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LMFBR BLANKET PHYSICS PROJECT
PROGRESS REPORT NO. 5

JUNE 30, 1974

DEPARTMENT OF NUCLEAR ENGINEERING
MASSACHUSETTS INSTITUTE OF TECHNOLOGY
CAMBRIDGE, MASSACHUSETTS 02139

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NUCLEAR ENGINEERING
READING ROOM - M.I.T.

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AEC Research and Development Contract

UC-34 Physics

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U.S. Atomic Energy Commission

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ABSTRACT

This is the fifth annual report of an experimental and analytical program for the investigation of the neutronics and photonics of benchmark mockups of LMFBR blankets.

During the period covered by the report, July 1, 1973 through June 30, 1974 work was devoted to completion of experimental work on Blanket Mockup No. 4, a three-assembly-row, steel-reflected blanket driven by a simulated demonstration-reactor core.

Extensive work was carried out on the measurement of gamma heating in the blanket and reflector regions of Mockup No. 4, primarily with state-of-the-art TLD methods.

Work was completed on the use of foil methods for epithermal neutron spectrometry. Calculations and parametric studies were continued in a number of areas.

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CHAPTER 1
INTRODUCTION

1.1 Foreword

This is the fifth annual report of the LMFBR Blanket Physics Project. This report covers work done since the last progress report, Reference (1), during the period from July 1, 1973 through June 30, 1974.

The MIT Blanket Research Project is part of the AEC's LMFBR development program, having as its primary objective the experimental investigation of clean, but realistic, benchmark mockups of the blanket-reflector region of large LMFBR reactors. The key experimental tool used in this work is the Blanket Test Facility at the MIT Research Reactor. The BTF contains a fission-converter plate tailored to deliver a neutron spectrum simulating LMFBR core leakage, which can be used to drive fast reactor blanket-reflector mockups.

Blanket subassemblies are constructed of uranium metal fuel rods, clad in carbon steel, surrounded by anhydrous sodium chromate. The homogenized mixture closely simulates UO_2 fuel, stainless steel clad and sodium metal coolant; all of the important heterogeneous effects are also closely simulated.

To date, four blankets have been investigated. Blanket No.1 was a borax-iron assembly used only for preliminary tests of system design performance; No.2 was a 3-subassembly-row, steel-reflected mockup of a typical large (1000 MWe) LMFBR design; and No. 3 was a 2-row, graphite-reflected mockup of an

advanced design. Blanket Mockup No.4 was similar to No.2, except that the converter has been modified to drive it with a spectrum typical of a smaller, demonstration-reactor-sized core.

1.2 Research Areas

During the report period work was carried out in the following areas:

- (1) State of the art TLD methods were applied to measure gamma heating in the blanket and reflector of Mockup No.4. This effort represents a substantial fraction of the project's efforts over the past year, involving a number of supporting analyses and experiments. (Chapter 2)
- (2) The last of a long series of evaluations of foil-method neutron spectrometry was completed. A multiple-foil stack of strong resonance absorber (gold) foils was used to infer the shape of the epithermal neutron energy spectrum in Mockup No.4. (Chapter 3)
- (3) Finally, a considerable variety of experiments and calculations of smaller scope were carried out in support of project objectives; these are reported in Chapters 4 and 5.

In the final chapter some general observations are made on the overall status of the projects efforts and future research goals are outlined. In this latter regard it should be noted that the MIT Research Reactor was shut down for

renovation in May 1974. Hence the experimental research program had to be curtailed in the final stages of FY 74, and the anticipated program for FY 75 will be strongly affected by how quickly the reactor can be returned to service.

1.3 Blanket Mockup No.4

Blanket Mockup No.4 is a 3-subassembly-row, steel reflected mockup driven by a simulated demonstration reactor core. The blanket and reflector regions are identical to Mockup No.2, which is described in detail in Ref. (2). Figure 1.1 and Table 1.1 show the major features of this assembly. The various chapters of this report describe experimental work or analytical and numerical calculations, most of which are centered about Blanket Mockup No. 4.

1.4 Staff

The project staff, including thesis students, during the report period was as follows:

M.J. Driscoll, Associate Professor of Nuclear Engineering
 I. Kaplan, Professor of Nuclear Engineering
 D. D. Lanning, Professor of Nuclear Engineering
 V.A. Miethé, Computer Operations Assistant
 A.T. Sipple, Engineering Assistant
 G.J. Brown, ScD. Student
 J.K. Chan, SM Student
 T.P. Choong, Research Assistant, SM Student (to Aug. 1973)
 G.A. Ducat, Research Assistant, ScD. Student (to Jan. 1974)
 O.K. Kadiroglu, ScD Student

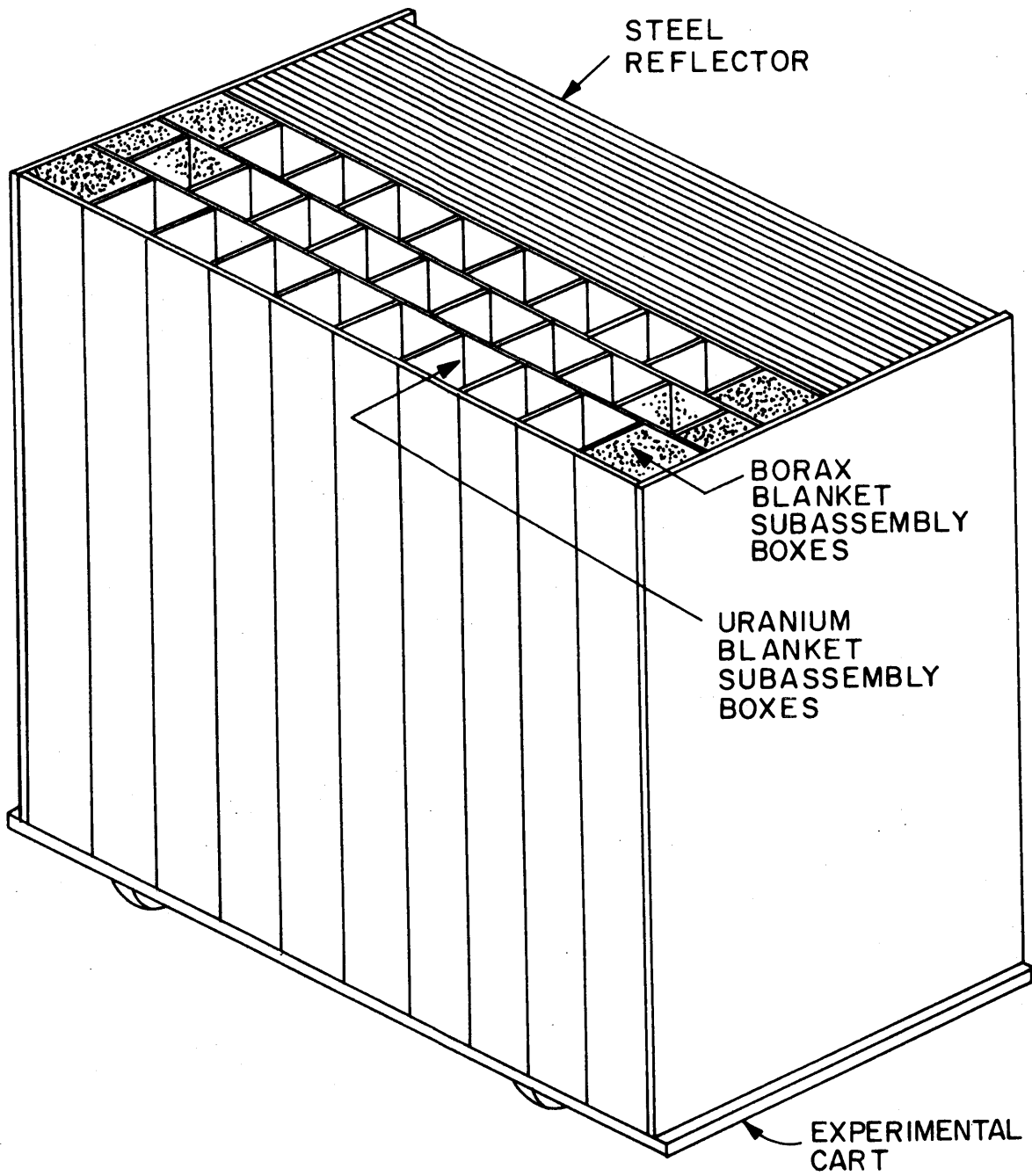


FIG. 1.1 SCHEMATIC VIEW OF BLANKET ASSEMBLY NO. 2

TABLE 1.1

Homogenized Atom Densities in Blanket No. 4

(Atoms/barn-cm)

Nuclide	Blanket No. 4	Equivalent Realistic Blanket*
U ²³⁵	0.000088	0.000016
U ²³⁸	0.008108	0.008131
O	0.016293	0.016293
Na	0.008128	0.008128
Cr	0.004064	0.003728
Fe	0.013750	0.012611
Ni	0.000000	0.001475
H	0.000073	0.000000
C	0.000096	0.000082

Nuclide	Steel Reflector
C	0.000590
Fe	0.084570

*Composed of 3.70 v/o depleted UO₂ (at 90% of the theoretical density), 20.7 v/o Type 316 stainless steel, 32.0 v/o sodium and 10.3 v/o void.

M.S. Kalra, Research Assistant, ScD Student

Y. Lukic, Research Assistant, Fall 1973

A.S. Leveckis, Undergraduate Laboratory Assistant (to May 15,1974)

R.E. Masterson, SM Student

P.A. Scheinert, Research Assistant, Nuclear Engineer's
Degree Student

J.I. Shin, Graduate Laboratory Assistant, Spring 1974

A. Tagishi, Research Assistant (since Jan. 1974)

M.K. Yeung, SM Student

1.5 References

- (1) LMFBR Blanket Physics Project Progress Report No.4
COO-2250-3, MITNE-149, June 30, 1973
- (2) T.C. Leung et al. "Neutronics of an LMFBR Blanket
Mock-Up", COO-3060-1, MITNE, Jan. 1972

CHAPTER 2

GAMMA HEATING MEASUREMENTS2.1 Introduction

During the past year the major new experimental effort has been the acquisition and application of a state-of-the-art capability for gamma heating measurements. A detailed topical report has been prepared⁽¹⁾ summarizing this work:

P.A. Scheinert and M.J. Driscoll, "Gamma Heating Measurements in Fast Breeder Reactor Blankets", COO-2250-10, MITNE-164, Aug. 1974 (est).

A condensed review of the work described in this report is presented in the sections which follow.

2.2 Methodology

After reviewing methods currently available for in-pile gamma dosimetry it was concluded that measurements based on the thermoluminescent (TLD) response of ${}^7\text{LiF}$ crystals possessed the best overall combination of desirable characteristics.⁽²⁾ This approach is also now being extensively employed for similar work on ANL fast critical facilities.⁽³⁾ Since the procedures employed at MIT were substantially the same as used at ANL and elsewhere, and since a commercial (Harshaw) readout device and TLD detectors were employed, we will not go into further detail on these topics here. Comments are appropriate on several aspects in which the

present work differed from or elaborated upon usual practice, in particular:

- (a) calculation of spectral response factors
- (b) calibration of TLD's
- (c) comparison with ionization-chamber-dosimeters
- (d) unfolding gamma spectra
- (e) measurement of spectral response factors

In addition to the above items, gamma heating traverses in Blanket Mockup No. 4 will be reported and a concluding section will discuss some of the improved techniques to be investigated in the future.

2.3 Spectral Response Factors

In the present instance we are interested not in the rate of energy deposition in the TLD material (LiF) itself, but in various blanket constituents: UO_2 , stainless steel and sodium. This requires careful design of the detector capsule: a wall material is selected consisting of (or simulating) the material in which gamma heating is to be measured; the wall must be thick enough so that an equilibrium spectrum of charged particles (electrons) exists inside the detector cavity, and not so thick that the ambient gamma photon spectrum is perturbed to a greater (or lesser) extent than in actual coolant, clad, or fuel; the small cavity is then filled with electron sensitive material - a TLD or ion chamber gas - to measure the energy deposition. Bragg-Gray theory is applied, under which the slowing down

spectrum is assumed characteristic of the surrounding medium and the stopping power is characteristic of the cavity material. Corrections must be applied for the fact that the TLD is not infinitesimally thin; in the present work Burlin's approach has been followed.

Reference (1) contains a detailed step-by-step derivation of the appropriate relations between cavity and wall response. Since Tuttle had already developed a computer program, RESPOND, for the numerical evaluation of the so-called "1/f" factor, we used his work as a starting point.⁽⁴⁾ Several specific differences between Tuttle's and our theoretical development were identified, however, and incorporated as changes in RESPOND. The revised program is listed in Reference (1).

The measured dose in the wall material is then given by:

$$D_w = \left(\frac{1}{f} \right)_x D_c \quad (1)$$

where $(1/f)_x$ is the computed correction factor and D_c is the dose in the cavity material - here a TLD, which is proportional to the thermoluminescent response in nanocoulombs. One additional step remains: the TLD must be calibrated in a gamma field having a known intensity and gamma spectrum. In the present work we used the uncollided first-flight gammas from a calibrated Co-60 source. Since the spectrum differs from that in the blanket mockup experiments it is necessary

to compute a $(1/f)_c$ factor appropriate to the calibration spectrum. Thus

$$D_w = \frac{\left(\frac{1}{f}\right)_x}{\left(\frac{1}{f}\right)_c} \cdot D_{CAL}(nc) \quad (2)$$

where $D_{CAL}(nc)$ = dose read from calibration curve, corresponding to the nc readout value from the TLD reader.

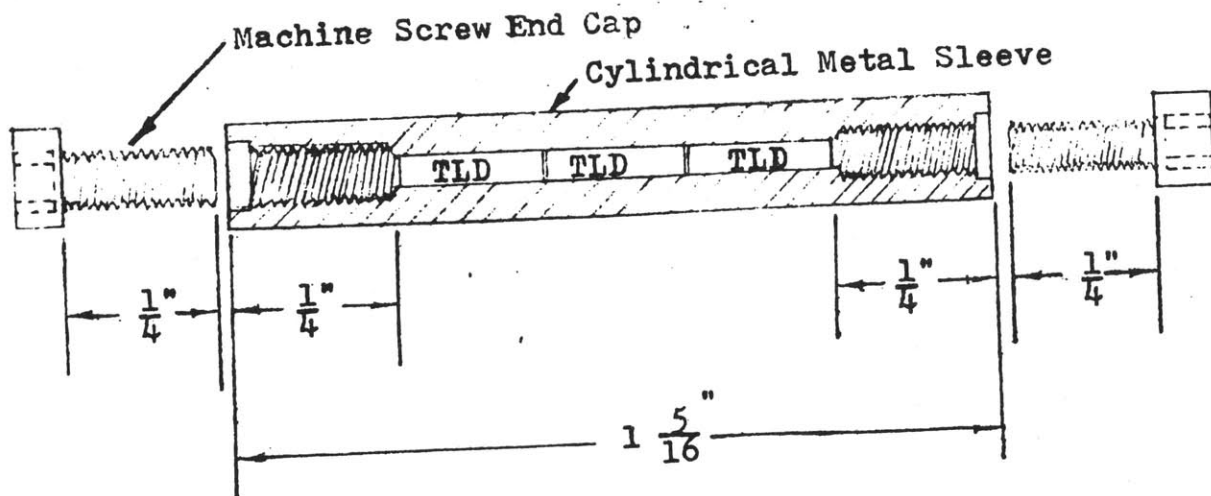
Note that there is some ambiguity in the determination even apart from that associated with the approximations involved in the derivation of the prescription for computation of $1/f$, namely one must know the gamma spectrum in the experimental facility in order to calculate $(1/f)_x$. This is not a particularly severe detriment in the present work, since we had available a coupled neutron-gamma cross section set which could be used for this purpose, and the $(1/f)$ corrections were generally not significantly different from unity.

In the present work three main capsule materials were employed: lead (simulating UO_2), aluminum (simulating sodium), and stainless steel. Several other materials were also used in an experiment to attempt unfolding of the ambient gamma spectrum. Table 2.1 lists the materials and key capsule dimensions, and Fig. 2.1 illustrates specific design details for the stainless steel capsule. All others were of similar

Table 2.1. As Built Dimensions (Inches) of TLD Capsules

<u>Material</u>	<u>Wall Thickness</u>	<u>Capsule Diameter</u>
Lead	0.070	0.125
Stainless Steel	0.070	0.187
Aluminum	0.152	0.350
Tungsten	0.036	0.118
Tin	0.102	0.250
Zirconium	0.099	0.244
Teflon	0.177	0.400

TLD Capsule
Stainless Steel Wall



- A. 3/16" Outside Diameter
- B. 0.064 Inside Diameter;
0.070" Wall Thickness:
- C. End Screws: 1/4" deep x 4-40

Features

- A. Bragg-Gray Cavity Design
- B. Holds three 1mm. Dia. x 6mm. TLD-700's
- C. Establishes Charged Particle Equilibrium

Fig. 2.1. Stainless Steel TLD Capsule

design, and each capsule held three TLD's (Harshaw TLD-700, 1 mm dia. x 6 mm long) as shown.

2.4 Calibration of TLD's and Experimental Results

All TLD's were individually calibrated in the same capsule (and location within the capsule) as used in subsequent experiments.

A calibration device, shown in Fig. 2.2, was constructed to hold the TLD capsules in a fixed position relative to a Co-60 source of approximately 70 millicuries, delivering ~ 95 rad/hr. at the inner ring of TLD's. The source was calibrated against an NBS-standardized Co-60 source (and subsequently checked against the Co-60 TLD irradiation facility at ORNL) to obtain absolute dose rate values used in preparing the TLD calibration curves.

The trefoil design was selected to minimize the contribution of scattered photons, which we found to be potentially appreciable from other designs, and difficult to calculate with precision. For like reasons, all calibrations were carried out with the device positioned at the center of a large vault to minimize the effects of room return.

Handling procedures for the TLD's with respect to cleaning, annealing and operation of the reader were investigated in some detail, and a standard procedure representative of state-of-the-art good practice was

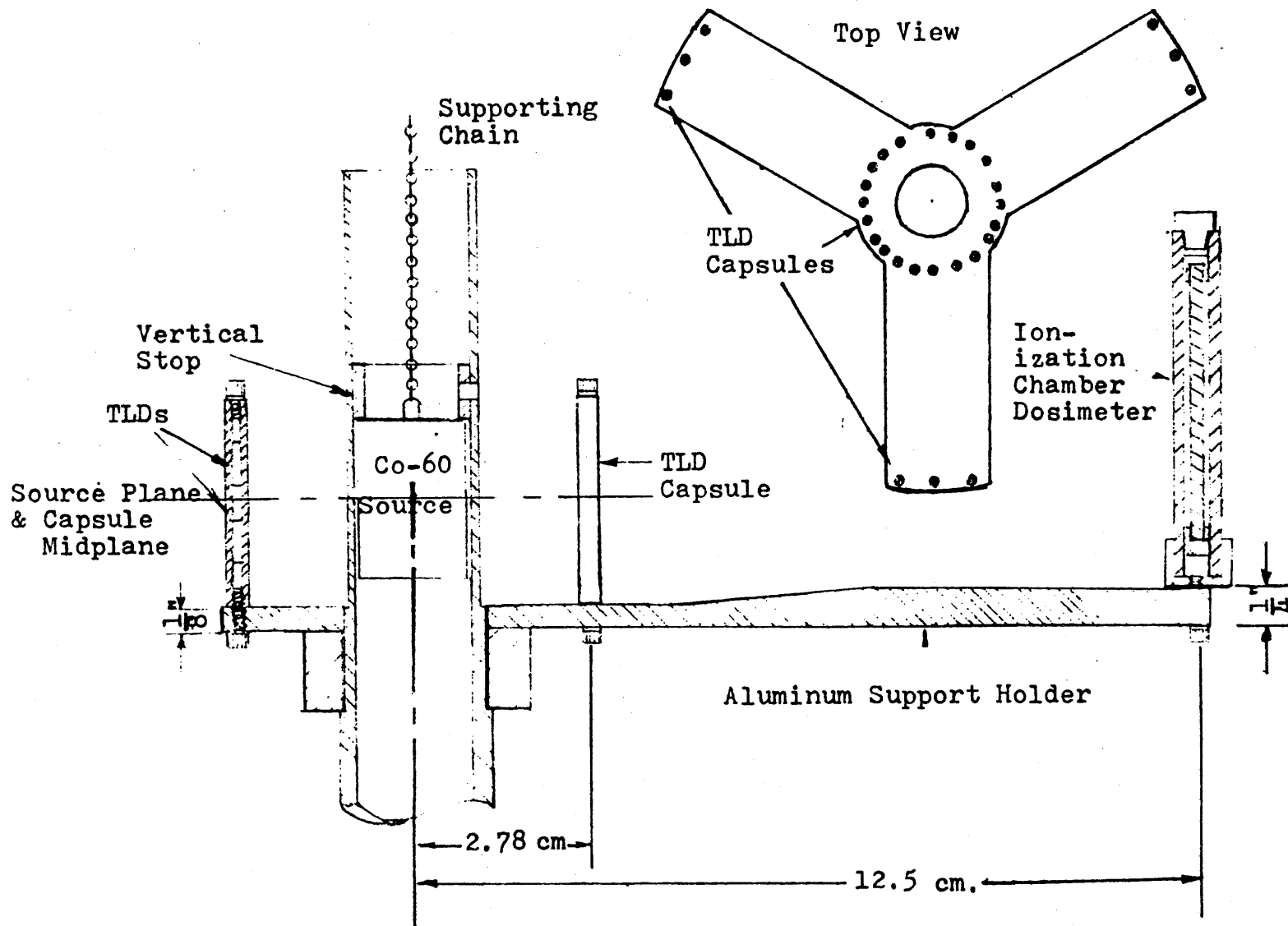


Fig. 2.2. Aluminum Irradiation Holder Used in M.I.T. Calibration Facility

developed (similar in most respects to procedures used at ANL). These procedures are documented in Ref. (1), and were followed religiously throughout.

After completion of their calibration, the TLD capsules were irradiated in Blanket Mockup No. 4 to obtain gamma heating traverses in UO_2 , SS, Na as a function of distance into the blanket and its steel reflector. Reference (5) describes the blanket and reflector construction, and the design of the traversing tubes. In the present runs the TLD capsules were held at the blanket centerline in notches cut out of steel foil-traversing rods. At the same time S_8P_1 calculations of the assembly were made with the ANISN program using a coupled neutron-gamma cross section set (22 neutron groups, 18 gamma groups) obtained from ORNL.⁽⁶⁾ The calculated results were used in the 1/f-factor determinations and to correct for neutron response by the TLD's.

Figure 2.3 shows a comparison of the measured and calculated absolute dose rates in UO_2 (lead) in Blanket Mockup No. 4. The good agreement is particularly gratifying when one considers that the calculation is based upon an absolute measurement of the thermal neutron source strength in the hohlraum region of the MIT Reactor: hence any errors in calculations of the thermal-to-fast-neutron converter assembly will also bias the results. Since ~90% of the blanket gammas originate in and are absorbed by the UO_2 ,

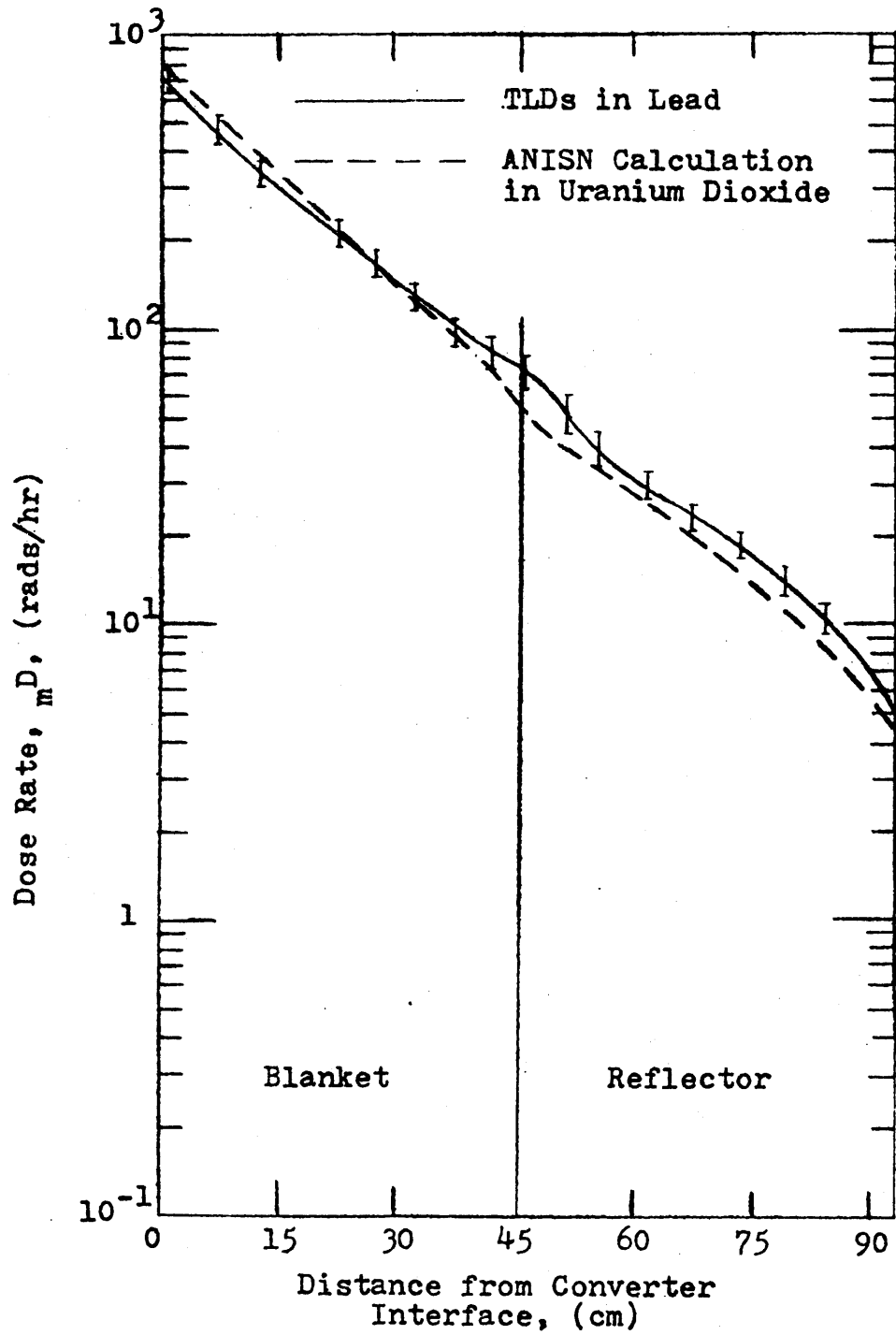


Fig. 2.3. Comparison of Absolute Dose Rate Traverses in Uranium Dioxide (lead)

this comparison also indicates a comparably good overall gamma balance. The corresponding measurements and calculations in steel and sodium do not agree as well, the C/E ratio ≤ 1.0 , but their contributions are so much less than the UO_2 that this discrepancy is not significant from a practical standpoint. Further, although the absolute values disagree, the relative shapes of the traverses are in good agreement.

2.5 Comparison with ICD Results

Because the TLD method is somewhat indirect and involves a physical phenomenon that is not yet fully understood, it was considered desirable to compare the TLD results to those from a more classical approach. Ionization Chamber Dosimeters (ICD's) were selected for this purpose. These devices are, in effect, merely capacitors which discharge in the presence of radiation. Figure 2.4 shows the ICD's designed and built for our experiments. They consisted of a steel rod inside a steel tube, separated at the ends by Ceresin wax insulators. In use the ICD's are charged to 300 volts prior to irradiation; then after irradiation an electrometer was used to measure the change in voltage, which is proportional to the energy deposited in the dosimeter.

The ICD's were calibrated in the same facility as the TLD's; just as with TLD's one must calculate 1/f-factor

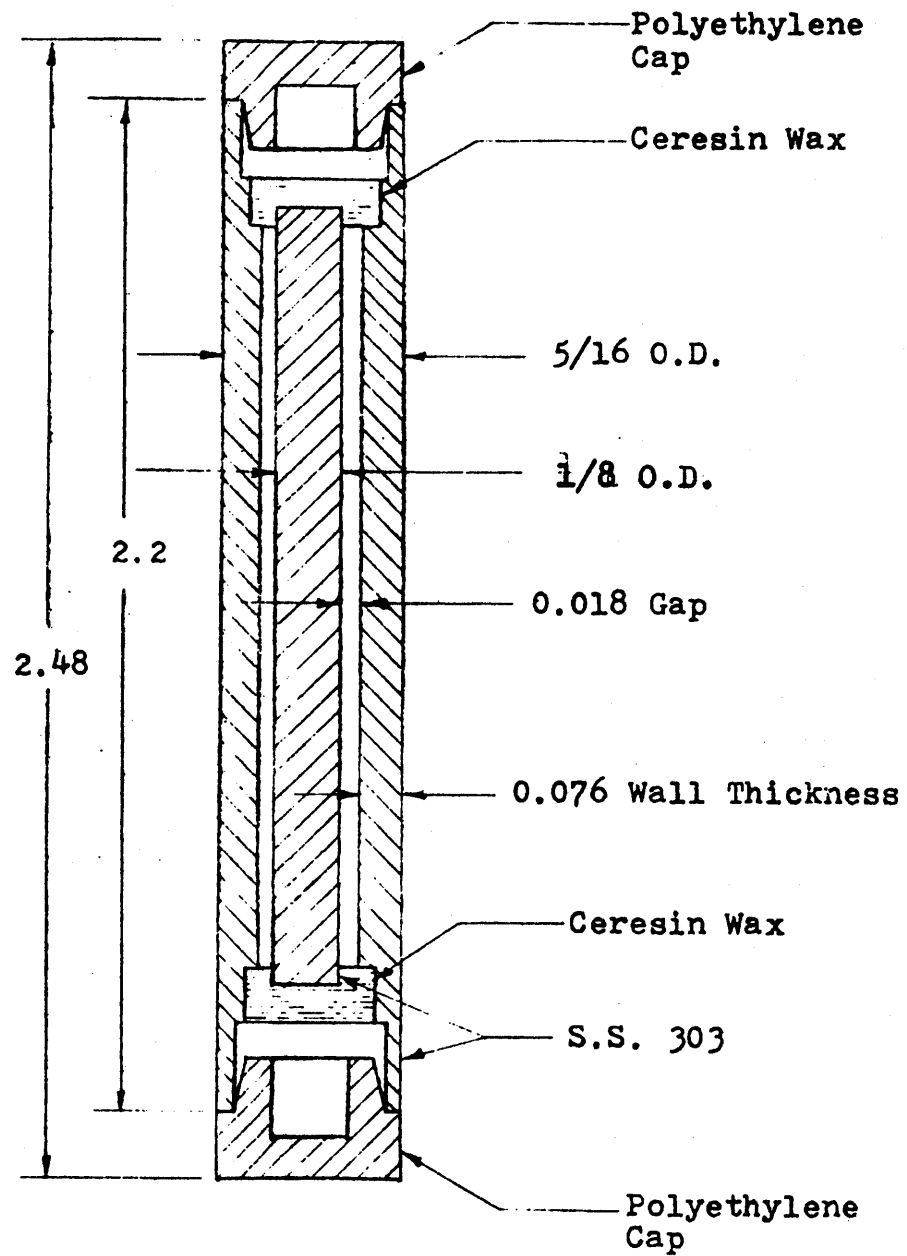


Fig. 2.4. Schematic of Ionization Chamber Dosimeter
(All Dimensions in Inches)

corrections for spectral effects. Several duplicate traverses were performed in the blanket mockup with these dosimeters. Figure 2.5 compares the ICD and TLD results, which are in good agreement considering the precision of both sets of data: $\pm 10\%$ for ICD's and $\pm 7\%$ for TLD's.

It was concluded that the far greater versatility of the TLD's, due to their small physical size and wide dose range, made them considerably superior to ICD's for future applications, and that TLD response is sufficiently well understood to provide useful gamma heating data.

2.6 Unfolding Gamma Spectra

If the heating rates in several materials and their appropriate multigroup cross sections are known, the gamma spectrum may in principle be found in a manner entirely analogous to the more familiar process of unfolding neutron spectra from foil activation data.

Capsules of stainless steel, tin, zirconium, tungsten and lead were irradiated at the center of Blanket Mockup No. 4 and the gamma heating rates determined using the Co-60 calibration facility and RESPOND program as before. These results were then processed by the MITSPECTRA neutron spectrum unfolding code (which is a simplified version of the RFSP code⁽⁷⁾ - an improved version of the SPECTRA code⁽⁸⁾).

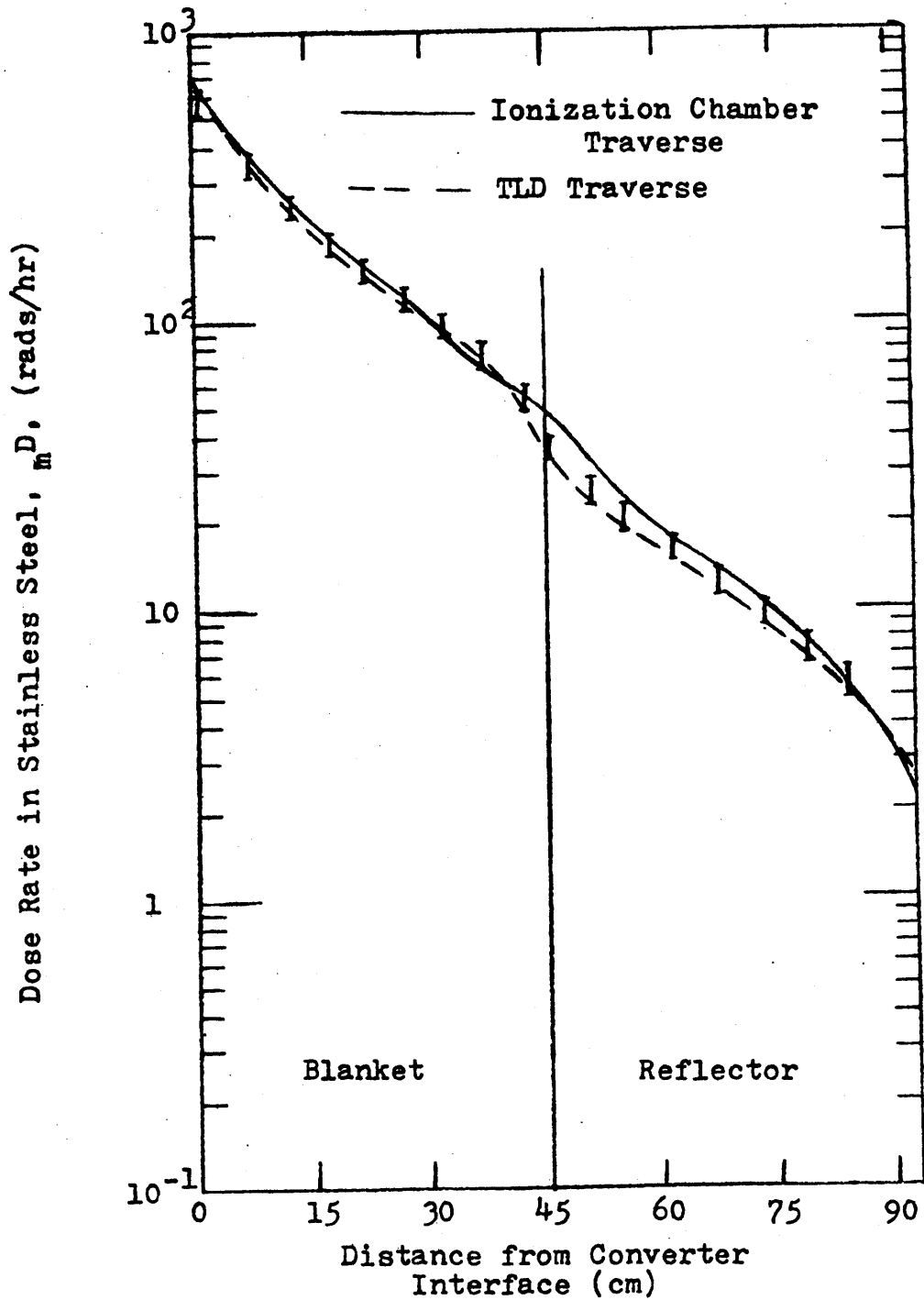


Fig. 2.5. Comparison of Absolute Dose Rates in Stainless Steel Measured with TLDs and Ionization Chamber Dosimeters.

The spectrum calculated by ANISN at the blanket midpoint is compared to the MITSPECTRA unfolding results in Table 2.2 and Fig. 2.6. Considering the fact that this was an initial effort, the results are in adequate agreement. This general approach appears to be very promising and additional work is recommended. Before much improvement can be realized, however, better precision for TLD readout must be established.

2.7 Measurement of Spectral Response Factors

If LiF is encapsulated in teflon, a very nearly "matched cavity" results. Hence a check on how well RESPOND calculates spectral response factors can be obtained by comparison of two capsules in the same location in an assembly, one with a teflon wall, the other with the material to be tested. This comparison was carried out for both stainless steel and lead relative to teflon, with the following results:

Dose Ratio Relative to Teflon/LiF

<u>Sleeve Material</u>	<u>Measured Ratio</u>	<u>Calculated Ratio</u>
Stainless Steel	0.970 \pm 0.136	1.056
Lead	1.333 \pm 0.187	1.480

The results agree within the experimental uncertainty, hence we may put some confidence in the use of RESPOND for (1/f) calculations. Again we were limited by the modest precision attainable using a state-of-the-art TLD readout device.

TABLE 2.2 Gamma Spectrum Unfolded
at Blanket Midpoint

1. Gamma Spectrum

<u>Gamma Group</u>	<u>E_{Max} (MeV)</u>	<u>MIT SPECTRA</u>	<u>ANISN*</u>	<u>% DEV.</u>
1	10.0	0.00131146	0.00101	+37.876
2	8.0	0.0128947	0.00640	+100.160
3	6.5	0.00678541	0.00463	+53.549
4	5.0	0.0210559	0.01218	+70.198
5	4.0	0.0869786	0.04171	+101.522
6	3.0	0.121072	0.06707	+70.838
7	2.5	0.171627	0.11238	+44.220
8	2.0	0.120574	0.10424	+10.252
9	1.66	0.0707146	0.06847	+4.242
10	1.33	0.103370	0.13662	-25.291
11	1.0	0.0737837	0.09804	-21.572
12	0.8	0.0613845	0.10531	-39.560
13	0.6	0.0441173	0.14949	-69.203
14	0.4	0.0479146	0.04861	+3.751
15	0.3	0.0373710	0.03201	+22.407
16	0.2	0.0175028	0.01091	+62.410
17	0.1	0.00146931	0.00085	+88.896
18	0.05	0.00007259	0.00006	+22.491
Total		1.0000	1.00000	

2. Capsule Dose Rates

<u>TLD Sleeve Material</u>	<u>Experimental Dose Rates (rads/hr.)</u>	<u>Calculated Dose Rates (rads/hr.)</u>	<u>% Dev.</u>
Fe	54.1	56.7	-4.82
Zr	56.9	59.5	-4.58
Sn	72.2	63.5	12.09
W	96.1	84.3	12.31
Pb	85.3	93.9	-10.11

* Calculated value used as initial guess to unfolding program

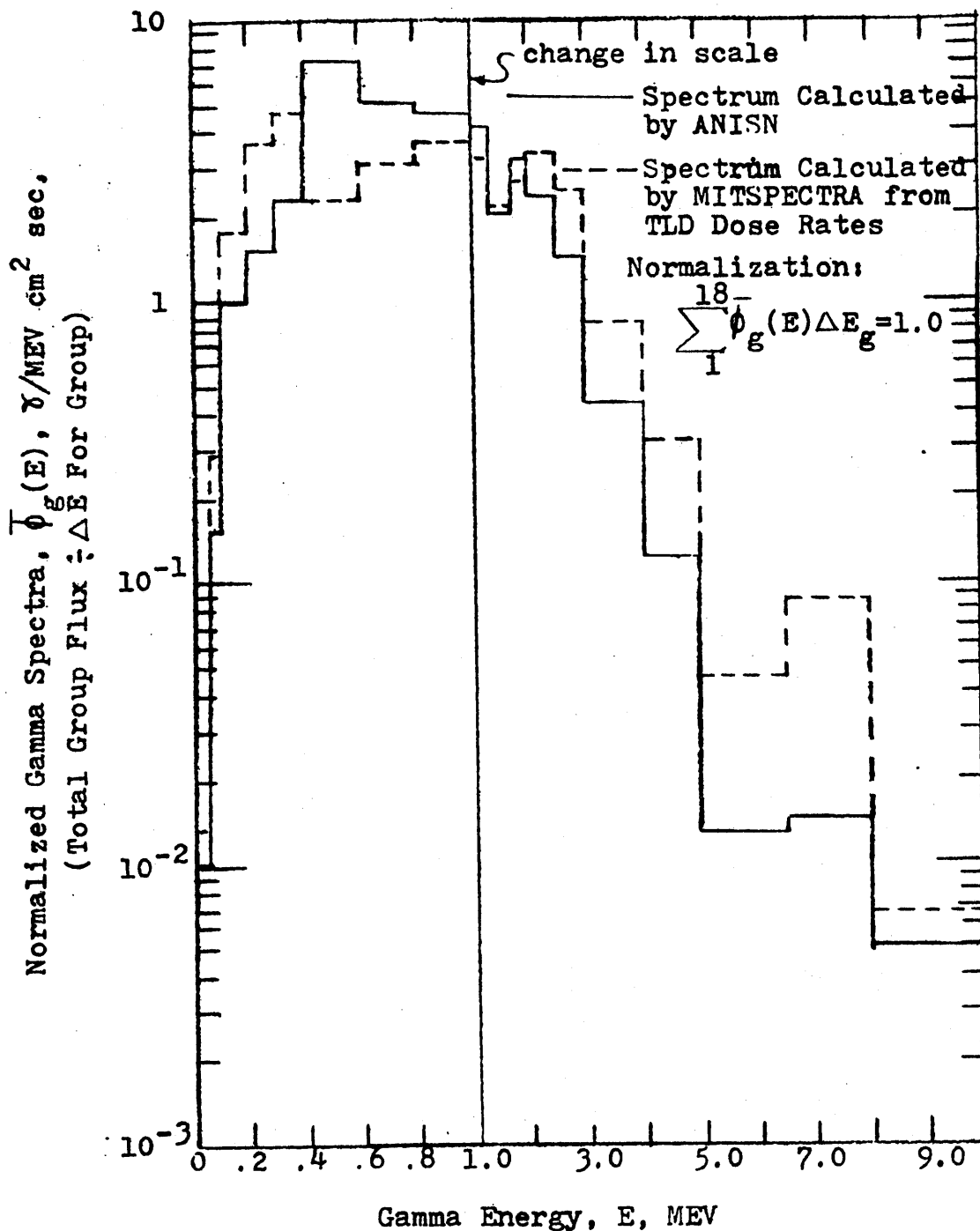


Fig. 2.6. Comparison of Spectra Calculated by ANISN and Unfolded by MITSPECTRA from Experimental Data

2.8 Summary, Conclusions and Recommendations

On the whole, gamma dosimetry using TLD detectors has been confirmed to be a useful and reasonably accurate means for gamma heating measurements in the blanket and reflector regions of fast reactors. It has been adopted as the reference method for all subsequent work at MIT. There are, however, a number of areas in which improvements should be sought:

(1) Increased precision in measurement of TLD response is an essential prerequisite to future work. Development of better TLD readout devices or upgrading of present commercial units is called for. The use of photon counting may offer one route to higher precision.⁽⁹⁾ Alternatively, an entirely new approach based on detection of radiophotoluminescent response may be profitable.⁽¹⁰⁾ Work along these lines will be investigated during the coming year at MIT.

(2) There are still a number of fine points associated with the various corrections applied to raw TLD response data which must be tidied up: neutron response, calculation of $1/f$ factors, calibration methods. Some of these items will undoubtedly be clarified in the joint ANL/ORNL/MIT intercomparison studies now underway.

(3) Parametric and sensitivity studies should be carried out using coupled neutron-gamma cross section

sets to better define the factors affecting the accuracy of state-of-the-art calculation methods. An investigation along these lines has been initiated at MIT.

(4) When better experimental precision has been attained, additional work should be carried out on gamma spectrum unfolding and measurement of $1/f$ -factors as discussed in sections 2.6 and 2.7 of this chapter.

(5) Additional measurements and calculations should be carried out in the reflector region of the MIT blanket mockups, since the C/E discrepancies appear to be consistently higher in this region.

As noted, work in this general area will continue to be an important task during the coming fiscal year.

2.9 References

- (1) P.A. Scheinert and M.J. Driscoll, "Gamma Heating Measurements in Fast Breeder Reactor Blankets", COO-2250-10, MITNE-164, August 1974.
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CHAPTER 3

A FOIL-STACK METHOD FOR EPITHERMAL NEUTRON SPECTROMETRY3.1 Introduction

For some time now, foil method neutron spectrometry has been a major sub-task within the overall project program. Work preceding that discussed here is described in Reference (1) and summarized in Reference (2); a more detailed report of the present research is available in Reference (3).

A variety of foil methods, whose characteristics are well-documented in the literature (4,5,6), have been applied for neutron spectrometry in reactor media. With the exception of sandwich-foil techniques (7,8), most of the common approaches have been investigated previously on the MIT Blanket Research Project. More recently, a variation of the sandwich-foil method, the stacked-foil technique (9,10,11) has been suggested as a worthwhile new approach. It was the purpose of the work summarized here to evaluate this method for epithermal neutron spectrometry in fast reactor applications.

In the usual sandwich method three foils of the same material are stacked together and irradiated. The difference between the exposed surface and the heavily-shielded center foil specific activities is attributed to resonance capture reactions; and hence this difference measures the neutron flux at specific resonance energies instead of over the whole spectrum. Each sandwich produces one data point, which corresponds to the flux at one or more dominant resonance energies;

by using several sandwiches made up of different materials and appropriate unfolding techniques, it is possible to infer values of the neutron flux at a half-dozen or so points distributed over the epithermal region.

The foil-stack method, on the other hand, uses a stack made up of a large number of foils of the same material. Analysis of the spatial distribution of the activities within the foil stack can also presumably be carried out to unfold the incident spectrum. As originally proposed by Nisle^{9,10,11} the analysis considers the superimposed effect of both resonance and non-resonance absorptions. However, in a fast reactor, most of the capture reactions occur above 1 kev: a region where instrumental spectrometry is superior to foil techniques. Thus the foil-stack method, as originally proposed, does not appear very attractive for either infra- or sub-kev spectrometry.

Weitzberg, however, in discussing his foil sandwich work⁸, mentioned use of more than three foils in a stack - which suggested that a combination of the sandwich and stack concepts might prove useful. This line of thought motivated the approach developed in the present research: a foil stack will be used, but the central heavily shielded foil activity will be subtracted from the other foil activities to accentuate the resonance activity contribution in the other foils in the stack and to suppress the non-resonant contributions. Analysis of the spatial distribution of activity differences within the stack should then permit the unfolding of a spectrum in

the 10 ev - 10 kev region dominated by resonance interactions.

The remaining sections of this chapter discuss the step-by-step procedures and analyses required to implement and evaluate this method.

3.2 Theoretical Considerations

The activity of the individual foils in a stack can be written in the form:

$$A_j - A_c = \sum_{g=1}^G (F_{gj} - F_{gc}) \sigma_g \phi_g, \quad j=1 \dots N \quad (3.1)$$

where

A_j = specific activity of j th foil (corrected for background, decay, etc.)

A_c = specific activity of center foil in stack

F_{gj} = self-shielding factor for the j th foil

F_{gc} = self-shielding factor for the center foil

σ_g = group g cross section for an infinitely dilute thin foil

ϕ_g = neutron group flux

G = total number of neutron groups

N = total number of foils

Note that because of the manner in which we have written the relation:

- (a) Groups in which there is no appreciable self-shielding drop out of the set of equations (since $F_{gj}=F_{gc}=1.0$)
- (b) Only $N/2$ (or $(N-1)/2$ for odd N) of the foil data are non-redundant because of the symmetry of the foil stack about the centerplane.

Infinitely dilute cross sections are familiar quantities which are tabulated in most published cross section sets or which can be calculated directly from published resonance parameters (e.g. ENDF) using available processing codes. Some comment is necessary, however, on the calculation of the self-shielding factors, F_{gj} .

As part of the present work, and as reported in detail in Reference (3), a computer program was developed to calculate foil self-shielding factors under the following conditions and approximations:

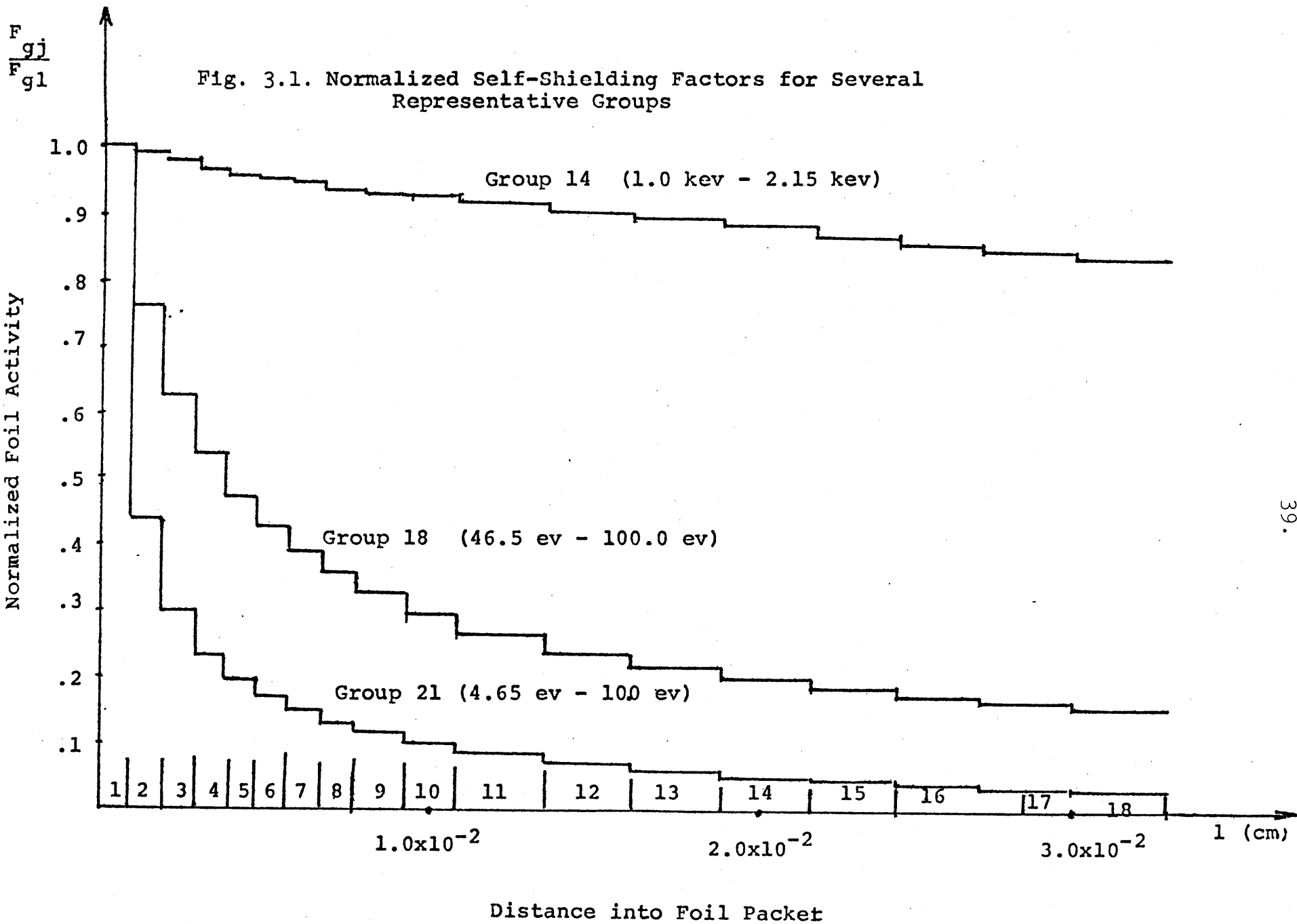
- (a) a volumetrically uniform isotropic source of (scattered) neutrons in an infinite medium surrounding a semi-infinite planar foil stack is considered. In this regard it should be noted that the development presented by Nisle^(9, 10, 11) appears applicable only to a uniform surface source.
- (b) the perturbation introduced by the stack of foils is ignored
- (c) the cross sections are described by the Breit-Wigner single-level formula, including the effect of Doppler-broadening. However, the effects of potential scattering were excluded (scatter-in is compensated by scatter-out).

The program was demonstrated to correctly predict $F_{gj} = 1$ for an infinitesimally thin foil, and to calculate the correct infinitely dilute σ_g values (using $1/E$ intra-group weighting for this test case only). The program was

then employed to calculate self-shielding factors for the stacks of 0.5 - 2.0 mil gold foils used in the present work (see section 3.3). The group structure employed was that of the 26-group ABBN cross section set.⁽¹²⁾ Several interesting results were made evident:

- (a) For the 12 groups above 4.65 kev the F_{gj} were found to be sufficiently close to unity to permit dropping these groups out of the summation in Eq.(3.1). Thus, as anticipated and as desired, the proposed method is sensitive to sub-kev range neutrons.
- (b) Groups 15, 16, 17, and 22 were found to have similar values of F_{gj} , as did 19, 20, and 23, 24, 25. Thus the foil stack method will not be sensitive to the detailed shape of the neutron spectrum within these three broad energy bands.
- (c) Groups 18 and 21 on the other hand exhibit distinctively different F_{gj} . Thus the foil-stack method should be fairly effective in inferring the overall shape of the spectrum but incapable of resolving any fine structure.

A plot of the self-shielding factors for the 14th (representative unresolved) group, 18th (representative resolved) group and the 21st (most heavily-shielded) group is shown in Fig. 3.1. Note that the self-shielding factor of each foil is normalized to that of the surface foil. It is clear from Fig. 3.1 that the largest changes in self-shielding -- hence the most significant in terms of discrimination among



various incident flux spectra -- occur in the first ten or so foils of the stack.

Having calculated self-shielding factors and infinite dilution cross-sections, we are ready to discuss the experimental procedures, the resulting data and the spectrum-unfolding analysis.

3.3 Experimental Procedure

Gold was selected as the foil material for a number of reasons.

- (a) Gold has desirable resonance structure for the present application and its cross section data are well known: 63 resolved resonance parameters in the range from a few ev to 1 kev are tabulated in ENDF/B. Its widespread use as a cross section standard, to which other materials are compared, also insures continuing attention to acquisition of improved data.
- (b) It has a relatively large capture cross section and only one isotope. The daughter product produced by neutron capture has a good combination of half-life, decay gamma yield and energy.
- (c) It is available in high purity (>99%) and ultra-thin foil form. Foil is available commercially, has comparatively good mechanical properties and is highly corrosion resistant.
- (d) Previous investigations using sandwich foil and foil stack techniques have invariably included gold

as a major or exclusively-employed detector material.

In brief, gold appeared to be so far the superior candidate that no further consideration was given to other possibilities.

The foil stack used in this work was made up of 104 gold foils, 0.5 inch in diameter, having thicknesses ranging from 0.5 mil (outermost foils) to 2 mils (inner foils).

The choices of thickness and diameter were motivated by several considerations. It was desirable to have the outer foils in the stack as thin as possible to achieve high spatial resolution, but 0.5 mil represents a practical lower limit (for unalloyed gold) since thinner foils are too easily damaged during handling. The total thickness of the stack is set by the requirement that significant resonance self-shielding be achieved as far up into the kev region as possible. While the largest possible radial extent is to be preferred, the foil diameter was restricted in the present case to fit available irradiation and counting facilities.

All foils were weighed, compressed into a stack and held together using mylar tape. The same foil stack was used in each run but irradiations were separated in time by at least six weeks to insure sufficient decay between runs. The foil stack was mounted in the central insert of the special test assembly constructed by Ortiz for proton recoil spectrometry in the MIT blankets.⁽¹³⁾ The stack was oriented with its midplane vertical and perpendicular to the face of the blanket to insure angular symmetry of the incident neutron current.

Irradiations were carried out in the second row of Blanket Mockup No. 4, in the assembly next to the central assembly of the 9-assembly row. Irradiation times were approximately 2 hours each. Two runs were made: one inside a cadmium can, the other inside aluminum.

In addition to the blanket runs, a calibration irradiation was carried out in the 1/E Calibration Facility in the MIT Reactor Hohlraum. (14)

All foils were counted using standard methods in the conventional sodium-iodide well-crystal set up described by Akalin. (15) The usual corrections for decay were made. Greater than 300,000 cpm were recorded for the least-activated, thinnest foil, thereby providing excellent counting statistics. The final step involved averaging the specific activity (cpm/mg) of symmetrically-placed foils on opposite sides of the stack midplane and subtracting the average specific activity of the centrally-located foil pair.

3.4 Data Analysis and Results

Because of the aforementioned inability of this method to respond to fine structure, an appropriately-constrained unfolding method had to be developed. A relation of the following form was assumed to hold for the neutron spectrum:

$$\Sigma_T(E) \varphi(E) = C E^m \quad (3.2)$$

where

$\Sigma_T(E)$ = total cross section of the homogenized ambient medium in which the foil stack was irradiated.

Equation (3.2) (in multigroup form) and Eq. (3.1) were solved iteratively by varying m until the mean square error between measured and calculated activity differences was minimized. We then compared the experimental results for m to the values of m determined by a least squares fit of Eq. (3.2) to the computed multigroup spectrum at the irradiation position (26 group ANISN, S_8 calculations).

The results were as follows: m was confirmed to be -1.0 in the $1/E$ Calibration Facility, and was found to be -0.10 in Blanket Mockup No.4, in good agreement with the fit to the 26-group calculations, which gave $m = -0.12$. In all cases the calculated foil-stack activity profiles reproduced the experimental data within experimental error.

3.5 Discussion and Conclusions

Although additional work could be done in several areas to improve the foil-stack method -- such as use of dilute gold alloy foils to improve spatial resolution -- it is clear that the usefulness of this approach is limited due to its inherent lack of energy resolution. Use of several foil stacks made up of different resonance absorbers could alleviate this defect, albeit with a significant increase in the amount of effort required. No further work along these lines is planned in the near future.

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CHAPTER 4

OTHER EXPERIMENTAL WORK4.1 Introduction

In this chapter two experimental topics will be summarized: measurements of U-238 capture near the blanket/reflector interface; and Blanket Mockup No. 4 foil traverses.

4.2 Interface Traverses

An interesting measurement was carried out during the past year to assess the effect of reduced U-238 self shielding near the blanket/reflector interface.

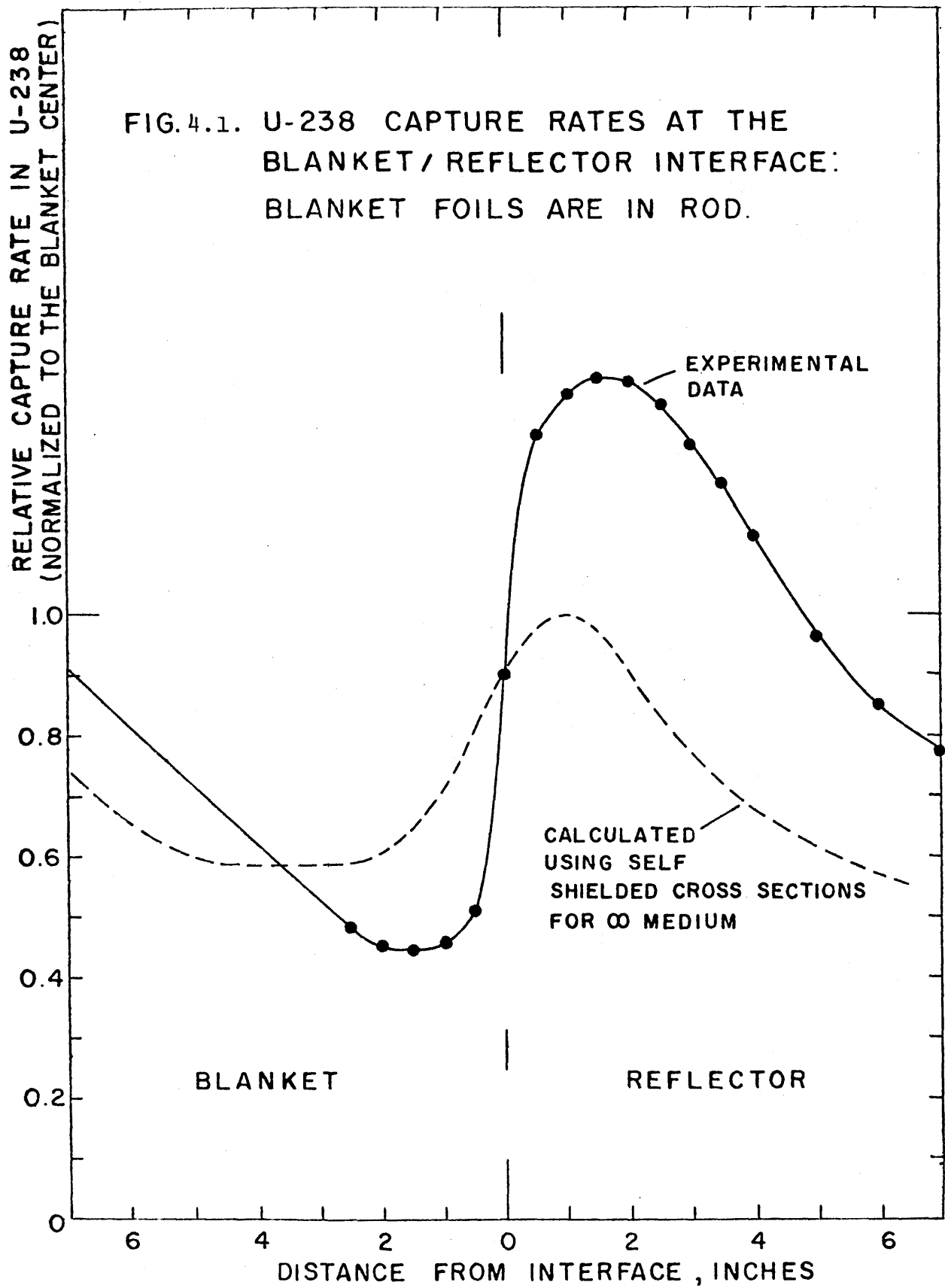
A standard uranium-metal fueled blanket assembly (see Ref. (1) for design details) was modified by drilling a hole into the side of the subassembly box at its vertical midplane, between the second and third rows of fuel pins. The hole extended approximately half-way through the six inch thick subassembly box. A thin steel-walled traversing tube, capped off at the end, was inserted into this hole and sealed to the subassembly box wall to form a traversing tube. The traversing tube was then loaded with a standard diameter length of fuel cladding containing a stack of 0.250 in. dia. metal fuel slugs with 0.250 in dia. depleted uranium foils sandwiched between them. The assembly was inserted into the 3rd row of the blanket with its traversing

tube lined up with a matching hole penetrating an extension of the 2 in. dia. steel cylinder which plugs the traversing hole drilled through the 18 in. thick steel reflector.

The hole in the reflector plug was filled with steel slugs, also with depleted U-238 foils sandwiched between them. Thus we now had the capability of measuring the capture rate in U-238 as a function of position in the region on either side of the blanket/reflector interface.

The depleted uranium foils in the special interface traversing set-up were irradiated following standard project procedures and counted for the induced Neptunium activity. Figure 4.1 shows the results. The points to the left of the interface correspond to U-238 capture in uranium fuel rods - as would occur in the real-life situation; the foils to the right of the interface, in the steel reflector, are of purely academic interest. We should also note here that the 0.25 in. dia. metal fuel rods were chosen to simulate the self shielding in the larger diameter UO_2 fuel rods usually specified for radial blankets: measurements were made in a special UO_2 fueled, sodium "cooled" assembly to confirm this hypothesis.

The results of greatest interest are the steep gradient and extremely high U-238 capture rate immediately adjacent to the interface. This indicates that the outboard fuel pins



in a blanket could experience a factor of 2 capture gradient across their diameter, and also that the plutonium buildup rate in these outermost pins could even exceed, by $\sim 10\%$, the rate in the innermost pins adjacent to the second row of blanket assemblies. The implications to the blanket designer are not entirely clear at this point, but this is a phenomenon worth becoming aware of. If more accurate calculations of these interface effects are required, then it is also clear that one must adopt a procedure which allows for reduced self shielding near the interface.

4.3 Blanket Mockup No. 4 Traverses

Data from the foil activation traverses performed on Blanket Mockup No. 4 have been compiled. Table 4.1 summarizes pertinent counting data. The equipment and procedures were in accordance with standard methods used by the project, documented in Refs. (1) and (2), which report similar data on Mockups No. 2 and 3. Table 4.2 lists the actual data. Note that all points are relative to the traverse position No. 5, at the "center" of the blanket. Some of this data has already been reported and analyzed by Wood (3) in his assessment of thorium blanket performance.

The most important use to which this data will be put in the future will be in an assessment of the ability of state-of-the-art calculational methods to adequately predict blanket breeding performance. In order to do this at least

TABLE 4.1. BLANKET MOCKUP NO. 4 ACTIVATION TRAVERSES:

TYPICAL DATA PERTINENT TO FOIL COUNTING

Reaction	Gamma Energy Detected (MeV)	Discriminator Settings (volts)*		Typical Counts Accumulated (Background subtracted)	Typical Irradiation Time (hours)	Waiting Time (hrs)	Counting Time (minutes)
		E _{min}	E _{max}				
Au ¹⁹⁷ (n,γ)Au ¹⁹⁸	B:	0.8	1.38	149,670	6.04	147.01	3
	R: 0.412	0.74	1.45	123,832	2.08	4.60	4
Mo ⁹⁸ (n,γ)Mo ⁹⁹	B:	1.56	2.05	32,444	10.02	65.42	9
	R: 0.78	1.55	2.60	22,437	14.00	5.67	11
Na ²³ (n,γ)Na ²⁴	B:	2.85	3.60	63,817	5.47	9.16	8
	R: 1.368	2.80	∞	56,525	6.04	4.73	5
Cr ⁵⁰ (n,γ)Cr ⁵¹	B:	0.58	1.05	11,395	7.95	7.4	9
	R: 0.322	0.53	0.98	2,306	3.04	.96	6
Mn ⁵⁵ (n,γ)Mn ⁵⁶	B:	1.70	2.50	30,000	6.71	29.79	8
	R: 0.845	1.62	2.55	85,008	3.04	17.13	5
In ¹¹⁵ (n,n')In ^{115m}	B:	0.66	0.98	51,703	10.02	10.80	8
	R: 0.335	0.63	0.95	6,421	12.00	11.22	9
U ²³⁵ (n,f)	B: 0.72 - ∞	1.68	∞	51,307	6.04	---	4
Pu ²³⁹ (n,f)	B: 0.72 - ∞	1.69	∞	35,437	5.00	---	--
Mn powder	R: 0.70 - ∞	1.58	∞	13,155	4.00	29.10	11
Th ²³² (n,f)	R: 0.72 - ∞	1.63	∞	190,314	5.53	3.46	9

*Corresponds to a calibration of approximately 0.43 MeV gamma energy per volt discriminator setting.

TABLE 4.2. BLANKET MOCKUP NO. 4: ACTIVATION TRAVERSES

	POSITION (from blanket- converter interface) cm	REACTION RATE \pm SDM			
		In ¹¹⁵ (n,n')In ^{115m}	Au ¹⁹⁷ (n, γ)Au ¹⁹⁸	Na ²³ (n, γ)Na ²⁴	Mo ⁹⁸ (n, γ)Mo ⁹⁹
blanket region	2.34	5.2305 \pm 0.3%	1.4170 \pm 0.5%	1.7915 \pm 13.4%	1.4330 \pm 3.0%
	7.50	3.3645 \pm 0.7%	1.3745 \pm 2.2%	1.5475 \pm 0.6%	1.3990 \pm 0.9%
	12.70	2.1785 \pm 0.9%	1.2940 \pm 1.4%	1.3835 \pm 0.1%	1.3075 \pm 0.3%
	17.40	1.4790 \pm 0.9%	1.1515 \pm 0.3%	1.2050 \pm 1.5%	1.1710 \pm 0.9%
	22.60	1.0000 ----	1.0000 ----	1.0000 ----	1.0000 ----
	27.80	.7240 \pm 0.6%	.8660 \pm 1.8%	.8475 \pm 1.7%	.8598 \pm 0.5%
	32.60	.5156 \pm 3.1%	.7410 \pm 0.3%	.7105 \pm 0.2%	.6999 \pm 1.5%
	37.60	.3844 \pm 4.2%	.6035 \pm 0.6%	.5709 \pm 0.5%	.5731 \pm 0.4%
	42.80	.2798 \pm 12.1%	.5920 \pm 0.4%	.5190 \pm 1.0%	.4881 \pm 1.5%
reflector region	48.2	.1710 \pm 0.9%	.6033 ----	1.4940 \pm 0.9%	.9821 \pm 0.0%
	50.8	.0933 \pm 4.6%	.6413 \pm 0.9%	1.4750 \pm 0.7%	.7519 \pm 2.4%
	52.3	.0568 \pm 3.0%	.5643 \pm 1.9%	1.3350 \pm 22.6%	.5664 \pm 0.2%
	58.4	.0375 \pm 17.7%	.4658 \pm 0.5%	.8805 \pm 27.9%	.4372 \pm 1.3%
	63.5	.0252 \pm 12.4%	.3581 \pm 0.7%	.7640 \pm 2.4%	.3312 \pm 0.4%
	68.5	.0180 \pm 28.5%	.2685 \pm 0.3%	.5750 \pm 1.8%	.2507 \pm 0.2%
	73.6	.0123 \pm 19.0%	.2005 \pm 0.6%	.4745 \pm 22.7%	.1885 \pm 0.8%
	78.7	.0091 \pm 24.2%	.1488 \pm 0.2%	.3570 \pm 27.7%	.1370 \pm 4.7%
	83.8	.0069 \pm 24.3%	.1056 \pm 1.1%	.2670 \pm 31.2%	.0985 \pm 2.6%

normalized to fifth position

TABLE 4.2. (Continued)

POSITION (from blanket- converter interface) cm	REACTION RATE \pm SDM			
	Mn ⁵⁵ (n, γ)Mn ⁵⁶	Cr ⁵⁰ (n, γ)Cr ⁵¹	U ²³⁵ (n,f)	Pu ²³⁹ (n,f)
2.34	1.2904 \pm 4.1%	1.6570 \pm 2.5%	1.7190 \pm 1.0%	1.9367
7.50	1.3220 \pm 0.4%	1.5270 \pm 1.5%	1.6128 \pm 2.7%	1.7399
blanket 12.70	1.2535 \pm 0.1%	1.3410 \pm 3.9%	1.3770 \pm 7.0%	1.4457
17.40	1.1315 \pm 1.9%	1.1910 \pm 2.4%	1.1846 \pm 7.8%	1.1339
region 22.60	1.0000 ----	1.0000 ----	1.0000 ----	1.0000
27.80	.8521 \pm 0.9%	.8665 \pm 0.7%	.8592 \pm 7.5%	.9518
32.60	.7325 \pm 2.3%	.7165 \pm 2.4%	.7240 \pm 1.2%	.7026
37.60	.6110 \pm 0.4%	.6050 \pm 5.2%	.6083 \pm 4.2%	.5304
42.80	.5305 \pm 0.2%	.5625 \pm 0.2%	.5620 \pm 3.1%	.4613
48.2	2.370	1.925 \pm 3.4%		
50.8	2.040	1.910 \pm 6.8%		
52.3	1.6200	1.645 \pm 13.4%		
reflector 58.4	1.290	1.335 \pm 8.8%		
63.5	1.000	1.000 ----		
region 68.5	.755	.741 \pm 5.1%		
73.6	.566	.592 \pm 20.6%		
78.7	.422	.453 \pm 4.6%		
83.8	.299	.323 \pm 18.1%		

normalized
to fifth
position

normalized to
fourteenth
position

those without SDM were obtained from only one run.

TABLE 4.2. (Concluded)

	POSITION (from blanket- converter interface) cm	REACTION RATES \pm SDM	
		Mn ⁵⁵ (n, γ)Mn ⁵⁶ (powdered)	Th ²³² (n,f)
	42.8	-----	1.3645 \pm 2.0%
	48.2	1.9827	1.0302 \pm 1.5%
Reflector	50.8	1.8701	1.0056 \pm 0.2%
Region	52.3	1.5130	1.0021 \pm 0.4%
	58.4	1.2860	.9987 \pm 0.1%
	63.5	1.0000	1.0000 ----
	68.5	.8520	.9953 \pm 1.1%
	73.6	.6618	.9958 \pm 0.3%
	78.7	.5132	.9951 \pm 0.7%
	83.8	.3935	.9951 \pm 0.9%

part of the data will have to be absolutely normalized, as can be seen from the following development.

Starting with the definition of blanket (external) breeding ratio:

$$b_x = \frac{\text{fissile Pu production rate in blanket}}{\text{fissile Pu destruction rate in reactor}} \quad (4.1)$$

and under the assumptions and approximations:

- (1) the beginning-of-life clean core and blanket are being simulated
- (2) all core fissile material is treated as equivalent to Pu-239
- (3) all fertile material is treated as equivalent to U-238,

we have,

$$b_x = \left[\begin{array}{c} \text{Core} \\ \text{term} \\ \frac{1}{1+\alpha_c^{49}} \end{array} \right] \cdot \left[\begin{array}{c} \text{Coupling} \\ \text{term} \\ F_{49c}^{25b} \end{array} \right] \cdot \left[\begin{array}{c} \text{Blanket} \\ \text{term} \\ C_b^* \end{array} \right] \quad (4.2)$$

where α_c^{49} = mean capture-to-fission ratio for the fissile species in the core

F_{49c}^{25b} = ratio of U-235 fission rate in blanket to fissile fission rate in core

$$C_b^* = \left(\frac{N_{28}}{N_{25}} \right)_b \cdot \left(\frac{\sigma_c^{28}}{\sigma_f^{25}} \right)_b, \text{ ratio of U-238 captures to U-235 fissions in the blanket}$$

Thus, at its simplest level, our concern is with measurement and calculation of $\frac{\sigma_c^{-28}}{\sigma_f^{-25}}$. Analysis of the already available data and measurement of additional absolute capture-to-fission ratios will be a prime objective of our future work.

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- (3) P.J. Wood and M.J. Driscoll, "Assessment of Thorium Blankets for Fast Breeder Reactors", COO-2250-2, MITNE-148, July 1973.

CHAPTER 5

PARAMETRIC AND OTHER STUDIES5.1 Introduction

To date most analytic and computational effort has been directed toward the design and analysis of experiments. However, with the recent shutdown of the MIT Reactor for renovation, the opportunity and need for increased effort in this area is evident. In addition the depth of our understanding has increased to the point where such studies can be focused on specific questions having an important pay off. So far work has been limited to investigation of the central issue in blanket mockup studies - the effect of core (converter assembly) characteristics on blanket performance. Information developed on this question reflects directly upon several questions of interest: the nature of (and need for) future experiments in which core (converter) characteristics are varied; the extent to which universal generalizations can be developed; and the amount of specific detail which must be incorporated into benchmark calculations centered around blanket mockup experiments. In this chapter results bearing on all of these points will be summarized.

5.2 Equivalent Source Study

The convenience of the MIT Blanket Mockups as benchmark assemblies would be enhanced if it were possible to simplify

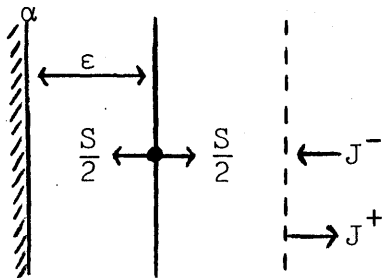
the treatment of the thermal-to-fast converter assembly.

At least two possibilities are available to us:

- (1) definition of an equivalent, conventional 2-zone cylindrical core
- (2) development of an equivalent source at the inboard surface of the blanket in slab geometry.

Forbe's work has dealt extensively with the first approach, which was a necessary step in establishing the validity of the Blanket Test Facility concept⁽¹⁾; the equivalent source approach is discussed below.

Consider a LHS boundary having albedo α , with a uniform isotropic source (of strength S) a small distance, ϵ , to its right:



The partial current in the positive direction is given by (for group g in a multigroup representation)

$$J_g^+ = \frac{S_g}{2} + \alpha_g (J_g^- + \frac{S_g}{2}) \quad (5.1)$$

Thus the net current becomes:

$$J_g = J_g^+ - J_g^- = \frac{S_g}{2} + \alpha_g (J_g^- + \frac{S_g}{2}) - J_g^-$$

or

$$J_g = \frac{S_g}{2} (1 + \alpha_g) + J_g^- (\alpha_g - 1) \quad (5.2)$$

Equation (5.2) can serve as the basis for an entire family of boundary conditions replacing the BTF converter. For example, if we take $\alpha_g = 1$, a particularly convenient choice, $S_g = J_g$. Multigroup calculations (26 group $S_g P_0$) were carried out using this and other variations. In the case cited, replacing the exact (anisotropic) J_g by an equivalent isotropic S_g gave fairly good results, as shown in Fig. 5.1, where the U-238 capture rate traverses are compared: the discrepancy is limited to within a few centimeters of the converter/blanket interface. Hence for mid-blanket or reflector comparisons, replacement of the converter by an equivalent source is acceptable; unfortunately overall blanket breeding performance is sensitive to reaction rates in that part of the blanket which is nearest the core (converter), thus even the modest discrepancies shown here, are to be avoided if possible. It may well be possible to perturb the source spectrum, S_g , to force better agreement with reaction rate traverses. However our inclination at present is to favor

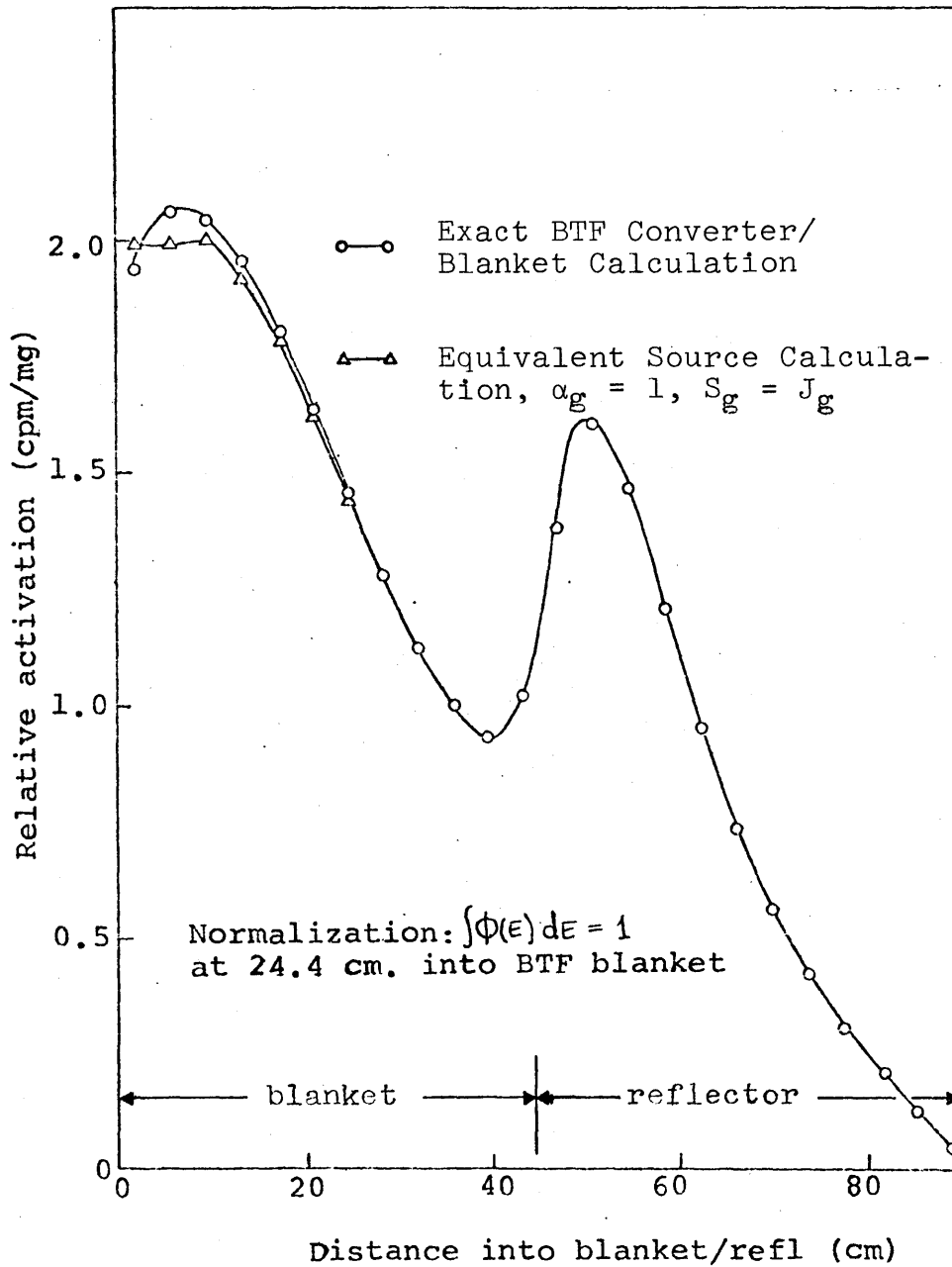


Fig. 5.1. Comparison of U-238 Capture Rates for Exact and Equivalent Source Calculations

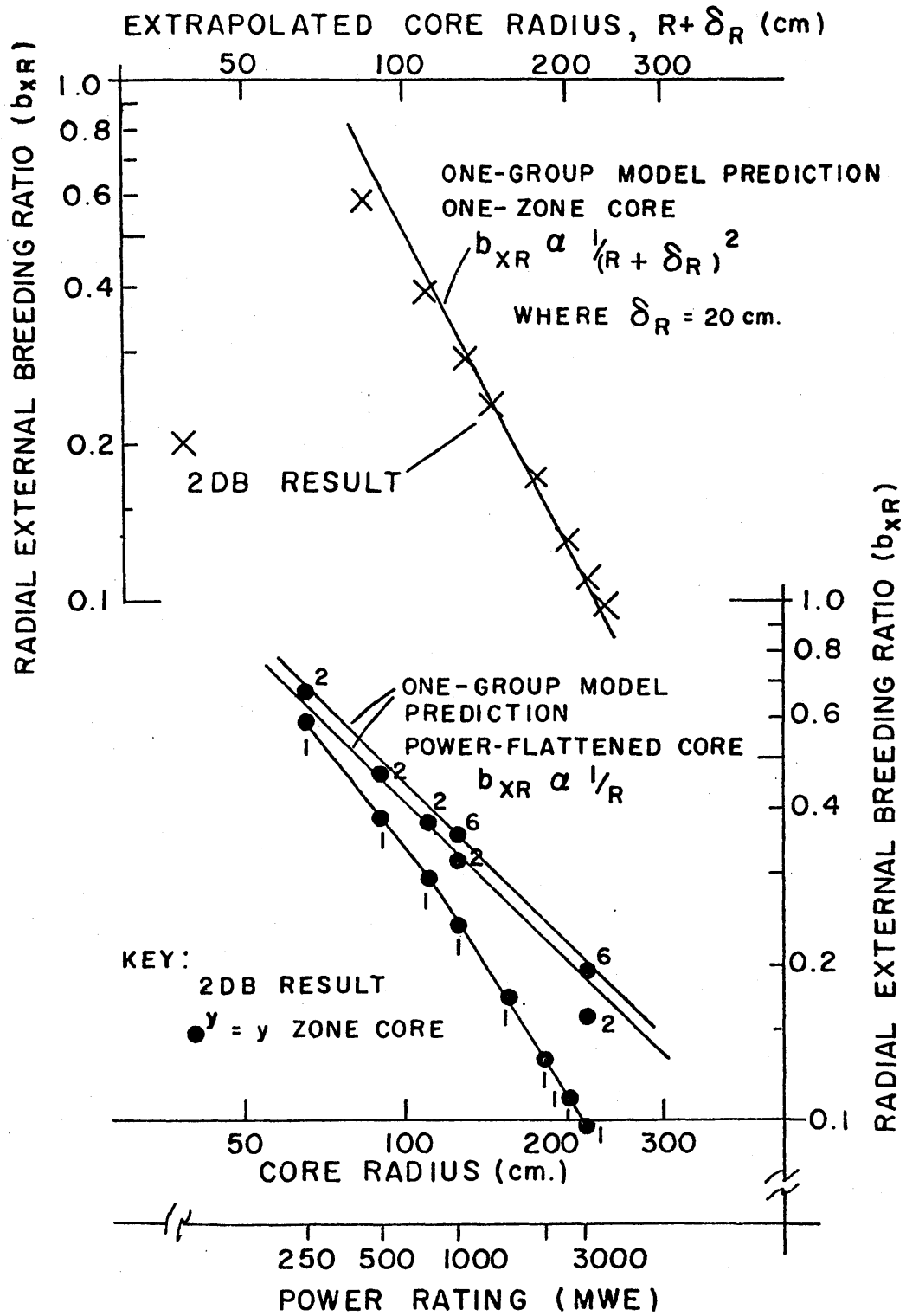
the less empirical approach of defining an equivalent cylindrical reactor for which the MIT blanket mockups may be considered either as radial or axial blankets.

5.3 Effect of Core Size on Blanket Performance

The question as to the effect of core size (reactor unit size) on blanket performance is central to the blanket designer. Some even feel that blankets will tend to decrease in thickness, and perhaps ultimately vanish on future LMFBR designs, or at the very least their performance will change significantly as reactors grow in size. On a more parochial level we are interested in knowing whether this aspect is an important point to be considered in planning future blanket mockup experiments.

Work has been initiated to look into this general problem. A simple one-group diffusion theory model of a radially-power-flattened core has been developed and compared to multi-zone multigroup reactor calculations. Preliminary results are shown in Fig. 5.2. Note that the radial blanket breeding ratio, which would decrease roughly as $(\text{radius})^{-2}$ for a single-enrichment core, tends to vary as $(\text{radius})^{-1}$ for radially-power-flattened cores and for cores of practical interest, having as few as two radial enrichment zones.

FIG. 5.2. THE EFFECT OF CORE-RADIUS AND ZONE-NUMBER ON RADIAL EXTERNAL BREEDING RATIO



Other initial results are equally informative: as core size increases, the plutonium production rate per kg. of blanket fuel tends to remain constant - hence blanket fuel cycle economics tends to remain relatively unaffected; and in radially-power-flattened cores the peripheral enrichment tends to remain nearly the same as core size increases - hence the spectrum during the blanket (and its resulting neutronic environment) also tends to remain relatively unaffected.

During the coming fiscal year this work will be completed and a topical report issued.

5.4 References

- (1) I.A. Forbes, et al., "Design, Construction and Evaluation of a Facility for the Simulation of Fast Reactor Blankets", MITNE-110, MIT-4105-2, Feb. 1970.

CHAPTER 6

SUMMARY, CONCLUSIONS AND FUTURE WORK6.1 Introduction

This is the fifth annual report of the LMFBR Blanket Physics Project at MIT. During the past year work has been concerned primarily with the following areas:

- (1) Gamma heating measurements in Blanket Mockup. No. 4, a three-subassembly-row, steel-reflected blanket driven by a simulated demonstration plant core leakage spectrum.
- (2) Completion of a number of experimental projects in advance of the shutdown of the MIT Reactor for renovation, which took place in May 1974.
- (3) Methods development work, chiefly in the area of gamma heating measurements.

6.2 Discussion

The most important conclusions which may be drawn from the past year's work are as follows:

- (1) State-of-the-art gamma heating measurements and calculations for the blanket region of fast reactors are in reasonably good agreement. Discrepancies are largest for the lighter nuclei

(sodium, iron) and in the reflector region. Improvements are still necessary in both areas. The precision of our TLD measurements has been disappointingly low (one sigma uncertainly $\sim \pm 8\%$); although ANL has reported better precision by even greater attention to detail, it would be highly desirable to devise a more forgiving procedure for TLD readout. Calculations are also not in a wholly satisfactory state: our present feeling is that the gamma source term (production of gammas by neutron interactions with nuclei) is the principal source of the error.

- (2) The problem of measuring sub-keV neutron spectra in experimental assemblies remains in an unsatisfactory state. Foil stacks and sandwiches of resonance absorbers can provide a rough idea of the general shape of the $\phi(E)$ envelope, but fine structure can only be incorporated by interjecting an undesirably large calculational component into the methodology.

6.3 Future Work

During the coming contract year, July 1, 1974 through July 30, 1975 work will have to be arranged to fit the projected schedule of MITR operations, concentrating on analytic/

numerical work in the interim and deferring further major experimental work until the reactor is once again available. Development of an improved TLD readout device can proceed independent of reactor availability, however, since Co-60 irradiation can be used to provide samples for test purposes.

Major areas of analytic interest are:

- (1) Evaluation of state-of-the-art gamma heating methods in fast reactor media, including parametric and sensitivity studies using the coupled neutron-gamma cross section set used to-date at MIT.
- (2) Development of a clear understanding of the effect of reactor size on the neutronic performance of its blankets. This will determine whether blanket studies can be completely divorced from the one key reactor core characteristic that is sure to change with time, as we go from demonstration to commercial LMFBR's.
- (3) Analysis of blanket compositions and configurations having the potential for improved breeding performance, including the development of a better understanding of the effect of blanket fuel management and economics on blanket design choices.

The priority items for future experimental work will include:

- (1) Completion of high precision measurements on Blanket Mockup No. 4 necessary to assessment of our state-of-the-art capability to predict breeding performance.
- (2) Initiation of experiments on Blanket Mockup No. 5, which will have steel reflector subassemblies (in place of the laminated steel sheet in Mockup No. 4), and can thereby facilitate investigation of the discrepancies in neutron and gamma transport noticed in our work on previous blanket mockups.

As a general observation, there is a growing consensus that blanket performance is constrained within a rather narrow range of capabilities, which means, among other things, that a smaller variety of assembly variations will be needed to confirm the adequacy of design methods. Attention has evolved from global considerations, such as assembly composition and thickness, to important local effects, such as self-shielding at the blanket/reflector interface and differences between in-rod and ex-rod fertile capture rates.

Appendix A

BIBLIOGRAPHY OF BLANKET PHYSICSPROJECT PUBLICATIONS

In this appendix are tabulated all publications associated with work performed in the MIT Blanket Physics Project. Sc.D. and Nuclear Engineer's theses are listed first, followed by S.M. and B.S. theses and then by other publications.

A.1 Doctoral and Engineer's Theses

(Also see section 3 for corresponding topical reports.)

Forbes, I.A.

Design, Construction and Evaluation of a Facility for the Simulation of Fast Reactor Blankets, Feb. 1970

Sheaffer, M.K.

A One-Group Method for Fast Reactor Calculations, Aug. 1970.

Tzanos, C.P.

Optimization of Material Distributions in Fast Breeder Reactors, Aug. 1971.

Kang, C.S.

Use of Gamma Spectroscopy for Neutronic Analysis of LMFBR Blankets, Nov. 1971.

Leung, T.C.

Neutronics of an LMFBR Blanket Mockup, Jan. 1972

Ortiz, N.R.

Instrumental Methods for Neutron Spectroscopy in the MIT Blanket Test Facility, May 1972.

A.1 Doctoral and Engineer's Theses (continued)

Brewer, S.T.

The Economics of Fuel Depletion in Fast Breeder
Reactor Blankets, Oct. 1972

Gregory, M.V.

Heterogeneous Effects in Fast Breeded Reactors, Dec. 1972.

Wood, P.J.

Assessment of Thorium Blankets for Fast Breeder Reactors,
July 1973.

Ducat, G.A.

The Parfait Blanket Concept for Fast Breeder Reactors,
Jan. 1974.

Brown, G.J.

Evaluation of High-Performance LMFBR Blanket
Configurations, May 1974.

Scheinert, P.A.

Gamma Heating Measurements in Fast Breeder Reactor
Blankets, (Engineer's Thesis), est. Aug. 1974.

A.2 S.M. and B.S. Theses

Ho, S.L.

Measurement of Fast and Epithermal Neutron Spectra
Using Foil Activation Techniques

S.M. Thesis, MIT Nucl. Eng. Dept., Jan. 1970

Mertens, P.G.

An Evaluation of a Subcritical Null-Reactivity Method
for Fast Reactor Applications

S.M. Thesis, MIT Nucl. Eng. Dept., May 1970

Westlake, W.J.

Heterogeneous Effects in LMFBR Blanket Fuel Elements

S.M. Thesis, MIT Nucl. Eng. Dept., June 1970

A.2 S.M. and B.S. Theses (continued)

Shupe, D.A.

The Feasibility of Inferring the Incident Neutron Spectrum from Prompt Capture Gamma-Ray Spectra

S.M. Thesis, MIT Physics Dept., Aug. 1970

Pant, A.

Feasibility Study of a Converter Assembly for Fusion Blankets Experiments

S.M. Thesis, MIT Nucl. Eng. Dept., Jan. 1971

Passman, N.A.

An Improved Foil Activation Method for Determination of Fast Neutron Spectra

S.M. Thesis, MIT Nucl. Eng. Dept., Jan. 1971

Forsberg, C.W.

Determination of Neutron Spectra by Prompt Gamma-Ray Spectrometry

M.S. Thesis, MIT Nucl. Eng. Dept., June 1971

Brown, G.J.

A Study of High-Albedo Reflectors for LMFBRs

S.M. Thesis, MIT Nucl. Eng. Dept., March 1972

Thompson, A.M.

Activation Profiles in Reactor Fuel Elements

B.S. Thesis, MIT Physics Dept., June 1972

Lal, D.

Determination of the Neutron Spectrum in the MITR Transistor Irradiation Facility

B.S. Thesis, MIT Chem. Eng. Dept., June 1972

Ho, S.Y-N.

Selection of Foil Materials for LMFBR Neutron Spectrometry

S.M. Thesis, MIT Nucl. Eng. Dept., May 1973

A.2 S.M. and B.S. Theses (continued)

Choong, T.P.

Fast Neutron Spectrometry in an LMFBR Blanket Reflector

S.M. Thesis, MIT Nucl. Eng. Dept., Aug. 1973

Kennerley, R.J.

Proton-Recoil Neutron Spectrometry in a Fast Reactor
Blanket

S.M. Thesis, MIT Nucl. Eng. Dept., Aug. 1973

Chan, J.K.

A Foil Method for Neutron Spectrometry in Fast Reactors

S.M. Thesis, MIT Nucl. Eng. Dept. , Jan. 1974

Yeung, M.K.

A Stacked-Foil Method for Epithermal Neutron Spectrometry

S.M. Thesis, MIT Nucl. Eng. Dept., Dec. 1974 (est.)

Masterson, R.E.

The Application of Perturbation Theory and Variational
Principles to Fast Reactor Fuel Management

S.M. Thesis, MIT Nucl. Eng. Dept., Sept. 1974 (est.)

A.3 Other PublicationsI.A. Forbes, M.J. Driscoll, T.J. Thompson, I. Kaplan and
D. D. LanningDesign, Construction and Evaluation of a Facility
for the Simulation of Fast Reactor Blankets

MIT-4105-2, MITNE-110, Feb. 1970

M.K. Sheaffer, M.J. Driscoll, and I Kaplan

A One-Group Method for Fast Reactor Calculations

MIT-4105-1, MITNE-108, Sept. 1970

A.3 Other Publications (continued)

I.A. Forbes, M.J. Driscoll, D.D. Lanning, I. Kaplan and
N.C. Rasmussen

LMFBR Blanket Physics Project Progress Report No.1

MIT-4105-3, MITNE-116, June 30, 1970

I.A. Forbes, M.J. Driscoll, T.J. Thompson, I. Kaplan and
D. D. Lanning

Design, Construction and Evaluation of an LMFBR
Blanket Test Facility

Trans. Am. Nucl. Soc., Vol.13, No.1, June 1970

S.T. Brewer, M.J. Driscoll and E.A. Mason

FBR Blanket Depletion Studies - Effect of Number of
Energy Groups

Trans. Am. Nucl. Soc., Vol. 13, No. 2, Nov. 1970

M.K. Sheaffer, M.J. Driscoll and I. Kaplan

A Simple One-Group Method for Fast Reactor Calculations

Trans. Am. Nucl. Soc., Vol. 14, No. 1, June 1971

T.C. Leung, M.J. Driscoll, I. Kaplan and D.D.Lanning

Measurements of Material Activation and Neutron
Spectra in an LMFBR Blanket Mockup

Trans. Am. Nucl. Soc., Vol. 14, No. 1, June 1971

S.T. Brewer, E.A. Mason and M.J. Driscoll

On the Economic Potential of FBR Blankets

Trans. Am. Nucl. Soc., Vol. 14, No. 1, June 1971

I.A. Forbes, M.J. Driscoll, N.C. Rasmussen, D.D. Lanning
and I. Kaplan

LMFBR Blanket Physics Project Progress Report No. 2

COO-3060-5, MITNE-131, June 1971

C.P. Tzanos, E.P. Gyftopoulos and M.J. Driscoll

Optimization of Material Distributions in Fast
Breeder Reactions

MIT-4105-6, MITNE-128, August 1971

A.3 Other Publications (continued)

T.C. Leung and M.J. Driscoll

A Simple Foil Method for LMFBR Spectrum Determination

Trans. Am. Nucl. Soc., Vol. 14, No. 2, Oct. 1971

C.S. Kang, N.C. Rasmussen and M.J. Driscoll

Use of Gamma Spectroscopy for Neutronic Analysis
of LMFBR Blankets

COO-3060-2, MITNE-130, Nov. 1971

T.C. Leung, M.J. Driscoll, I. Kaplan and D.D. Lanning

Neutronics of an LMFBR Blanket Mockup

COO-3060-1, MITNE-127, Jan. 1972

N.R. Ortiz, I.C. Rickard, M.J. Driscoll and N.C. Rasmussen

Instrumental Methods for Neutron Spectroscopy in the
MIT Blanket Test Facility

COO-3060-3, MITNE-129, May 1972

V.C. Rogers, I.A. Forbes and M.J. Driscoll

Heterogeneity Effects in the MIT-BTF Blanket No. 2

Trans. Am. Nucl. Soc., Vol. 15, No.1, June 1972

S.T. Brewer, E.A. Mason and M.J. Driscoll

The Economics of Fuel Depletion in Fast Breeder
Reactor Blankets

COO-3060-4, MITNE-123, Nov. 1972

M.K. Sheaffer, M.J. Driscoll and I. Kaplan

A One-Group Method for Fast Reactor Calculations

Nucl. Sci. Eng., Vol. 48, p. 459 (1972)

C.P. Tzanos, E.P. Gyftopoulos and M.J. Driscoll

Optimization of Material Distributions in Fast Reactor
Cores

Nucl. Sci. Eng., Vol. 52, p. 84 (1973)

A.3 Other Publications (continued)

M.J. Driscoll, D.D. Lanning and I. Kaplan

LMFBR Blanket Physics Project Progress Report No. 3

COO-3060-6, MITNE-143, June 1972

M.V. Gregory, M.J. Driscoll and D.D. Lanning

Heterogeneous Effects in Fast Breeder Reactors

COO-2250-1, MITNE-142, Jan. 1973

P.J. Wood and M.J. Driscoll

Assessment of Thorium Blankets for Fast Breeder Reactors

COO-2250-2, MITNE-148, July 1973

G.A. Ducat, M.J. Driscoll and N.E. Todreas

The Parfait Blanket Concept for Fast Breeder Reactors

Trans. Am. Nucl. Soc., Vol. 16, No. 1, June 1973

G.A. Ducat, M.J. Driscoll and N.E. Todreas

The Parfait Blanket Concept for Fast Breeder Reactors

COO-2250-5, MITNE-157, Jan. 1974

G.J. Brown and M.J. Driscoll

Evaluation of High-Performance LMFBR Blanket Configurations

COO-2250-4, MITNE-150, May 1974

P.A. Scheinert and M.J. Driscoll

Gamma Heating Measurements in Fast Breeder Reactor Blankets

COO-2250-10, MITNE-164, (est. Aug. 1974)

A.3 Other Publications (continued)

P.J. Wood and M.J. Driscoll

Economic Characterization of Breeder Reactor
Blanket Performance

Trans. Am. Nucl. Soc., Vol. 17, Nov. 1973

P.J. Wood and M.J. Driscoll

The Economics of Thorium Blankets for Fast Breeder
Reactors

Trans. Am. Nucl. Soc., Vol. 17, Nov. 1973

M.V. Gregory, M.J. Driscoll, D.D. Lanning

Heterogeneous Effects on Reactivity of Fast Breeder
Reactors

Trans. Am. Nucl. Soc., Vol. 17, Nov. 1973