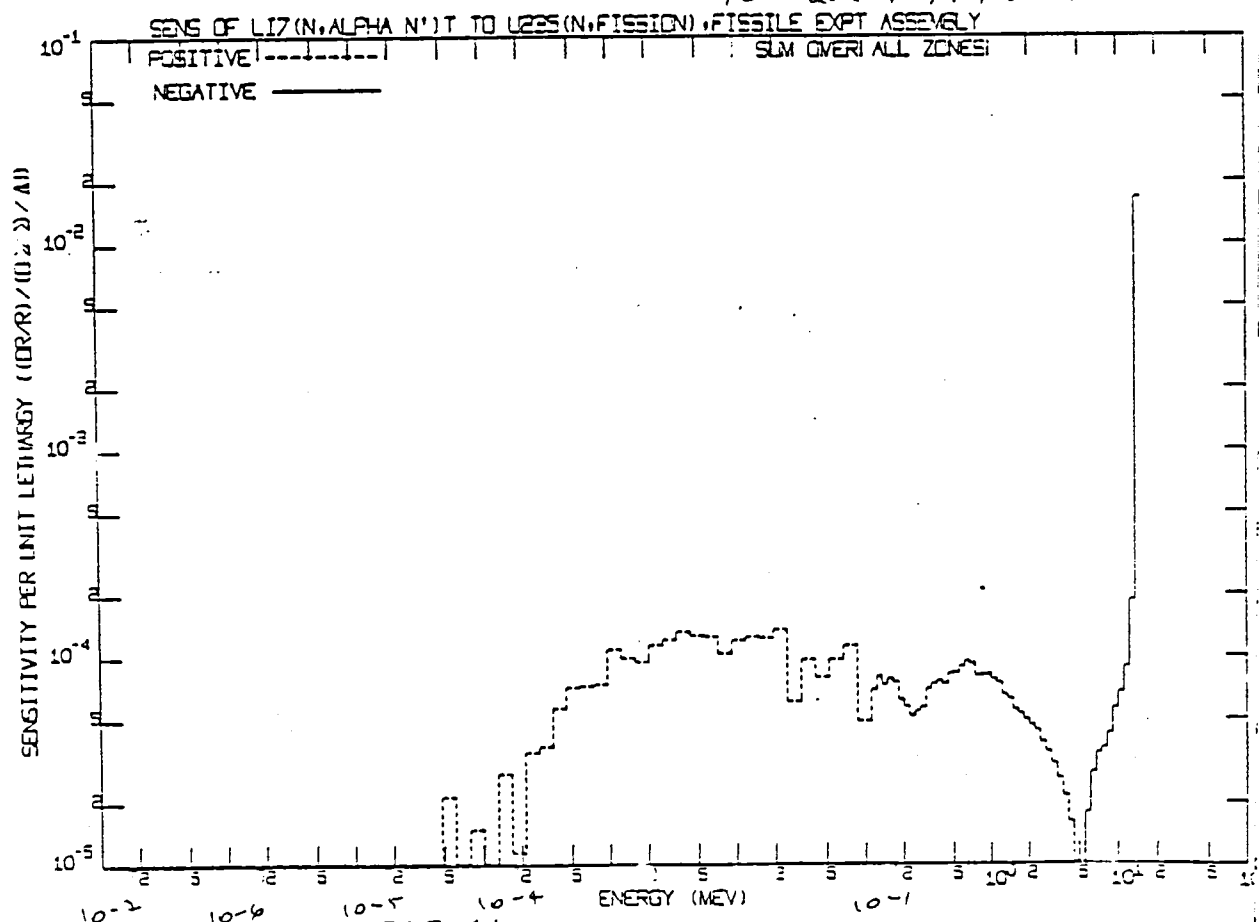
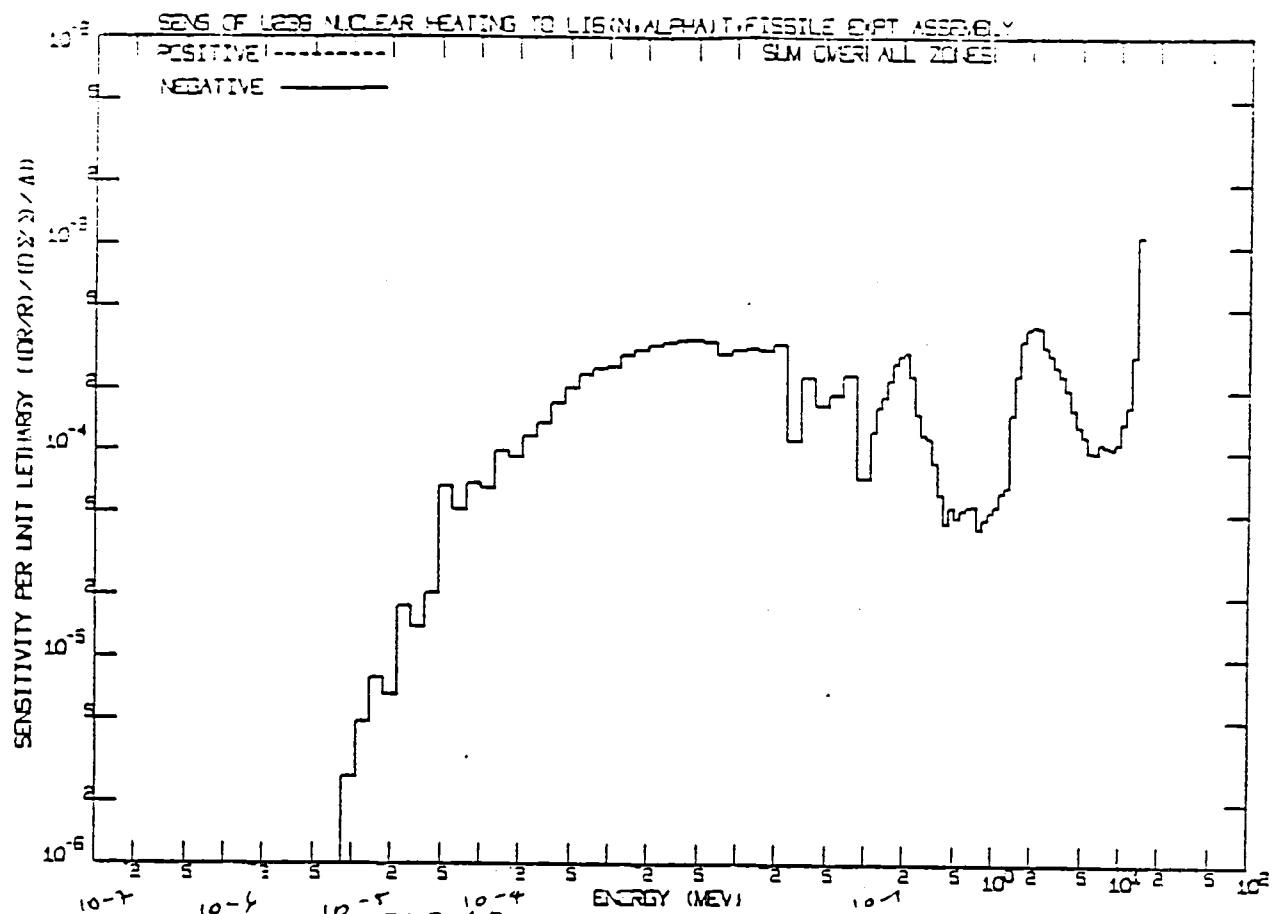


FIG. 12 Sensitivity of ${}^{238}\text{U}$ heating to ${}^6\text{Li}(n, \alpha)T$ in FR.



CHAPTER 9

THEORY AND RESULTS OF UNCERTAINTY ANALYSIS

9.1 INTRODUCTION

Nuclear design calculations have always been limited in their accuracy by computational methods, idealisation of the geometrical arrangement and the inaccuracy in the cross section nuclear data. However, after a period of extensive refinements in calculational methods and computer codes, the main uncertainty is believed to be in the basic nuclear data. One of the main aims of transport sensitivity analysis is to determine the uncertainties in the results of transport calculations due to data uncertainties. Many such studies have been done in the past for fast breeder reactors, including the economic impact of the uncertainties, for example by Greebler (1970) and Usachev et al (1973).

An attempt has been made here to calculate the uncertainties in certain reactor performance parameters due to the uncertainties in the nuclear cross sections used in the transport calculation. Indeed, the estimation of these uncertainties is potentially one of the most useful aspects of the approach to sensitivity studies presented here. The uncertainty analysis is presented for both the non fissile and fissile reactors as well as the two associated integral experiments. The sensitivities of the reactor parameters to the cross sections contained in each of the four design studies are folded (in the manner described next) with the estimated energy dependent cross section uncertainties. These uncertainties are held in the form of covariance matrices, which contain energy group partial cross section uncertainties

and the correlations (a measure of the degree of closeness or linkage between two quantities) between these two partial cross sections. In the parlance of uncertainty work we say that this covariance matrix contains the correlated cross section errors of one partial with respect to that of another partial. It should be made very clear at this point that the correlations used in this work have been estimated. No available correlation files exist to date. The energy group partial errors have either been guessed or obtained from various sources. The uncertainty is calculated for each element or isotope separately. Consider one of the partials making up the element, w , say. Due to the measurement process this partial will be correlated in some way to every other partial in the same element, x , y and z , say, either positively or negatively. Correlations between partials of different isotopes have not been considered in this analysis. The covariance matrix between the partials w and x is then folded with the relevant sensitivity coefficients of the reactor parameter in question to the partials w and x . In this way we obtain the variance of the reactor parameter to the correlated uncertainties due to the partials w and x . This process is repeated between the rest of the isotopes partial cross sections.

The presence of correlations in the cross section uncertainties means that the covariance matrices are non diagonal. The covariance matrices are square of dimension equal to the number of energy groups. The number of such covariance matrices is equal to the sum of the number of elements times the number of partials in each element. Correlations between uncertainties in the same kind of cross section at different energies occurs naturally where the shape of the energy dependence of the cross section is well known but the absolute

normalisation has larger errors. For example capture and fission cross sections in the few hundred Kev have often been measured using $^{10}\text{B}(n,\alpha)$ or $\text{Li6}(n,\alpha)\text{T}$ standard cross sections. At higher energies the ^{235}U fission and (n,p) reactions have been employed. In other cases, energy measurements such as the β decay of ^{252}Cf or 2200 m sec^{-1} are used. Any significant uncertainties in the input data lead to the measured values. Such uncertainties can be reduced by comparing experimental results, if the same standard cross section is used. Moreover, the uncertainty in evaluated cross sections and correlations among the various cross sections measured to the same standards. Several authors have recognised the sensitivity of computer calculations to reactor parameters to the correlation properties of the input data.

The uncertainty analysis, derived and presented in sensitivity theory, deals strictly in terms of the uncertainties in reactor parameters due to uncertainties in the same basic cross section data. This is called a sensitivity analysis (Weisbin 1976) and a mathematical derivation is given next. However, a posteriori uncertainty analysis which includes integral experiments and adds the results of experiments and the calculational modelling to a priori procedure. Together, both approaches can be made of the important sources of uncertainty, which can then be eliminated to avoid overly conservative design procedures. The inter-

and methods are treated as if the results of a new series of correlated differential measurements were to be added to an already existing (ie the a priori) file of differential data. Using a procedure developed by Gandini (1973), the a posteriori set of mean values and covariances for a differential cross section data file can be derived from the a priori file and its covariance matrix by minimising an expectation or distribution function (representing the combined dataset) which represents both differential and integral results. For further mathematical details and discussion see Weisbin (1978).

9.2 BASIC COVARIANCE FORMULATION

In an ideal situation we have the measurements of many different nuclear cross section data sets from laboratories all over the world ; the partial cross section values and the evaluators error information for each partial and over the complete energy range. A reference set of cross section values is obtained from all these different sets (n , say) of cross sections. The reference set is obtained from all the n sets by taking the expectation or average value of the energy dependent cross sections. For the sake of argument we assume that the cross sections are in the same energy group structure from all the laboratories. The reference set is the set of cross section values used in the computational codes ; in this work it was the ENDF/B-4 cross sections.

To estimate the variance in a computed result R , which has been calculated using the reference set, and which has statistical uncertainties in it, a statistical analogue computation can be made to evaluate the following defining

expression for the variance of the parameter R,

$$V_R = E \left[(R_n - E(R_n))^2 \right] \quad (9.1)$$

Here the variance of R, denoted by V_R , is the expectation value of samplings of the value of R (the nth sample having a value R_n) based on the statistical nature of the uncertainties of the data base, and $E(R_n)$ is the average or expectation value of R for all the samples. In computational terms a sampling from uncertainty distributions for the whole cross section data base is made and a transport calculation using the sampled cross sections is carried out to get each R_n . The sampling process consists of picking out the energy dependent cross section value for each partial cross section from a normal distribution having a specified variance V_{σ_i} and an average value $E(\bar{\sigma}_i)$ for the ith cross section element.

In terms of differences between R and the average value $E(R_n)$ resulting from using a particular sampled cross section data base, we have

$$V_R = E(\delta R_n^2) \quad (9.2)$$

$$\text{where } \delta R_n = R_n - E(R_n) \quad (9.3)$$

Rewriting this expression in terms of the R 's being explicit functions of cross sections resulting from the sampling procedure (ie the cross section set for all $\xi_{i,n}$),

$$\delta R_n = R_n(\xi_{i,n}) - E(R_n(\xi_{i,n})) \quad (9.4)$$

where the i subscript refers to and labels the partial cross sections of the cross section set in the nth sample. The energy group index has been suppressed but is implicit in all the following equations. If we assume a linear relationship to exist between the data elements and the result (a condition that was

carefully preserved in the derivation of the sensitivity function), then we can express δR_n in terms of sensitivity coefficients as,

$$\delta R_n = \sum_i \frac{\partial R}{\partial \xi_i} \delta \xi_{i,n} \quad (9.5)$$

where $\delta \xi_{i,n}$ is given as

$$\delta \xi_{i,n} = \xi_{i,n} - E(\xi_{i,n}) \quad (9.6)$$

The variance of R , V_R , being formally obtained as the expectation value of δR^2 can now be written as,

$$V_R = E(\delta R_n)^2 = E \left[\left(\sum_i \frac{\partial R}{\partial \xi_i} \delta \xi_{i,n} \right)^2 \right] \quad (9.7)$$

The partial derivative is equal to the value of $\partial R / \partial \xi_i$ computed with all $\xi_i = E(\xi_{i,n})$, the mean cross section values of all the n cross section sets. Since $\partial R / \partial \xi_i$ and $\partial R / \partial \xi_j$ are constants over the statistical ensemble, ie they are assumed to be invariant with n (the sensitivity profiles are strongly design dependent and so are constant over the design considered) we can put,

$$V_R = \sum_{i,j} E \left[\frac{\partial R}{\partial \xi_i} \frac{\partial R}{\partial \xi_j} \delta \xi_{i,n} \delta \xi_{j,n} \right] \quad (9.8)$$

$$= \sum_{i,j} \frac{\partial R}{\partial \xi_i} \frac{\partial R}{\partial \xi_j} E(\delta \xi_{i,n} \delta \xi_{j,n}) \quad (9.9)$$

The expectation value of the product of $\delta \xi_{i,n}$ and $\delta \xi_{j,n}$ over a joint probability density function $f(\xi_{i,n} \xi_{j,n})$,

$$\begin{aligned} \text{COV}(\xi_{i,n} \xi_{j,n}) &= E(\delta \xi_{i,n} \delta \xi_{j,n}) \\ &= \iint f(\xi_{i,n} \xi_{j,n}) (\xi_{i,n} - E(\xi_{i,n})) (\xi_{j,n} - E(\xi_{j,n})) d\xi_{i,n} d\xi_{j,n} \end{aligned} \quad (9.10)$$

So we can write,

$$V_R = \sum_{i,j} \frac{\partial R}{\partial \xi_i} \frac{\partial R}{\partial \xi_j} \text{COV}(\xi_{i,n} \xi_{j,n}) \quad (9.11)$$

which in terms of sensitivity coefficients, P , and relative

covariances RCOV can be written as,

$$U_R = \sum_{i,j} P_{\xi_i} P_{\xi_j} RCOV(\xi_{i,n} \xi_{j,n}) \quad (9.12)$$

where U_R is the relative uncertainty $(\delta R_n / R)^2$.

Finally, the relative correlation matrix, $RCORR(\xi_{i,n} \xi_{j,n})$, can be re-expressed as the correlation matrix and the relative standard deviation, $RSD(\xi_{i,n})$, of cross section $\xi_{i,n}$,

$$U_R = \sum_{i,j} P_{\xi_i} P_{\xi_j} CORR(\xi_{i,n} \xi_{j,n}) RSD(\xi_{i,n}) RSD(\xi_{j,n}) \quad (9.13)$$

which can finally be written as,

$$U_R = \sum_{i,j} P_{\xi_i} P_{\xi_j} CORR(\xi_{i,n} \xi_{j,n}) \frac{SD(\xi_{i,n})}{\xi_{i,n}} \frac{SD(\xi_{j,n})}{\xi_{j,n}} \quad (9.14)$$

where $SD(\xi_{i,n})$ denotes the standard deviation or error of cross section $\xi_{i,n}$.

The final problem variance is thus given in terms of sensitivity coefficients, a correlation matrix and relative cross section standard deviations. The sensitivity coefficients are calculated in the course of solving the transport equation to obtain the angular fluxs and using a mean value set of cross section data, denoted by $\xi_i = E(\xi_{i,n})$. The partial derivatives at these mean values are the required mean sensitivity coefficients. The covariance matrix is likewise evaluated using variances and covariances of the basic data around their mean.

9.3 ESTIMATION OF CROSS SECTION COVARIANCE MATRICES

Without actual cross section covariance matrices being available with which to do an uncertainty analysis, it is fairly difficult (but necessary) to try and estimate sensible correlations between the partials (Perey et al 1975 and Perey 1978). The correlations could really be made rather arbitrary

and anything that one desires (though between -1 and +1, of course) but care is needed here otherwise very nonsensical answers will result. There are two problems to be careful about here. Firstly, (Weaver 1983) the choice of the correlation matrix elements must be made carefully. For instance, if (matrix elements) A and B are fully correlated and B and C are fully correlated, then A and C must also be fully correlated. Secondly, as regards the full correlation matrices between partial cross sections (of the same isotope) it is impossible to have all of them positively correlated to each other. As far as cross section measurement is concerned, due to the measurement process (where some partials are derived from others), it is physically unreal for there to be full positive correlation between all partial cross sections of the same nuclide, whatever the magnitude of the correlation.

It should be mentioned that the uncertainty analysis in this work only deals with the correlations between partials of the same isotope, not between partials of different isotopes. A covariance matrix between two partials contains information on the energy dependent cross section errors of each of the partials and also contains information on the correlations or 'measure of closeness' between energy groups from the two partials. The energy dependent cross section errors have been obtained from a variety of sources and have the form of relative percentage uncertainties at particular energies. These errors are really estimated errors and are given for broad ranges of energy rather than for each separate energy group. The magnitudes of the errors are shown in table 1. Note that the effects of uncertainties in secondary energy and angular distributions have not been included in this work. The uncertainty analysis here

only deals with the uncertainties in the collisional loss cross section and therefore does not consider the effects of all nuclear data uncertainties simultaneously. The methodology for secondary neutron analysis is well developed and can be found in papers by Gerstl et al (1980) and Gerstl (1980).

For the correlation information guesses had to be made. As mentioned previously, the ENDF tape contains all sorts of cross section information on each of the isotopes. File 1 is the most important in that it gives general background information on each element and how each partial cross section was found, either by experimentally measuring, theoretically calculating or by deriving it via the addition or subtraction of other partial cross sections. With this data the correlations were constructed. The total cross section is known quite accurately and the nonelastic cross section can be found fairly easily by adding up all the inelastic level scattering cross sections (1st, 2nd, 3rd.....continuum) and the nonelastic cross sections ((n,p), (n,alpha) etc). These partial cross sections can, in turn, be found by experiment. In ENDF it is found that the elastic scattering cross section was calculated by subtracting the nonelastic from the total cross section. So as far as the uncertainty analysis is concerned, the elastic and nonelastic reactions are negatively correlated. The degree or magnitude of correlation is unknown and had to be estimated rather arbitrarily. It is usually found that the (n,2n) and (n,3n) partials are negatively correlated. This comes about because in a cross section measurement of the (n,2n), (n,3n) type reactions it is the total multiplication that is measured. Subtracting this result from the theoretically determined (n,2n) cross section, the remainder is considered to be the (n,3n) cross

section. In summary, for any given isotope, the elastic scattering cross section was considered to be negatively correlated with all other partials and the $(n,2n)$ is considered to be negatively correlated to the $(n,3n)$.

Using the above positive and negative correlations, three types of correlation matrix were chosen ; 0.0, 0.5 and 1.0 correlation. Each had its main diagonal (running from top left to bottom right) as 1.0. This means whatever the partials, the correlation is +1.0 between the same energy groups. The off diagonal matrix elements, giving linkage between different energy groups, were given values of 0.0, 0.5 and 1.0 respectively. This was done so that a range of possible variances would be given, since none of the correlations are accurate. The correlation between elastic scattering and other partials and between $(n,2n)$ and $(n,3n)$ is the same as the magnitudes given above, but the signs are now negative.

A program has been written which calculates the standard deviation or error of a given reactor parameter due to the correlated uncertainties between partials of a given isotope, according to equation 9.14. The correlation $CORR(\xi_{i,n}, \xi_{j,n})$ was guessed and the relative cross section errors $SD(\xi_{i,n})/\xi_{i,n}$ were obtained from various sources. The variance of each correlated partial is calculated and summed to yield the variance due to the correlated partial cross sections for the isotope. These variances are added quadratically to yield the total variance of the reactor parameter. Taking its square root and multiplying by 100 gives the relative percentage error of the parameter to all the correlated cross section uncertainties in the system. The partials chosen for the analysis are shown in table 2.

likely errors. Such an analysis produces a great deal of information in the form of variances between partials. So it was decided to only present the total errors on some selected reactor parameters for the three strengths of correlation in this chapter. The variances between partials are summed for each isotope to give the uncertainty due to an isotope. In turn, the isotope errors are summed to give the total error.

It should be noted that although some responses show high sensitivity to a particular cross section change, the uncertainty analysis, which includes the correlation between cross section uncertainties can lead to different conclusions. The partial cross sections considered in this analysis are given in table 2. These are also the partial cross sections which will subsequently be adjusted.

Tables 3 and 4 give a summary of the errors on some reactor parameters due to different strengths of correlation between cross section uncertainties. The results are presented for the non fissile and fissile reactors as well as for the two associated integral experiments. As a general comment on tables 3 and 4, it is evident that no error has decreased as the correlation strength between cross section uncertainties has increased. Also, no total error or isotope error has come out zero or negative. In the Los Alamos sensitivity and uncertainty code SENSIT-1D (Gerstl 1980), any negative variances are automatically reset to zero. These two results come about because nowhere have the negative correlations been made too high or too strong. In all cases the high energy dependent responses $\text{Li7}(n,n' \alpha)T$ and $\text{Fe}(n,p)$ show the greatest uncertainties. This comes from a combination of the rather large uncertainties

9.4 SOURCE OF CROSS SECTION UNCERTAINTIES

Energy group dependent cross section uncertainties had to be obtained or guessed for the partial cross sections of seven isotopes ; Li6, Pb, Fe, Li7, F19, U238 and U235. For Fe and Pb the uncertainties were obtained from a report by Muir and LaBauve (1981). This report contains the covariance (correlation and uncertainty) matrices between partials of the same nuclide, in a 30 group energy structure, for nine elements and probably represents the best covariance estimates that can presently be obtained. However, for this work, only the uncertainties were used and they were expanded to cover the 100 group structure of DLC37F. The U235 and U238 uncertainties were obtained from papers by Salvatores (1973), McKnight et al (1974), Pazy (1974), Bhat (1976) and Salmi et al (1978). The (n,2n) and (n,3n) reactions of uranium are given the same uncertainties as those for Pb. On all threshold type reactions, when estimating uncertainties, the highest error has been placed at the threshold energy. The Li6 and Li7 uncertainties were taken from Reupke (1978) and Alsmiller et al (1974). For the former case, the errors had to be converted from the point energy values to that of broad groups. Some of the data for fluorine was also taken from the latter source ; the rest of it was estimated by comparison with Pb and Fe uncertainties and from the neutron cross section book by Garber et al (1976).

9.5 DISCUSSION AND RESULTS OF UNCERTAINTY ANALYSIS

The results of the uncertainty analysis are now presented. Since no accurate covariance matrices were available, a range of correlations was taken to try and obtain rough estimates of

of the partials at these high energies and because of the very large sensitivity values of the two reaction rates. The following discussion is broken up into the uncertainty analyses of the non fissile and fissile systems, respectively.

a) Non fissile uncertainties

In the case of $\text{Li7}(n,n' \alpha)\text{T}$ for the non fissile designs, the Pb contributes nearly double as much to the uncertainty (65%) as that of the Li7 isotope (34%). The $\text{Pb}(n,2n)$ partial totally dominates the processes within the non fissile designs and it is the correlation of $\text{Pb}(n,2n)$ with itself that causes the huge uncertainty. The reason is a combination of the large $\text{Pb}(n,2n)$ cross section uncertainties (60% near threshold) and the high sensitivity of $\text{Li7}(n,n' \alpha)\text{T}$ to $\text{Pb}(n,2n)$, particularly in the highest energy group. This high sensitivity is saying that any change in $\text{Pb}(n,2n)$ drastically changes the $\text{Li7}(n,n' \alpha)\text{T}$ reaction rate. The total sensitivity coefficient of the above is 0.8. The Li7 contribution to uncertainty comes through on the direct term of the sensitivity of $\text{Li7}(n,n' \alpha)\text{T}$ to $\text{Li7}(n,n' \alpha)\text{T}$. The total sensitivity coefficient for this is 0.98 but its cross section errors are much less, resulting in a lower overall uncertainty.

The error in the hydrogen gas production via the $\text{Fe}(n,p)$ reaction for the non fissile reactor is the biggest uncertainty in all this work. The reason is mostly due to the correlated uncertainty of the $\text{Pb}(n,2n)$ partial with itself. In the same way as the $\text{Li7}(n,n' \alpha)\text{T}$ reaction, there is also some contribution to uncertainty from the high sensitivity of $\text{Fe}(n,p)$ to $\text{Fe}(n,p)$, and also from the negative sensitivity of $\text{Fe}(n,p)$ to Fe total inelastic. The Fe total inelastic scattering cross section was

given fairly large errors (table 1) and this offsets its slightly smaller sensitivity. However, both these effects only contribute about $1/2$ % to the total $\text{Fe}(n,p)$ uncertainty whilst the Pb partials account for over 99% of the error. The total sensitivity coefficient of $\text{Fe}(n,p)$ to $\text{Pb}(n,2n)$ is very large, over -350% for a 100% increase in $\text{Pb}(n,2n)$. When this high sensitivity is folded in with the large errors on the $\text{Pb}(n,2n)$ partial, the result is a very high uncertainty. The $\text{Fe}(n,p)$ reaction is extremely sensitive to $\text{Pb}(n,2n)$. The sensitivity of $\text{Fe}(n,p)$ to Pb elastic shows a similar sensitivity, but the Pb elastic cross section uncertainties are much less. The NFIE has much lower sensitivities of $\text{Fe}(n,p)$ to the Pb partials simply because so much lead was removed in constructing the integral experiment. Indeed in going from the non fissile reactor (NFR) to the non fissile integral experiment (NFIE) nearly 49% of the effective thickness of the Pb was removed. The sensitivity of $\text{Fe}(n,p)$ to $\text{Pb}(n,2n)$ decreased by over a factor 4 for the 50% reduction in the lead thickness of the above change. The reason for the exaggerated sensitivity decrease is that a good proportion of the lead in the integral experiment was put in a softer neutron flux, far from the source.

The $\text{Li6}(n,\alpha)\text{T}$ breeding rate and the Li6 nuclear heating rate have the lowest uncertainties. This is a combination of low sensitivities and small cross section uncertainties in the Li6 partial. Since Li6 neutron heating is dominated by the $\text{Li6}(n,\alpha)\text{T}$ cross section below about 0.1 MeV, the reason for its low uncertainty is obvious. The uncertainty in $\text{Li6}(n,\alpha)\text{T}$ at a correlation of zero is so low mainly because of the positive and negative interplay of the Pb partials. At this correlation the positive variance contribution of $\text{Pb}(n,2n)$ on $\text{Pb}(n,2n)$ and

Pb(n,3n) on Pb(n,3n) is mostly cancelled by the negative variance contribution of Pb(n,2n) on Pb(n,3n) and Pb(n,2n) on Pb total inelastic. The jump in uncertainty from a correlation of 0.0 to 0.5 is nearly a factor 3. This is due to mostly to the change in sign (small negative to a slightly higher positive) of Pb(n,gamma) on Pb total inelastic.

It is a general feature of the non fissile reactor in going from correlation zero to 0.5, that the uncertainties show their largest increase. This clearly indicates that some type of off diagonal correlation must be included in this work if sensible looking answers are required. Probably the best type of off diagonal correlation value to use is in the range 0.2-0.7. However, in the work presented here, this recalculation has not been made. This uncertainty increase occurs to a much lesser extent in the NFIE. The increase in uncertainty in going from correlation 0.5 to 1.0 is smaller for both reactor and integral experiment assemblies. Considering that Pb is so important in the non fissile reactor, it is a bit surprising that the uncertainty of Pb heating is so low. The low uncertainty is due to the fact that the positive and negative variances have roughly the same magnitude and are tending to cancel, and also because sensitivities are generally fairly low anyway. In turn, the reason for the low sensitivities is because the lead is so dominant that any change in other materials (non lead ones) has very little effect indeed. The partial variances of Fe are typically of the order of 10^{-12} . The positive effect of the Pb(n,2n) on Pb(n,2n) has been cancelled by the negative effect of the Pb(n,2n) on Pb total inelastic. The Pb heating uncertainties are higher in the NFIE because there is less lead, and so other materials now have more of an effect on this heating parameter.

In all of the NFR uncertainty calculations the iron plays a very minor role. The only time it plays any consequence is via the direct term in the sensitivity of $\text{Fe}(n,p)$ to $\text{Fe}(n,p)$. The Li6 isotope plays about the same role as the iron. Because of its higher isotopic abundance, Li7 plays a slightly more important role. But it is the lead that dominates the reactor uncertainties and particularly that due to $\text{Pb}(n,2n)$ and $\text{Pb}(n,3n)$. The former owes its importance to the high sensitivities that it produces whilst the latter via its high cross section uncertainty. Playing a slightly lesser role is the Pb total inelastic partial. In the NFIE the fluorine has about the same importance as the iron. It can be seen that, in general, uncertainties in the integral experiment are higher than those in the reactor. This is due simply to the reduction in the iron and lead - the system is optically thinner and non Fe or Pb system changes have more effect on the neutron dynamics than in the case of the reactor.

In summary, it appears that most of the lead cross section partials need reevaluation. In particular, the $(n,2n)$, $(n,3n)$ and total inelastic scattering cross sections need to be known more accurately. However, even with the above recommendation, it must be remembered that the errors in table 1 were chosen to be conservative and represent upper limits of the cross section errors.

b) Fissile uncertainties

As far as the uncertainty of the $\text{Li7}(n,n' \alpha)\text{T}$ response is concerned, the fissile reactor exhibits a similar mode of behaviour as the NFR except that Pb is replaced with uranium as

the dominant material and the $Pb(n,2n)$ is replaced by the $^{238}U(n,3n)$ partial as the dominant cross section. The importance of this partial is a combination of its high cross section uncertainty (60%) and the high sensitivity of $Li7(n,n' \alpha)T$ to $^{238}U(n,3n)$. The total coefficient for this is -0.14. The sensitivity of $Li7(n,n' \alpha)T$ to $^{238}U(n,fission)$ is twice as big, -0.29, but because the $^{238}U(n,f)$ cross section is more accurately known than the $(n,3n)$, the uncertainty is lower. The above statement is true for the uncertainties of all reactor parameters, except for $^{238}U(n,f)$ where the very high sensitivity of $^{238}U(n,f)$ to $^{238}U(n,f)$, with a total coefficient of 1.15, via the direct term in the sensitivity equation. This offsets the high cross section errors of $(n,3n)$ to make the $^{238}U(n,f)$ partial itself the dominant cross section in producing the $^{238}U(n,f)$ fission rate reactor uncertainty.

As for the non fissile case, the change in uncertainty in going from no correlation to an off diagonal correlation of 0.5, is quite noticeable. Again, this indicates that some type of off diagonal correlation needs to be included to get sensible looking answers. The amount of increase in uncertainty with this change is particularly noticeable with the $^{235}U(n,fission)$ and heating response. The increase in error is about a factor 3.5 for both these responses. This increase is due solely to over a ten fold increase in the partial variance of $^{235}U(n,f)$ to $^{235}U(n,f)$. Therefore the uncertainty of the ^{235}U responses are very sensitive to the type of correlation used. The similarity of the uncertainties of the $Li6(n,\alpha)T$, $Li7(n,n' \alpha)T$, $^{238}U(n,f)$ and $^{235}U(n,f)$ responses to their heating rates is due solely to the former four partials being the dominant cross sections in making up the heating rates. The $Li6$ parameters again show the

smallest uncertainties. The fissile reactor (FR) uncertainties are larger than the fissile integral experiment (FIE) ones because of the higher isotopic concentration of Li6 in the enriched lithium.

Although the mean free path thickness of uranium is the same between the reactor and experiment, the ^{238}U uncertainties are slightly higher and the ^{235}U uncertainties slightly lower for the FIE design when compared to the FR. The reason for the $^{238}\text{U}(n,f)$ and ^{238}U heating uncertainty increases is due simply to the placing of half the uranium zone at a halfway position in the integral slab. The high energy biased ^{238}U responses are now more responsive or perturbed more than they were in the FR case. Neutrons causing a contribution in ^{238}U in the 2nd half of the uranium block must travel through 18.9cm of LiF and 6cm of Fe, and so the ^{238}U rates are naturally more sensitive to the other cross section changes. This higher sensitivity is carried through to effect the uncertainties. The ^{235}U responses decrease in going from the FR to FIE are larger than the ^{238}U increase. The reason for the decrease is due to a decrease in the sensitivity of $^{235}\text{U}(n,f)$ and ^{235}U neutron heating to $^{238}\text{U}(n,f)$ and ^{238}U neutron heating. The physical reason is not because any uranium has been removed but because half of it has been moved to a different neutron environment, one that contains a great deal more thermal neutrons. Therefore, there is more ^{235}U reactions occurring, with correspondingly more independence from ^{238}U , and therefore having less sensitivity.

For both fissile assemblies, as in the non fissile case, the Li7 and Fe neutron nuclear heating responses show some of the highest uncertainties. Taking the FIE, the ^{238}U and the Fe

contribute mostly to the production of the $\text{Li7}(n,n' \alpha)\text{T}$ uncertainty, the ^{238}U via the high partial variance of $^{238}\text{U}(n,f)$ to $^{238}\text{U}(n,f)$ which originates from the high negative sensitivity (-0.13) of $\text{Li7}(n,n' \alpha)\text{T}$ to $^{238}\text{U}(n,f)$. Both are high energy reactions and compete strongly for neutrons. Neutrons resulting from a $^{238}\text{U}(n,f)$ fission reaction are fairly unlikely to have energies above 3.8 Mev, the $\text{Li7}(n,n' \alpha)\text{T}$ threshold. The Fe contributes to the $\text{Li7}(n,n' \alpha)\text{T}$ uncertainty mostly via the Fe inelastic partial. The sensitivity of $\text{Li7}(n,n' \alpha)\text{T}$ is highly negative (-0.23) and in addition the Fe inelastic scattering cross section has been given a relative percentage uncertainty of 20%, in the highest 5 energy groups, just where the sensitivity of $\text{Li7}(n,n' \alpha)\text{T}$ to Fe inelastic is at its most negative.

The $\text{Li7}(n,n' \alpha)\text{T}$ reaction also contributes to its own uncertainty via the direct term in the sensitivity of $\text{Li7}(n,n' \alpha)\text{T}$ to $\text{Li7}(n,n' \alpha)\text{T}$, though the error is only about half that produced by the Fe and ^{238}U . The Fe heating (again, this discussion is on the FIE) uncertainty comes mostly from the ^{238}U , but particularly from the Fe. This is in contrast to the NFIE where the Pb dominated in contributing to the Fe heating uncertainty and so there is now no competing lead in the fissile case. Additionally, for the fissile case the spectrum is a lot harder (because of the uranium) and the Fe assumes a greater importance ; that is, the various responses in the fissile assembly are more sensitive to the iron than in the non fissile designs. The uncertainty caused by the Fe originates from the high sensitivities of Fe heating to $\text{Fe}(n,p)$, 0.37, and Fe heating to Fe inelastic, -0.14,. In addition, both these partials have fairly high cross section uncertainties attached to them. As for the Li7 heating, the uncertainty in Fe heating from the uranium

204

comes from the partial variance of $^{238}\text{U}(n,f)$ to $^{238}\text{U}(n,f)$.

It can be seen that the Fe plays a more important role in the fissile designs than in the non fissile ones. This is due not only to the spectrum being harder, with the Fe therefore having more 'bite', but also a greater thickness of Fe was used in the cladding. The ^{238}U isotope is the most important material, mainly because of its position close to the source and because of its dominant interactions with 14.1 MeV neutrons via the fission process. The fluorine plays a more important role in the fissile design, whilst Li6 and Li7 have about the same importance in the experiment, where the lithium is in its natural isotopic state, as in the non fissile designs.

On the basis of the above, the only cross sections needing reevaluation are the Fe inelastic levels (though only the total scattering cross section was used here) and the Fe(n,p) partial. The fairly high uncertainties caused by the $^{238}\text{U}(n,f)$ partial are due to its dominance and importance in the fissile assemblies rather than its cross section uncertainties.

Pb elastic	5% 1-30	3% 30-59	2% 60-100
Pb (n,2n)	15% 1-7	60% 8	
Pb (n,3n)	60% 1		
Pb (n,gamma)	30% 1-3	15% 4-32	10% 33-61 5% 62-100
Pb inelastic	30% 1-5	10% 6-26	20% 27-32 40% 33
Li6 elastic	2% 1-100		
Li6 (n,2n)alpha	50% 1-9		
Li6 (n,n')alpha	20% 1-22		
Li6 (n,gamma)	5% 1-100		
Li6 (n,p)	10% 1-16		
Li6 (n,alpha)T	10% 1-30	5% 31-100	
Li6 inelastic	20% 1-22		
Fe elastic	8% 1-32	1% 33-70	5% 71-100
Fe (n,2n)	10% 1-2	60% 3-4	
Fe (n,n')alpha	20% 1-2	60% 3-4	
Fe (n,n')proton	20% 1-4		
Fe (n,gamma)	25% 1-36	10% 37-70	5% 71-100
Fe (n,p)	5% 1-9	15% 10-19	5% 20-33 20% 34
Fe (n,d)	40% 1-4	60% 5	
Fe (n,T)	40% 1		
Fe (n,He3)	40% 1-2		
Fe (n,alpha)	18% 1-16	25% 17	
Fe inelastic	20% 1-5	8% 6-28	20% 29
Li7 elastic	10% 1-20	7% 21-100	
Li7 (n,n')	50% 1-6		
Li7 (n,n')alpha	50% 1-4		
Li7 (n,n' alpha)T	10% 1-17		
Li7 (n,gamma)	5% 1-100		
Li7 (n,d)	15% 1-6		
Li7 inelastic	15% 1-20	10% 21-33	20% 34
Fl9 elastic	5% 1-100		
Fl9 (n,2n)	30% 1-3	60% 4	
Fl9 (n,n')alpha	40% 1-2	20% 3-11	
Fl9 (n,n')proton	40% 1-2	20% 3-6	
Fl9 (n,gamma)	10% 1-30	30% 31-100	
Fl9 (n,p)	40% 1-2	20% 3-13	
Fl9 (n,d)	40% 1-2	20% 3-9	
Fl9 (n,t)	40% 1-2	20% 3-7	
Fl9 (n,alpha)	20% 1-12	40% 13-23	
Fl9 inelastic	5% 1-49		
U238 elastic	10% 1-100		
U238 (n,2n)	15% 1-6	60% 7-9	
U238 (n,3n)	60% 1-3		
U238 fission	6% 1-53		
U238 (n,gamma)	5% 1-100		
U238 inelastic	15% 1-53		
U235 elastic	10% 1-100		
U235 (n,2n)	15% 1-8	60% 9-11	
U235 (n,3n)	60% 1-3		
U235 fission	10% 1-10	3% 11-31	7% 32-100
U235 (n,gamma)	5% 1-100		

TABLE 1.

RELATIVE UNCERTAINTIES
OF CROSS SECTIONS (GROUPS ARE INCLUSIVE)

Li6 elastic	Fe(n,2n)alpha	Fe(n,2n)	Fe(n,2n)
Li6 (n,n')alpha	Li6 (n,n')alpha	Fe(n,alpha)	Fe(n,alpha)
Li6 (n,gamma)	Li6 (n,gamma)	Fe inelastic	Fe inelastic
Li6 (n,proton)	Li6 (n,proton)		
Li6 (n,alpha)T	Li6 (n,alpha)T		
Li6 inelastic	Li6 inelastic		
Li7 elastic	Li7 elastic		
Li7 (n,2n)	Li7 (n,2n)		
Li7 (n,2n)alpha	Li7 (n,2n)alpha		
Li7 (n,alpha n')T	Li7 (n,alpha n')T		
Li7 (n,gamma)	Li7 (n,gamma)		
Li7 (n,deuterium)	Li7 (n,deuterium)		
Li7 inelastic	Li7 inelastic		
Pb elastic	Pb elastic		
Pb (n,2n)	Pb (n,2n)		
Pb (n,3n)	Pb (n,3n)		
Pb (n,gamma)	Pb (n,gamma)		
Pb inelastic	Pb inelastic		
Fe elastic	Fe elastic		
Fe (n,2n)	Fe (n,2n)		
Fe (n,n')alpha	Fe (n,n')alpha		
Fe (n,n')proton	Fe (n,n')proton		
Fe (n,gamma)	Fe (n,gamma)		
Fe (n,proton)	Fe (n,proton)		
Fe (n,deuterium)	Fe (n,deuterium)		

TABLE 2. PARTIAL CROSS SECTIONS USED FOR
UNCERTAINTY ANALYSIS

NFIE UNCERTAINTIES				FR UNCERTAINTIES			
	<u>0.0</u>	<u>0.5</u>	<u>1.0</u>		<u>0.0</u>	<u>0.5</u>	<u>1.0</u>
Li6(n,alpha)T	0.422	1.127	1.53	Li6(n,alpha)T	2.22	3.35	4.18
Li7(n,n' alpha)T	12.20	14.02	15.6	Li7(n,n' alpha)T	8.66	11.22	13.30
Li6 heating	0.413	1.394	1.93	U238(n,fission)	3.36	4.80	5.91
Pb heating	1.940	3.023	3.809	U235(n,fission)	3.47	8.20	11.10
Fe heating	5.39	7.92	9.816	Li6 heating	1.69	2.52	3.13
Li7 heating	3.64	5.777	7.300	Li7 heating	5.65	7.48	8.94
Fe(n,p)	45.990	51.12	56.15	Fe heating	6.23	6.77	7.28
SS DPA	3.930	6.555	8.392	U238 heating	3.40	4.72	5.74
Pb DPA	2.41	2.62	2.81	U235 heating	3.42	8.51	11.54
				Fe(n,p)	6.81	7.84	8.74
				Li6 DPA	2.83	2.85	2.87
NFIE UNCERTAINTIES				FIE UNCERTAINTIES			
	<u>0.0</u>	<u>0.5</u>	<u>1.0</u>				
Li6(n,alpha)T	2.78	3.59	4.24				
Li7(n,n' alpha)T	11.78	13.3	14.6		<u>0.0</u>	<u>0.5</u>	<u>1.0</u>
				Li6(n,alpha)T	1.95	2.37	2.72

THREE DIMENSIONAL SENSITIVITY

10.1 INTRODUCTION

Monte Carlo methods are fast becoming a significant tool in sensitivity analysis and perturbation calculations (Rief 1977). This is because, in principle, a full capability Monte Carlo code is the most accurate method for analysing neutronics problems, particularly where ducts, pipes and inhomogeneities are of primary importance in a reactor design. The use of the method is enhanced by three exclusive features which no other method has (Dubi and Rief 1980). Firstly, the Monte Carlo method is not limited to first order perturbation theory and so a complete set of sensitivity profiles may be obtained in one run. This is in contrast to deterministic methods, based on first order perturbation theory, where each sensitivity profile, centred at a given value, of the perturbed parameter, requires a separate forward and adjoint calculation. Secondly, complicated realistic three dimensional structures can be studied. Thirdly, group averaging errors can be avoided by the use of point nuclear data, so that when it comes to cross section adjustment, the adjustments should represent shortcomings in the basic data, rather than difficulties in the group averaging process.

To this end, several Monte Carlo codes are now available, TIMOC (Rief 1975), DUCKPOND (Hall 1980), MCNP (Los Alamos 1979) and TRIPOLI (Dejonghe 1980) which have the option of performing a perturbation analysis. The perturbation algorithm is based on correlated tracking and usually includes the sampling of sensitivity profiles. In this approach, the forward equation is

solved and estimates of the perturbation effects are made by correlated sampling in the same tracks for both perturbed and unperturbed cases. The effect of the perturbation (or the sensitivity) is calculated at each collision by weight factors that correct for sampling from the unperturbed collision density functions. In addition, the flux and sensitivity calculations by the track length estimator (to remedy the infinite variance problem) is used to allow for the scoring at discrete points. Rief uses the above technique to compare results with the ANISN/SWANLAKE procedure for a point source in a sphere and for a transmission calculation through iron slabs. Good agreement was found for both cases.

Since none of the above codes with the perturbation option was readily available, it was decided to use the Monte Carlo code MORSE (Emmett 1975) with SWANLAKE to obtain some sort of 3-dimensional picture, in the same way that ANISN and SWANLAKE are used for the 1-dimensional case. It should be mentioned, though, that the work performed in this chapter is not really a true 3-dimensional sensitivity analysis. The effort has been directed into obtaining information on the effect of the sensitivity profiles when the fluxes are scored in a finite system containing correct neutron leakage. In this way a real neutronic model of the integral experiment is being constructed and direct comparisons between MORSE and ANISN representations can then be made.

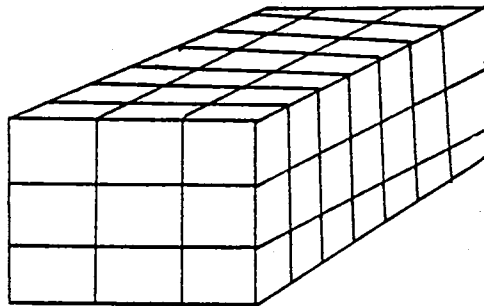
Remembering that the integral experiments were neutronically constructed from fluxes and sensitivity profiles from the 1-dimensional code ANISN, the question that this section of work is trying to answer is what is the level of sensitivity profile

or neutronics distortion near the integral assembly edges. If the amount of distortion is high then the sensitivity tuning between the reactor and its associated experiment is no longer meaningful at these spatial positions. In this case, foil response measurements at the experiment edges, with the subsequent cross section adjustments, will not be relevant to the reactor and the experimentalist should not make foil response measurements at these positions.

It is to be expected, though, that even if edge effects are severe, foil measurements made near the centre of the experiment, where the fluxes calculated by MORSE and ANISN will be at their closest, will be most relevant to the fusion reactor.

10.2 OVERVIEW AND METHODOLOGY

Since both integral experiments have a slab geometry, it was decided to cut up the assembly into parallelepiped shapes and to score the forward and adjoint scalar fluxes in these regions. Thus the assembly is broken up into the following form.



The ANISN spatial dimensions and interval boundaries were preserved for the MORSE integral experiments and they were used to define the geometry in the x direction. The y and z transverse dimensions are 80cm X 80cm for the non fissile integral

experiment (NFIE) and $100\text{cm} \times 100\text{cm}$ for the fissile integral experiment (FIE). Both integral assemblies were cut up into 891 spatial zones or boxes. It would probably be impractical to go above this number due to peripheral space and execution time restrictions, though the accuracy on the fluxes would be improved. The slabs were left in their bare states and no reflective boundary conditions were imposed. Neutrons leaking from the surface are not returned. An isotropic point source, emitting neutrons at an energy of 14.1 Mev, was located 1cm from the first surface and along the centre line of the experiment. In order to lessen the execution time, the Russian roulette technique was used. Any particle with a weight below a specified value has Russian roulette played on it. If it survives it has its weight reset to one, otherwise it is killed. The coupled 100 neutron and 21 gamma group DLC37F cross section set was used in all this work. It was decided to use track flux scoring since the volume integration over the spatial zone is carried inherently in the flux and such fluxes are directly compatible with the zone ones calculated by ANISN. Also, the track scoring requires less computer time than the point scoring though the accuracy on the obtained fluxes will be lower.

The MORSE program had to be amended slightly to write the forward and adjoint fluxes to disk and to reorder them into the correct energy and zone form required by SWANLAKE. A forward and adjoint MORSE run is required for each response located at each spatial position. A separate forward run is needed because the existence of the response foil perturbs the flux near it. As outlined in chapter 4, the foil detector chosen was the $\text{Ni}(n,p)$ response function. This was used in both integral experiments at a central and an edge position at a particular spatial location.

In addition, the NFIE has been further analysed by a $^{232}\text{Th}(n,\text{fission})$ response foil. For the calculations on the NFIE the foils were given an area of 25cm^{-2} ($5\text{cm} \times 5\text{cm}$) and a thickness of 0.3cm . This is bigger than an experimentalist would use but it does give a high count with a lower error. The weight of the Ni and Th foils with these dimensions is 67.0g and 87.0g , respectively. In order to get a more physical representation, the Ni foil used on the FIE was given an area of 1cm^{-2} ($1\text{cm} \times 1\text{cm}$) and a thickness of 0.3cm . It weighs 2.7g . The smaller area results in fewer counts, with increased error, and so for the FIE the weight at which Russian roulette is played was increased from 10^{-3} to 10^{-2} . The corresponding Russian roulette weight in the NFIE is 10^{-3} . An additional benefit in using these two foils is that there are no direct or fissile gain terms to be calculated, thus reducing the execution time.

The methodology for calculating the sensitivities is as follows. A spatial position in the integral experiment was chosen for a foil. For both MORSE and ANISN, a forward run was made, giving its forward flux and response, and an adjoint run giving the adjoint flux. The adjoint source used was the foil response function, spatially located at the same position as the foil. Because the interval boundaries in an ANISN run are set, the foil has to completely fill the spatial position. For this reason the density of foil material used in the ANISN run was adjusted so that it presented the mean free path thickness as that presented by the foil in the MORSE geometry in the spatial x direction. The forward and adjoint fluxes are folded in the usual way to produce the sensitivity profiles. To obtain the necessary statistical accuracy for this work, many particle histories had to be tracked. For all runs always over $200,000$

neutrons or adjunctions were followed, giving errors on fluxes of between 0.5% for group 1 and about 20-50% for low energy groups. Finally, some minor modifications had to be made to the SWANLAKE code. The angular adjoint flux reordering process for energy and angle was removed because an intermediate program handles the energy reordering, and the angular reordering is redundant since the fluxes from MORSE are scalar. The volume integration was removed because the track fluxes from MORSE inherently carry the volume parameter. To get these fluxes into the correct form for the sensitivity equation, one of the fluxes had to be divided by volume. Lastly, the spatial flux averaging process was removed. SWANLAKE normally takes the interval boundary fluxes and averages them to form the mid interval boundary flux. However, because the MORSE fluxes are already zone averaged, this process was disabled. In summary, it must be restated that the following is not a proper 3-dimensional sensitivity analysis but rather a gross approximation to obtain edge effect information on sensitivity profiles. Also shown and discussed here are the ANISN and MORSE integral parameters.

10.3 NON FISSILE 3-DIMENSIONAL SENSITIVITY

The neutron and gamma integral parameters for the non fissile assembly are given in table 1 and a direct comparison is made with the results obtained from the ANISN code. Both sources were isotropic and ANISN was run in the P_3S_4 mode. Just over half a million case histories were followed to obtain these results, and the statistical error is less than 0.1% for all the results. The relative percentage reduction between the ANISN and MORSE results is also shown. Due to the correct leakage representation in

215

MORSE, all the ANISN results are higher. The largest differences occur in the lowest energy reaction rates. This is due to the fact that the interaction probability is less at lower neutron energies and so there is more neutron leakage. At high energies a neutron sees essentially the same neutronic environment and mean free path thicknesses. The biggest decrease is seen in the $\text{Li6}(n,\alpha)\text{T}$ reaction and the smallest in the $\text{Li7}(n,n'\alpha)\text{T}$ activation rate.

The total integrated sensitivity coefficients of the $\text{Ni}(n,p)$ and $^{232}\text{Th}(n,\text{fission})$ foils to some selected cross section partials are shown in tables 2, 3, 4 and 5. The ANISN interval 19 quoted in the table corresponds to a real distance 8.7cm from the source (7.7cm from the first surface), and interval 29 as 15.1cm from the source (14.1cm from the first surface). Therefore, these are the spatial positions that an experimentalist must place the $\text{Ni}(n,p)$ and $^{232}\text{Th}(n,f)$ foils in order to compare the experimental and calculational results.

The big difference between the ANISN and MORSE results occurs when the response foil is placed to one side of the assembly, and the adjoint flux has many zero entries. The ANISN adjoint fluxes are always finite. These zero entries occurred in the MORSE zones on the opposite side to the response foil and at all zones furthest from it. With many more restarts these zero entries would become finite but with the available resources it was not felt wise to do this. The total sensitivity coefficients obtained by MORSE for centrally positioned response foils in tables 2 and 4 can be directly compared to the corresponding coefficients in tables 6 and 7 which have been evaluated from ANISN fluxes. It is meaningless to compare the off centre MORSE

foil sensitivities, since the adjoint source used to compute the two sensitivities is in a different spatial position.

In general the sensitivities in table 2 are between 100 and 1000 times lower than those of table 6, although the difference between the sensitivities for the $Pb(n,2n)$ partial is only about a factor 30. Yet again, this is indicative of the dominance of this partial. The reason for the low MORSE sensitivities is because it is so much harder to score a neutron reaction in a foil in a leaky MORSE geometry. Therefore, because there are fewer neutrons around with which to make a score, then the foil response does necessarily become more sensitive to those neutrons still remaining. All the MORSE sensitivities are negative, but some of the ANISN sensitivities to the elastic partials show a positive value. This is due directly to the reflection of neutrons from positions in the assembly beyond the foil position. The sensitivity is always negative in the zones near the source because any increase in the system materials reduces the number of neutrons contributing to the high energy biased $Ni(n,p)$ and $^{232}Th(n,f)$ responses.

A detailed energy breakdown of the MORSE sensitivities shows that all the elastic partials have positive sensitivities in the top 3 or 4 energy groups, but that this effect is swamped by the negative values at lower energies. The sensitivities of tables 3 and 5 show even greater sensitivity than tables 2 and 4. The reason is again one of difficulty in neutrons producing a score. Because of this higher sensitivity, the values of tables 3 and 5 have roughly the same magnitudes as those given in tables 6 and 7. However, to calculate the amount of sensitivity profile distortion, with neutron leakage correctly taken into account,

the shapes of some relevant profiles must be compared. Figures 1, 2 and 3 give the sensitivities of the $\text{Ni}(n,p)$ foil located in interval 19 to the Li6 elastic, $\text{Li6}(n,\alpha)T$ and $\text{Pb}(n,2n)$ partials. These have been obtained from ANISN forward and adjoint fluxes. Figures 4, 5 and 6 give the corresponding profiles for the $\text{Ni}(n,p)$ foil in a centre position to the same partials. The forward and adjoint fluxes come from MORSE. The similarity of the shape between the two profiles is excellent and reaffirms the known fact that the neutronic environments for an infinite slab and that for the central position of a finite sized slab should be quite similar. Therefore foil measurements made by an experimentalist at the central position in the NFIE can be directly compared to the results obtained in this thesis by the 1-dimensional code ANISN.

Finally, figures 7, 8 and 9 give the same sensitivity profiles as figures 4, 5 and 6 except that the $\text{Ni}(n,p)$ foil is now positioned at one of the edges. Because of the foils remoter location, the number of neutrons scoring in it are less and therefore the accuracy on the flux is much lower. The spikes on the profiles of figures 7, 8 and 9 are a direct consequence of the errors on the flux being higher for the edge located foil. Apart from this additional structure, the profiles of figures 7, 8 and 9 are quite similar to that of figures 1, 2 and 3. Therefore there has been very little sensitivity profile distortion at the outer edges of the NFIE, and the neutronic environments of the slab centre and edges are therefore quite similar.

Although not given in detail, the results of the $^{232}\text{Th}(n,f)$ foil analysis will be briefly mentioned. As for the $\text{Ni}(n,p)$

case, the sensitivity profile of the fissile foil in the slab central position compared to the profiles evaluated by ANISN is quite similar. Some similarity of neutronics is lost by the $^{232}\text{Th}(n,f)$ being located at the edge position. Interval 29 corresponds to 14.1 cm from the first position of neutronic interaction and it is expected that few fast neutrons would appear at this spatial position. Due to large errors in these sensitivity profiles no definite conclusion can be made.

Finally, a comparison is given in table 8 of the $\text{Ni}(n,p)$ and $^{232}\text{Th}(n,f)$ activation rates obtained from MORSE and ANISN (and used for the SWANLAKE runs). Also the calculated uncertainties of the responses calculated by the ANISN code are shown.

10.4 FISSILE 3-DIMENSIONAL SENSITIVITY

The fissile integral parameters are given in table 9. Like the non fissile case, over half a million neutron case histories were tracked to obtain these results. The errors on these results are less than 1%. The relative percentage reduction is also shown. Again the greatest change occurs in those responses which are low energy biased. However, in general the differences between ANISN and MORSE are smaller for the fissile case than the non fissile case. This can be explained simply by the fact that the average flux is a lot harder due to the fission process. The interaction probability is higher at high energies with the corresponding reduction in leakage, and so again the neutronic environments calculated by ANISN and MORSE are at their closest. In addition the number of neutrons in the fissile assembly is a lot greater than in the non fissile case, which means on average there is a greater chance of a low energy interaction occurring.

This results in the reduction of the low energy response functions being less than would otherwise have occurred. The fissile reactivity, dominated by the $^{238}\text{U}(n,f)$ reaction, is roughly the same for both assemblies.

Table 10 gives the total integrated sensitivity coefficients of the $\text{Ni}(n,p)$ foil in interval 32 at a centre and an off centre position to some selected partials. These were obtained from MORSE forward and adjoint fluxes. The corresponding ANISN sensitivities are given in table 11. Interval 32 corresponds to a distance 13.7cm from the source (12.7cm from the first wall surface). It should be remembered that the above MORSE results are obtained from a $\text{Ni}(n,p)$ foil of area 1cm^{-2} . The results show the same very large sensitivities of the off centre $\text{Ni}(n,p)$ foil as for the non fissile case. These are very large sensitivities indeed and indicate the extreme difficulty that neutrons of high energy have in scoring in the Ni foil at that position. Table 11 also shows the positive zone sensitivities due to neutrons being reflected back towards the foil from other parts of the assembly. Again, as for the NFIE case, a detailed energy breakdown shows the top few energy groups for the elastic partials have positive sensitivities.

Figures 10, 11 and 12 give the profiles obtained from an ANISN run of the sensitivities of $\text{Ni}(n,p)$ at a spatial position 13.7cm from the source to the $^{238}\text{U}(n,f)$, $^{235}\text{U}(n,f)$ and $\text{Li6}(n,\alpha)\text{T}$ partials. Figures 13, 14 and 15 give the corresponding sensitivities obtained from the MORSE fluxes with the $\text{Ni}(n,p)$ foil positioned centrally. The difficulty in comparing these profiles is that the errors on the MORSE fluxes are quite large, particularly at the lowest energies, and that many MORSE fluxes

are zero. These errors are carried through to the sensitivity profiles. Comparing the above 6 figures it can be seen that the sign of the sensitivities do correspond but that the profile shapes do not. This is because too few neutrons have been tracked to get a high statistical accuracy, even though over half a million particles were followed. It is difficult to draw a definite conclusion from these graphs but it can be seen that the $\text{Ni}(n,p)$ in the central position of the assembly is responding in roughly the same sort of way as that calculated by ANISN. This is to be expected since the flux is quite hard, with the $\text{Ni}(n,p)$ response being high energy biased, and the neutronic environments are very similar. Much computer time would be needed to get good statistical accuracy on these fluxes, but it was not felt justified to proceed any further here. Figures 13 and 14 both show the same positive sensitivity peak as that of figures 10 and 11. This is the fission spectrum effect, and is due to neutrons appearing at these energies from the fission process. Neutrons appearing at this energy range, above the $\text{Ni}(n,p)$ threshold of about 1 MeV, have a high chance of contributing to the $\text{Ni}(n,p)$ reaction rate. Hence the positive sensitivity. Figures 16, 17 and 18 show the sensitivities of the $\text{Ni}(n,p)$ foil in the off centre position. As can be seen, these sensitivities are about a 100 times bigger than for the foil in a central position. A great deal more structure is shown on these profiles. Apart from the obvious reason that the number of decades between the 14.1 MeV group and the next highest peak is less in figures 16, 17 and 18, the greatest amount of structure is due to the large number of neutrons tracked. Over 700,000 particle case histories were followed and even then the $\text{Ni}(n,p)$ response had a 30% error on it. The fission spectrum effects are clearly shown in figures 16

and 17, though the actual sensitivity values seem to be somewhat random. Although, due to the large uncertainties, no definite conclusion can be reached as to the profile distortion in the fissile assembly, the MORSE and SWANLAKE results are predicting the same signs as the ANISN and SWANLAKE sensitivities.

The Ni(n,p) foil responses and correlated uncertainties for the fissile integral experiment are given in table 8. To give the Ni(n,p) foil work more resources, no $^{232}\text{Th}(n,f)$ foil calculations were made on this assembly.

	ANISN SLAB P ₂ S ₂	MORSE BARE SLAB	PERCENTAGE REDUCTION	LI6 ELASTIC LI6 (N,PROTON) LI6 (N,ALPHA)T	-1.6226E-06 -2.5867E-07 -1.5185E-06	LI6 ELASTIC LI6 (N,PROTON) LI6 (N,ALPHA)T	-8.1583E-04 -8.1896E-05 -5.3451E-04
Li6(n,alpha)T	9.4606E-2	5.30151E-2	44%	PB ELASTIC PB (N,2N) PB (N,3N)	-1.9820E-05 -1.0031E-02 -3.8295E-04	PB ELASTIC PB (N,2N) PB (N,3N)	-1.3029E-02 -1.9960E+00 -6.8610E-02
Li7(n,alpha n')T	9.8715E-3	9.68700E-3	2%	FE ELASTIC FE (N,DUETERIUM) FE INELASTIC	-1.0172E-05 -1.0563E-05 -3.6086E-04	FE ELASTIC FE (N,DUETERIUM) FE INELASTIC	-5.2719E-03 -2.7274E-03 -1.2301E-01
Fe neutron heating	1.2078E+5	1.13320E+5	6%	LI7 ELASTIC LI7 (N,N' ALPHA)T LI7 INELASTIC	-2.7302E-05 -2.3315E-05 -3.0038E-05	LI7 ELASTIC LI7 (N,N' ALPHA)T LI7 INELASTIC	-1.3249E-02 -8.4735E-03 -1.1559E-02
Pb neutron heating	1.0058E+5	7.93820E+4	21%				
Li6 neutron heating	4.7699E+5	2.76005E+5	42%				
Li7 neutron heating	1.6623E+5	1.56520E+5	6%				
19F neutron heating	1.6120E+5	1.74133E+5	8%				
Fe gamma heating	2.7003E+5	2.12020E+5	27%				
Pb gamma heating	2.2869E+6	1.94440E+6	15%				
Li6 gamma heating	1.3304E+3	1.18450E+3	11%				
Li7 gamma heating	1.5315E+4	1.36350E+4	12%				
F19 gamma heating	5.2882E+4	4.69450E+4	11%				

TABLE 2
SENS OF NI FOIL AT CENTRE
POSITION IN NFIE INTERVAL 19

TABLE 3
SENS OF NI FOIL AT OFF CENTRE
POSITION IN NFIE INTERVAL 19

LI6 ELASTIC LI6 (N,PROTON) LI6 (N,ALPHA)T	-1.1081E-05 -5.8327E-07 -5.7298E-06	LI6 ELASTIC LI6 (N,PROTON) LI6 (N,ALPHA)T	-4.0793E-03 -7.1887E-05 -1.9788E-03
PB ELASTIC PB (N,2N) PB (N,3N)	-1.9821E-04 -3.8011E-02 -1.8517E-03	PB ELASTIC PB (N,2N) PB (N,3N)	-3.2711E-02 -1.8849E+00 -8.2975E-02
FE ELASTIC FE (N,DUETERIUM) FE INELASTIC	-4.8571E-05 -5.0377E-05 -1.0288E-03	FE ELASTIC FE (N,DUETERIUM) FE INELASTIC	-1.8366E-03 -2.9903E-03 -1.9514E-01
LI7 ELASTIC LI7 (N,N' ALPHA)T LI7 INELASTIC	-1.6265E-04 -1.4517E-04 -1.4517E-04	LI7 ELASTIC LI7 (N,N' ALPHA)T LI7 INELASTIC	-6.0578E-02 -1.1198E-02 -2.5533E-02

TABLE 4
SENS OF TH FOIL AT CENTRE
POSITION IN NFIE INTERVAL 29

TABLE 5
SENS OF TH FOIL AT OFF CENTRE
POSITION IN NFIE INTERVAL 29

SENSITIVITY OF NI(N,P) AND 232TH(N,F) FOILS AT VARIOUS POSITIONS
IN THE NON FISSION INTEGRAL EXPERIMENT
SENSITIVITIES ARE EVALUATED FROM MORSE FLUXS

TABLE 1. COMPARING MORSE AND ANISN NFIE PERFORMANCE
PARAMETERS.
SOURCE IS ISOTROPIC IN BOTH CASES.

SENSITIVITY COEFFICIENTS OF NI FOIL IN INT 19
TO L16, P0, P8, L17, F19 PARTIAL CROSS SECTIONS
NON FISSIONE INTEGRAL EXPERIMENT

	ZONE 1	ZONE 7	ZONE 11	ZONE 15	ZONE 19	SUM
L16 ELASTIC	-1.29E-04	4.67E-04	6.28E-05	8.72E-06	1.43E-06	-1.89E-04
L16(N,2N)ALPHA	-3.75E-03	-6.16E-06	-7.93E-07	-1.08E-07	-1.51E-08	-3.82E-04
L16(N,N')ALPHA	-1.73E-03	6.17E-05	5.68E-06	5.65E-07	8.21E-08	-1.36E-03
L16(N,GAMMA)	-1.46E-07	-7.11E-09	-9.23E-10	-1.22E-10	-1.35E-11	-1.14E-07
L16(N,PROTON)	-9.36E-05	-6.14E-06	-7.31E-07	-9.13E-08	-1.07E-08	-9.96E-05
L16(N,ALPHA)T	-5.39E-04	-6.02E-05	-8.66E-06	-1.22E-06	-1.27E-07	-6.09E-04
L16 INELASTIC	-1.43E-03	6.38E-05	5.91E-06	5.92E-07	8.52E-08	-1.36E-03
L17 ELASTIC	-1.19E-02	5.65E-03	7.57E-04	1.05E-04	1.65E-05	-5.40E-01
L17(N,2N)	-1.38E-03	-2.37E-05	-3.04E-06	-4.13E-07	-5.76E-08	-1.41E-03
L17(N,2N)ALPHA	-2.45E-03	-7.91E-05	-9.30E-06	-1.13E-06	-1.36E-07	-2.44E-03
L17(N,N')ALPHA	-8.88E-03	9.52E-04	1.01E-04	1.12E-05	1.25E-06	-7.91E-03
L17(N,GAMMA)	-1.77E-06	-7.75E-08	-1.00E-08	-1.32E-09	-1.47E-10	-1.46E-06
L17(N,DEUTERIUM)	-7.86E-04	-2.96E-05	-3.48E-06	-4.25E-07	-5.06E-08	-8.19E-04
L17 INELASTIC	-1.17E-02	1.13E-03	1.16E-04	1.26E-05	1.72E-06	-1.05E-02
F19 ELASTIC	-4.31E-03	6.26E-03	8.72E-04	1.24E-04	1.98E-05	-2.96E-03
F19(N,2N)	-3.48E-03	-7.19E-05	-8.48E-06	-1.05E-06	-1.33E-07	-3.57E-03
F19(N,N')ALPHA	-2.95E-03	7.15E-05	5.64E-06	3.98E-07	1.02E-08	-1.97E-03
F19(N,N')PROTON	-8.53E-04	-2.41E-05	-2.90E-06	-3.63E-07	-4.51E-08	-8.80E-04
F19(N,GAMMA)	-2.29E-06	-2.41E-07	-3.42E-08	-4.78E-09	-5.05E-10	-2.57E-06
F19(N,PROTON)	-2.13E-03	-1.17E-04	-1.39E-05	-1.75E-06	-2.11E-07	-2.27E-03
F19(N,DEUTERIUM)	-1.11E-03	4.75E-05	5.63E-06	6.97E-07	8.33E-08	-1.16E-03
F19(N,TRITON)	-6.80E-04	-2.78E-05	-3.29E-06	-4.07E-07	-4.91E-08	-7.11E-04
F19(N,ALPHA)	-5.14E-03	-3.58E-04	-4.25E-05	-5.31E-06	-6.35E-07	-5.55E-03
F19 INELASTIC	-3.83E-02	3.63E-04	-3.98E-06	-4.87E-06	3.34E-07	-3.80E-02
P0 ELASTIC	-2.17E-01	2.13E-01	2.49E-02	3.62E-03	5.24E-04	-2.56E-02
P0(N,2N)	-3.54E-01	-1.12E-02	-1.72E-03	-2.47E-04	-3.53E-05	-3.67E-01
P0(N,3N)	-1.17E-02	-1.33E-03	-1.33E-04	-1.55E-05	-1.81E-06	-1.32E-02
P0(N,GAMMA)	-6.13E-05	-5.14E-05	-6.08E-06	-8.97E-07	-1.21E-07	-1.28E-04
P0 INELASTIC	-5.83E-02	-2.50E-02	-2.70E-03	-3.77E-04	-4.80E-05	-8.64E-02
FE ELASTIC	0.0	0.0	0.0	0.0	0.0	2.70E-02
FE(N,2N)	0.0	0.0	0.0	0.0	0.0	-2.87E-02
FE(N,N')ALPHA	0.0	0.0	0.0	0.0	0.0	-2.42E-04
FE(N,N')PROTON	0.0	0.0	0.0	0.0	0.0	-2.63E-03
FE(N,GAMMA)	0.0	0.0	0.0	0.0	0.0	-5.22E-05
FE(N,PROTON)	0.0	0.0	0.0	0.0	0.0	-1.12E-02
FE(N,DEUTERIUM)	0.0	0.0	0.0	0.0	0.0	-1.42E-03
FE(N,TRITON)	0.0	0.0	0.0	0.0	0.0	-2.25E-05
FE(N,HELIUM 3)	0.0	0.0	0.0	0.0	0.0	-2.38E-05
FE(N,ALPHA)	0.0	0.0	0.0	0.0	0.0	-3.35E-03
FE INELASTIC	0.0	0.0	0.0	0.0	0.0	-4.87E-02

TABLE 6

SENSITIVITY COEFFICIENTS OF NI FOIL IN INT 29
TO L16, P0, P8, L17, F19 PARTIAL CROSS SECTIONS
NON FISSIONE INTEGRAL EXPERIMENT

	ZONE 1	ZONE 7	ZONE 11	ZONE 15	ZONE 19	SUM
L16 ELASTIC	-3.73E-03	1.97E-03	1.44E-04	2.42E-05	4.15E-06	-1.50E-03
L16(N,2N)ALPHA	-1.76E-04	1.42E-05	1.26E-06	1.34E-07	1.07E-08	-1.61E-04
L16(N,N')ALPHA	-2.28E-03	1.46E-04	4.19E-06	9.96E-08	1.69E-07	-2.13E-03
L16(N,GAMMA)	-8.74E-08	-5.15E-08	-5.13E-09	-7.46E-10	-8.11E-11	-1.45E-07
L16(N,PROTON)	-5.96E-05	-3.19E-05	-2.88E-06	-3.86E-07	-4.75E-08	-9.47E-05
L16(N,ALPHA)T	-4.70E-04	-6.04E-05	-6.88E-05	-9.86E-06	-1.01E-06	-1.15E-03
L16 INELASTIC	-2.10E-03	1.57E-04	4.66E-06	1.81E-07	1.80E-07	-2.14E-03
L17 ELASTIC	-4.95E-02	2.19E-02	1.51E-03	2.16E-04	5.08E-05	-2.50E-02
L17(N,2N)	-1.38E-03	4.71E-05	3.95E-06	3.97E-07	2.63E-08	-1.31E-03
L17(N,2N)ALPHA	-2.46E-03	-3.14E-04	-1.13E-05	-3.95E-06	-4.63E-07	-2.81E-03
L17(N,N')ALPHA	-1.87E-02	3.84E-03	3.28E-04	4.31E-05	5.79E-06	-1.45E-02
L17(N,GAMMA)	-9.74E-07	-5.53E-07	-5.70E-08	-7.94E-09	-8.67E-10	-1.59E-06
L17(N,DEUTERIUM)	-7.83E-04	-1.58E-04	-1.47E-05	-1.82E-06	-2.11E-07	-9.57E-04
L17 INELASTIC	-2.47E-02	2.29E-03	9.29E-05	6.89E-06	3.89E-06	-2.23E-02
F19 ELASTIC	-4.06E-02	1.29E-02	2.88E-03	4.14E-04	8.01E-05	-4.29E-01
F19(N,2N)	-3.50E-03	1.42E-04	5.23E-06	2.77E-07	-1.95E-08	-3.35E-03
F19(N,N')ALPHA	-3.21E-03	3.35E-04	1.21E-05	4.11E-06	5.02E-07	-2.84E-03
F19(N,N')PROTON	-8.69E-04	-8.31E-05	-3.55E-06	-1.07E-06	-1.29E-07	-9.62E-04
F19(N,GAMMA)	-1.92E-06	-2.30E-06	-2.51E-07	-3.61E-08	-3.73E-09	-4.51E-06
F19(N,PROTON)	-1.58E-03	-6.29E-04	-5.28E-05	-6.84E-06	-8.39E-07	-2.27E-03
F19(N,DEUTERIUM)	-1.02E-03	-2.56E-04	-2.29E-05	-2.89E-06	-1.42E-07	-1.10E-03
F19(N,TRITON)	-6.40E-04	-1.45E-04	-1.36E-05	-1.71E-06	-2.01E-07	-8.05E-04
F19(N,ALPHA)	-2.91E-03	-1.83E-03	-1.59E-04	-2.11E-05	-2.67E-06	-4.92E-03
F19 INELASTIC	-5.43E-02	-5.56E-03	-8.46E-04	-1.10E-04	-5.62E-06	-6.09E-02
P0 ELASTIC	-1.08E-01	-2.08E-01	1.00E-01	1.45E-02	2.27E-01	-1.94E-01
P0(N,2N)	-3.47E-01	-3.10E-01	2.39E-01	3.29E-04	3.46E-05	-6.34E-01
P0(N,3N)	-1.17E-02	-1.11E-02	-6.30E-04	-7.07E-05	-8.10E-06	-2.19E-02
P0(N,GAMMA)	-6.77E-05	-1.93E-04	-5.05E-05	-7.24E-06	-9.92E-07	-1.19E-04
P0 INELASTIC	-6.33E-02	-1.18E-01	-1.65E-02	-2.41E-03	-3.14E-04	-2.01E-01
FE ELASTIC	0.0	0.0	0.0	0.0	0.0	-7.95E-03
FE(N,2N)	0.0	0.0	0.0	0.0	0.0	-4.10E-02
FE(N,N')ALPHA	0.0	0.0	0.0	0.0	0.0	-3.96E-04
FE(N,N')PROTON	0.0	0.0	0.0	0.0	0.0	-4.11E-03
FE(N,GAMMA)	0.0	0.0	0.0	0.0	0.0	-1.14E-04
FE(N,PROTON)	0.0	0.0	0.0	0.0	0.0	-1.65E-02
FE(N,DEUTERIUM)	0.0	0.0	0.0	0.0	0.0	-2.11E-03
FE(N,TRITON)	0.0	0.0	0.0	0.0	0.0	-5.42E-05
FE(N,HELIUM 3)	0.0	0.0	0.0	0.0	0.0	-3.97E-05
FE(N,ALPHA)	0.0	0.0	0.0	0.0	0.0	-5.00E-03
FE INELASTIC	0.0	0.0	0.0	0.0	0.0	-1.00E-01

TABLE 7

	ANISN	MORSE	ERROR	ANISN SLAB P ₃ S ₈	MORSE BARE SLAB	PERCENTAGE REDUCTION
²³⁵ U(n,p) in centre position of interval 19, NFIE	7.4166E-3	5.4651E-4	3.4%	0.26356	0.1857	30%
²³⁵ U(n,p) in off centre position of interval 19, NFIE	7.4166E-3	1.3232E-7	9.0%	3.3846E-2	3.3621E-2	0.6%
²³⁵ U(n,p) in off centre position of interval 19, NFIE	7.4166E-3	1.3232E-7	9.0%	0.134513	0.132961	1.15%
²³⁵ U(n,p) in off centre position of interval 19, NFIE	7.4166E-3	1.3232E-7	9.0%	7.45931E-3	5.4747E-3	27.0%
²³⁵ U(n,p) in off centre position of interval 19, NFIE	7.4166E-3	1.3232E-7	9.0%	1.3183E+6	9.1525E+5	31%
²³⁵ U(n,p) in off centre position of interval 19, NFIE	7.4166E-3	1.3232E-7	9.0%	4.73717E+5	4.633E+5	2.0%
²³⁵ U(n,p) in off centre position of interval 19, NFIE	7.4166E-3	1.3232E-7	9.0%	2.5528E+5	2.517E+5	1.4%
²³⁵ U(n,p) in off centre position of interval 19, NFIE	7.4166E-3	1.3232E-7	9.0%	2.5024E+7	2.4717E+7	1.2%
²³⁵ U(n,p) in off centre position of interval 19, NFIE	7.4166E-3	1.3232E-7	9.0%	1.19142E+6	8.891E+5	25.0%
²³⁵ U(n,p) in off centre position of interval 19, NFIE	7.4166E-3	1.3232E-7	9.0%	8.42739E+3	6.139E+3	27.0%
²³⁵ U(n,p) in off centre position of interval 19, NFIE	7.4166E-3	1.3232E-7	9.0%	1.11939E+5	4.1392E+4	63.0%
²³⁵ U(n,p) in off centre position of interval 19, NFIE	7.4166E-3	1.3232E-7	9.0%	7.83628E+5	6.723E+5	14.2%
²³⁵ U(n,p) in off centre position of interval 19, NFIE	7.4166E-3	1.3232E-7	9.0%	5.06647E+5	4.9076E+5	3.0%
²³⁵ U(n,p) in off centre position of interval 19, NFIE	7.4166E-3	1.3232E-7	9.0%	3.84752E+5	3.2624E+5	14.0%
²³⁵ U(n,p) in off centre position of interval 19, NFIE	7.4166E-3	1.3232E-7	9.0%	0.16	0.138	14.0%

TABLE 9. COMPARING ANISN AND MORSE FIE PERFORMANCE
PARAMETERS.
SOURCE IS AN ISOTROPIC ONE.

TABLE 8(a). COMPARING ANISN AND MORSE RESPONSES.

²³⁵ U(n,p) foil in interval 19 of NFR	0.0	0.5	1.0
²³⁵ U(n,p) foil in interval 19 of NFR	4.69	5.39	6.03
²³⁵ U(n,p) foil in interval 29 of NFR	8.981	9.928	10.79
²³⁵ U(n,p) foil in interval 32 of FIE	10.75	13.55	15.87

TABLE 8(b). FOIL UNCERTAINTY RESULTS.

SENSITIVITY COEFFICIENTS OF NI(N,P) FOIL IN INT 32 TO L16,
U238, U235, FE, LI7, F19 PARTIAL CROSS SECTIONS, FISSION INTEGRAL EXPERIMENT

	ZONE 3	ZONE 5	ZONE 7	ZONE 9	ZONE 11	SUM
16 ELASTIC	-2.71E-02	1.30E-02	1.23E-03	9.41E-05	8.61E-06	-1.28E-02
16(N,2N) ALPHA	-3.89E-03	-6.40E-04	-3.14E-05	-2.06E-06	-1.67E-07	-4.57E-03
16(N,N') ALPHA	-2.21E-02	-2.14E-03	-6.44E-05	-5.42E-06	-4.66E-07	-2.44E-02
16(N,GAMMA)	-7.53E-07	-2.92E-07	-1.55E-08	-1.85E-09	-2.48E-10	-1.06E-06
16(N,PROTON)	-6.43E-04	3.32E-04	-1.45E-05	-8.51E-07	-6.20E-08	-9.91E-04
16(N,ALPHA) T	-2.87E-03	-1.71E-03	-1.46E-04	-3.26E-05	-5.28E-06	-4.77E-03
16 INELASTIC	-2.23E-02	-2.11E-03	-6.30E-05	-5.39E-06	-4.67E-07	-2.45E-02
17 ELASTIC	-4.13E-01	1.82E-01	1.76E-02	1.35E-03	1.23E-04	-2.12E-01
17(N,2N)	-1.65E-02	-2.72E-03	-1.32E-04	-8.63E-06	-6.97E-07	-1.94E-02
17(N,2N) ALPHA	-2.47E-02	-3.94E-03	-1.67E-04	-9.90E-06	-7.28E-07	-2.89E-02
17(N,ALPHA) T	-2.12E-01	-5.29E-03	-2.17E-04	-4.21E-05	-4.96E-06	-2.17E-01
17(N,GAMMA)	-9.77E-06	-3.73E-06	-2.09E-07	-2.73E-08	-3.90E-09	-1.37E-05
17(N,DEUTERIUM)	-7.68E-03	-1.40E-03	-5.93E-05	-3.58E-06	-2.70E-07	-9.15E-03
17 INELASTIC	-2.59E-01	-2.27E-05	5.61E-04	1.37E-05	-4.95E-08	-2.58E-01
19 ELASTIC	-3.42E-01	1.96E-01	1.73E-02	1.33E-03	1.23E-04	-1.27E-01
19(N,2N)	-3.79E-02	-4.76E-03	-2.12E-04	-1.26E-05	-9.38E-07	-4.29E-02
19(N,N') ALPHA	-3.66E-02	-2.95E-03	-2.09E-04	-1.79E-05	-1.69E-06	-3.98E-02
19(N,N') PROTON	-8.99E-03	-1.42E-03	-6.31E-05	-3.89E-06	-2.98E-07	-1.05E-02
19(N,GAMMA)	-1.78E-05	-1.00E-05	-1.61E-06	-4.65E-07	-9.37E-08	-3.00E-05
19(N,PROTON)	-1.89E-02	-7.96E-03	-3.65E-04	-2.27E-05	-1.78E-06	-2.73E-02
19(N,DEUTERIUM)	-1.06E-02	-2.66E-03	-1.18E-04	-7.34E-06	-5.74E-07	-1.34E-02
19(N,TRITON)	-6.57E-03	-1.48E-03	-6.49E-05	-4.01E-06	-3.11E-07	-8.12E-03
19(N,ALPHA)	-4.00E-02	-2.44E-02	-1.08E-03	-6.42E-05	-4.78E-06	-6.56E-02
19 INELASTIC	-5.53E-01	-5.21E-02	-1.45E-03	-1.01E-04	-8.23E-06	-6.07E-01
ZONE 2	ZONE 4	ZONE 6	ZONE 10	ZONE 12	SUM	
2 ELASTIC	-2.54E-01	-1.31E-01	3.85E-02	2.76E-04	3.01E-05	-3.46E-01
2(N,2N)	-1.61E-01	-1.20E-01	-2.63E-03	-1.08E-05	-8.10E-07	-2.84E-01
2(N,N') ALPHA	-1.30E-03	-8.75E-04	-1.86E-05	-7.01E-08	-5.13E-09	-2.19E-03
2(N,N') PROTON	-1.46E-02	-1.06E-02	-2.34E-04	-9.73E-07	-7.44E-08	-2.54E-02
2(N,GAMMA)	-1.09E-04	-1.99E-04	-1.44E-05	-1.07E-06	-1.63E-07	-3.24E-04
2(N,PROTON)	-4.51E-02	-5.39E-02	-1.88E-03	-8.66E-06	-7.03E-07	-1.01E-01
2(N,DEUTERIUM)	-7.03E-03	-5.68E-03	-1.38E-04	-5.77E-07	-4.52E-08	-1.29E-02
2(N,TRITUM)	-1.70E-04	-1.14E-04	-2.44E-06	-9.18E-09	-6.81E-10	-2.87E-04
2(N,HELIUM 3)	-1.24E-04	-8.42E-05	-1.81E-06	-6.85E-09	-5.10E-10	-2.11E-04
2(N,ALPHA)	-1.44E-02	-1.57E-02	-5.04E-04	-2.34E-06	-1.91E-07	-3.06E-02
2 INELASTIC	-2.47E-01	-3.83E-01	-9.90E-03	-5.13E-05	-3.83E-06	-6.39E-01

ZONE 1 ZONE 8 SUM

238 ELASTIC	-4.65E-01	5.26E-03	-4.59E-01
238(N,2N)	-1.83E-01	-7.34E-04	-1.83E-01
238(N,3N)	-1.22E-01	-1.31E-04	-1.22E-01
238 TOTAL FISSION	-2.22E-01	4.75E-04	-2.22E-01
238(N,GAMMA)	-6.14E-04	-1.21E-04	-7.35E-04
238 TOTAL INELASTIC	-7.67E-02	-7.65E-04	-7.75E-02
235 ELASTIC	-3.39E-03	3.96E-05	-3.35E-03
235(N,2N)	-4.53E-04	-1.65E-06	-4.55E-04
235(N,3N)	-6.04E-05	-5.92E-08	-6.05E-05
235 TOTAL FISSION	-1.79E-03	2.96E-04	-1.49E-03
235(N,GAMMA)	-8.35E-06	-1.44E-06	-9.79E-06

TABLE 11.

SENSITIVITY OF NI(N,P) FOIL AT A CENTRE AND OFF CENTRE POSITION
IN THE FISSION INTEGRAL EXPERIMENT
SENSITIVITIES ARE EVALUATED FROM MORSE FLUXS

TABLE 10.

L16 ELASTIC	-1.1573E-4	L16 ELASTIC	-2.1245E-1
L16(N,PROTON)	-2.3644E-5	L16(N,PROTON)	-7.2398E-2
L16(N,ALPHA) T	-8.7547E-5	L16(N,ALPHA) T	-7.2223E-2
FE ELASTIC	-1.2239E-4	FE ELASTIC	-2.0234E-1
FE(N,DEUTERIUM)	-2.4435E-4	FE(N,DEUTERIUM)	-3.3086E-2
FE INELASTIC	-8.2975E-4	FE INELASTIC	-5.7291E-2
L17 ELASTIC	-2.0150E-3	L17 ELASTIC	-3.4901E+0
L17(N,N' ALPHA) T	-7.5884E-3	L17(N,N' ALPHA) T	-4.4444E+0
L17 INELASTIC	-8.1702E-3	L17 INELASTIC	-6.3990E+0
U238 ELASTIC	-5.6280E-5	U238 ELASTIC	-1.4485E-2
U238(N,3N)	-9.8608E-3	U238(N,3N)	-8.2171E-2
U238 TOTAL FISSION	-1.6511E-2	U238 TOTAL FISSION	-2.0690E+0
U235 ELASTIC	-3.4512E-7	U235 ELASTIC	-9.5954E-5
U235(N,3N)	-4.9069E-6	U235(N,3N)	-9.8345E-4
U235 TOTAL FISSION	-2.1593E-4	U235 TOTAL FISSION	-2.1040E-2

SENS OF NI FOIL AT CENTRE
POSITION IN FIE AT INT 32
SENS OF NI FOIL AT OFF CENTRE
POSITION IN FIE AT INT 32

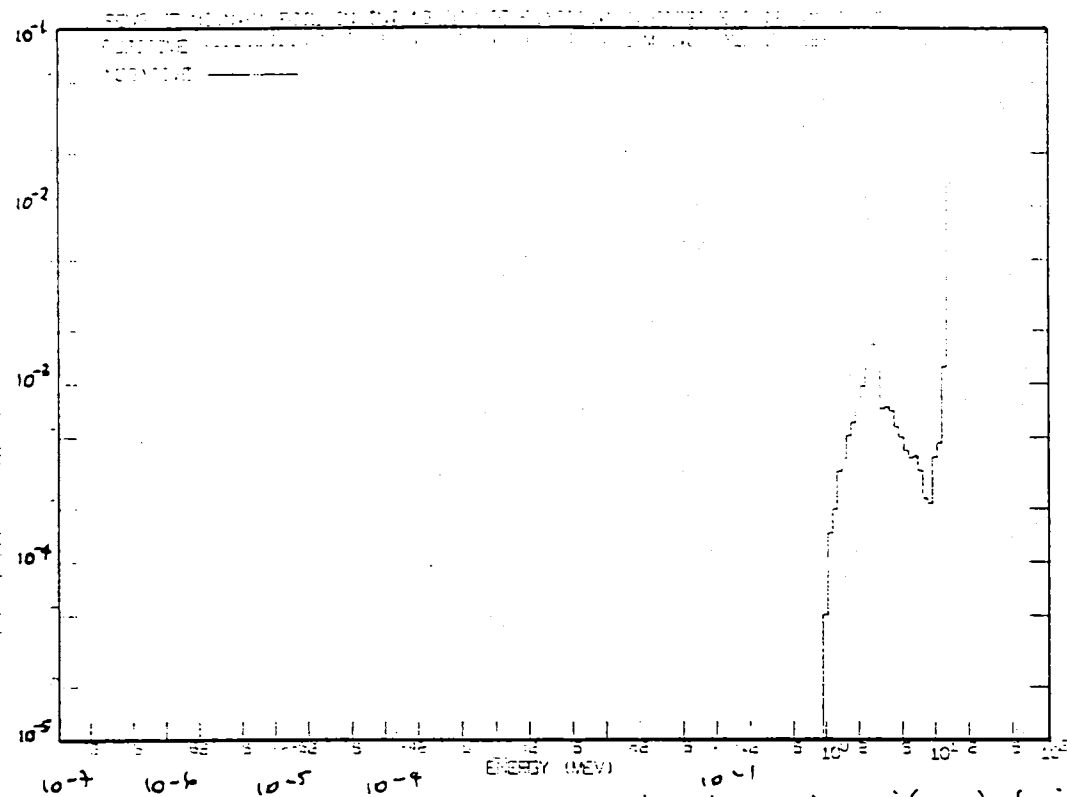


FIG 1. sensitivity of Ni(n,p) foil in interval 19 to 6Li elastic, NFIE.

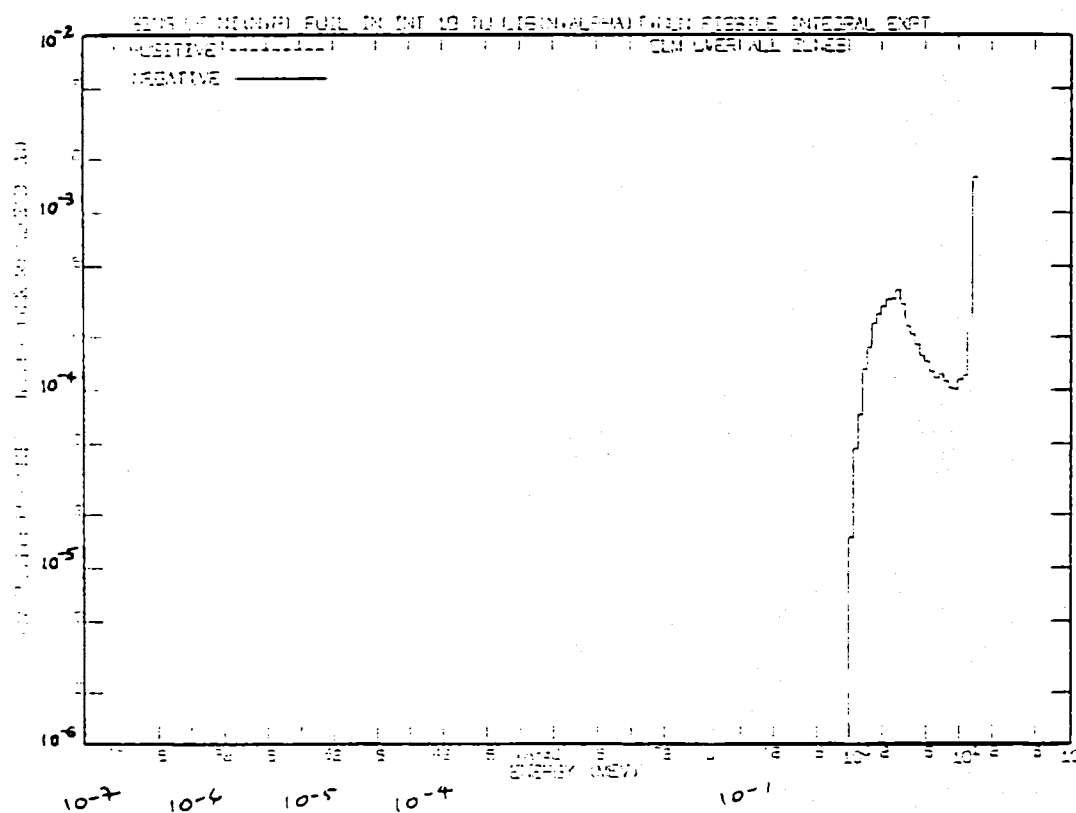


FIG 2. sensitivity of Ni(n,p) foil in interval 19 to 6Li(n,α)T, NFIE.

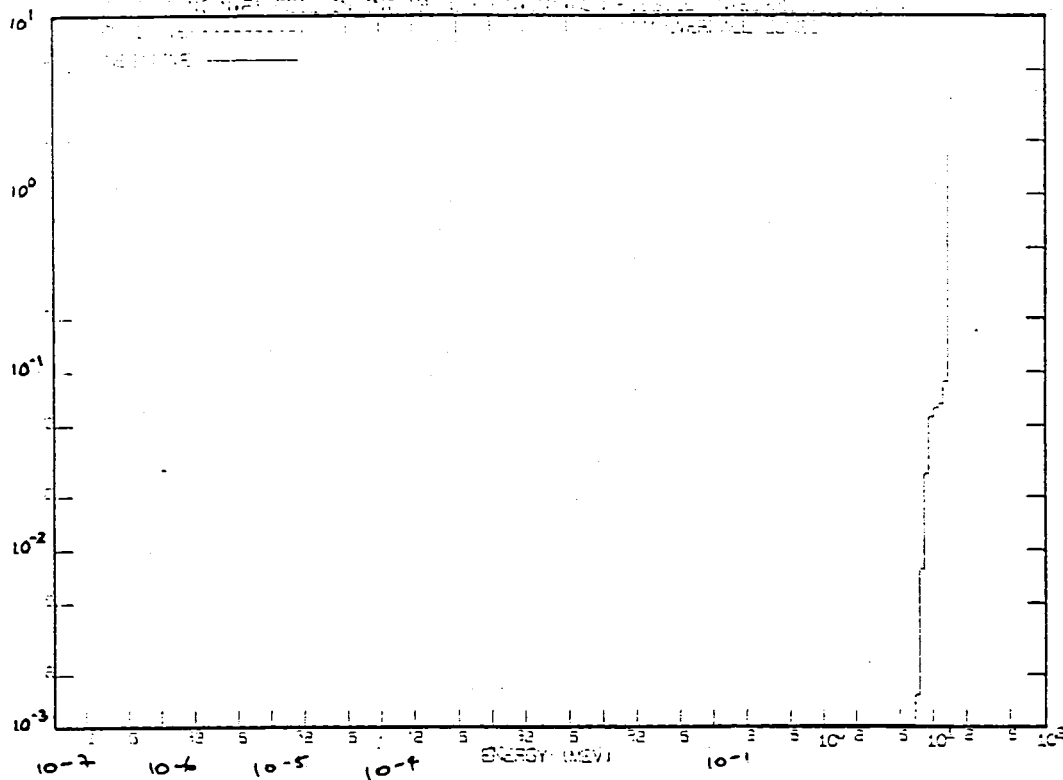


FIG 3. sensitivity of Ni(u,p) foil in
interval 19 to Pb(n,2n)
in NFIE.

Sensitivity of Ni(u,p) foil.
in a centre position to
Pb(n,2n) in NFIE.

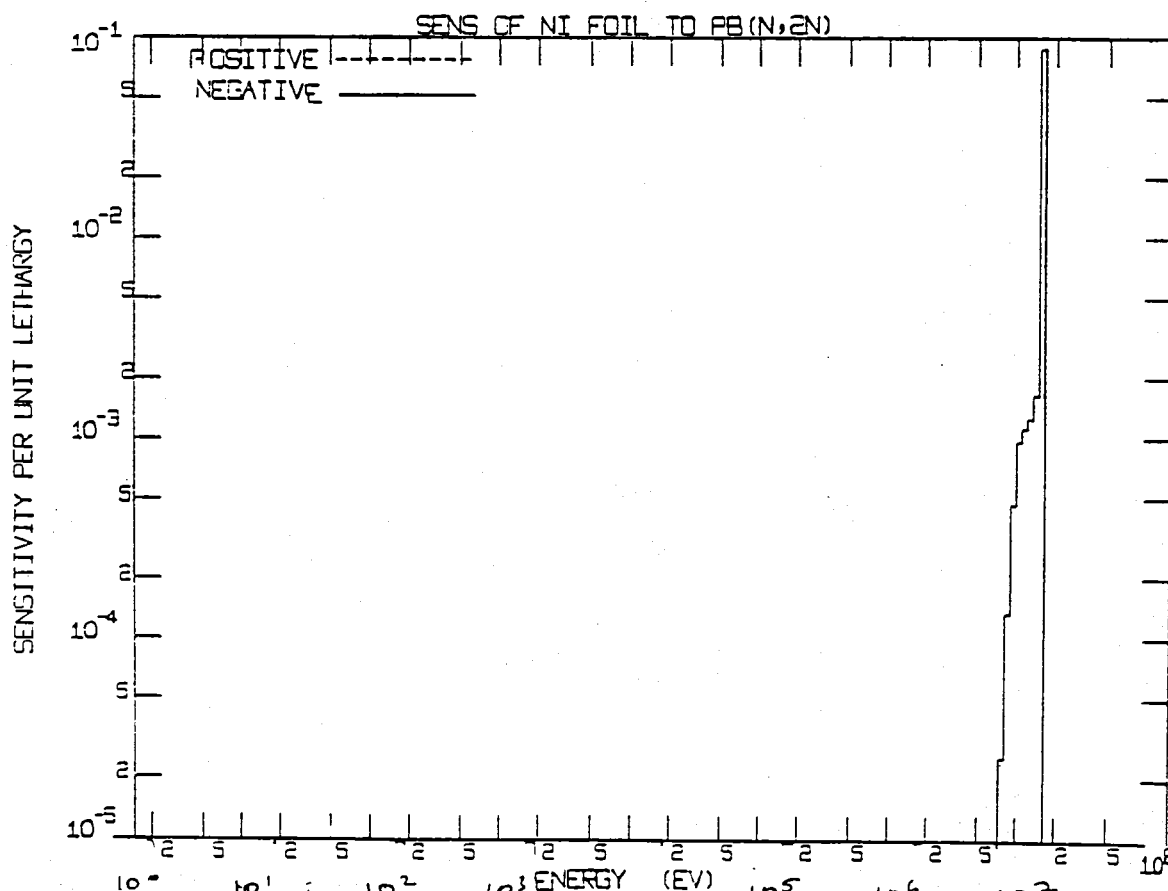


FIG 4.

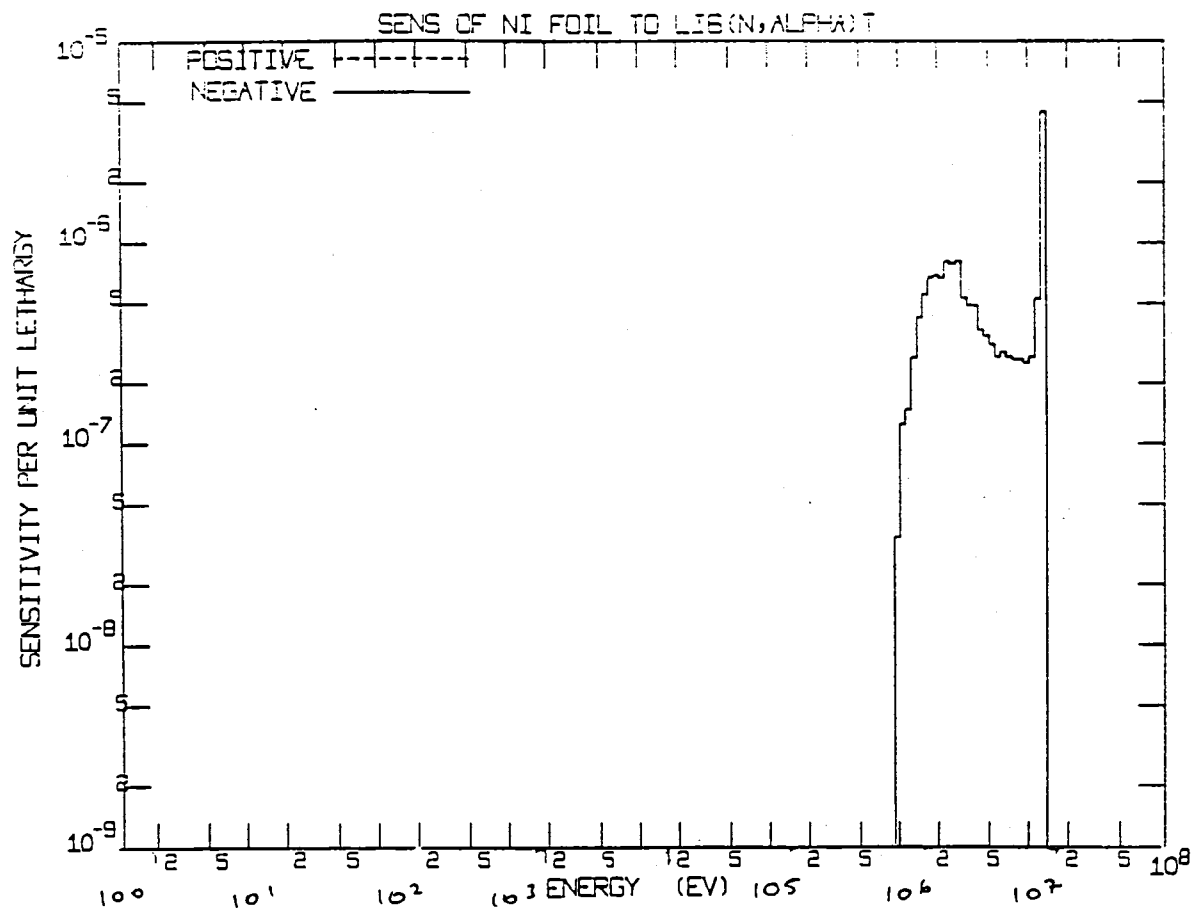
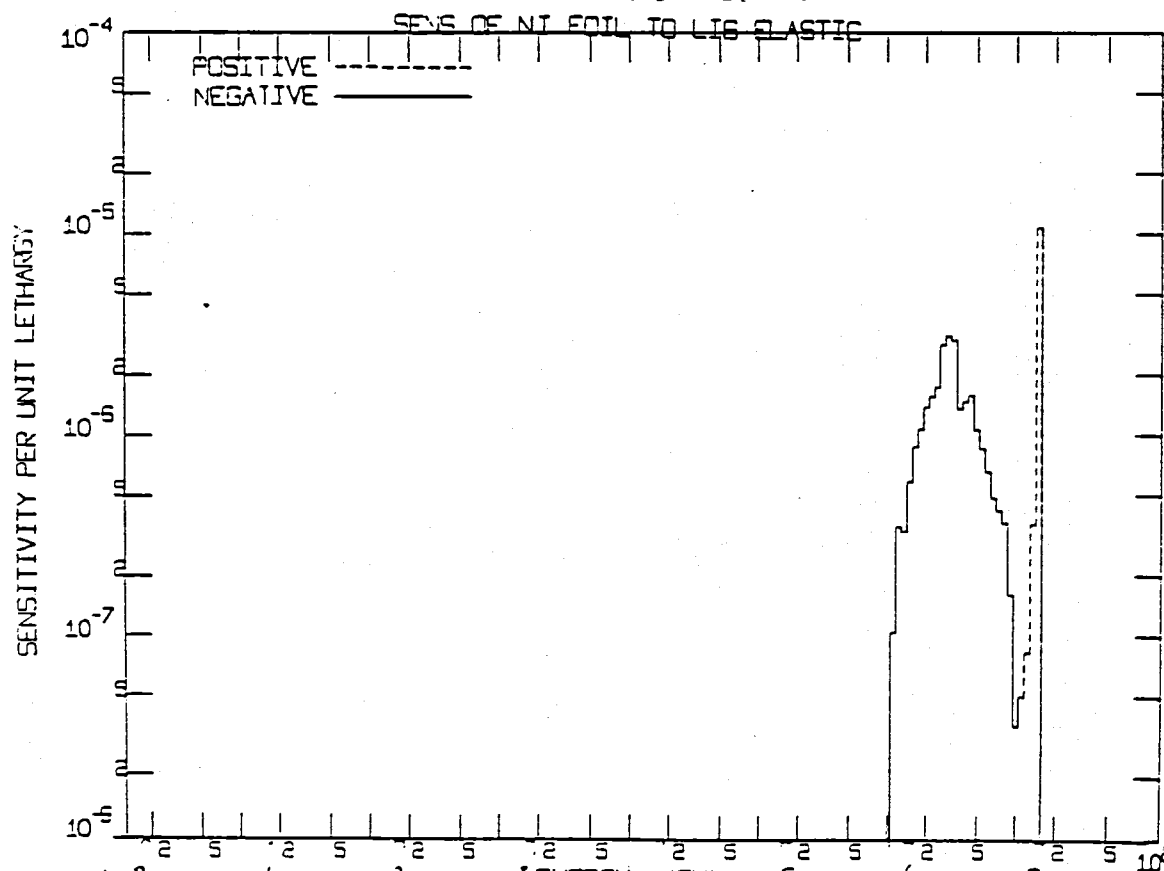


FIG 5. sensitivity of Ni(4,p) foil
in a centre position
to $\text{Li}(4,\alpha)\text{T}$, in NFIE

FIG 6. sensitivity of Ni(4,p) foil.
in a centre position to
 Li elastic in NFIE



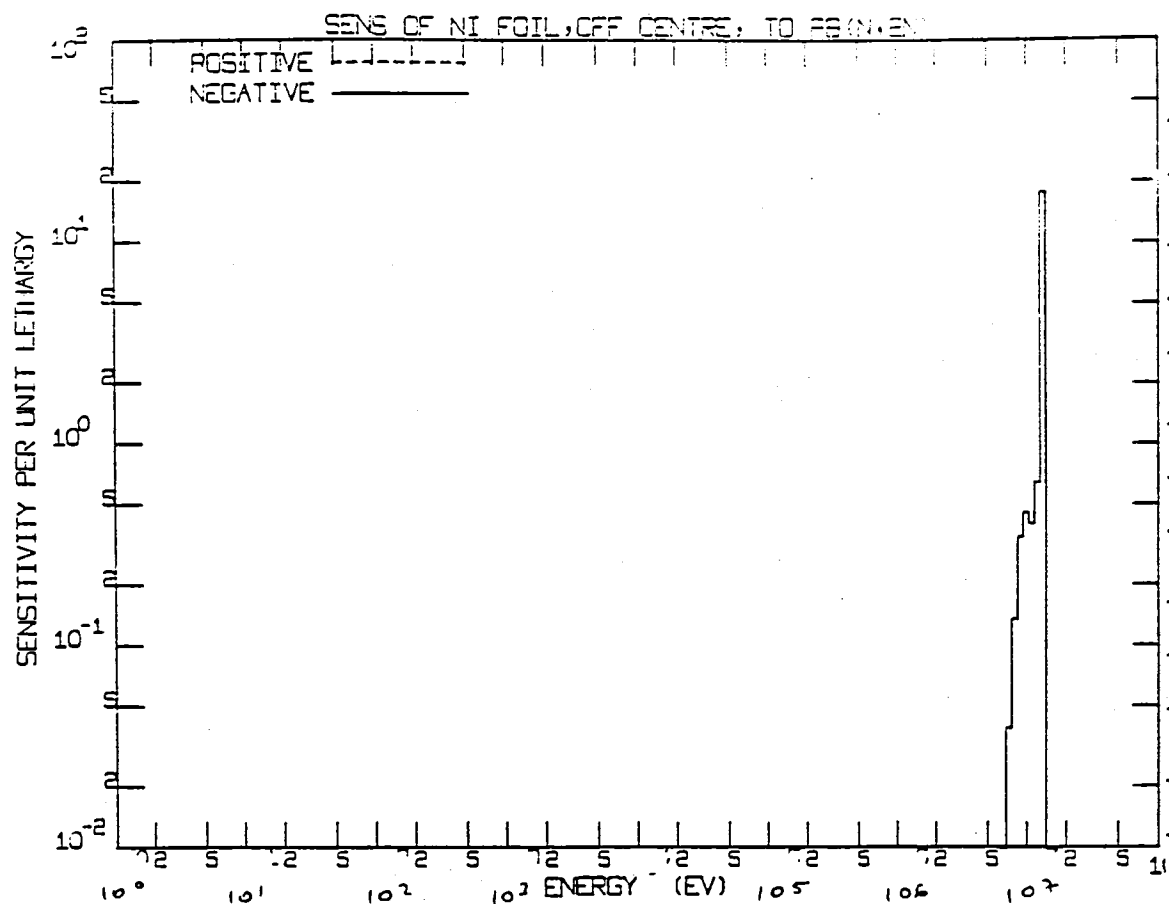
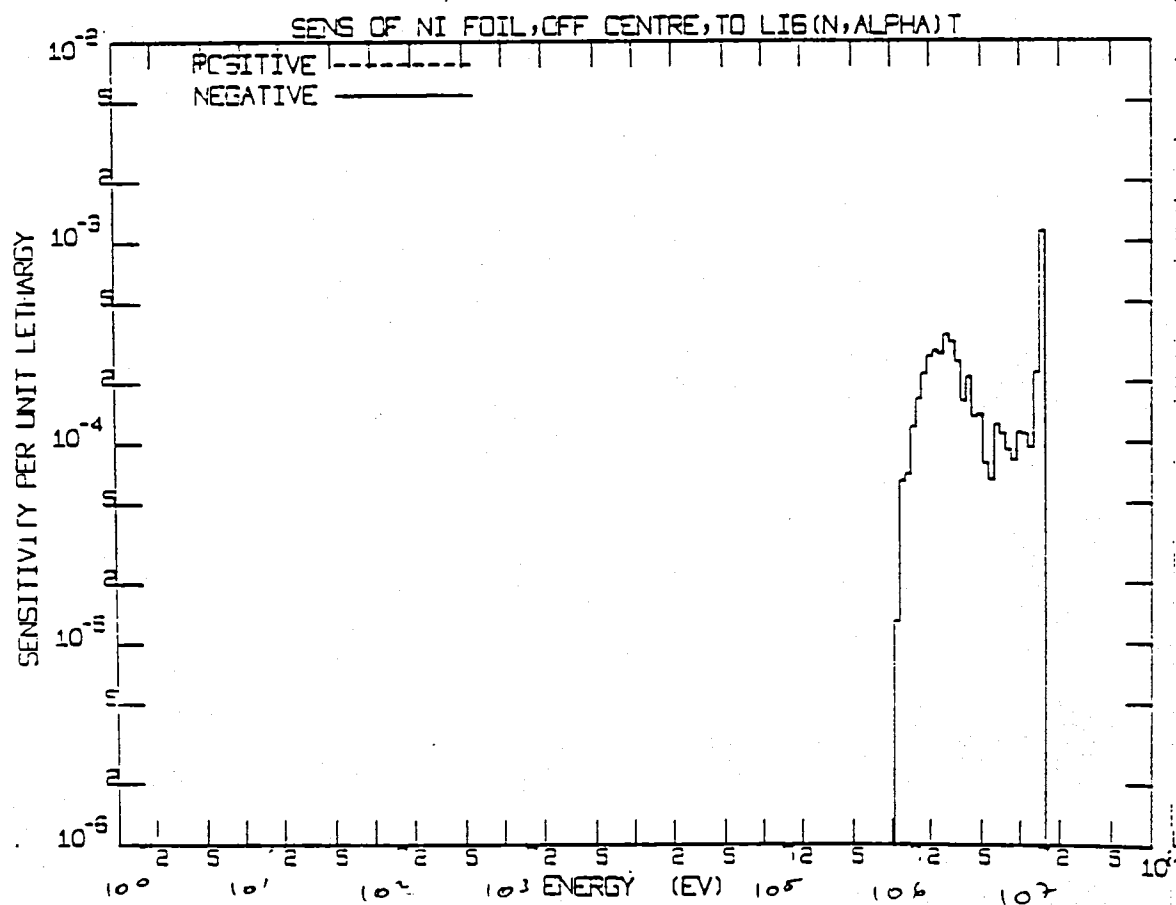


FIG 7. sensitivity of $Ni(u,p)$ foil in an off-centre position to $Pb(u,2n)$ in NFIE

FIG 8. sensitivity of $Ni(u,p)$ foil in an off-centre position to $Li(u,\alpha)$ in NFIE



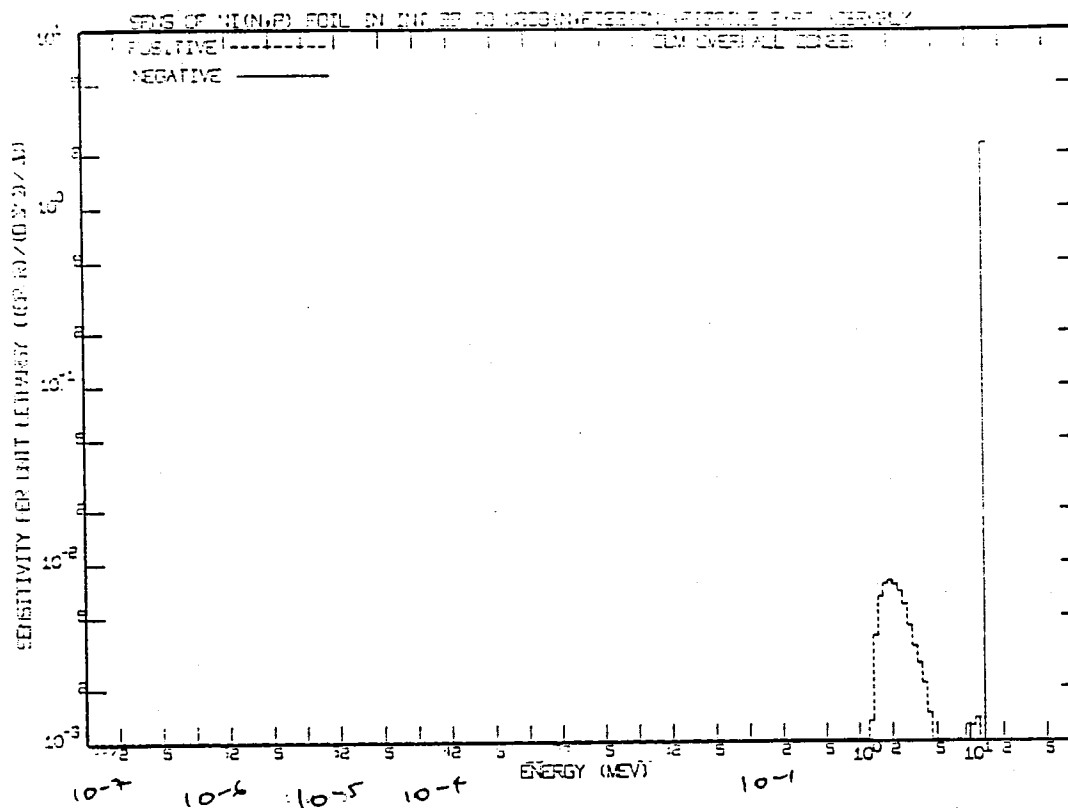
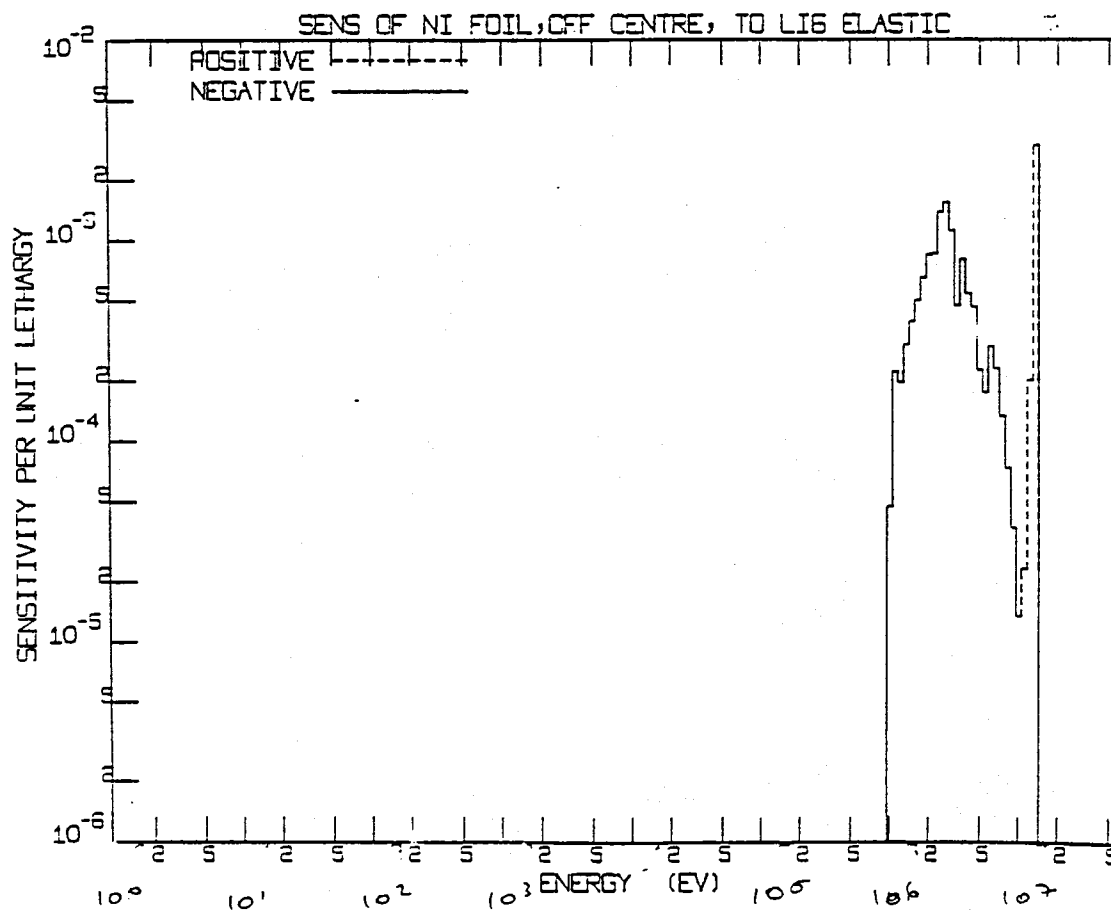
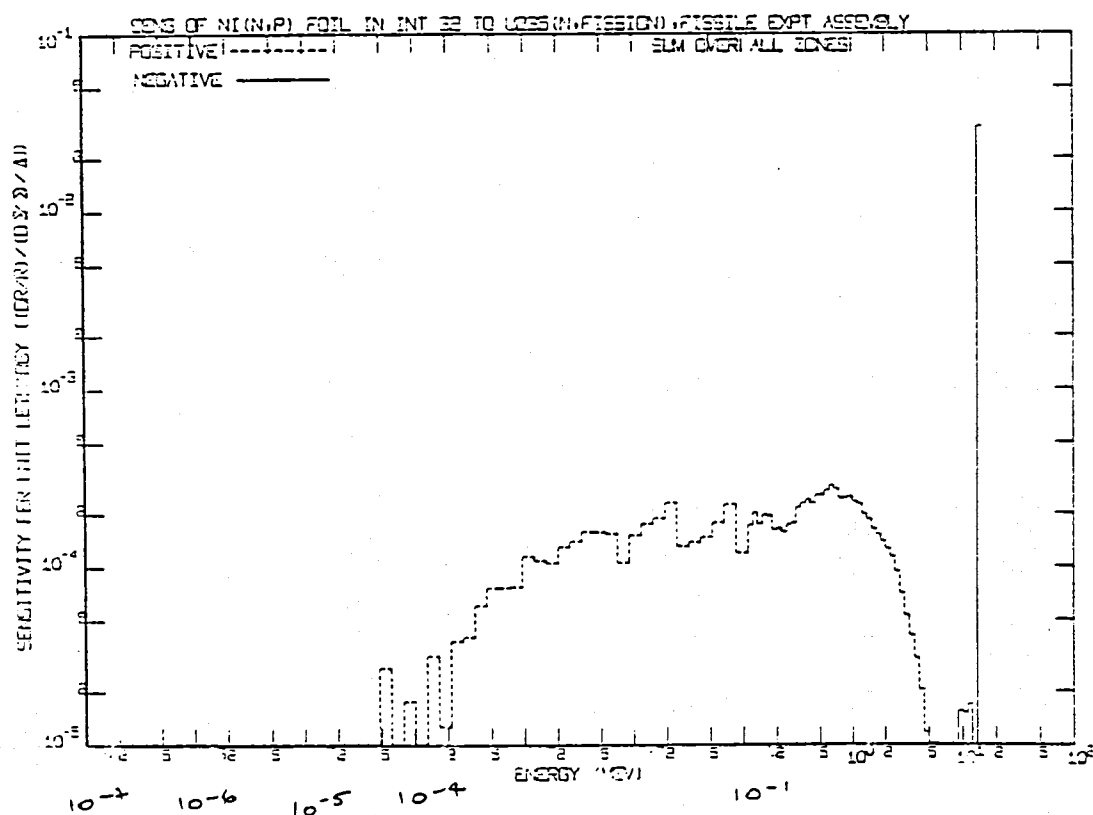
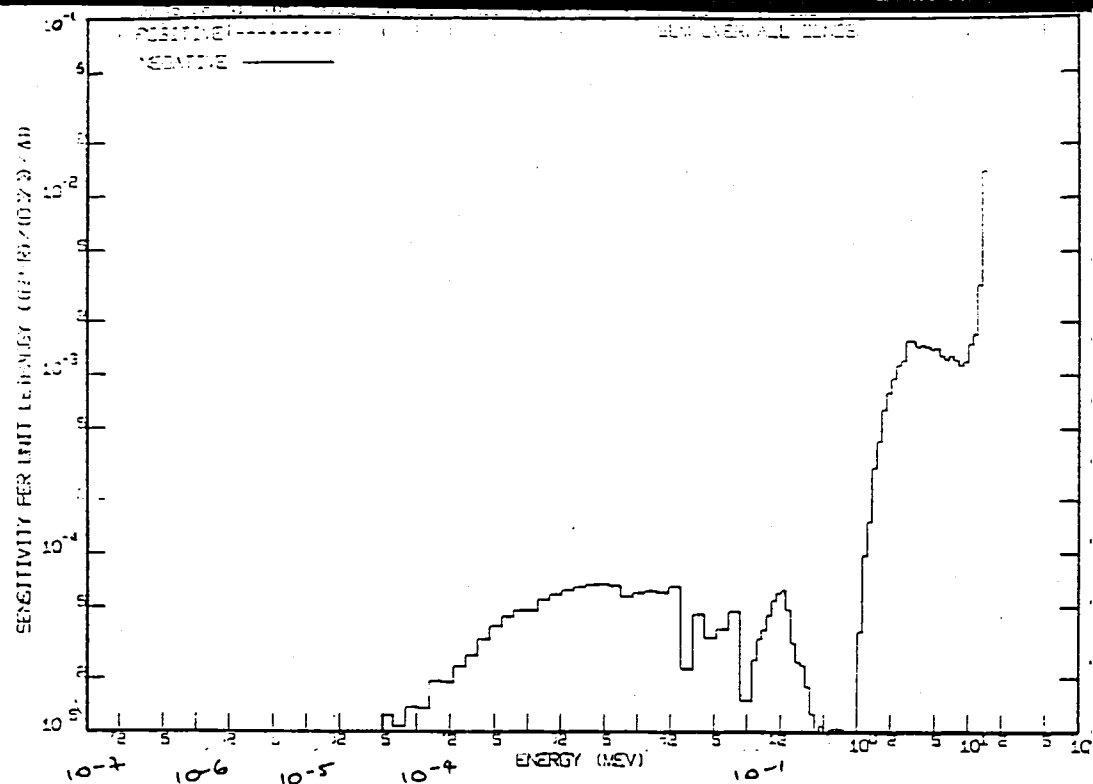


FIG 10. sensitivity of Ni(n,p) foil in interval 32 to ²³⁸U(n,f) in F

FIG 9. sensitivity of Ni(n,p) foil in an off centre position to elastic, NF





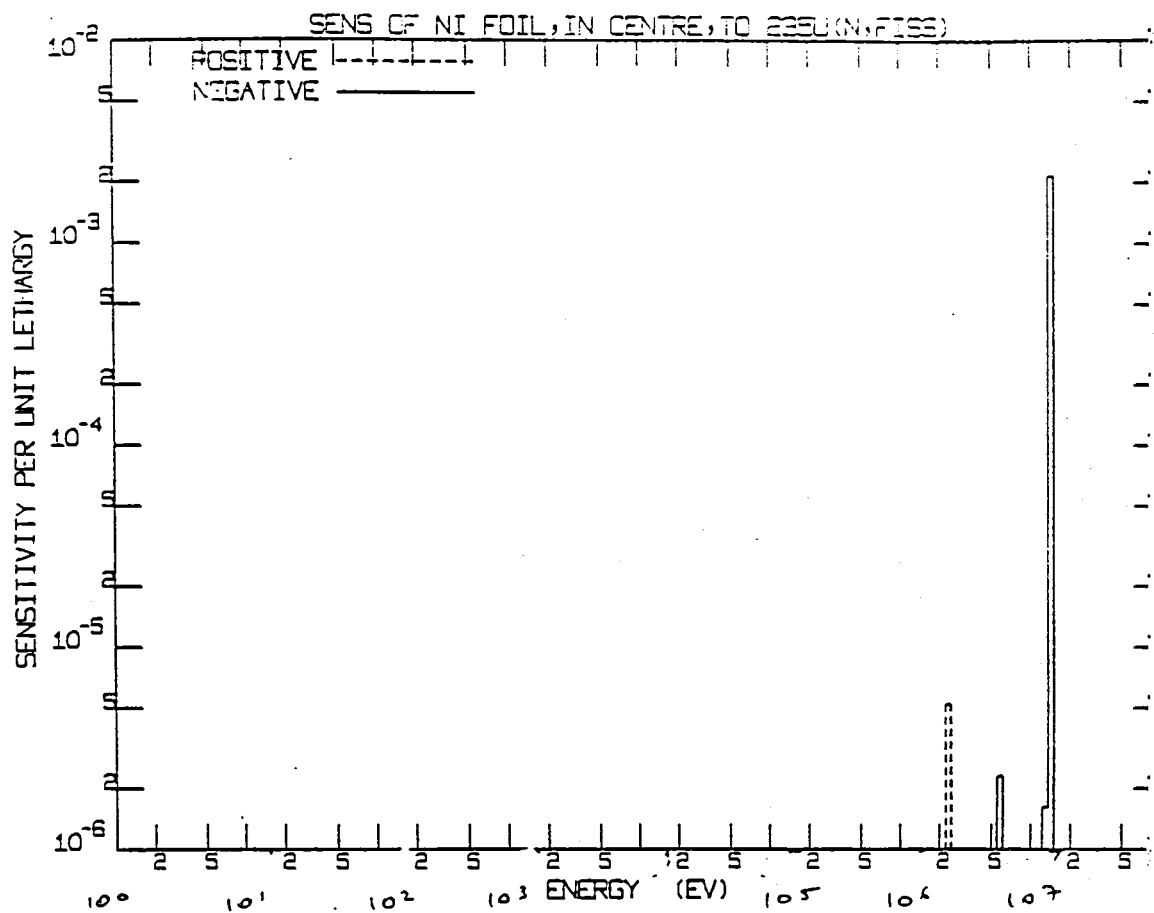
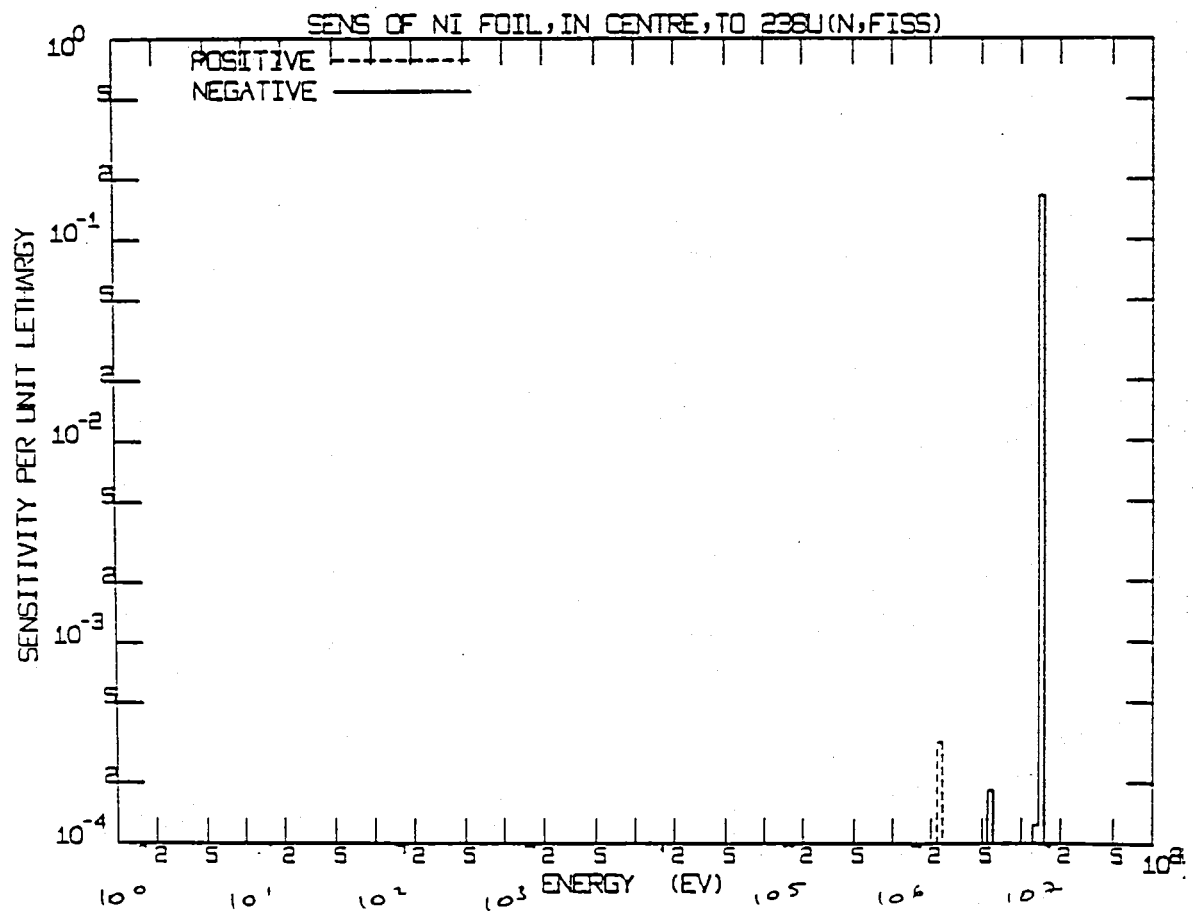


FIG 14. Sensitivity of Ni(u,p) foil
in a centre position to
 $^{235}\text{U}(\text{u},\text{f})$ in FIE.

sensitivity of Ni(u,p) foil in a
FIG 13. centre position to $^{238}\text{U}(\text{u},\text{f})$ in FIE



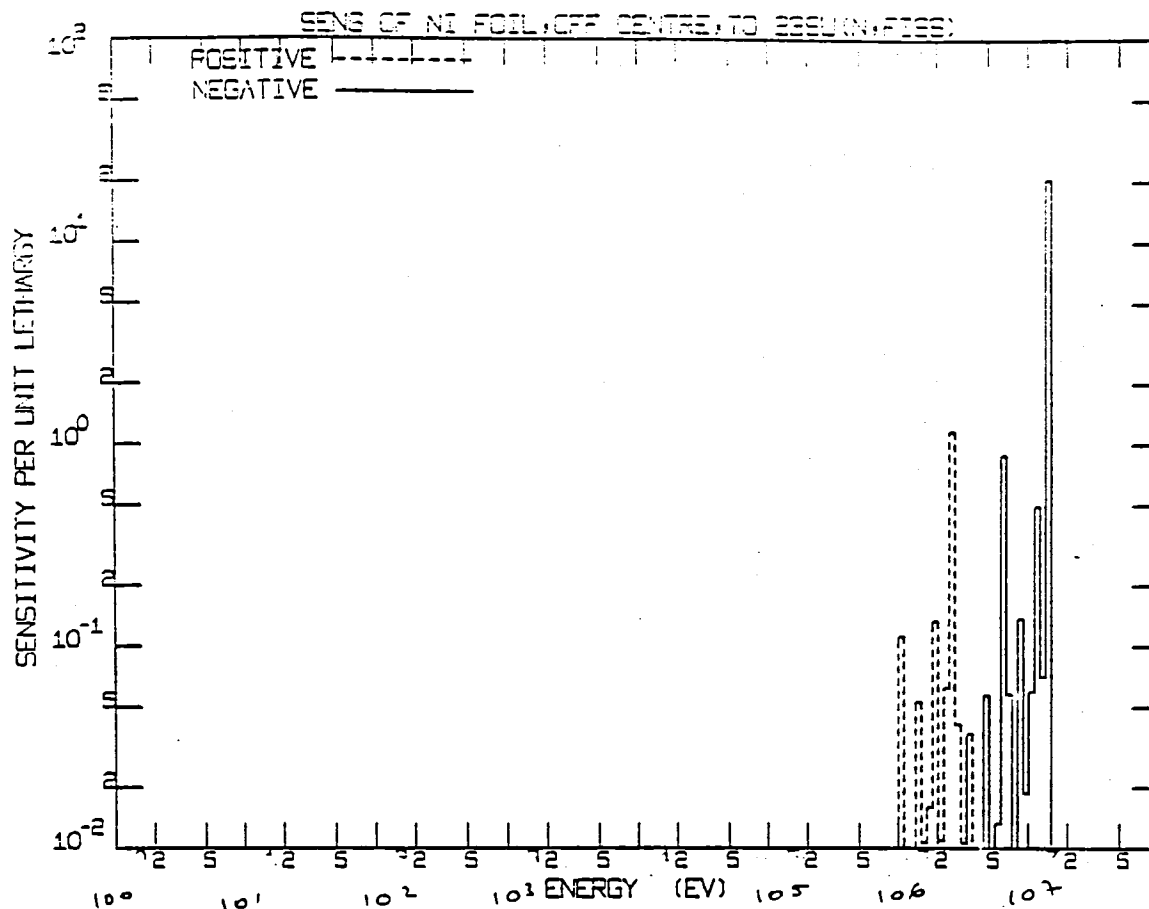
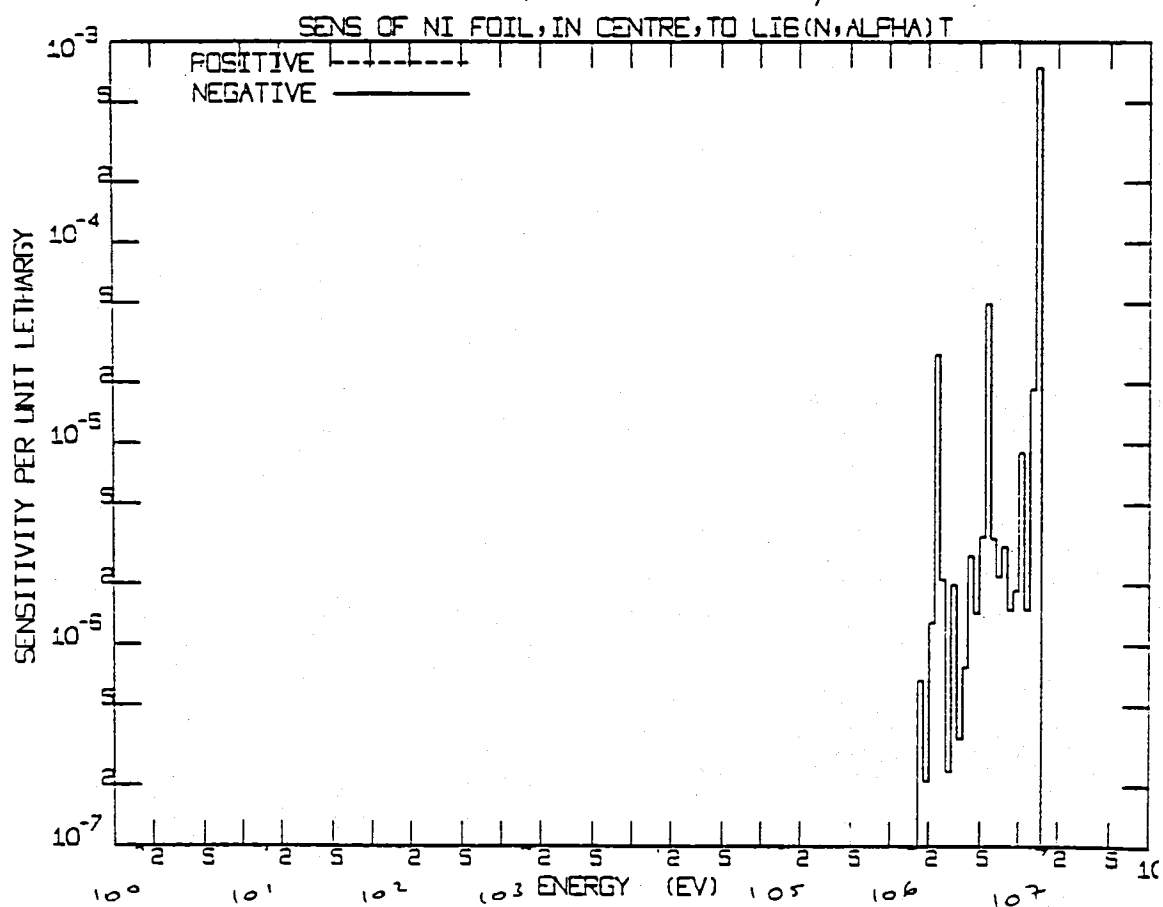


FIG 15. sensitivity of Ni(n,p) foil in
interval 32 in a centre position
to $^{238}\text{U}(\text{n},\text{f})$, FIE.



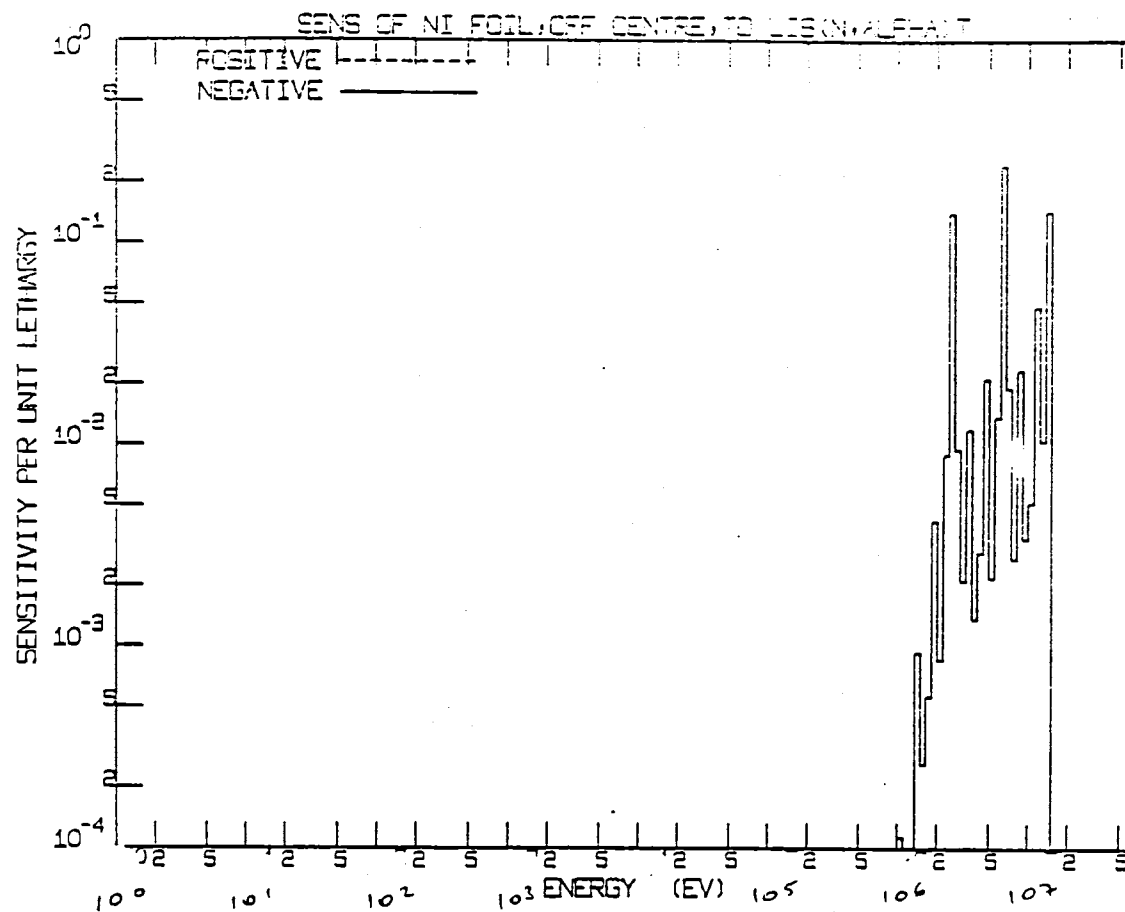
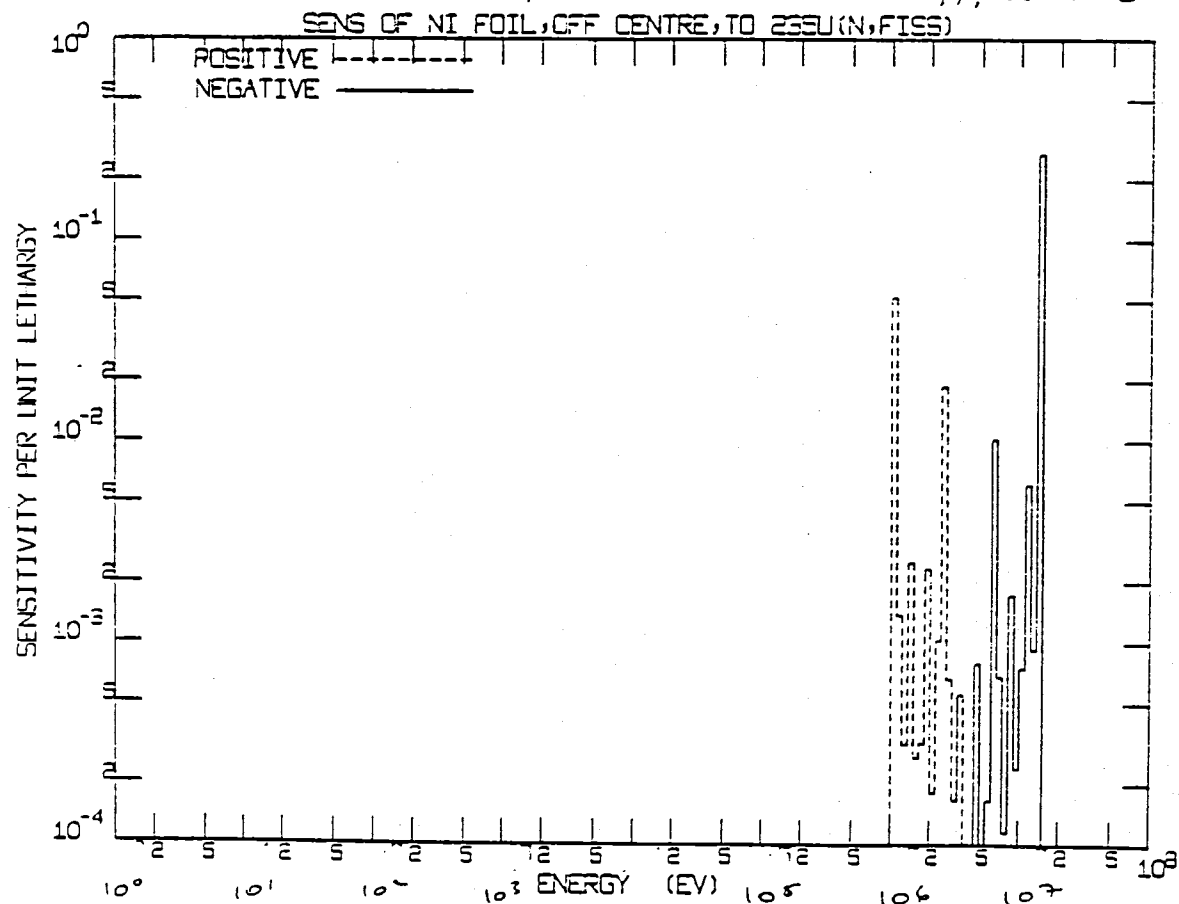


FIG 18. sensitivity of Ni(n,p) foil in interval 32 in an off centre position to ${}^6\text{Li}(n,\alpha)T$, in FIE

FIG 17. sensitivity of Ni(n,p) foil in interval 32 in an off centre position to ${}^{235}\text{U}(n,f)$, in FIE.



CONCLUSIONS AND FURTHER WORK

The main aim of this work is to form the first half (computational) of a cross section adjustment scheme. The relevant responses and sensitivities have been obtained which will ultimately allow the 100 energy group structure form of the ENDF/B-4 cross sections of the Li6, Li7, Fe, 19F, Pb, U238 and U235 isotopes to be adjusted by the least squares sensitivity method. The sensitivities are held on tape and are readily available.

Two fusion reactors have been designed, the HIBALL non fissile design (Badger 1981) and a fissile reactor using a low gain (~ 8) pellet target (Sim 1984). Both these reactors have been analysed using sensitivity theory. The non fissile and fissile integral experiments associated with these reactors have been designed using 1-dimensional transport theory, which will contain all the modelling errors, discrete ordinates errors and cross section uncertainties in such a method. A summary of the materials and their thicknesses for both integral experiments is given in table 1. The two fusion reactors and their associated integral experiments have had an uncertainty analysis made on them. In addition, for the integral experiments, a correct 3-dimensional representation with particle leakage has been made using a Monte Carlo code. Three dimensional sensitivity coefficients have also been derived from this work, as well as sensitivity information pertinent to the response foils which will be used in the cross section adjustment.

254

As can be seen by comparing the system parameters of the 3-dimensional work (chapter 10) with the 1-dimensional work for the non fissile (chapter 7) and fissile (chapter 8) experiments, the 1-dimensional work leaves a lot to be desired. The parameters for the experiments are reduced, and for some responses very much so. Perhaps the biggest approximation used in this work was in the 1-dimensional representations of the slab, which cannot accurately take account of side leakages. However, in designing these integral experiments the effort was devoted to constructing such assemblies which could physically easily be built, and there was no way of being able to use any other geometry except a slab. Such a slab assembly will need little, if any, structural support. Some work has been done (though not presented) on a spherical integral experiment of the non fissile reactor and the tuning between the sensitivity profiles was excellent, except for a few resonance absorption peaks due to the fluorine. Thus the adjusted cross sections obtained from this integral assembly, by comparing real and computational integral foil response measurements, can be used with confidence for a neutronic recalculation on the fusion reactor. The biggest difference in the sensitivity profiles between the two reactors and their experiments is due to the change in geometry, with the increased neutron leakage in the slab assemblies. However, even with the above considerations, the radiation and neutronics environment between the reactors and experiments is quite similar, as can be seen by the likeness of the sensitivity profiles shown in chapter 10.

As for the non fissile reactor (NFR), the lead in the non fissile integral experiment (NFIE) design dominates the particle softening and transport processes. It is the lead that produces

the flux spectrum of the NFR and so it had to be retained in fairly large amounts in the NFIE to force the spectrum to have the required similar shape. Indeed, most of the NFR and NFIE dynamics can be explained with reference to $Pb(n,2n)$, Pb elastic, $Pb(n,3n)$ and Pb inelastic cross sections. Since the amount of lead had to be reduced, because of the weight constraint, with the resulting reduction in the number of low energy neutrons, extra sets of iron slabs were placed between the LiF zones. These iron zones were stripped from the reflector so there was no gain in weight in making this modification. A bonus here is that the stainless steel cladding, surrounding the LiF powder, is catered for neutronically. The presence of the fluorine introduces some extra absorption structure (in the 10-40 Kev energy range) but is very useful for additional particle downscattering. It appears that the tuning is quite good at energies above about 2-3 MeV. This is due to there being less neutron leakage at higher energies, because the total cross section is generally higher at these energies. At very low energies the tuning is not so exact, and in particular the amount of flux at these energies is much less in the NFIE due to the higher leakage. The weight of the NFIE is 3 metric tonnes which is low enough that no additional structural support will be needed in the containing room.

For both the fissile reactor (FR) and the fissile integral experiment (FIE), the uranium is dominant in determining the radiation dynamics. This is a function of its position physically close to the source and of its high neutron multiplication cross sections ($(n,2n)$, $(n,3n)$ but $(n,fission)$ mostly) which determines the energy dependent shape of the flux spectrum entering the rest of the assembly. The biggest

difference between the reactor and experiment is that half of the uranium in the integral experiment was put in the central region, in the environment of a softer neutron flux. In this way the neutron economy could be increased and the amount of iron reflector could be reduced whilst keeping the total amount of uranium constant. It could be argued that the placing of half the uranium in the 2nd zone makes the ^{238}U isotope redundant. Whilst this is largely true, it should be remembered that it is only the similarity of sensitivity profiles that is desired - a condition which has been carefully preserved in this work. In addition the fission process in the ^{235}U isotope ensures that the average neutron energy remains quite high at the mid point of the experiment. The weight of the FIE is 3.1 metric tonnes. The experiment uses natural LiF and natural iron slabs as a neutron moderator. The tuning between the FR and FIE is very good down to about an energy of 1 Kev. Below this energy there is rapid flux attenuation due to the presence of the enriched lithium in the reactor, which accounts for the profile difference. The enriched LiF of the FIE shows roughly the same profiles as that of the FR but it is not envisaged that the fissile assembly will be built with the enriched material.

As regards the uncertainty analysis, the errors of the reactor parameters are in general not too large for any of the neutronic systems. This is in view of the fact that approximate 1-dimensional sensitivity theory was used with estimated correlation files and very conservative cross section errors. However, it must be remembered that for the reactors, these are only partial 'total' errors, since not all the elements have had their sensitivities obtained, only those which will subsequently be used as a material in an integral experiment.

257

From the results of this uncertainty analysis, the cross section partials needing re-evaluation are $Pb(n,2n)$, $Pb(n,3n)$, the Pb inelastic scattering levels, the Fe inelastic scattering levels and the $Fe(n,proton)$ cross section partial. In addition, because of the isotope dependence of the nuclear energy release (Q value) of nuclear reactions, individual neutron reaction cross sections must be known by isotope. In particular, problems occurred when trying to process the kerma factors for the lead partials. The direct term on the $Pb(n,3n)$ partial was omitted because it gave a negative kerma factor even when the ENDF/B-4 Q value was used in the calculations. The uncertainty analysis work presented in this thesis is called a standard error analysis because no secondary energy distribution (SEDs) uncertainties have been included and it is only the production cross sections of the secondaries which have been used to obtain the results. Very roughly, for a 121 group standard cross section set, 84% of all non zero table positions pertain to the description of secondary energy distributions and only 16% to the actual reaction cross sections. The SEDs error formalism is now well developed (chapter 9) but as yet the complete files, or otherwise, for the SEDs are non existent. Effort should be directed into the evaluation and release of SED uncertainty files since, without the inclusion of these uncertainties, the error analysis presented here must be viewed as a first order and rather poor approximation.

As discussed in chapter 10, the profile distortion gives a measure as to how the neutronic environments of ANISN (from the fluxes of which the integral experiments were constructed) and MORSE compare. The MORSE results correspond to the real world - within the bounds of accuracy the MORSE results are those that an

experimentalist would obtain if the assembly were to be physically built and the same measurements performed on it. From the chapter 10 results it is seen that the amount of profile distortion between the MORSE and ANISN results is quite small, particularly when comparing the ANISN foil results to the MORSE results with the foil located in a central position. The main reason for the profile distortion being so small is the high energy response nature of the foils being used. Since the neutronic environments are very similar for the ANISN and MORSE representations at the higher energies (less leakage), it is to be expected that high energy response functions (ie $Ni(n,p)$ and $^{232}Th(n,f)$) would show the same sort of behaviour. Since the FIE spectrum is harder than that of the NFIE, and the foil sensitivity comparison between ANISN and MORSE is so close for the NFIE, it is to be expected that the foil sensitivities will be even closer for the FIE case. The results show that this small profile distortion is definitely true for the NFIE and would probably be true for the FIE, though because of statistical uncertainties (ie the foil is so small) no definite conclusions can be reached. If a low energy response function had been used as a foil then the ANISN and MORSE profiles would be very dissimilar at these energies. Therefore, in summary, it can be said that the ANISN results are a good representation of the behaviour of the integral experiments calculated by the MORSE code and that the sensitivities obtained from it can be used for a cross section adjustment scheme with confidence. Additionally, as seen in chapters 7 and 8, the degree of tuning, a measure of the likeness of the neutronic environments of the experiment and the reactor, is also quite close between the designed experiments and the associated fusion reactors. Hence, data (cross section) adjustments obtained from using the ANISN and SWANLAKE calculated

25

sensitivities and responses made on the integral experiments, will be relevant to the fusion reactors.

Finally mention should be made of a few uncertainties that are inherent in this type of work. The 1-dimensional code ANISN can only use one type of fission spectrum during any one particular run. Such fissile results are likely to be quite dependent on the fission spectrum used, and so it was decided to make some direct calculations to determine the effect of the spectrum. Table 2 gives some response changes for two different fission spectrums. The comparison is between the ENDF/B-4 spectrum (labelled 1) used throughout this work, and the fission spectrum (labelled 2) that was used to analyse and design the fissile reactor (Sim 1984). This latter spectrum comes from Joanou and Dudek. The FR contains a high reactivity and shows a substantial change in its reactor parameters between the two spectra. The changes for the FIE are not particularly large. This large change gives cause for concern and effort should be directed into obtaining a more standard fission spectrum.

The final recommendation, and the most important, is that the integral experiments are built. Not only can cross section adjustment then take place but also the direct 'calculation against experiment' comparison can be made, yielding the inaccuracy or otherwise of the calculational models. Finally, the above integral experiments should have their photon field fluxes analysed and have pertinent sensitivities obtained for when a photon detector becomes available at Birmingham. In summary, it could be said that this work has uncovered more problems than solutions (eg the need for SED error files), but at least it has highlighted the most important questions to be asked.

		<u>NFIE</u>	<u>FIE</u>
zone	1	Pb, 5.9cm	U, 2.4cm
	2	Fe, 0.5cm	Fe, 2cm
	3	LiF, 1.3cm	LiF, 6.3cm
	4	Fe, 0.5cm	Fe, 2cm
	5	Pb, 5.9cm	LiF, 6.3cm
	6	Fe, 0.5cm	Fe, 2cm
	7	LiF, 1.3cm	LiF, 6.3
	8	Fe, 0.5cm	U, 2.4cm
	9	Pb, 5.9cm	LiF, 6.3cm
	10	Fe, 0.5cm	Fe, 2cm
	11	LiF, 1.3cm	LiF, 6.3cm
	12	Fe, 0.5cm	Fe, 2cm
	13	Pb, 5.9cm	LiF, 6.3cm
	14	Fe, 0.5cm	Fe, 5cm
	15	LiF, 1.3cm	
	16	Fe, 0.5cm	
	17	Pb, 5.9cm	
	18	Fe, 0.5cm	
	19	LiF, 1.3cm	
	20	Fe, 0.5cm	
	21	Fe, 10.0cm	
weight		3.0 tonnes	3.1 tonnes
(y,z)		(80,80) cm	(100,100) cm
thickness (total)		51.0cm	58.6cm

TABLE 1. INTEGRAL EXPERIMENT MATERIALS AND THICKNESSES.

	Li6(n,alpha)T	Li7(n,n' alpha)T	238U(n,f)	235U(n,
FR (1)	1.5081	0.085225	0.39185	0.04827
FR (2)	1.6654	0.09548	0.4493	0.05389
FIE (1)	0.49003	0.13101	0.18257	0.01110
FIE (2)	0.48884	0.13043	0.18060	0.01120

TABLE 2. COMPARING FISSION SPECTRUMS.

APPENDIX 1

MODIFIED SWANLAKE INPUT DATA

The input data needed to run the modified SWANLAKE input code is now presented. The changes in the input data are switches to tell the code whether to include the fissile term, the direct gain term or whether to write out the sensitivity profile onto tape or disk. The complete input data file is presented.

The data may be input in either conventional FIDO (\$ or *) or free form (\$\$ or **) format in the order listed. The \$ units call for integer input and the * units require floating point input. In the data below, the bracketted numbers refer to the number of entries required for each unit or section. Normal FIDO shorthand, such as F for fill, R for repeat etc are allowed. Note the placement of the T's required at the end of every input data set.

The input data now follows.

1\$\$ (16)

- 1) IGM number of energy groups
- 2) ITL cross section table length (usually IGM+3)
- 3) INT number of spatial intervals
- 4) ISN order of Sn quadrature used in ANISN
- 5) ISCT order of Pn cross section expansion used in ANISN
- 6) NZONE number of spatial zones
- 7) NE number of cross section sets to be analysed
- 8) ITAPE $\mp$ (1 or 2), (calculate ϕ/ϕ^* or read ϕ/ϕ^* from unit 7)
if ITAPE < 0, then 13** array is read
- 9) ISIG (1 or 2), read cross sections from (cards or tape)

- 10) IGOM (1 or 2 or 3), (plane or cylindrical or spherical) geometry
- 11) IPRNT (0 or 1), (dont print or print) cross sections
- 12) IPLZN (0 or 1), (dont print or print) P_2 predictions, P_2
sensitivity matrix and fission sensitivity matrix
- 13) IARNG $\pm$ (1 or 2 or 3), sensitivity matrix printed in
(ANISN of MORSE or both) format. For a given
energy group I, ANISN prints ϕ_T in position
3, then $I' \rightarrow I$, while MORSE prints ϕ_T in
position 3, then $I \rightarrow I'$. If IARNG < 0 then
sensitivity profile per unit lethargy for each
cross section set is written onto disk on
channel 10
- 14) IPLOT (0 or 1 or 2 or 3), plot sensitivity profiles (none
or for neutron cross sections or for gamma cross sections
or for both), summed over all spatial zones
- 15) IPLTZ (0 or 1), (dont plot or plot), sensitivity profiles
for each separate spatial zone
- 16) ILNG last energy group for first particle type
(normally last neutron group)

2** (1)

integrated response, R ,for the problem investigated

T

If ITAPE= $\pm$ 1 then 4**, 5**, 6** and 7\$\$ arrays are required.

If ITAPE= $\pm$ 2 then next input will be the neutron energy
group boundaries, 8**

4** (NOA) Sn weights, where NOA=ISN+1 for plane or sphere and
NOA=ISN(ISN+4)/4 for cylinder

5** (NOA) Sn cosines

T

6** (INT+1) radii by interval boundaries, in centimetres
 7\$\$ (NZONE+1) interval boundary numbers corresponding to
 zone boundaries

T

8** (ILNG+1) neutron energy group boundaries in Mev,
 high to low
 9** (IGM-ILNG+1) gamma energy group boundaries in Mev,
 high to low

T

The following data (title cards, 10\$\$, 11** or 11\$\$, 12**,
 13** ,fission spectrum) must be input for each separate
 cross section set to be analysed INE=1 to NE.

Title card for cross section set INE in 20A4 format.

(this appears at the beginning of the sensitivity analysis
 output, and on the plot for the first particle type)

Title card for gamma sensitivity profile plot, 20A4 format.

(this is only input if IPLOT=3 and it appears on the
 second plot)

10\$\$ (1) Order of Pn cross section expansion for
 cross section set INE

this number is carried as the variable LISN

T

11** (IGM*ITL(LISN+1)) cross sections for cross section
 set INE, input on cards (ISIG=1)

T

OR

11\$\$ (LISN+1) library ID numbers for cross section set INE
 to be input from channel 8 (ISIG=2)

T

12** (NZONE) number density by spatial zone for
cross section set INE

T

13** (IGM*NZONE) macroscopic activity cross sections
for each energy group
for each spatial zone of the response function

T

(IGM) Fission spectrum, by energy group, high to low,
in E12.5 format
this must occur in every SWANLAKE datafile
regardless of whether
or not a fissile nuclide is used

Title card for neutron summation plot (if NE > 1)

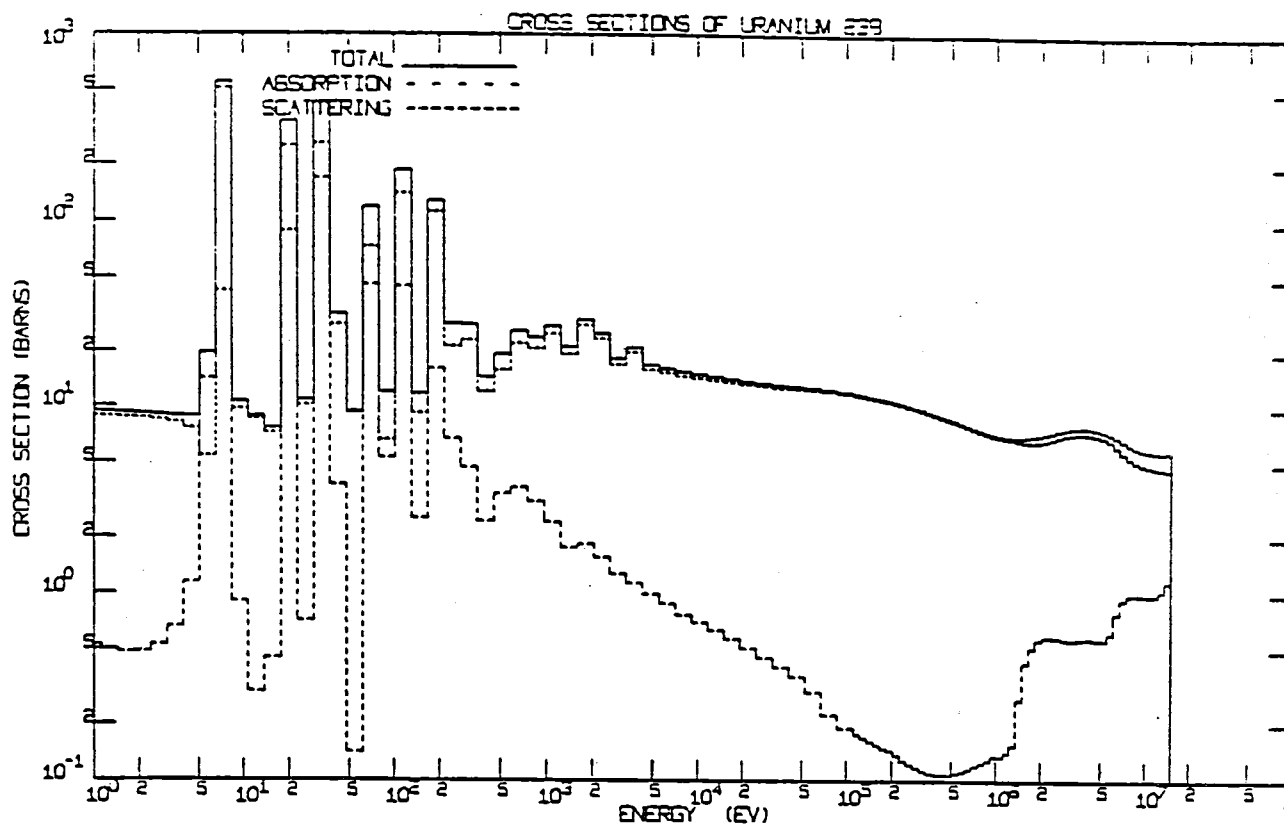
Title card for gamma summation plot (if NE > 1 and IPLOT =3)

The blank common requirments are specified in the main routine
of SWANLAKE.

```
COMMON D(XXX)
COMMON /FISS/D1(YYY)
IBLCOM = XXX
JBLCOM = YYY
```

where YYY and XXX specify the amount of blank common required for
the fissile and the non fissile calculations, respectively.
Formulae can be used to find the exact blank common requirment
but experience shows that for the analysis of one cross section
set, a P_0 cross section expansion needs $XXX = 100,100$ while a P_3
would need $XXX = 150,000$ words of blank common. It is found that
a fissile blank common specification of $YYY = 50,000$ is ample in
all cases.

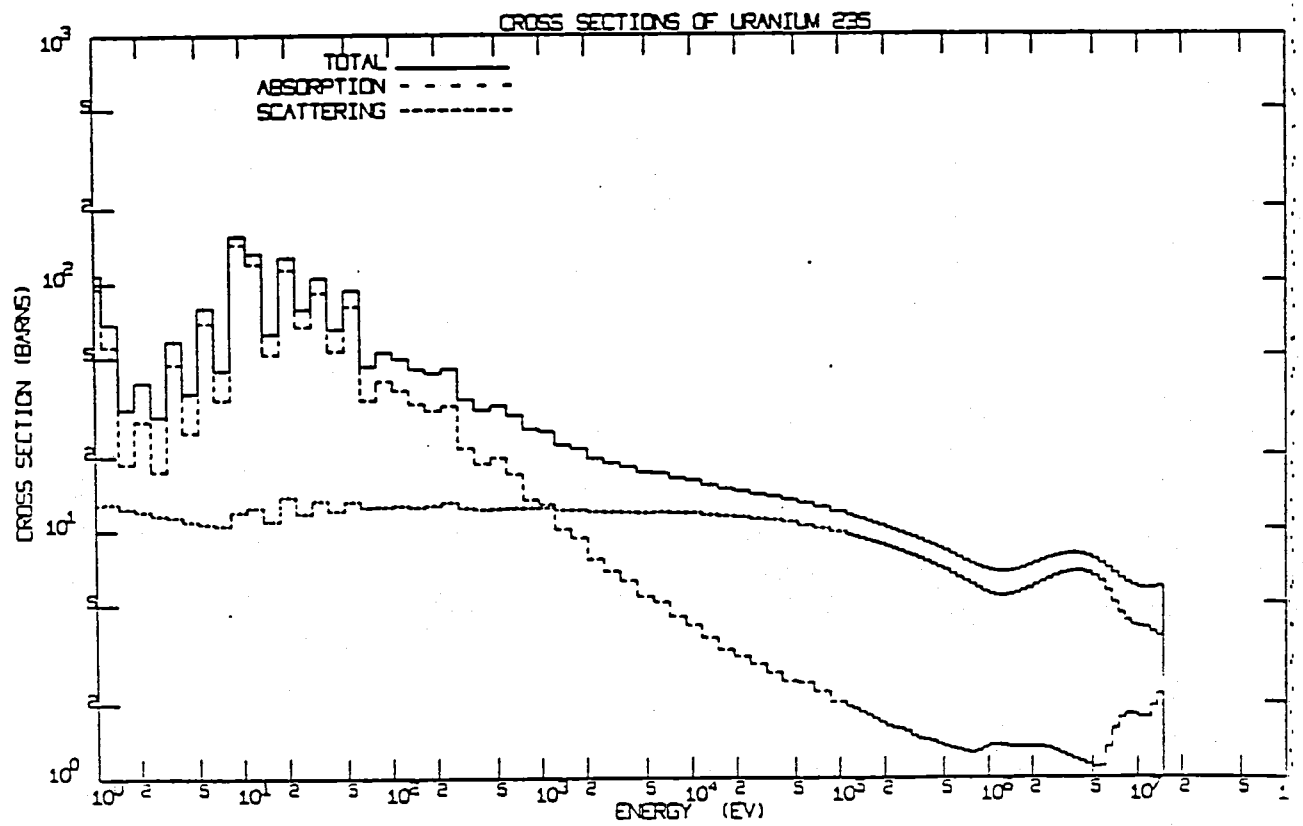
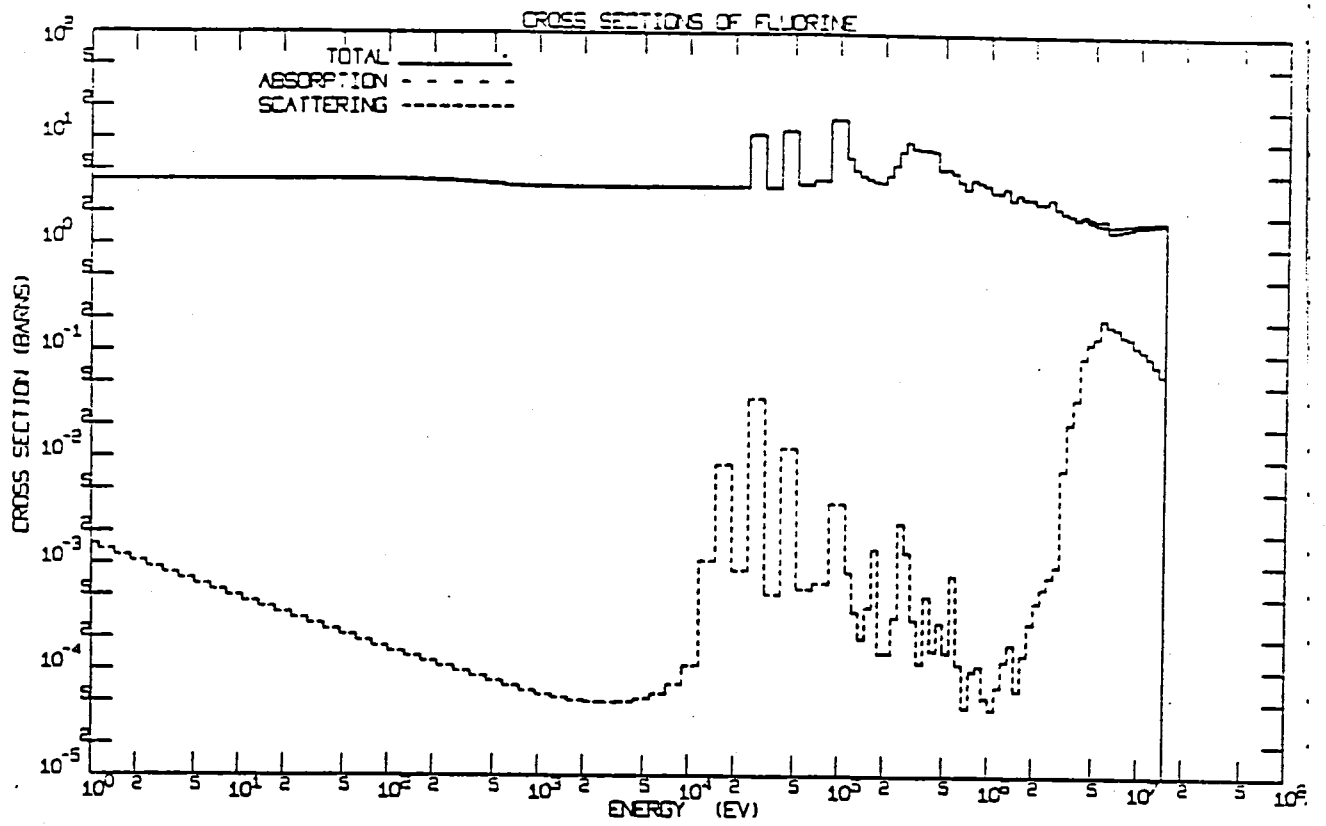
In addition to the standard discs and datasets required by the SWANLAKE code, detailed on pp 14-15 of the manual, an unformatted dataset will be needed on channel 10 if the sensitivity profile is to be stored.

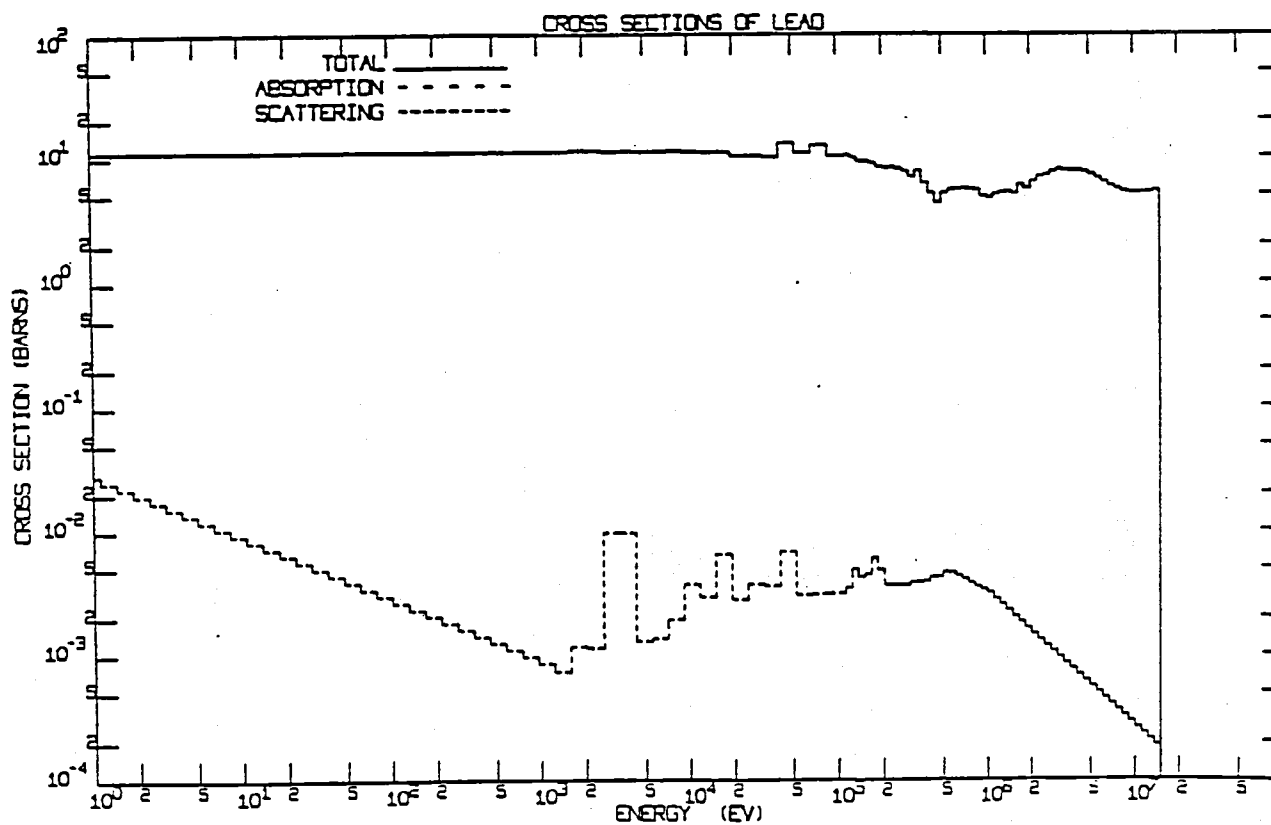
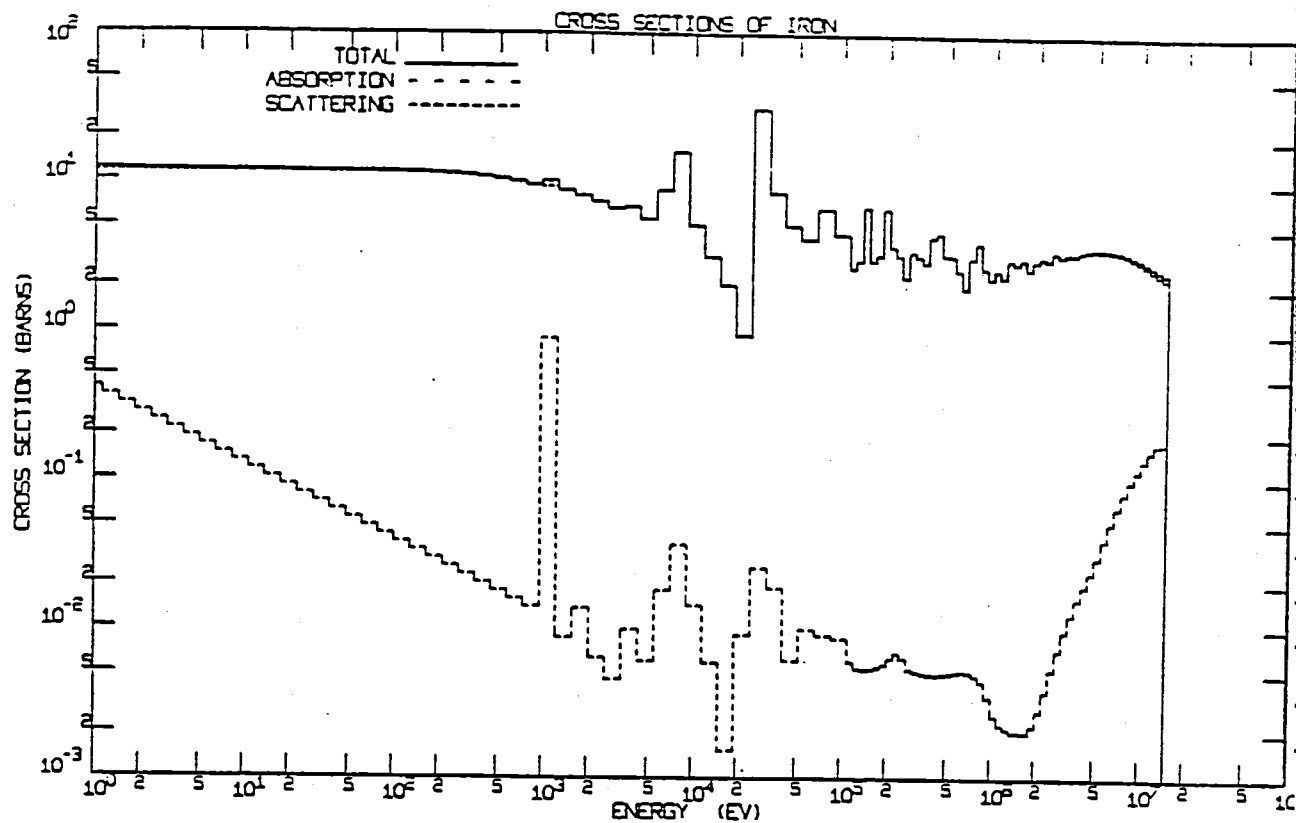


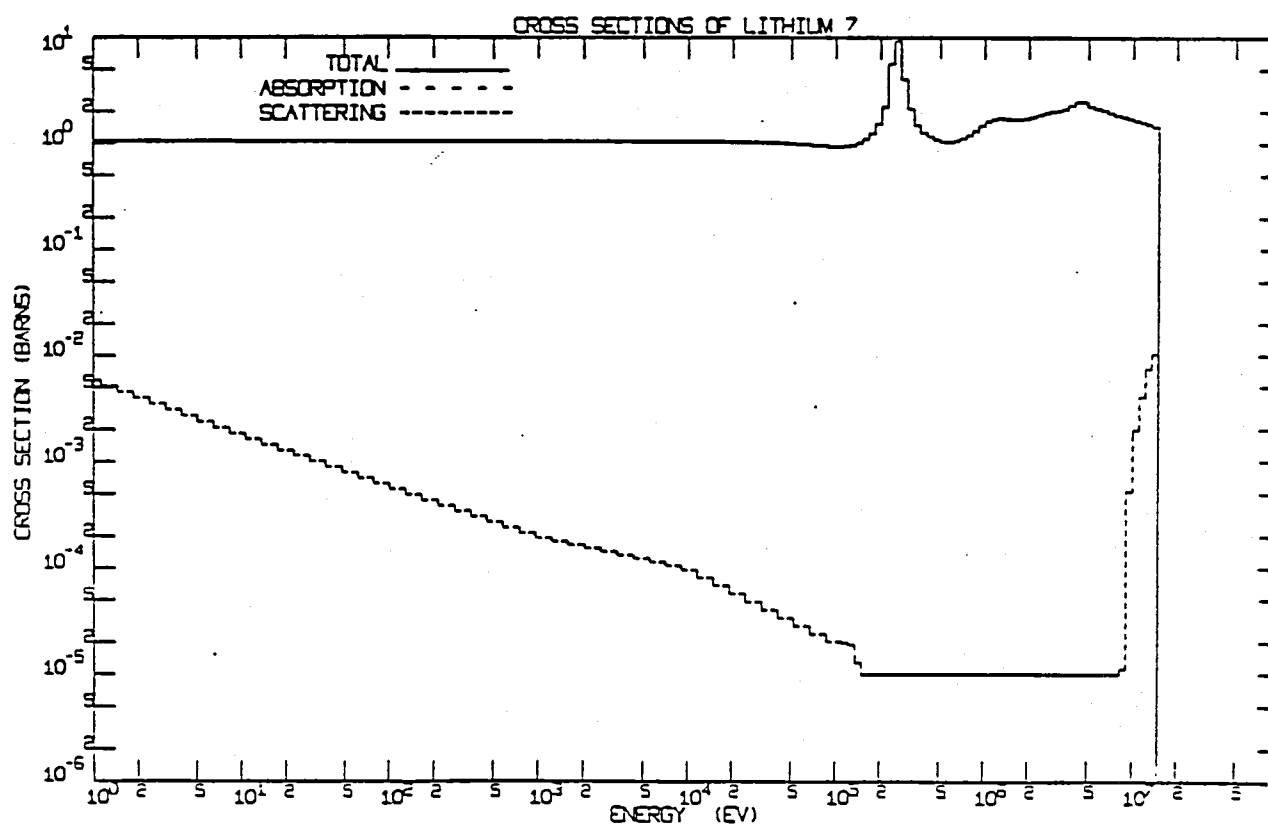
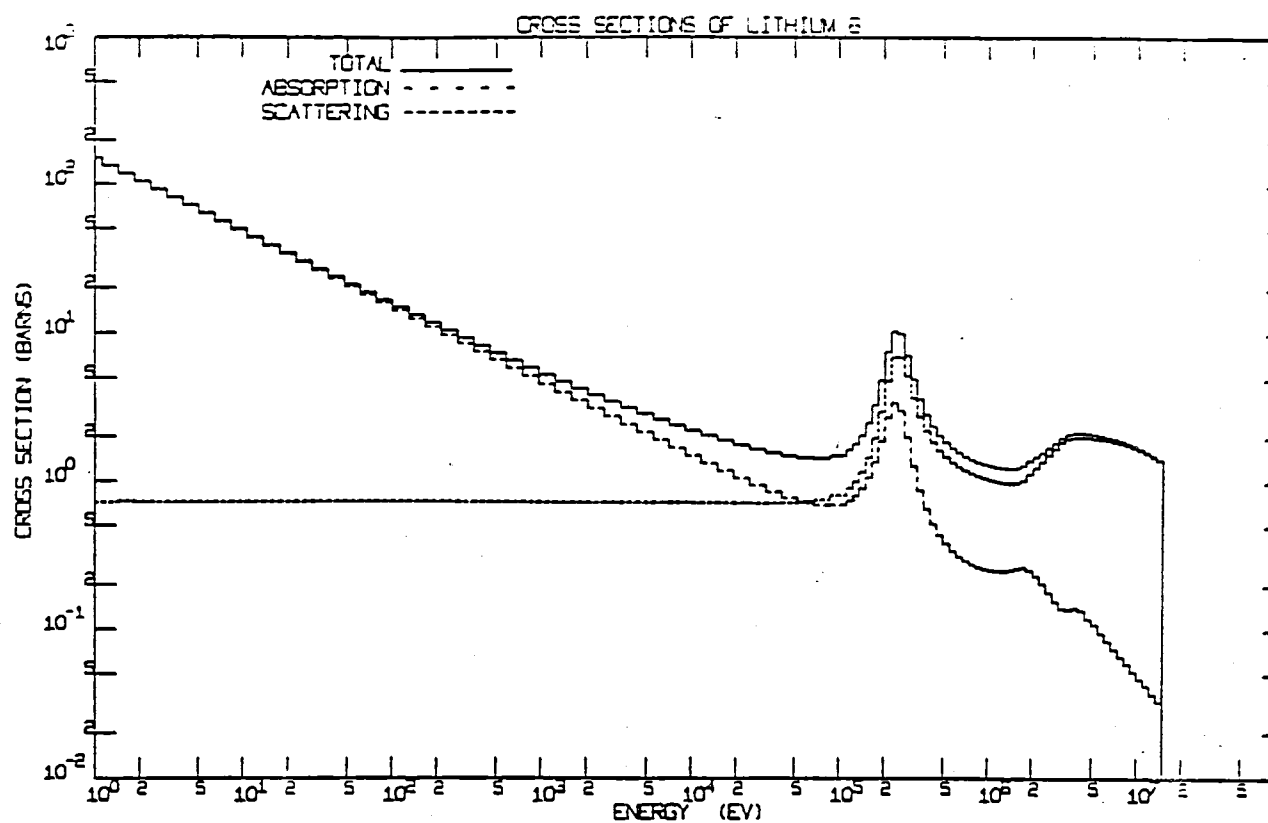
APPENDIX 2

SOME CROSS SECTIONS

Some elastic, inelastic and total cross sections are now shown. The energy group structure is the 100 ENDF/B-4 cross section set. The cross sections for Li6, Li7, Fe, Pb, P19, U238 and U235 are shown.







APPENDIX 3

THE MEANING OF THE ADJOINT FLUX

The neutron distribution in a reactor can never be observed, only the effects of the distribution are observable. This observable may be a temperature reading, an electrical impulse or any other quantity, but it is only a manifestation of some portion of the particle distribution that is perceived. In this context the adjoint flux can be considered to be a second independent distribution that relates the effect that the particle distribution at each point in phase space has on the observable of interest.

Thus the adjoint flux can be physically interpreted as giving a quantitative measure as to the importance of a certain particle in contributing to a certain detector response. For example, consider some reactor system. In the forward or normal run a unit source emits neutrons at a certain rate and produces a certain response or reaction rate R at the detector at the phase space point $(r_d, E_d, \Omega_d, t_f)$, after a time t_f has elapsed. Calculating the adjoint flux, $\phi^*(r_o, E_o, \Omega_o, t_o)$, for this type of detector defined at $(r_d, E_d, \Omega_d, t_f)$ at the phase space point $(r_o, E_o, \Omega_o, t_o)$ gives a measure as to the response that a neutron born at position r_o , at time t_o , energy E_o and emitted at an angle Ω_o would produce in the detector defined by $(r_d, E_d, \Omega_d, t_f)$ after a measuring time t_f . Thus the adjunction (adjoint particle) is important to the (future) observable reading.

One immediate use of the adjoint function is apparent. If it is desired to determine the response of a given detector to

neutrons of many energies, it is not necessary to perform calculations of the neutron flux for each neutron source or energy. A single calculation of the adjoint will suffice to compute the detector response for any source.

There are a number of important and interesting differences between the behaviour of neutrons and adjunctions. The time direction is reversed in the adjoint Boltzmann equation. In the forward case neutrons can be considered to be emitted from $t=0$ to $t=t_f$, some final time. In the adjoint case adjunctions are emitted from $t=t_f$ to $t=0$. The length of time is the same but it is reversed in direction.

The computational codes used in this work use adjoint sources which emit adjunctions but because an adjoint source is really a particle detector of some kind (eg a threshold foil) it should be remembered that it is physically a particle sink.

It is of interest to compare the forward and adjoint time dependent equations. The forward equation may be regarded as defining an initial value problem. Given $\phi(v, E, \Omega, t=0)$ the transport equation can be used to determine at all subsequent times whereas the adjoint equation can be thought of as defining a final value problem. That is to say, if ϕ^* is given at some final time $t=t_f$, the values of ϕ^* may be found at earlier times by integrating the adjoint equation backward in time. In physical terms this means that the activity of a detector (or adjoint source) at some particular time affects the importance for activating the detector of neutrons at all earlier times but it can have no effect on the importance of neutrons at later times. For the forward flux, however, the situation is the opposite in that the source at any given time will have no effect

on the flux at earlier times but it affects the flux at later times.

Another important point to note is that if the forward Boltzmann equation is written with the boundary condition of no incoming flux into the reactor system from the 'outside world' then the corresponding adjoint boundary condition is that of zero outgoing adjoint flux. It should be noted that an outgoing adjoint flux is treated as an incoming flux by neutronics codes because of the time reversal. Physically this means that a neutron leaving the reactor free surface has no importance, since by the definition of a free surface it cannot return. In most reactor calculations it is usually assumed that the outside world is a void - thus the above statement is saying that there is zero chance of a lost neutron having an interaction in a vacuum and so returning to the reactor to contribute to the chosen response.

There is a certain symmetry involved between a forward and adjoint computational run. In the forward run the source normalisation is 1.0 whilst the detector response is the macroscopic cross section of the detector nuclide. In the adjoint run the source is the forward runs response function, which is left unnormalised, whilst the response is 1.0. Thus the sources and responses are reversed in going from a forward to an adjoint run. Thus if in the forward run the source has a normalisation of 2.0 then the response used in the adjoint run must be 2.0. The forward source has units of neutrons per cm^{-3} per sec^{-1} and the adjoint has units of response per neutron per cm^{-1} . For both MORSE and ANISN the adjoint source should be input in this form, as the codes automatically include the adjoint source volume.

An adjoint calculation involves the transport of fictitious particles called adjunctons. Adjunctons are absorbed and scattered in interactions with atomic nuclei in a similar way as that of neutrons. However, it must be made clear that an adjoint run does not preserve particle balance like a forward run does because of the cross section matrix inversion. The quantity of importance is preserved. An adjoint run has its energy direction reversed to that of the forward run - the cross section matrices are inverted for all the system materials. Thus adjunctons are upscattered in energy in collisions. This inversion is carried out automatically by the computational codes. In MORSE the forward run has the neutron histories weight modified by the non absorption factor (p 262, MORSE manual),

$$\xi_s^{g'}(r) / \xi_e^{g'}(r)$$

after a neutron-nucleus interaction at position r at an incident neutron energy g' . In the adjoint case the adjunctons are forced to scatter (as in the forward case) and the weight of the adjuncton is modified by the factor (p 278, MORSE manual),

$$\sum_g \int d\bar{n} \xi_s^{g \rightarrow g'}(\bar{r}, \bar{n} \rightarrow \bar{n}') / \xi_e^{g'}(\bar{r})$$

where the forward run energy group structure is retained so that group g' is higher in energy than g . Thus the weight factor is now no longer a simple non absorption probability and may assume values in excess of unity even if no capture or multiplication reactions are present.

One important discrepancy between ANISN and MORSE should be pointed out. It is seen that the adjoint fluxes from ANISN are always finite whilst some of those of MORSE may be zero, even after the tracking of many adjunctons. The results obtained from

an ANISN adjoint run are for a system that effectively has had an infinite number of adjunctons tracked in it. Thus every $(r \in \mathcal{R})$ position will have had some sort of reaction in which the resulting neutrons will have contributed to the detector response either directly or by its progeny (particles which can trace their life cycle back to the original neutron through fissioning, scattering etc). MORSE, however, uses a statistical method and so always for a finite number of tracked adjunctons, there will be $(r \in \mathcal{R})$ positions where the adjoint flux is zero.

Finally, it is interesting to note that in systems with a great deal of fission the adjoint run converges very much faster than the forward run, thereby saving on computer time. The reason is due to the difference in the quantity that supplies the amount of upscattering that the computational code needs to account for. In the forward case this is $\nu(\epsilon) \zeta_f(\epsilon)$ whereas in the adjoint case it is $\chi(\epsilon)$. The energy dependent shape of these two distributions are markedly different, thus explaining the difference in the rate of convergence.

APPENDIX 4

SOME DEFINITIONS AND EXPLANATIONS

THE COULOMB BARRIER

Before the fusion process can occur the repulsive electrostatic potential (the Coulomb barrier) between the nuclei of the reacting ions must be overcome. Enough energy must be supplied to the reacting ions (in the form of kinetic energy) in order that the two nuclei come within range of each others attractive nuclear strong force. For the DT process this occurs for ion kinetic energies of the order of 300 keV. In practice in a plasma, however, fusion reactions will occur at energies below this. The quantum mechanical tunneling effect, for one ion to tunnel through the potential barrier of another is always present, and the fact that in a system of any particles with a Maxwellian energy spectrum means there will always be a few particles at much higher energies than other particles.

THE IGNITION TEMPERATURE

In order for a fusion system to become self sustaining, fusion reactions must take place at a rate sufficiently high to balance energy dissipating processes such as particle loss, Bremsstrahlung, cyclotron and synchrotron radiation losses. In the case of the DT reaction, each fusion reaction supplies to the plasma only the energy of the ${}^4_2\text{He}$ ion, ie 3.5 MeV, since the neutron easily escapes the plasma. An important energy-dissipating process is the emission of electromagnetic radiation

by the electrons in the plasma. Bremsstrahlung radiation is the principal source of radiation loss and occurs when an electron is accelerated in the field of an ion. In the case of a fusion plasma, this radiation is of relatively low energy and therefore not very penetrating compared to, say, gamma radiation. However, the density of the plasma is so low that the photons escape the plasma region and their energy is lost in the sense of being unavailable to help sustain the fusion reaction.

The simplest fusion ignition condition is obtained by setting the fusion energy release rate in the plasma equal to the energy loss rate due to Bremsstrahlung. For the DT reaction in a 50-50 DT mixture, assuming ion and electron temperatures are equal, an ignition temperature of 54 million K (corresponding to $kT=4.6$ keV) is obtained.

SECOND CHANCE FISSION

As one approaches the threshold energy of the fission process in ^{238}U , the fission cross section increases steeply and then reaches a plateau. The rapid rise indicates first chance fission and can be written as $(n, \text{fission})$. At higher energies, around 6.5 MeV for ^{238}U , the cross section presents another steep rise corresponding to the threshold for second chance fission. This is in fact a $(n, n' \text{ fission})$ reaction in which the residual nucleus resulting from inelastic neutron scattering is formed with an excitation energy close to, or greater than, the barrier height in that nucleus, which can then decay by fission with a significant probability.

Above the threshold for second chance fission the total fission cross section reaches a plateau whose value corresponds to the sum of first chance and second chance fission cross sections. At still higher energies, about 14 MeV, third chance fission can occur.

REFERENCES

- ABDEL-KHALIK S.I. , Nuclear Technology/Fusion , pp 53-72,
January 1983
- ALSMILLER R.G. et al, ORNL-TM-4696, October 1974
- ALSMILLER R.G. et al, ORNL-TM-5198, 1976
- BADGER B. et al, reports UWFD-450 and KFK-3202, 1981
- BARTINE D.E. et al, Oak Ridge, ORNL-TM-3809, 1973
- BARTINE D.E. et al, Nuc. Sci. Eng. 53, pp 304-318, 1974(b)
- BARTINE D.E. et al, Nuc. Sci. Eng. 55, pp 147-167, 1974(a)
- BHAT M.R., US report, ANL-76-90, JUNE 1976
- BELL G.I et al, Nuclear Reactor Theory, Van Nostrand,
N.Y., 1970
- BEYNON T.D. et al, pp 543-562, IAEA-207, October 1976
- BEYNON T.D. et al, Dept. of Physics report, Birmingham, 1978
- BEYNON T.D. et al, Journal of Physics D, vol 12, pp 1453-1462,
1979
- BEYNON T.D., proc. 24th Scottish Univ. summer sch. in physics,
pp 271-302, 1982
- CAMPBELL C.C. et al, Nucl. Data for Reactors, Vol 2, pp 391-425,
IAEA, Vienna, (1970)
- CECCHINI G. et al, Proc 3rd Intern. Conf. Peaceful uses ,
At. Energy, Geneva, Vol 2, 388 (1964)
- CURTIS R.H., Ph.D thesis, Queen Mary College, London Univ., 1978
- DEJONGHE G. et al, Oak Ridge, ORNL-RSIC/44, 1980
- DRAHT J.B, Nuc. Sci. Eng., 62, pp 117-129, 1977
- DRAKE M.K. (editor), BNL report, BNL-50274, 1970
- DUDERSTADT J. et al, Transport Theory, John Wiley
and sons, N.Y., 1979
- DUBI A. et al, Nucl. Data and Benchmarks for Reac. Shielding,
Proc. of a Specialists Meeting, NEA, Paris, 1980

- DUBI A. et al, LA-8813-MS, Los Alamos, 1981
- EMMETT M., Neutron Physics Division, ORNL-TM-4972, 1975
- ENGLE W.W Jr, A users manual for ANISN, K-1693,
Computer centre, Union Carbide Corp., 1967
- GANDINI A., Jour. of Nucl. Energy, vol 21, 755, 1967
- GANDINI A., parts 1,2 and 3, RT/FI(73)5, RT/FI(73)22, RT/FI(74)3,
Comitato Nazionale Energy Nucleare, Casaccia,
Italy, 1973,1974
- GARBER D. et al, Neutron Cross Sections, vol 2, Brookhaven,
N.Y., 1976
- GERSTL S.A., Los Alamos, LA-6118-MS, 1975
- GERSTL S.A. et al, Los Alamos, LA-8333-MS, 1980
- GERSTL S.A., Los Alamos, LA-8498-MS, 1980
- GERSTL S.A. et al, Nuc. Sci. Eng., 62, pp 137-156, 1977
- GOLDSTEIN H.A., Nucl. Reactor Shielding Int. Conf. on Reactor
Shielding, 5th, Knoxville, Tenn, 1977, pp 75-90
- GREEBLER P. et al, Nuc. Data for Reactors, Proc. of Conf.,
Helsinki, IAEA, vol 1, pp 17-33, 1970
- GREENE N. et al, Oak Ridge, ORNL-TM-3706, 1976
- HAIGHT R.C. et al, Nuc. Sci. Eng., 61, pp 53-59, 1976
- HALL M., Oak Ridge, OPNL-RSIC/44, 1980
- HANSON G.E. et al, LA-3810, Los Alamos, 1967
- HUANG L. et al, Trans. Am. Nuc. Soc., 18, 365, 1974
- HUMI M. et al, Proc. 3rd Intern. Conf. Peaceful uses At. Energy,
Geneva, vol 2, 398, 1964
- HUMMEL H.H et al, Nuc. Sci. Eng., 54, pp 35-46, 1974
- JOANOU G. et al, General Atomic report 4265
- KUMAR A. et al, Jour. Fusion Energy, vol 3, no. 3,
pp 185-197, 1983
- KEEFE D., Ann. Rev. Nucl. Part. Sci., vol 32, pp 391-441, 1982

- LATHROP K.D, Reactor Tech., 15, pp 107-134, 1972
- LAWSON J.D., AERE-GP/M, no. 185, AERE Harwell, 1955
- LAWSON J.D., Proc. Phys. Soc. of London, B 70, 6, 1957
- LAWSON J.D., Congress of the French Soc. of Phys.,
Grenoble, 1983
- LEWINS J., chapter 2, Importance - the adjoint flux,
Pergamon press, N.Y., 1965
- Los Alamos, LA-7396-MS, 1971
- MARKOVSKII D. et al, IAE-2579, 1975
- McKNIGHT R. et al, Advanced Reactors; physics, design and
economics, Pergamon press, 1975, pp 620-629
- MUIR D.W. et al, Tech. of Cont. Nucl. Fusion, pp 910-917, 1976
- MUIR D.W. et al, LA-8733-MS, 1981
- OBLOW E.M. et al, ORNL-TM-4110, 1973
- OBLOW E.M., Nuc. Sci. Eng., 59, no. 2, pp 187-189, 1975
- OBLOW E.M., ORNL-TM-5176, 1976
- OBLOW E.M. et al, Nuc. Sci. Eng., 75, pp 88-110, 1980
- PARKER K. et al, in Nuclear Data for Reactors, vol 2, pp 293-307,
IAEA, Vienna, 1967
- PAZY A. et al, Nuc. Sci. Eng., 55, pp 280-295, 1974
- PENDLEBURY E., in Neutron Cross Sections and Technology,
CONF-710301, vol 1, 1, US At. Energy Comm., 1971
- PEREY F. et al, in Advanced Reactors; physics, design and
economics, Pergamon press, pp 578-591, 1975
- PEREY F., Rev. of Theory and App. of Sens. and Uncert. Analysis,
Proc. of Seminar Workshop, pp 311-318, Tennessee,
1978
- PERKINS L.J. et al, Nuc. Sci. Eng., 78, pp 30-43, 1981
- REUPKE W.A., The consistency of differential and integral thermo-
nuclear neutronics data, Ph.D thesis, LA-7067-T, 1978

RIBE F.L., Reviews of Modern Physics, 47, 1, 1975

RIBE F.L., Nuclear Technology, 34, 179, 1977

RIEF H., Proc. of Specialists Meeting on Sensitivity studies and
Shielding Benchmarks, 68, OECD, Paris, 1975

RIEF H., Nuclear Reactor Shielding Int. Conf. on Reactor
Shielding, 5th, Knoxville, Tennessee, Science press, 1977

ROSE R.P, Jour. of Fusion Energy, pp 185-196, vol 1, no. 2, 1981

RSIC, DLC2, 100 group neutron cross section data based on
ENDF/B-4, 1972

RSIC, DLC37F, Coupled 100 group neutron and 21 gamma ray cross
cross section for EPR neutronics,
RSIC data library collection, 1974

SALMI et al, Proc. Conf. on the Review of the Theory and App. of
Sensitivity, and Uncertainty Analysis, 49,
Tennessee, 1978

SALVATORES M., Nuc. Sci. Eng., 50, pp 347-353, 1973

SANTORO R.T. et al, ORNL-TM-5777, July 1981

SCHMIDT J., in Nucl. Data for Reactors, vol 2, pp 279-291,
IAEA, Vienna, 1967

SCOTT M.C et al, UK Nuclear Data Program, pp 90-93,
Jan-Dec, UKNDC(83)P109, 1982

SCOTT M.C., private communication, April 1984

SEKI Y. et al, Jour. Nucl. Sci. Tech., 14 part 2,
pp 97-107, 1977a

SEKI Y. et al, Jour. Nucl. Sci. Tech., 14 part 3,
pp 210-225, 1977b

SEKI Y. et al, Nuc. Sci. Eng., 73, pp 87-93, 1979

SIM B.S., Ph.D thesis, Dept. of Physics, Birmingham Univ., 1984

SIMONS E. et al, Los Alamos, LA-6272-MS, 1976

TAYLOR N.P, Dept. of Physics, report 02/81, 1981

TOBIAS M.L. et al, Oak Ridge, ORNL-TM-4184, 1973(a)

TOBIAS M.L. et al, Oak Ridge, ORNL-TM-4200, 1973(b)

TOBIAS M.L. et al, Oak Ridge, ORNL-TM-4201, 1974

USACHEV L.N., Jour. Nucl. Energy, parts A/B, 18, pp 571-583, 1964

USACHEV L.N., Proc. of the First Intern. Conf. on the Peaceful
uses of Atomic Energy, Geneva, 5, 503, 1956

USACHEV L.N. et al, App. of Nucl. Data in Science and Tech.,
Symposium, Paris, 1973, IAEA/SM-170/91

WEALE J. et al, Jour. of Nucl. Energy, parts A/B, vol 14,
91, 1961

WEAVER D.R., A Covariances cook book, Dept. of Physics
report, 1983

WEISBIN C.R. et al, Nuc. Sci. Eng., 66, pp 307-333, 1978

WEISBIN C.R. et al, Oak Ridge, ORNL-TM-4847, August 1975

WEISBIN C.R. et al, Oak Ridge, ORNL/TM-5563, 1976

WYMAN M.E., USAEC Report LA-2234(Rev), Los Alamos, 1972

YEIVIN Y. et al, lecture notes on cross section adjustment
procedures, commission of the European communities,
EUR 6041 EN, 1978

YOUSEFF M.Z. et al, Univ. of Wisconsin report, UWFD-308, 1979

YOUSEFF M.Z. et al, Nucl. Tech./Fusion, vol 2, pp 648-666, 1982