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NEUTRONICS OF AN LMFBR BLANKET MOCK-UP

by

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

January, 1972

Massachusetts Institute of Technology
Department of Nuclear Engineering
Cambridge, Massachusetts 02139
AEC Research and Development Report
UC-34 Physics

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MASSACHUSETTS INSTITUTE OF TECHNOLOGY
DEPARTMENT OF NUCLEAR ENGINEERING

Cambridge, Massachusetts

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ABSTRACT

Experimental measurements using foil activation techniques were performed on an LMFBR blanket mock-up, made up of subassemblies of 0.25-inch-diameter uranium metal fuel rods, clad in low-carbon steel, and surrounded by anhydrous sodium chromate (Na_2CrO_4) powder. This work was carried out in the MIT Blanket Test Facility which employs a converter assembly to transform the highly thermalized neutrons from the reactor's thermal column into a spectrum typical of LMFBR core leakage neutrons. Vertical and horizontal traverses were made in the blanket mock-up to confirm that an energy-independent buckling characterized the transverse leakage.

Axial reaction rate traverses of numerous materials including $\text{U}^{238}(\text{n}, \gamma)$, $\text{U}^{238}(\text{n}, \text{f})$, $\text{U}^{235}(\text{n}, \text{f})$ and $\text{Pu}^{239}(\text{n}, \text{f})$ were made throughout the blanket and reflector regions. Experimental results were compared with 26-group, S_8 calculations. In general, the agreement was good except in the iron reflector. The degree to which the U^{238} capture cross section was self-shielded was found to be the factor having the greatest affect on the calculations.

Multiple-foil packets containing resonance and threshold absorbers were irradiated at various locations within the blanket and neutron spectra unfolded from measured activities using a method based on reactor slowing-down theory. The unfolded blanket spectra were found to be harder than calculated using shielded U^{238} cross sections, but softer than calculated using unshielded U^{238} cross sections.

ACKNOWLEDGMENTS

The success of the MIT Fast Reactor Blanket Research Project is due to the support of the U.S. Atomic Energy Commission and to the contribution of a number of individuals. The work described in this report has been performed primarily by the principal author, T. C. Leung, who has submitted substantially the same report in partial fulfillment of the requirements for the Sc. D. degree at MIT.

The Staffs of the MIT Reactor, Machine Shop, Electronic Shop and Radiation Protection Office have provided much assistance and advice during the experimental portion of this work. Messrs. Albert Supple and George Sullivan have been of invaluable assistance in the setup of experiments and the operation and maintenance of the Blanket Test Facility. Particular thanks are due to N. Ortiz for his advice in regard to the instrumentation.

Special thanks are also due to Dr. I. Forbes (now at Lowell Institute of Technology) who has given much assistance in the calculational aspects of this work. All computer calculations were done on the IBM-360 at the MIT Information Processing Service Center.

TABLE OF CONTENTS

	<u>Page</u>
Chapter I. Introduction	12
1.1 Foreword	12
1.1.1 Motivation for Blanket Research	12
1.1.2 Current Status of Experimental LMFBR Blanket Research	13
1.1.3 History of the MIT Effort	14
1.2 Objective of Present Work	14
1.3 Organization of This Report	16
Chapter II. Experimental Facility	17
2.1 The Blanket Test Facility (BTF)	17
2.2 The BTF Blanket Mock-Up No. 2	23
2.2.1 General Description	23
2.2.2 Subassembly Description	26
2.2.3 Blanket Composition	31
2.3 Preliminary Calculations	36
2.3.1 Neutron Spectrum Calculations for Blanket Mock-Up No. 2	36
2.3.2 Comparison with a Reference LMFBR	38
2.3.3 Improved Multigroup Calculations	42
2.4 Sensitivity Studies	45
2.4.1 S_2 , S_4 vs. S_8 Calculations	45
2.4.2 The Effect of Hydrogen in the Sodium Chromate	47
2.4.3 Variation in Chromate Density	47
2.4.4 Effect of Using 1% U^{235} vs. Pu^{239} Fuel	50
2.4.5 Effects of Fuel Enrichment in Blanket No. 2	52
2.4.6 General Comments on Sensitivity Studies	56
Chapter III. Buckling Measurements	57
3.1 Introduction	57
3.2 Experimental Procedure	59
3.2.1 Foil Irradiation	59
3.2.2 Counting Equipment	62
3.2.3 Foil-Counting Techniques	62
3.2.4 Corrected Gamma Activities	66

	<u>Page</u>
3.3 Results of Measurements	68
3.3.1 Gold, Indium and Molybdenum Vertical Activation Traverses in Blanket No. 2	68
3.3.2 Gold, Indium and Molybdenum Horizontal Activation Traverses in Blanket No. 2	71
3.3.3 Buckling Results	75
3.4 Summary	78
Chapter IV. Reaction Rate Measurements	80
4.1 Introduction	80
4.2 Experimental Procedure	81
4.2.1 Description of Foil Materials	81
4.2.2 Foil Irradiation	81
4.2.3 Foil Counting Techniques	85
4.2.3.1 Capture Reaction Rate Measurements	86
4.2.3.2 Threshold Reaction Rate Measurements	87
4.2.3.3 Fission Reaction Rate Measurements	87
4.2.4 Corrected Activities	88
4.2.5 Sources of Errors in Foil Activation	92
4.3 Experimental Results	95
4.4 Effect of Self-Shielded U^{238} Cross Sections on Reaction Rate Calculations	103
4.5 The Influence of the Weighting Spectrum	108
4.6 Neutron Balance	111
4.6.1 Introduction	111
4.6.2 Theoretical Calculations	111
4.6.3 Experimental Verification	113
4.6.4 Results of Blanket Neutron Balance	115
4.7 Comparison of BTF Blanket No. 2 Spectral Indices to ZPPR	118
4.8 Summary	123
Chapter V. Neutron Spectrum Measurements	125
5.1 Introduction	125
5.2 The Unfolding Method	127

	<u>Page</u>
5.2.1 Description of the Unfolding Process	127
5.2.2 Testing the Unfolding Method – Computer Experiments	129
5.2.3 Flux Sensitivity to Input Activities	133
5.3 Foil Activation	134
5.3.1 Choice of Detector Foils	134
5.3.2 Foil Irradiation and Counting Procedure	134
5.3.3 Corrected Gamma Activities	136
5.4 Results of Spectrum Measurements in BTF Blanket Mock-Up No. 2	138
5.5 Summary	145
Chapter VI. Conclusions and Recommendations	148
6.1 Summary	148
6.2 Future Work	151
6.3 Conclusion	154
Appendix A. Additional Data on Blanket Mock-Up No. 2	155
Appendix B. Listing of Foil Activities for the Buckling Measurements	158
Appendix C. Correction Schemes for Elastic Downscattering Cross Sections	163
Appendix D. The DRISTAN Code	168
Appendix E. Error Analysis	176
Appendix F. References	183

LIST OF FIGURES

	<u>Page</u>
2.1 View of MIT Research Reactor Showing the Hohlraum and Blanket Test Facility Irradiation Room	19
2.2 Schematic Cross Section View of Hohlraum and Blanket Test Facility	20
2.3 Schematic Plan View of the Hohlraum and Blanket Test Facility	21
2.4 Schematic View of Blanket Assembly No. 2	24
2.5 Plan View of Blanket Assembly Showing the Traversing Tube Positions	25
2.6 Subassembly Grid Plate for Blanket Mock-Up No. 2	27
2.7 Blanket No. 2 Unit Cell	28
2.8 Schematic Cross Section View of Blanket No. 2 Subassembly	30
2.9 Comparison of the 16-Group Blanket Flux Distributions with ZPPR Core vs. BTF Converter Driving the Blanket	40
2.10 Multigroup Spectra at 24.75 cm. into BTF Blanket Mock-Up No. 2 and Reference LMFBR Blanket	41
2.11 Comparison of BTF Blanket No. 2 Spectra at $Z = 24.75$ cm for 16- and 26-Group ANISN S_8 Calculations	44
2.12 Percentage Deviations in Blanket No. 2 Reaction Rate Calculations for S_2 , S_4 vs. S_8 Options in ANISN	46
2.13 Percentage Error in Reaction Rates in Blanket No. 2 as a Function of the Sodium Chromate Water Content	48
2.14 Percentage Changes in Blanket No. 2 Reaction Rates as a Function of the Sodium Chromate Loadings	49
2.15 Percentage Change in Blanket No. 2 Reaction Rate Distribution When 1% Pu^{239} Fuel is Substituted for 1% U^{235} Fuel	53
2.16 Percent Change in Both Unshielded and Shielded U^{238} Activation as the Enrichment Changes from 1.07% to 0.19%.	55

	<u>Page</u>
3.1 Schematic View of the Converter and Blanket Assembly Showing the Foil Irradiation Positions	61
3.2 Well-Type Scintillation Counting System	63
3.3 Calibration of the Counting System	64
3.4 Gold Vertical Activation Traverses in Blanket No. 2	69
3.5 Vertical Activation Traverses in Blanket No. 2	70
3.6 Vertical Activation Ratios in Blanket No. 2	72
3.7 Gold Horizontal Activation Traverses in Blanket No. 2	73
3.8 Horizontal Activation Traverses in Blanket No. 2	74
3.9 Horizontal Activation Ratios in Blanket No. 2	76
4.1 Typical Foil Holder Rods	84
4.2 Sensitivity of Gamma Activity to Foil Thickness	93
4.3 Axial Capture Rate Distributions	96
4.4 Additional Axial Capture Rate Distributions	98
4.5 Axial Threshold Reaction Rate Distribution, $\text{In}^{115}(\text{n}, \text{n}')$	99
4.6 Axial Threshold Reaction Rate Distribution, $\text{U}^{238}(\text{n}, \text{f})$	100
4.7 Axial Fission Rate Distributions	102
4.8 Effect of U^{238} Self-Shielded Cross Sections on Reaction Rate Calculations	105
4.9 Effect of Using U^{238} Self-Shielded Cross Sections on Pu^{239} Fission Rate Calculations	106
4.10 Comparison of Shielded (In-Rod) and Unshielded (Ex-Rod) U^{238} Foil Traverses	107
4.11 Effect of Different Weighting Spectra on Reaction Rate Calculations	110

	<u>Page</u>
5.1 Comparison of ϕ_i^{ANISN} and ϕ_i^{DRIS} Spectra in the Blanket Region, R = 124.75 cm	131
5.2 Comparison of ϕ_i^{ANISN} and ϕ_i^{DRIS} Spectra in the Reflector, R = 155.6 cm	132
5.3 Neutron Spectra in Blanket No. 2	141
5.4 Blanket No. 2 Neutron Spectra Unfolded by DRISTAN at Z = 7.5 cm	143
5.5 Intercomparison of Neutron Spectra at 7.5 cm into Blanket No. 2	144
5.6 Effect on the Neutron Spectra Calculations in Blanket No. 2, at Z = 7.5 cm, Using Shielded and Unshielded U ²³⁸ Cross Sections	146
A.1 Blanket Subassembly Numbering	156

LIST OF TABLES

	<u>Page</u>
2.1 Subassembly Component Weights	29
2.2 Homogenized Atom Densities in BTF Blanket No. 2	33
2.3a. Comparison of Blanket Atom Densities	34
2.3b. Comparison of BTF Blanket and ZPPR Blanket Atom Densities	35
2.4 Computer Model of Blanket No. 2	37
2.5 Reference LMFBR-ZPPR Core Coupled with Blanket Mock-Up No. 2	39
2.6 Group Structures of Multigroup Cross-Section Sets	43
2.7 Comparison of Blanket Spectra at $Z = 24.4$ cm, Using 1% Pu^{239} vs. U^{235} Fuel	51
2.8 Comparison of Blanket Spectra at $Z = 24.4$ cm for Two Different Enrichment Fuels	54
3.1 26-Group Activation Cross Sections for Gold, Indium and Molybdenum	58
3.2 Description of Foil Materials	60
3.3 Typical Data for Foil Counting in Buckling Measurements	67
4.1 Physical Characteristics of Foil Materials	82
4.2 Nuclear Properties of Foil Materials	83
4.3 Typical Data Pertinent to Foil Counting	89
4.4 Results of Axial (Z) Reaction Rate Traverses	90
4.5 Additional Axial (Z) Reaction Rate Traverses	91
4.6 Comparison of U^{238} Unshielded and Shielded Cross Sections	104
4.7 Listing of Parameters Used to Obtain $R_{\alpha\beta}/R_{\text{UC}}$	116
4.8 Neutron Balance for BTF Blanket Mock-Up No. 2	117

	<u>Page</u>
4.9 Comparison of Fission Ratios Between BTF Blanket Mock-Up No. 2 and Fast Critical Assemblies	120
4.10 Comparison of Fission Ratios Between ZPPR and BTF Blankets	121
5.1 The Variation of Flux Characterization Parameters with Position	133
5.2 Properties of Foil Materials Used in Blanket Spectrum Measurements	135
5.3 Measured Spectrum-Packet Foil Activities in BTF Blanket Mock-Up No. 2	137
5.4 Corrected Spectrum-Packet Foil Activities in BTF Blanket Mock-Up No. 2	139
5.5 BTF Blanket Mock-Up No. 2 Spectra Unfolded by DRISTAN	140
A.1 Subassembly Loadings of Sodium Chromate in Blanket Mock-Up No. 2	155
A.2 Calculation of Atom Densities in Blanket Mock-Up No. 2	157
B.1 Data for Gold Vertical Activation Traverses in Blanket No. 2	158
B.2 Data for Indium Vertical Activation Traverses in Blanket No. 2	159
B.3 Data for Molybdenum Activity Traverses in Blanket No. 2	160
B.4 Data for Horizontal Activation Traverses in Blanket No. 2	161
B.5 Data for Horizontal Activation Traverses in Blanket No. 2	162
C.1 α_i Values for $\phi(u) \propto E$ Weighting Spectrum	166
C.2 α_i Values for Mid-Blanket Weighting Spectrum	167
E.1 The Standard Deviation of A Repeated $Au^{197}(n,\gamma)$ Traverse	182

Chapter I

INTRODUCTION

1.1 FOREWORD

1.1.1 Motivation for Blanket Research

The optimization of the blanket region plays a very important role in the development of an economical LMFBR since as much as one-third to one-half of the fissile breeding and three-quarters of the fertile inventory occur in the blanket region. Thus, only the proper choice of blanket composition and configuration can lead to high breeding performance and minimum fuel cycle and power cost of future LMFBRs.

Unfortunately, there are substantial difficulties involved in purely theoretical studies of neutron propagation in blanket media. Multigroup calculations for blanket optimization, based mainly on methods and data developed for core studies, are likely to be inadequate. Prediction of reactor physics parameters is considerably more complex and less reliable in the blanket region than it is in the core, mainly because of the severe spectral degradation occurring in the blanket. It has therefore become increasingly evident that a systematic experimental study of the blanket region is highly desirable.

"Clean" or "benchmark" blanket experiments, with realistic blanket compositions, can serve as a useful tool for the evaluation of important neutron cross-section data. The wide range of spectra encountered in the blanket and its reflector and the abrupt spatial attenuation provide a sensitive test of both the capture and scattering cross

sections used in multigroup cross-section sets. In addition to providing a test of calculation techniques and cross-section data, blanket experiments can provide experimental data and empirical design guidelines relevant to the optimization of fast sodium cooled reactor blankets.

1.1.2 Current Status of Experimental LMFBR Blanket Research

Until recently, experimental work on fast reactor blanket design has not received much attention. Most of the fast reactor physics experiments performed on critical assemblies, such as ZPR III, ZPR IV, VERA, ZEBRA and SNEAK (A1, W1) have been core-oriented. In many of these assemblies no blanket was used, or a crude approximation to realistic blanket compositions was made, and reflector material at the outer periphery of the blanket was seldom included.

Since December 1969, a series of critical experiments performed on the ZPPR and ZPR VI assemblies, under the Demonstration Reactor Benchmark Program (D1), has been under way in order to support the initial phases of LMFBR demonstration plant design. Although strong emphasis is placed on operational and safety-related measurements, in the core, studies on blanket properties and configurations have also been included. The axial and radial blanket configuration and the composition of the ZPPR assembly are closely representative of those planned for power reactor design. In addition, a sodium-stainless steel reflector is provided external to both axial blankets to simulate, in part, the reflective effects of the corresponding region in the power reactor. Similarly, iron blocks are used in two rows to simulate a radial reflector. The build-up of fissile material in the blanket is also simulated and the resulting effects upon flux distributions, etc. are investigated. In order

to relate the major parameters measured in the basic plate-fueled system to those that exist in a power reactor configuration, experiments on pin-geometry are also performed.

1.1.3 History of the MIT Effort

The research program at MIT is focused on the blanket as a semi-independent entity, i.e., without having to tie the blanket mock-ups to a fast reactor core. The Blanket Test Facility, constructed for this purpose, was designed to simulate the neutron energy spectra found at the periphery of fast reactor cores in order to drive mock-ups of fast reactor blankets. Construction of the MIT Blanket Test Facility was begun in July 1968, funded by an MIT research grant. In July 1969, support under USAEC Contract AT(30-1) 4105 commenced.

The MIT Fast Reactor Blanket Research Project is directed toward partial fulfillment of goals outlined in the LMFBR Program Plan - Physics, WASH-1109 (W2), primary under Task 9.2.2, Item 9. Specifically, a series of blanket experiments is to be carried out which are both "clean" enough to serve as "benchmark" cases for blanket physics calculations and at the same time realistic enough to provide empirical design guidance. Experimental measurements on blankets, namely, the measurement of reaction rates and neutron energy spectra in the blanket and the determination of the effects of blanket and outer reflector composition and thickness, have been assigned Priority 1 by the AEC.

1.2 OBJECTIVE OF PRESENT WORK

It was the objective of the present work to provide experimental data and to test theoretical methods on BTF Blanket Mock-Up No. 2 (described in Chap. II) which is the first of a series of realistic mock-ups of typical

LMFBR blanket configurations and compositions. It was also intended that standard techniques (D2), developed on this blanket mock-up, could be used for future experimental research on subsequent blanket mock-ups.

Experimental measurements using foil activation techniques were made in the blanket mock-up in the following three major areas:

a) Buckling Measurements. Vertical and horizontal traverses were made with gold, molybdenum and indium foils to determine whether an energy-independent buckling was achieved to characterize the transverse leakage.

b) Capture and Fission Rate Measurements. Foil activation measurements, including U^{238} capture and fission, Pu^{239} fission and Fe, Na, Cr capture, were made through the axial (simulating radial) direction of the blanket, and in some cases deep into the reflector region. The materials in question were all typical of those used as fuel, coolant or structure in real LMFBR blankets. Other reaction rates, such as $Au^{197}(n, \gamma)$, $Mo^{98}(n, \gamma)$ and $In^{115}(n, n')$, were also measured in the blanket.

c) Neutron Spectral Measurements. Multiple-foil packets containing resonance and threshold foils were irradiated at various locations within the blanket, and neutron spectra were unfolded from the measured activities.

In addition to the data analysis necessary to express the experimental results in their most useful form, the following theoretical work was carried out as a support effort to interpret the significance of the data to LMFBR program objectives:

a) Slab geometry calculations were made using a multigroup computer code, ANISN (E1, E2, B1), to obtain neutron spectra and activation traverses for comparison with the experimental data.

b) Calculations made with various modified multigroup cross-section sets – namely, (i) self-shielded vs. unshielded and (ii) results using different spectrum-weighted sets – were compared with experimental results.

1.3 ORGANIZATION OF THIS REPORT

A brief description of the BTF and initial evaluation tests are reviewed in the first section of Chapter II. The construction of Blanket Mock-Up No. 2 and detailed matching of the homogenized composition with that of a realistic LMFBR blanket are then discussed.

The application of foil activation techniques for the buckling measurements to confirm the existence of a characteristic transverse buckling will be discussed in Chapter III. Results of capture and fission reaction rate measurements are reported in Chapter IV. Comparison of multigroup calculations with experimental data is included. The neutron balance for the entire blanket is also discussed in this chapter.

The technique developed for unfolding blanket spectra from foil activities is introduced in Chapter V. The comparison between the blanket spectrum measurements and multigroup calculations is then discussed.

The final chapter discusses the experimental data obtained from Blanket No. 2 and comments on the discrepancies between theoretical and experimental results. Suggestions pertinent to future blanket work are then presented.

Chapter II

EXPERIMENTAL FACILITY

The design features and capabilities of the Blanket Test Facility are central to the present blanket research. Brief descriptions of the Blanket Test Facility and Blanket Mock-Up No. 1, which was used for evaluating the performance of the facility, are presented in Section 2.1 of this chapter. A detailed description of the BTF and of the initial evaluation tests can be found in reference (F1). The construction of Blanket Mock-Up No. 2, designed to simulate the nuclide compositions of a realistic LMFBR blanket, is described in Section 2.2. The detailed matching of the homogenized composition of this blanket mock-up with that of a realistic blanket is then discussed. Finally, reference numerical calculations of the assembly characteristics predicted by application of multigroup codes are described.

2.1 THE BLANKET TEST FACILITY (BTF)

In the design of the Blanket Test Facility, it was desirable that the neutron flux in the slab geometry of the blanket assembly should have the same spatial shape in the Z-direction as the neutron flux in the cylindrical geometry of an LMFBR radial blanket in the R-direction, in order to increase the practical value of the experimental data. Thus, the transverse dimensions of both converter and blanket assembly were selected to match leakage for a hypothetical LMFBR so that axial traverses in the blanket mock-up would correspond to radial traverses in the LMFBR. The details of the slab simulation of cylindrical geometry are discussed in reference (F1).

The Blanket Test Facility employs a converter assembly, made up of slightly enriched uranium fuel rods arranged in a tightly packed slab array, designed to transform the highly thermalized neutrons from the MIT reactor thermal column into a spectrum typical of LMFBR core-leakage neutrons. Figure 2.1 shows the location of the Blanket Test Facility, at the rear of the graphite-lined hohlraum of the MIT Reactor.

The cross-section and plan views of the facility are shown in Figs. 2.2 and 2.3, respectively. The BTF irradiation region is defined by a five-sided aluminum box, 6 ft by 6 ft by 6 ft. The aluminum liner box is seam-welded to maintain the sealed nature of the hohlraum and prevent the leakage of A^{41} from the hohlraum into the irradiation region and the reactor building. Boral sheet is attached to the top, sides and floor of the box to reduce back-scattering of thermalized neutrons from the surrounding concrete shielding. The shielding for the irradiation region is provided by four heavy concrete shield blocks, two stationary and two portable. A set of rails, extending from the front of the irradiation region out to the reactor building containment wall, permits the insertion of the cart-mounted experimental blanket mock-up assemblies into the irradiation region.

For the operation of the Blanket Test Facility the two portable shielding blocks, weighing approximately 15 tons each, are removed with the reactor building's overhead crane to provide access to the irradiation region. The converter assembly is rolled to the front of the irradiation region, the blanket mock-up assembly is installed directly behind it, and the concrete shield doors are replaced. Opening the lead shutters and steel doors allows the neutrons from the MIT Reactor to enter the hohlraum. The thermalized neutrons in the hohlraum are then

FIG. 2.1 VIEW OF M.I.T. RESEARCH REACTOR SHOWING THE HOHLRAUM AND BLANKET FACILITY IRRADIATION ROOM.

FIG. 2.2 SCHEMATIC CROSS SECTION VIEW OF HOHLRAUM AND BLANKET TEST FACILITY

FIG. 2.3 SCHEMATIC PLAN VIEW OF THE HOHLRAUM AND BLANKET TEST FACILITY

transformed by the converter into a spectrum typical of LMFBR core-leakage neutrons and fed into the blanket assembly.

Before initiating studies on simulated fast reactor blanket mock-ups, the Blanket Test Facility was tested by driving a low cost, easily constructed iron-borax assembly, Blanket Mock-Up No. 1, 55.4 in. by 60.8 in. by 30.5 in. This assembly was a parallelepiped, consisting of 63 subassemblies. The subassemblies were loaded with a mixture of low carbon steel punchings and anhydrous borax powder to provide a composition of approximately 50% iron and 50% $\text{Na}_2\text{B}_4\text{O}_7$. This mixture contains oxygen, sodium and iron (typical blanket constituents) and simulates the neutronic properties of uranium with iron and boron so that the neutron spectra and overall spatial flux shapes generated are quite similar to those in realistic blanket assemblies. Apart from dose measurements to check the biological shielding, accessibility following irradiation, and the residual activation of the converter and blanket assemblies, other experiments were also performed to test for correct converter spectrum and source shape generation, for correct transverse equilibrium and leakage effects in the blanket, and finally for streaming and back-scattering effects from the surrounding concrete shielding. The details of the evaluation tests are provided in reference (F1).

In summary, the BTF at MIT makes possible the experimental study of fast reactor blanket mock-ups without the need for a fast reactor core. A second advantage is that approximately a factor of ten less total blanket material is required in the BTF than for a fast critical facility. In addition, the BTF is safe and convenient to operate since no critical system is involved. This facility, when operated at a converter power of 50 watts, will permit blanket experiments at an equivalent core power level of approximately 500 watts.

2.2 THE BTF BLANKET MOCK-UP NO. 2

The BTF Blanket Mock-Up No. 2 consists of uranium metal rods clad in carbon steel tubes, surrounded by anhydrous sodium chromate (Na_2CrO_4), and is designed such that the atom densities are similar to a typical LMFBR radial blanket which contains UO_2 rods in stainless steel cladding and cooled by liquid sodium.

2.2.1 General Description

Figure 2.4 shows BTF Blanket Mock-Up No. 2 on its cart. The support structure for the assembly consists of two pieces of 1 inch X 60 inches X 39 inches and one piece of 1 inch X 62-7/16 inches X 58-1/4 inches hot rolled, mild steel plates welded to make an H-frame. The front half of the H-frame contains three rows of subassemblies and the back half is filled with 18 (1 in. X 60 in. X 58-1/4 in.) hot rolled, mild steel plates acting as a neutron reflector. There are 29 full subassemblies (5-13/16 in. X 5-13/16 in. X 60 in.) and two half-size subassemblies (2-13/16 in. X 5-13/16 in. X 60 in.). Twenty-five of the central subassemblies are filled with steel-clad uranium rods and sodium chromate. The remaining peripheral subassemblies are filled with a mixture of iron and borax.

A total of 26 foil tubes are provided for the irradiation of various foils in the blanket in the axial and transverse directions (see Fig. 2.5). The foil tubes are mild steel tubing of 7/16-inch O.D., 0.028-inch wall thickness, and 58-inch length and are held in place by the top and bottom grid plates in each subassembly. A 2-1/8-inch-diameter hole, 3 inches below mid-plane, is also drilled through the 18-inch steel reflector for extraction of neutron and prompt gamma spectra from the blanket.

FIG. 2.4 SCHEMATIC VIEW OF BLANKET ASSEMBLY NO. 2

FIG.2.5 PLAN VIEW OF BLANKET ASSEMBLY SHOWING THE TRAVERSING TUBE POSITIONS

In addition, this hole also allows the insertion of a foil holder rod for foil activation measurements in the reflector region. When not in use for beam extraction, the hole is filled with a cylindrical steel plug.

2.2.2 Subassembly Description

Each subassembly box has a wall thickness of approximately $3/32$ inch and a seal-welded bottom closure plate. It contains 121 fuel rods arranged in an 11×11 square lattice whose pitch is 0.511 inch (Figs. 2.6 and 2.7). The uranium metal rods are 0.250 inch in diameter and 48 inches in length. Sixty of the rods have a U^{235} enrichment of 1.016%, and sixty-one have a U^{235} enrichment of 1.134%; the two enrichments are loaded in a checkerboard pattern within the subassembly box. A total of 3,025 fuel rods were fabricated at MIT by recladding the uranium metal rods in low carbon mild-steel tubing. The clad tubing dimensions are $5/16$ -inch O.D., 0.018-inch wall thickness, and 50-inch length. Each end of the tube is closed by a press-fitted steel plug, $1/2$ inch long by $9/32$ inch O.D.; and this arrangement leaves a one-inch free space in the tube to allow for dimensional variations and fuel expansion. The fuel rods are held in place by aluminum bottom and top grid plates, $1/4$ inch in thickness.

The inter-rod volume is filled with anhydrous sodium chromate powder, Na_2CrO_4 , (technical grade) which has been dried and ground. Since the presence of any water in the sodium chromate would tend to soften the neutron energy spectrum in the simulated blanket (see Section 2.4.1), it was decided to reduce the water content to ≤ 0.1 w/o by drying. This was done by baking the sodium chromate in 5-Kg. batches for eight hours in a domestic electric oven heated to $400^\circ F$. (The dehydration temperature of sodium chromate is about $300^\circ F$.) Since the

FIG. 2.6 SUBASSEMBLY GRID PLATE FOR BLANKET No.2

FIG. 2.7 BLANKET NO. 2 UNIT CELL

bake-dried sodium chromate was hard and crusty, it was impossible to load it directly into the subassemblies. The sodium chromate had to be reground into a fine powder, using a commercial coffee grinder, so that it could fill the subassembly box more easily and give a more uniform loading with less voids. Finally, after loading the dried and ground sodium chromate, each subassembly was sealed by a 0.035-inch steel top plate which was epoxied in place so that the subassembly would be airtight and watertight (Fig. 2.8).

Table 2.1 shows a breakdown of the subassembly weight (F2). The weights of U^{238} and U^{235} in each subassembly are 88.35 kg and 0.95 kg, respectively. The total of 121 clad tubes (each 50 inches in length) is 28.6 pounds. The average loading of sodium chromate in one subassembly is 31.11 kg. The loadings in different subassemblies vary from 30.51 kg to 31.80 kg, or only about $\pm 2\%$ maximum deviation from the mean. The subassembly loadings of sodium chromate are listed in Table A.1, Appendix A. The effects on the blanket calculations due to the variation in chromate loading are discussed later in Section 2.4.3.

TABLE 2.1

Subassembly Component Weights

Uranium metal	89.30 kg
Na_2CrO_4	31.11 kg
Cladding	13.00 kg
Subassembly box	26.55 kg
Grid plate support tubes	0.91 kg
Grid plates	0.36 kg
Total	161.23 kg

FIG. 2.8 SCHEMATIC CROSS SECTION VIEW OF BLANKET NO. 2 SUBASSEMBLY

2.2.3 Blanket Composition

The calculation of nuclide concentrations is based on homogenizing the various materials in the subassembly over a unit volume, 5.92 inches by 5.92 inches (the cross-sectional area of a subassembly) by 48.0 inches (the active height of uranium fuel). The uranium rods, cladding, sodium chromate and support tubes, except for the top and bottom grid plates, are included in this volume; hence, the nuclide densities thus calculated represent those in the central portion of the blanket. The calculation was made in the usual manner, using the relation:

$$N_i = \frac{w_i A}{V M_i} \quad (2.1)$$

where

N_i = atom density of the i-th material

w_i = weight of the i-th material in the defined unit volume

V = the defined unit volume

M_i = molecular weight of the i-th material

A = Avogadro's number

Table A.2 in Appendix A lists the values of N_i , w_i , and M_i for the Blanket No. 2 constituents. The amount of carbon in our low carbon steel cladding and subassembly sheath is about 0.15 w/o. Other impurities in the steel, such as manganese and nickel, were negligible. The water content in the sodium chromate was found to be 0.10 ± 0.02 w/o by the crucible drying method and confirmed to be ≤ 0.1 w/o by the prompt activation analysis method. The effect of hydrogen content on the blanket spectrum has been analyzed by I. Forbes and reported in the LMFBR Blanket Physics Project Progress Report No. 1 (F2). His results will be discussed further in Section 2.4.2.

The homogenized atom densities in Blanket No. 2 are given in Table 2.2, where they are compared with the atom densities in an "equivalent realistic blanket" composed of 37.0 v/o depleted UO_2 (at 90% of theoretical density), 20.7 v/o Type 316 stainless steel (71.2 w/o Fe, 20.0 w/o Cr and 8.8 w/o Ni), 32 v/o sodium and 10.3 v/o void. It is evident that Blanket No. 2 provides a realistic blanket composition in all respects, with the exception of the small hydrogen content.

The atom densities of Blanket No. 2 are compared with those of typical LMFBR blankets in Table 2.3a. In general, it has a composition representative of most typical projected 1000-MWE blanket designs (G1, G2, W3, B2, J1, A2).

In Table 2.3b, the atom densities of Blanket No. 2 are compared with those of the radial and axial blankets of the ZPPR assembly planned for the AEC's Demonstration Reactor Benchmark Program (D1). Although Blanket No. 2 has a radial blanket-like configuration, its sodium content is closer to that of the ZPPR axial blanket. The sum of chromium and iron densities in Blanket No. 2 is higher than the sum of chromium, iron and nickel atom densities in the ZPPR blankets. Unlike real blankets which contain depleted uranium fuel, Blanket No. 2 has a slightly higher enrichment in uranium (1% enriched). The sensitivity to use of 1% enriched fuel as opposed to depleted fuel (0.2% enriched) will be discussed in Section 2.4.5.

TABLE 2.2
Homogenized Atom Densities in B. T. F. Blanket No. 2

Nuclide	Blanket No. 2	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

* Composed of 37.0 v/o depleted UO₂ (at 90% of theoretical density), 20.7 v/o Type 316 stainless steel, 32.0 v/o sodium and 10.3 v/o void.

TABLE 2.3a. Comparison of Blanket Atom Densities
(Nuclei/cc $\times 10^{-24}$)

	B & W	JAERI #2	JAERI #1	CE (Radial)	CE (Axial)	GE (Radial)	GE (Axial)	AI (Inner Radial)	AI (Outer Radial)	AI (Axial)	BTF #2
U	.0091	.0124	.0099	.0127	.0093	.0099	.0050	.0124	.0130	.0064	.0082
Na	.0090	.0064	.0099	.0082	.0102	.0082	.0117	.0076	.0074	.0127	.0081
O	.0183	.0249	.0199	.0253	.0186	.0198	.0099	.0249	.0261	.0128	.0163
Fe	.0106	.0126	.0100	.0092	.0080	.0108	.0118	.0096	.0082	.0119	.0138
Cr	.0025	.0030	.0024	.0020	.0018	.0026	.0028	.0023	.0020	.0028	.0041
Ni	.0012	.0015	.0012	.0014	.0012	.0013	.0014	.0011	.0010	.0014	—
Cr+Ni	.0037	.0045	.0036	.0034	.0030	.0039	.0042	.0034	.0030	.0042	.0041
Fe + Cr+N _i	.0143	.0171	.0136	.0126	.0110	.0147	.0160	.0130	.0112	.0161	.0179

B & W = Babcock and Wilcox (B2)

JAERI = Japan Atomic Energy Research Institute (J1)

CE = Combustion Engineering

GE = General Electric (G1, G2)

AI = Atomics International (A2)

BTF = MIT Blanket Test Facility

TABLE 2.3b. Additional Comparison of Blanket Atom Densities
(10^{24} Nuclei/cc)

Isotope	B. T. F. Blanket Mock-Up No. 2	ZPPR Benchmark Demonstration Reactor (D1)	
		Radial Blanket	Axial Blanket
U ²³⁵	0.000088	0.000026	0.000016
U ²³⁸	0.008108	0.01099	0.00708
O	0.016293	0.01993	0.0141
Na	0.008128	0.00589	0.008899
Fe	0.013750	0.00778	0.009670
Cr	0.004064	0.00230	0.002290
Ni	—	0.00091	0.000913
C	0.000096	0.0009955	—
H	0.000073*	—	—

* Assuming 0.1 w/o H₂O in Na₂CrO₄ powder.

2.3 PRELIMINARY CALCULATIONS

As part of the analytic effort in support of the design of Blanket Mock-Up No. 2, multigroup calculations were carried out to confirm that all major variables were correctly accounted for. The neutron spectra calculated for Blanket Mock-Up No. 2 are presented in Section 2.3.1 of this chapter, and the comparison of the neutron spectra with those of a typical fast reactor system is discussed in Section 2.3.2. Sensitivity studies pertinent to the design are discussed in Section 2.4.

2.3.1 Neutron Spectrum Calculations for BTF Blanket Mock-Up No. 2

Multigroup calculations on the BTF Blanket Mock-Up No. 2 were performed with the one-dimensional transport theory code, ANISN (B1, E1, E2). All work herein was based on the S_8 option. Table 2.4 lists the converter and blanket data used as standard input for the program. At first, the 16-group modified Hansen-Roach cross-section set (H1) was used for the calculation. The inner boundary condition at the plane where the converter interfaces with the graphite-lined cavity or "hohlraum" of the MITR was simulated by a vacuum boundary condition and a thermal Maxwellian source in the numerical calculations. This thermal source flux (groups 15 and 16) diffuses through the graphite and produces a fission source in the first few centimeters of the fuel region. The graphite external moderator and the bulk of the fuel region then provide the correct slowing-down and absorption characteristics to yield the desired leakage spectrum at the converter-blanket interface, similar to that leaking from a typical LMFBR core. The aluminum cladding of the UO_2 fuel in the converter provides a rough approximation to the transport properties of sodium coolant; this stratagem was adopted from early critical assembly practice where half-density aluminum was successfully used as a substitute for sodium.

TABLE 2.4. Computer Model of Blanket No. 2

Converter Assembly				Blanket No. 2		
Active Fuel Height	48.0 inches			Effective Height	60.0 inches	
Active Fuel Width	60.0 inches			Effective Width	74.0 inches	
Graphite Region Thickness	8.000 inches			Blanket Thickness	17.76 inches	
First Fuel Region Thickness	3.150 inches			Reflector Thickness	18.00 inches	
Second Fuel Region Thickness	3.715 inches					
Boral Sheet Thickness	0.125 inch					
Atom Densities (nuclei/cc $\times 10^{24}$)						
Material	Graphite Region	First Fuel Region	Second Fuel Region	Boral Sheet	Blanket No. 2	Iron Reflector
U ²³⁵		0.000286	0.000158		0.000088	
U ²³⁸		0.013952	0.014080		0.008108	
B				0.026239		
C	0.083110			0.006560	0.000096	0.00059
Al		0.012221	0.012221	0.044263		
O		0.028477	0.028477		0.016293	
Na					0.008128	
Cr					0.004064	
Fe					0.013750	0.08457
H					0.000073*	

* 0.1 w/o H₂O in the sodium chromate.

2.3.2 Comparison with a Reference LMFBR

It was necessary to have a reference LMFBR design so that the predicted multigroup flux distributions in the axial direction of the simulated blanket assembly in slab geometry could be compared with those in the radial direction of a typical LMFBR blanket in cylindrical geometry. The reference LMFBR was chosen to be a ZPPR (D1) two-region core coupled with a radial blanket whose material composition is the same as our simulated fast reactor blanket (Table 2.2). Table 2.5 gives the dimensions and material compositions of the reference LMFBR. Dimensions and material compositions for this reference LMFBR are typical of recent 1000-MWe designs (G1, G2, W3, B2, J1, A2).

Figure 2.9 is a comparison of the 16-group flux distribution in the simulated blanket and the radial blanket of the reference LMFBR blanket normalized by setting $\int \phi(E) dE = 1$ at a distance of 24.75 cm into both blankets. Only the first nine groups are shown because neutrons having energies below 100 eV (groups 10 through 16) constitute only a negligibly small fraction of the total. The two flux distributions are in good agreement, especially in energy groups four through eight (0.9 MeV to 0.55 keV) in which over 90% of the total neutrons in the blanket lie. Such discrepancies as exist tend to be larger at the core-(converter) blanket interface and the blanket-reflector interface. The former discrepancy is attributed to material composition mismatch and the latter to the inability to perfectly simulate cylindrical geometry by using a parallelepiped.

Figure 2.10 shows the comparison of the two different blanket spectra at 24.75 cm into the blanket (at about the center of the blanket). The flux spectrum shapes for the two cases are in good agreement. The discrepancies become larger in both the low energy (eV region) and the

TABLE 2.5. Reference LMFBR - ZPPR Core Coupled with Blanket Mock-Up No. 2

DIMENSIONS

Core Zone 1	0.0 - 71.4 cm
Core Zone 2	71.4 - 101.0 cm
Blanket	101.0 - 146.1 cm
Reflector	146.1 - 191.8 cm
Effective Height of Core = 128.4 cm	

ATOM DENSITIES ($\times 10^{24}$ nuclei/cc)				
Isotope	Core Zone 1	Core Zone 2	Blanket	Reflector
Pu ²³⁹	0.00085295	0.0012794	—	—
Pu ²⁴¹	0.00001687	0.0000253	—	—
U ²³⁵	0.00001245	0.0000116	0.000090	—
Pu ²⁴⁰	0.00011640	0.0001746	—	—
U ²³⁸	0.005573	0.0051885	0.008100	—
O	0.013532	0.0122450	0.016200	—
Na	0.008899	0.008899	0.010700	—
Fe	0.012120	0.014520	0.014506	0.08457
Cr	0.002585	0.002732	0.001876	—
Ni	0.001048	0.001116	0.000915	—
Mo	0.0002195	0.000329	—	—
C	—	—	0.000091	0.00059

FIG. 2.9 COMPARISON OF THE 16 GROUP BLANKET FLUX DISTRIBUTIONS WITH ZPPR CORE VS BTF CONVERTER DRIVING THE BLANKET.

FIG. 2.10 MULTIGROUP SPECTRA AT 24.75 CM. INTO
BTF BLANKET MOCK-UP NO.2 AND REFERENCE
LMFBR BLANKET

high energy (MeV region) tails of the spectra. Such discrepancies are not important because the extremely high and low energy neutrons constitute only a small fraction of the total neutrons.

The converter assembly of the BTF was designed to transform the highly thermalized neutrons from the reactor thermal column into a spectrum typical of LMFBR core leakage neutrons. The above numerical calculations show that, except for minor discrepancies, the LMFBR Blanket Mock-Up No. 2 when driven by the converter of the BTF indeed produces internal neutron spectra similar to those of a fast reactor blanket driven by a fast reactor core.

2.3.3 Improved Multigroup Calculations

The original multigroup calculations for the design of the BTF and the Blanket Mock-Up No. 2 were performed with the modified 16-group Hansen-Roach cross-section set. Improved analysis of the simulated blanket was achieved by increasing the number of energy groups from 16 to 26. The 26-group ABBN, also called "Bondarenko" or "Russian," cross-section set (A3) was used for all later analysis instead of the 16-group Hansen-Roach LASL set. Since there are ten more energy groups in the Russian set, a much more detailed analysis of the blanket can be made. Table 2.6 lists the group structure of both the Hansen-Roach 16-group and the ABBN 26-group cross-section sets. Note that the latter has fully twice as many groups above 1 keV, which is the range of current interest. The ABBN set also has a softer built-in weighting spectrum (actually " $1/E$ ") than the LASL set.

Figure 2.11 shows the comparison of the BTF Blanket No. 2 spectra at 24.75 cm into the blanket for the 16- and 26-group S_8 ANISN calculations. Unshielded U^{238} cross sections have been used in both cases.

TABLE 2.6

Group Structures of Multigroup Cross-Section Sets

ABBN		HANSEN-ROACH	
Group No.	Lower Energy Boundary	Group No.	Lower Energy Boundary
1	6.5 MeV	1	3.0 MeV
2	4.0 "	2	1.4 "
3	2.5 "	3	0.9 "
4	1.4 "	4	0.4 "
5	0.8 "	5	0.1 "
6	0.4 "	6	17.0 keV
7	0.2 "	7	3.0 "
8	0.1 "	8	550.0 eV
9	46.6 keV	9	100.0 "
10	21.5 "	10	30.0 "
11	10.0 "	11	10.0 "
12	4.65 "	12	3.0 "
13	2.15 "	13	1.0 "
14	1.0 "	14	0.4 "
15	465.0 eV	15	0.1 "
16	215.0 "	16	Thermal
17	100.0 "		
18	46.5 "		
19	21.5 "		
20	10.0 "		
21	4.65 "		
22	2.15 "		
23	1.0 "		
24	0.465 "		
25	0.215 "		
26	Thermal

FIG.2.11 COMPARISON OF BTF BLANKET No.2 SPECTRA AT $Z = 24.75$ CM. FOR 16- AND 26- GROUP ANISH S_8 CALCULATIONS.

The normalization is such that $\int \phi(E) dE = 1$ at $Z = 24.75$ cm into the blanket. The numbers in the figure indicate the corresponding energy groups. The general spectral shapes are in good agreement, but the advantage of the 26-group cross-section set can be seen in that it shows the characteristic dip due to sodium elastic scattering near 3 keV, whereas in the 16-group case this feature is obscured.

2.4 SENSITIVITY STUDIES

Sensitivity studies pertinent to the design and to the general assessment of results are to be discussed in this section. The five items that have been investigated are: a) S_2, S_4 vs. S_8 calculations, b) the effect of hydrogen in the sodium chromate, c) variation in chromate density, d) effect of using 1% Pu^{239} vs. U^{235} fuel, and e) effects of fuel enrichment in Blanket No. 2.

2.4.1 S_2, S_4 vs. S_8 Calculations

A sensitivity study of the effect on reaction rates of using different S_N options in the one-dimensional ANISN calculation has been performed. The comparison is extended up to S_8 since the use of S_N options higher than S_8 was found not to improve the results significantly.

Figure 2.12 shows the percent differences using S_2, S_4 and S_8 options to calculate U^{238} capture (a typical capture reaction) and U^{238} fission (a typical threshold reaction) rates through the blanket and steel reflector regions in BTF Blanket Mock-Up No. 2. It is evident that the S_2 and S_4 options predict higher U^{238} fission rates but lower U^{238} capture rates in the blanket region than the S_8 option. On the other hand, in the reflector region, lower U^{238} fission rates and higher capture rates are predicted by the S_2 and S_4 options. In general, the S_2 and S_4 options predict harder spectra in the blanket but softer spectra in the reflector.

FIG.2.12 PERCENTAGE DEVIATIONS IN BLANKET NO.2
 REACTION RATE CALCULATIONS FOR S_2, S_4 VS
 S_8 OPTIONS IN ANISN.

2.4.2 The Effect of Hydrogen in the Sodium Chromate

Excessive water content of the sodium chromate in Blanket No. 2 can have a significant effect on the spectrum in the blanket and therefore on the foil activities measured in the axial direction in the blanket (equivalent to the radial direction in an LMFBR).

The ANISN code was used to make 26-group, 1D, S_8 calculations of the $U^{238}(n, \gamma)$ capture rate and $U^{238}(n, f)$ fission rate through the blanket and steel reflector of Blanket No. 2. The U^{238} capture (a typical capture reaction) and U^{238} fission (a typical threshold reaction) rates were calculated for chromate water contents of 0.0 w/o, 0.1 w/o, 0.2 w/o and 0.5 w/o. Figure 2.13 shows, for a given sodium chromate water content, the amount by which the U^{238} capture and fission rates through Blanket No. 2 differ from the zero water content case. It is seen that the error is reduced to about the limit of precision in the experimental determination of the activities only when the water content of the sodium chromate is reduced to 0.1 w/o or less.

2.4.3 Variation in Chromate Density

Since the loadings of sodium chromate in different subassemblies were not exactly equal but varied from 30.51 kg to 31.80 kg or about 2% from the mean (Section 2.2.2), both the blanket spectrum and reaction rates could be affected. The effect of variation in chromate density on the U^{238} capture and U^{238} fission rates (typical capture and threshold reaction rates, respectively) was calculated by using the 26-group, S_8 option in the 1D, ANISN code. Figure 2.14 shows the percent changes in the U^{238} capture and fission rates through Blanket No. 2 at chromate loadings equal to -2% and 2% from the mean. It is seen that there is a

FIG 2.13 PERCENTAGE ERROR IN REACTION RATES IN BLANKET NO.2 AS A FUNCTION OF THE SODIUM CROMATE WATER CONTENT

FIG.2.14 PERCENTAGE CHANGES IN BLANKET NO.2 REACTION RATES AS A FUNCTION OF THE SODIUM CHROMATE LOADINGS.

slight increase in the capture rate but decrease in the fission rate as the chromate density is increased from -2% to 2%. This can easily be explained by the fact that the presence of more sodium chromate tends to soften the blanket spectrum, and hence to increase the activation of material with large, low energy (n, γ) cross sections (e.g., the U^{238} capture reaction), but tends to decrease the activation of materials with high energy thresholds (e.g., the U^{238} fission reaction).

However, since the percent changes in the capture and fission reaction rates caused by the variation of chromate density are small ($\leq 1\%$) compared to the limit of precision in the experimental determination of the activities ($\approx 2\%$), the effect of variation in chromate density can, in general, be ignored. In addition, since the subassemblies having the largest deviation from the mean loading were placed near the outside of the blanket (see Appendix A for sodium chromate loading pattern), the effect of variation in chromate density in the central region of the blanket was even further reduced.

2.4.4 Effect of Using 1% U^{235} vs. Pu^{239} Fuel

The effect of using 1% U^{235} fuel instead of Pu^{239} fuel on the blanket spectra and reactions rates has been studied, again by calculations using the ANISN code.

Table 2.7 shows the spectrum at $Z = 24.4$ cm in Blanket No. 2 for the two different cases. It is seen that the blanket spectrum using Pu^{239} fuel is softer than that using U^{235} fuel: a decrease in neutron flux at energies above 4.65 keV (group 12) but an increase in neutron flux at lower energies. This means that using Pu^{239} fuel would increase the reaction rate of materials with large, low energy cross sections but would

TABLE 2.7

Comparison of Blanket Spectra at $Z = 24.4$ cm,
Using 1% U^{235} vs. Pu^{239} Fuel

Energy Group*	Group Fluxes, Arbitrary Normalization	
	1% U^{235} Fuel	1% Pu^{239} Fuel
1	1.43×10^5	1.38×10^5
2	7.95×10^5	7.63×10^5
3	1.54×10^6	1.47×10^6
4	3.67×10^6	3.52×10^6
5	5.74×10^6	5.53×10^6
6	1.69×10^7	1.65×10^7
7	2.75×10^7	2.70×10^7
8	3.26×10^7	3.22×10^7
9	3.52×10^7	3.49×10^7
10	2.72×10^7	2.72×10^7
11	3.10×10^7	3.11×10^7
12	1.77×10^7	1.77×10^7
13	4.39×10^6	4.40×10^6
14	1.61×10^7	1.63×10^7
15	1.26×10^7	1.30×10^7
16	9.01×10^6	9.32×10^6
17	5.50×10^6	5.72×10^6
18	2.45×10^6	2.48×10^6
19	1.05×10^6	1.11×10^6
20	3.09×10^5	3.09×10^5
21	8.40×10^4	8.48×10^4
22	8.40×10^4	8.85×10^4
23	6.61×10^4	7.34×10^4
24	4.16×10^4	3.81×10^4
25	1.76×10^4	3.39×10^4
26	4.18×10^3	1.59×10^4

* For equivalent energies, see Table 2.6.

decrease the activation of materials with high energy thresholds.

Figure 2.15 shows the percent increase in the U^{238} capture rate and the percent decrease in the U^{238} fission rate in Blanket No. 2 when 1% Pu^{239} fuel is used instead of U^{235} fuel.

2.4.5 Effects of Fuel Enrichment in Blanket No. 2

ANISN runs were performed for several different enrichments of the Blanket No. 2 fuel. In particular, a comparison of the spectra and activation traverses for the current 1.074% fuel enrichment and for depleted fuel (0.190%) has been made.

Table 2.8 shows the spectrum at $Z = 24.4$ cm in Blanket No. 2 for the two enrichments. It is seen that decreasing the blanket fuel enrichment from 1.07% to 0.19% decreases the flux above 1 keV and increases the flux below this energy. This means that the effect of decreasing the fuel enrichment to 0.19% would be to increase the activation of materials with large, low energy (n, γ) cross sections and to decrease the activation of materials with predominating high energy resonances or threshold reactions.

Figure 2.16 shows the changes in both unshielded and shielded U^{238} activation through the blanket and reflector as the enrichment is decreased from 1.07% to 0.19% (the U^{238} cross sections used for the multigroup flux calculations were shielded in all cases). It is seen that the unshielded U^{238} activation (equivalent to foils irradiated outside the fuel rods) increases in the blanket and decreases in the reflector. The shielded U^{238} activation (required for breeding calculations), on the other hand, decreases in both the blanket and the reflector. Other reaction rates such as $Au^{197}(n, \gamma)$ and $Mo^{98}(n, \gamma)$ behave in a manner similar to the unshielded U^{238} case.

FIG.2.15 PERCENTAGE CHANGE IN BLANKET NO.2 REACTION RATE DISTRIBUTION WHEN 1% ^{239}Pu FUEL IS SUBSTITUTED FOR 1% ^{235}U FUEL.

TABLE 2.8
Comparison of Blanket Spectra at $Z = 24.4$ cm
for Two Different Enrichment Fuels

Energy Group*	Group Fluxes, Arbitrary Normalization	
	1.074% Enriched U	0.190% Enriched U
1	1.432×10^5	8.106×10^5
2	7.948×10^5	4.408×10^5
3	1.534×10^6	7.754×10^5
4	3.671×10^6	2.035×10^6
5	5.738×10^6	3.450×10^6
6	1.696×10^7	1.201×10^7
7	2.750×10^7	2.160×10^7
8	3.261×10^7	2.708×10^7
9	3.522×10^7	3.035×10^7
10	2.722×10^7	2.400×10^7
11	3.103×10^7	2.827×10^7
12	1.764×10^7	1.636×10^7
13	4.384×10^6	4.086×10^6
14	1.608×10^7	1.548×10^7
15	1.263×10^7	1.270×10^7
16	9.006×10^6	9.578×10^6
17	5.494×10^6	6.228×10^6
18	2.446×10^6	2.942×10^6
19	1.054×10^6	1.379×10^6
20	3.086×10^5	4.254×10^5
21	8.392×10^4	1.182×10^5
22	8.384×10^4	1.160×10^5
23	6.608×10^4	9.508×10^4
24	4.161×10^4	6.498×10^4
25	1.762×10^4	3.422×10^4
26	4.183×10^3	1.170×10^4

*For equivalent energies, see Table 2.6.

FIG.2.16 PERCENT CHANGE IN BOTH UNSHIELDED AND SHIELDED ^{238}U ACTIVATION AS THE ENRICHMENT CHANGES FROM 1.07% TO 0.19%.

2.4.6 General Comments on Sensitivity Studies

A number of general comments on the results of the sensitivity studies are given below:

a) Tolerances on Blanket No. 2 in terms of water content and local density variations have been adequately controlled: the small effects due to water contamination can be taken into account using multigroup calculations; and the density variations can be neglected.

b) The non-negligible U^{235} content could be thought of as a rough approximation to bred Pu^{239} in a highly irradiated blanket. However, it is unlikely that the in-out blanket fuel management scheme necessary to achieve uniform fissile content would be employed in LMFBR's.

c) The 1.07% enriched fuel would probably be a good approximation to a natural uranium blanket at beginning of life, but LMFBR's are more likely to use depleted uranium, in which case results can be affected by up to around 10%. In view of this and the considerations cited in b) above, it is preferable to think of Blanket Mock-Up No. 2 as a clean, well-defined benchmark mock-up rather than as an "exact" mock-up of a given LMFBR blanket at some point in its life.

Finally, it is worth anticipating subsequent results to the extent of noting that the effects of U^{238} heterogeneous self-shielding overshadow all of the effects considered here.

So far, all the numerical calculations performed on the BTF Blanket Mock-Up rest on the assumption of the validity of one-dimensional analysis. Therefore, the first task is to confirm the existence of a transverse buckling in the blanket. This has been done experimentally and is discussed in the next chapter.

Chapter III

BUCKLING MEASUREMENTS

3.1 INTRODUCTION

In the neutronic studies of the simulated LMFBR blanket, the loss of neutrons by leakage in the lateral (horizontal and vertical) directions is an important part of the neutron balance for several reasons. As was noted previously, this leakage is carefully tailored to enable simulation of cylindrical geometry in a slab system. Furthermore, it is essential that this leakage be characterized by a buckling formulation in order to validate the one-dimensional calculation methods used to design the experiment and interpret the data.

The determination of the lateral buckling factor by foil activation traverses will be described in this chapter. Some conclusions drawn from these foil activation traverses will also be discussed.

Horizontal and vertical (X- and Y-directions, see Fig. 3.1) activation traverses were made with various foil materials at varying distances into the blanket assembly to determine the horizontal and vertical flux shapes in the blanket and to permit calculation of the transverse leakage, or buckling. The foil materials used were gold, indium and molybdenum. Table 3.1 lists the 26-group activation cross sections for gold, indium and molybdenum used in this work. The $\text{Au}^{197}(n, \gamma)\text{Au}^{198}$ reaction has relatively large cross sections at both high and low neutron energies; hence, the gold activities serve as a neutron detector over the whole neutron energy spectrum. On the other

TABLE 3.1

26-Group Activation Cross Sections for Gold, Indium and Molybdenum

Group	E_L	ACTIVATION CROSS SECTION, BARNS		
		$Au^{197}(n,\gamma)Au^{198}^*$	$In^{115}(n,n')In^{115m}^*$	$Mo^{98}(n,\gamma)Mo^{99}^{**}$
1	6.5 MeV	0.014	0.299	0.020
2	4.0 "	0.018	0.346	0.007
3	2.5 "	0.032	0.349	0.015
4	1.4 "	0.063	0.259	0.020
5	0.8 "	0.100	0.0801	0.028
6	0.4 "	0.144	0.0122	0.042
7	0.2 "	0.237	0.0006	0.052
8	0.1 "	0.339	0.0	0.070
9	46.6 keV	0.049		0.10
10	21.5 "	0.076		0.15
11	10.0 "	1.143		0.26
12	4.65 "	1.838		0.45
13	2.15 "	2.85		0.55
14	1.0 "	6.29		0.70
15	465.0 eV	9.77		1.70
16	215.0 "	12.57		1.50
17	100.0 "	12.60		3.30
18	46.5 "	25.17		11.10
19	21.5 "	9.76		10.00
20	10.0 "	6.19		0.64
21	4.65 "	706.05		0.16
22	2.15 "	470.83		0.24
23	1.0 "	25.67		0.35
24	0.465 "	23.63		0.52
25	0.215 "	32.45		0.76
26	Thermal	87.74		2.70

*The gold and indium activation cross sections were obtained by collapsing the 620-group activation cross sections of the SAND-II (M1) library over a typical LMFBR core spectrum.

**The molybdenum activation cross sections are those of the ABBN, 26-group cross-section set.

hand, the 300-keV threshold reaction of indium, $\text{In}^{115}(n, n')\text{In}^{115m}$, has large cross sections at energies in the MeV region, and therefore indium is suitable as an indicator of high energy neutron behavior. The $\text{Mo}^{98}(n, \gamma)\text{Mo}^{99}$ reaction has a very small cross section at high energies but a relatively large cross section in the eV to keV energy region. For this reason, molybdenum was chosen as a detector of eV to keV energy neutrons.

3.2 EXPERIMENTAL PROCEDURE

3.2.1 Foil Irradiation

The size and weight of the gold, indium and molybdenum foils are listed in Table 3.2. Each foil was taped into a recessed slot (0.254-inch diameter, 60 mil deep) milled in 5/16-inch-diameter, low carbon steel, foil-holder rods. A single thickness of 3-mil Mylar tape was used to hold each foil in place. (The perturbation effect of Mylar tape on foil activity will be discussed in Section 4.2.5.) The foil holders were inserted into the 3/8-inch-I.D. steel foil-holder tubes in the blanket sub-assembly prior to irradiation. A set of foil holders was lined up in the axial Z-direction, positioned at $Z = 0.92, 2.96, 5.00, 6.84, 8.88, 10.92, 12.76, 14.80$ and 16.84 inches, respectively, from the converter-blanket interface. Another set of foil holders, all of which were 8.88 inches from the converter-blanket interface, was lined up in the horizontal X-direction, positioned at $X = \pm 1.5, \pm 4.5, \pm 7.5, \pm 10.5, \pm 13.5, \pm 16.5, \pm 19.5, \pm 22.5$ and ± 25.5 inches from the center line of the blanket assembly. Figures 2.5 and 3.1 show schematic views of the foil irradiation positions in the Blanket Mock-Up No. 2. All the foil holders were oriented with the foil facing the converter assembly. The typical irradiation time for the foils ranged from four to ten hours.

2

TABLE 3.2
Description of Foil Materials

Foil Material	Reaction	Diameter (Inch)	Thickness (Mil)	Weight (gm)	Half Life of Radioactive Isotope
Gold	$\text{Au}^{197}(\text{n}, \gamma)\text{Au}^{198}$	1/8	10	0.038	2.8 days
Indium	$\text{In}^{115}(\text{n}, \text{n}')\text{In}^{115\text{m}}$	1/4	10	0.060	4.5 hours
Molybdenum	$\text{Mo}^{98}(\text{n}, \gamma)\text{Mo}^{99}$	1/8	25	0.057	66 hours

FIG. 3.1 SCHEMATIC VIEW OF THE CONVERTER AND BLANKET ASSEMBLY SHOWING THE FOIL IRRADIATION POSITIONS

3.2.2 Counting Equipment

The well-type scintillation counting system, used to measure the gamma activities of irradiated detector foils, is shown schematically in Fig. 3.2. The same system was also used for counting foils for axial traverses described in Chapter IV. A well-type, thallium-activated, sodium iodide crystal was used because of its intrinsically high efficiency. This 2-inch-diameter crystal, with photo-multiplier tube and preamplifier, was surrounded by 3 inches of iron shielding and 8 inches of lead shielding to maintain a very low background. The high voltage supply for the photo-multiplier tube and the low voltage for the preamplifier were provided by the voltage supply, Hewlett-Packard Model 5551A. A single-channel pulse height analyzer was used: Hewlett-Packard Model HP-5201L scaler-timer. The scaler-timer unit had both preset time or count options and also a 6-digit readout. The dead-time of the counting system was 2.4 microseconds.

The radioisotopes Co^{60} , Mn^{54} , Co^{57} , and Cs^{137} were used to calibrate the counting system. Figure 3.3 shows the calibration curve. Before each foil counting session, the calibration on the counting system was checked to ensure that there had been no drift in voltage.

3.2.3 Foil Counting Techniques

The counting techniques for each of the foil materials used in the buckling measurements are discussed in this section, which concludes with a summary table (Table 3.3) showing the pertinent characteristics of the different foil counting methods.

Gold Foil Counting. In the buckling measurement, since far more traverses were made with gold foils than with other detector foils, many

FIG. 3.2 WELL-TYPE SCINTILLATION COUNTING SYSTEM.

FIG. 3.3 CALIBRATION OF THE COUNTING SYSTEM

more irradiated gold foils had to be counted. Consequently, instead of using the well-type crystal counting system, the MIT Mark I counter with an automatic sample changer was used. A thallium-activated sodium iodide crystal was also used as the detector. The Mark I counting system will not be described here, but reference is made to publications of the MIT Lattice Project (S1, S2).

The activation product, Au^{198} , in the irradiated gold foil emits primarily gamma rays of energy 412 keV. The single channel analyzer settings were set to admit the desired 412-keV gamma-ray photopeak. After counting, the raw data were printed out and punched on paper tape by the automatic system. A computer code ACTVTY, (L1), was used to calculate the foil activity in counts per minute per milligram, after correction for sample changing time, counter dead-time, background and decay.

Indium Foil Counting. Detection of the $\text{In}^{115}(\text{n}, \text{n}')\text{In}^{115\text{m}}$ reaction is complicated by the competing neutron capture reaction, $\text{In}^{115}(\text{n}, \gamma)\text{In}^{116}$. The In^{116} activity consists of gamma rays with energies greater than the 335-keV. The gamma rays are from the $\text{In}^{115\text{m}}$ activity, so that there can be Compton-effect counts from In^{116} activity underneath the 335-keV photopeak. The resonance integral for the capture reaction $\text{In}^{115}(\text{n}, \gamma)\text{In}^{116}$ is about 2640 barns; as a result, measurements of the $\text{In}^{115\text{m}}$ activity (4.5-hour half-life) was postponed until at least 8 hours after the end of the irradiation to allow decay of the In^{116} activity (54-min. half-life).

The indium foils were counted individually with the well-type scintillation counting system described in Section 3.2.2. The 335-keV gamma ray from $\text{In}^{115\text{m}}$ was detected, and the channel width of the single channel analyzer was set approximately to 120 keV, straddling the photopeak.

Molybdenum Foil Counting. The molybdenum foils were counted in the same counting system as the indium foils. The 0.78-MeV gamma ray of Mo^{99} was counted with a channel width setting of approximately 130 keV.

Table 3.3 summarizes the counting characteristics of the foil materials used in the buckling measurements: the gamma energies detected, discriminator settings of the analyzer, typical counts accumulated, counting time and irradiation time.

3.2.4 Corrected Gamma Activities

The "raw" counts were corrected for room and residual foil background activities, for weights and for sample decay time according to the equation:

$$A = \frac{\lambda (C-B)}{w \left(1 - e^{-\lambda t_i}\right) \left(e^{-\lambda t_w}\right) \left(1 - e^{-\lambda t_c}\right)} \quad (3.1)$$

where

A = corrected activity (in counts/min-mg)

C = number of counts accumulated after dead-time correction

B = room and residual foil background

λ = decay constant of the radionuclide

w = weight of foil

t_i = irradiation time

t_w = wait time after removal from the flux until the beginning of foil counting

t_c = counting time.

The corrected gamma activities of the gold, indium and molybdenum horizontal and vertical traverses in BTF Blanket Mock-Up No. 2 are

TABLE 3.3

Typical Data for Foil Counting in Buckling Measurements

Reaction	Gamma Energy Detected (MeV)	Discriminator Settings (volts)		Typical Counts Accumulated (Background Subtracted)	Counting Time (min)	Typical Irradiation Time (hr)
		E _{min}	E _{max}			
Au ¹⁹⁷ (n,γ)Au ¹⁹⁸	0.412	2.24*	3.64*	100,520	5.0	6
Mo ⁹⁸ (n,γ)Mo ⁹⁹	0.78	1.75	2.15	14,413	2.0	4
In ¹¹⁵ (n,n')In ^{115m}	0.335	0.70	1.10	44,690	20.0	4

*MARK I counter settings.

listed in Appendix B. These results are also presented graphically in the next section.

3.3 RESULTS OF MEASUREMENTS

3.3.1 Gold, Indium and Molybdenum Vertical Activation Traverses in Blanket No. 2

Figure 3.4 shows the vertical (Y-direction) activation traverses for gold. (See Fig. 3.1 for the coordinates of the blanket system.) The vertical traverses were made 1.5 inches off the horizontal center line ($X = 1.5$ inches) to avoid the interface of the staggered subassemblies. The five different vertical activation traverses were made at various depths (in the axial, Z, direction) into the blanket. The circles shown in the figures indicate the experimental values, and the standard deviation lies within the diameter of these circles. The solid line is the fitted curve (by the least-square method) over all the experimental points. The gold activation vertical traverses are fit quite well by the relation

$$A(Y) = A(0) \cos \left(\frac{\pi Y}{60} \right), \quad (3.2)$$

where A is the gold activity and Y is in inches. The small discrepancies between measured activities and the fitted cosine curve indicate that the loading of Blanket Mock-Up No. 2 is fairly uniform. In Blanket Assembly No. 1, the iron and borax test blanket, large discrepancies between measured activities and the fitted cosine curve were attributed to non-uniform loading in that blanket assembly.

The indium and molybdenum vertical activation traverses are shown in Fig. 3.5. Again, both these activation traverses have the same cosine shape as that of gold,

$$A(Y) = A(0) \cos \left(\frac{\pi Y}{60} \right), \quad (3.2)$$

FIG.3.4 GOLD VERTICAL ACTIVATION TRAVERSES IN BLANKET NO. 2

FIG.3.5 VERTICAL ACTIVATION TRAVERSES IN BLANKET NO.2

where A is the indium or molybdenum activity and Y is in inches. Since the foils are chosen to selectively detect neutrons of different energies, the activity curves suggest that neutrons of different energies diffuse transversely in the same fundamental flux mode.

Figure 3.6 shows the vertical gold-to-indium, indium-to-molybdenum, and gold-to-molybdenum activation ratios at various depths into the blanket. It is seen that the activation ratios are constant from $X = -15$ inches to $X = 15$ inches, indicating that vertical spectral equilibrium is reached over a large central region of the blanket. Also, these activation ratios change as a function of depth into the blanket. The increase in the gold-to-indium and gold-to-molybdenum activation ratios and the decrease in the indium-to-molybdenum activation ratio with depth into the blanket confirm the expected softening of the neutron spectrum.

3.3.2 Gold, Indium and Molybdenum Horizontal Activation Traverses in Blanket No. 2

Similarly, horizontal (X-direction) activation traverses were made with gold, indium and molybdenum to determine the horizontal distribution of the neutron flux in the blanket. Figure 3.7 shows the horizontal activation traverses for gold at different depths into the blanket in the axial, Z, direction. These traverses were made 3.0 inches above and below the mid-plane of the blanket. Figure 3.8 shows the horizontal activation traverses for indium and molybdenum in the front of the blanket, $Z = 0.0$ inch and 8.88 inches into the blanket. All the measured foil activities can be fitted by cosine curves of the following form,

$$A(X) = A(0) \cos \left(\frac{\pi X}{74} \right), \quad (3.3)$$

where A is the foil activity and X is in inches.

FIG. 3.6 VERTICAL ACTIVATION RATIOS IN BLANKET NO. 2

FIG.3.7 GOLD HORIZONTAL ACTIVATION TRAVERSES IN BLANKET NO. 2

FIG. 3.8 HORIZONTAL ACTIVATION TRAVERSES IN BLANKET NO. 2

The foil activation ratios of different horizontal traverses are shown in Fig. 3.9. The same conclusion as to spectral softening with depth can be drawn from the horizontal traverses as in the case of the vertical direction traverses.

3.3.3 Buckling Results

The horizontal and vertical activation traverses with gold, indium and molybdenum foils in the blanket assembly show that the horizontal and vertical spatial distributions of the neutron flux in the blanket assembly have the desired cosine dependence; furthermore, these cosine distributions have constant width and height through the thickness of the blanket. Hence, the neutron flux in the blanket can be described by the relation

$$\phi(X, Y, Z, E) = \cos \left(\frac{\pi X}{74} \right) \cos \left(\frac{\pi Y}{60} \right) \phi(Z, E), \quad (3.4)$$

and the lateral leakage from the assembly can be accounted for with a constant buckling factor. The calculation of the buckling factor, B^2 , is made by using the relationship,

$$B^2 = \left(\frac{\pi}{H} \right)^2 + \left(\frac{\pi}{A} \right)^2, \quad (3.5)$$

where H, A are the experimentally determined values of the effective height and width of the neutron flux distribution in the blanket, respectively. Numerically,

$$H = 60 \text{ inches} \pm 1.5 \text{ inches} = 152.4 \pm 3.8 \text{ cm},$$

$$A = 74 \text{ inches} \pm 1.5 \text{ inches} = 188.0 \pm 3.8 \text{ cm},$$

and

$$B^2 = 0.000704 \pm 0.000028 \text{ cm}^{-2}.$$

FIG.3.9 HORIZONTAL ACTIVATION RATIOS IN BLANKET NO. 2

This experimentally determined buckling, B^2 , has a probable error (Appendix E) of about $\pm 4.0\%$, but this uncertainty will affect the neutron balance in the entire blanket only slightly. The leakage neutrons in the lateral directions account for only about 16% of the total losses (6.4% in the horizontal, X, direction and 9.6% in the vertical, Y, direction); therefore, the $\pm 4.0\%$ uncertainty in the transverse buckling will contribute an uncertainty of only $\pm 0.54\%$ to the blanket neutron balance. In addition, calculations with the ANISN code showed that an increase in the lateral buckling by as much as 20% had no appreciable effect on the axial reaction rate distribution in the blanket. In other words, the reaction rates in the blanket are not sensitive to the transverse leakages.

It is desirable to know what size of cylindrical system Blanket No. 2 corresponds to. The relationship governing slab simulation of cylindrical geometry blanket is given by the following equation from reference (F1):

$$A(Z) \cong \frac{\pi\sqrt{2} (Z+R)}{\sqrt{1+2\beta(Z+R)}} \quad (3.6)$$

where

$$\beta^2 = \alpha^2 + \left(\frac{\pi}{H}\right)^2, \quad \alpha^2 = \frac{\Sigma_a - \nu\Sigma_f}{D}, \quad (3.7)$$

and

A = effective width of the blanket in slab geometry,

R = radius of core,

Z = distance into the blanket.

For a radial blanket composed of 50 v/o depleted UO_2 , 30 v/o sodium and 20 v/o stainless steel, having an effective height of 138 cm, we get $\beta = 0.0409$ (by Eq. 3.7). Next, we take the measured value of $A(Z)$ to be the value at the middle of the 45-cm-thick Blanket No. 2, i.e., $A(Z) = 188$ cm at $Z = 22.5$ cm. Then, by Eq. 3.6, the calculated core radius, R ,

at $Z = 0.0$ cm, is 135 cm. This result means that Blanket No. 2 mocks up the radial blanket of a LMFBR with a core radius, $R = 135$ cm.

Although this core size is larger than that of the two-zone ZPPR-2 core ($R \cong 100$ cm), it is typical of the size of large 1000-MWe LMFBR cores.

On the other hand, Blanket No. 2 can also be considered to be simulating an axial blanket. In that case,

$$\left(\frac{\pi}{H}\right)^2 + \left(\frac{\pi}{A}\right)^2 = \left(\frac{2.405}{\tilde{R}}\right)^2 \quad (3.8)$$

where

H = effective height of blanket mock-up,

A = effective width of blanket mock-up,

$\tilde{R}$ = extrapolated radius of reactor core.

If we use the measured values for Blanket No. 2, $A = 188$ cm and $H = 152$ cm, the extrapolated radius $\tilde{R}$ is calculated to be 90.5 cm so that the actual radius of the core is $R \cong 72$ cm. Hence, Blanket No. 2 mocks up the axial blanket of an LMFBR with a 72-cm core radius. This value is close to that of the single-zone demonstration benchmark reactor (ZPR 6, Assembly 7), (Z1), which has a core radius of 80 cm.

3.4 SUMMARY

It was shown in this chapter, first, that the transverse leakage of Blanket No. 2 can be characterized by a constant buckling factor, $B^2 = 0.000704 \pm 0.000028 \text{ cm}^{-2}$. This result validated the use of the one-dimensional calculation method used to design the experiments and interpret the data in the blanket.

Second, Blanket No. 2 can be considered as mocking up a radial blanket with a 135-cm core radius, or as an axial blanket having a core

radius of 72 cm. Thus, the results of reaction rate and spectra measurements in Blanket No. 2 (Chaps. IV and V) were therefore applicable to both axial and radial blankets although the blanket was specifically optimized for a radial blanket.

Chapter IV

REACTION RATE MEASUREMENTS

4.1 INTRODUCTION

Measurements of fission and capture rates in fast reactor blanket mock-ups are important because of their direct relation to practical parameters in realistic blankets, such as the rates of energy and plutonium production. In addition, comparison of reaction rate measurements with theoretical predictions are particularly useful in checking calculational models and for testing cross-section data.

Reaction rate traverses in the BTF Blanket Mock-Up No. 2 were made in the axial Z-direction, simulating the radial direction of actual LMFBR blankets. The three types of reaction rates studied were as follows:

- a) Capture reaction rates: $\text{Au}^{197}(n, \gamma)$, $\text{U}^{238}(n, \gamma)$, $\text{Mo}^{98}(n, \gamma)$, $\text{Cr}^{50}(n, \gamma)$ and $\text{Na}^{23}(n, \gamma)$.
- b) Threshold reaction rates: $\text{U}^{238}(n, f)$ and $\text{In}^{115}(n, n')$.
- c) Fission reaction rates: $\text{U}^{235}(n, f)$ and $\text{Pu}^{239}(n, f)$.

The techniques employed in measuring reaction rates in the blanket assembly are described in Section 4.2 of this chapter. Results of reaction rate measurements and comparisons with theoretical prediction are presented in Section 4.3. Factors affecting the reaction rate calculations in the blanket assembly are also discussed in Sections 4.4 and 4.5. A neutron balance for the entire blanket assembly has been calculated and

verified by measurement of a few key reaction rates. The results of the neutron balance are given in Section 4.6.

4.2 EXPERIMENTAL PROCEDURE

4.2.1 Description of Foil Materials

Table 4.1 lists the physical characteristics of the detector foil materials used for the reaction rate measurements in BTF Blanket Mock-Up No. 2. Most foil materials (gold, molybdenum, indium, uranium and plutonium) were in metallic disc form. Only two of the foil materials used, sodium carbonate and chromium, were in powder form. These powdered foil materials were contained in polyethylene capsules (1 inch long, 0.250-inch O.D. and 0.210-inch I.D.) which were heat-sealed before irradiation to prevent leakage. The plutonium foils were enclosed in aluminum cans having a wall thickness of 0.020 inch because of the high radiological toxicity of plutonium.

The nuclear properties of the foil materials used in the reaction rate measurements are listed in Table 4.2, where $T_{\frac{1}{2}}$ is the half-life of the product nuclide and the (E_{γ}) are the energies of the gamma rays of the product nuclide; also given is the corresponding percent yield per disintegration ($\% \gamma$).

4.2.2 Foil Irradiation

In the reaction rate measurements on foil materials in disc form, including the enclosed plutonium foils, the foils were simply taped into recessed spots (0.254-inch diameter, 60 mils deep) milled in 5/16-inch-diameter, low carbon steel foil-holder rods. A single layer of Mylar tape, 3 mils thick, was used to hold each foil or capsule in place. A typical foil holder is shown in Fig. 4.1. For the powdered foil materials,

TABLE 4.1. Physical Characteristics of Foil Materials

Reaction	Foil Material	Description		Typical Weight (mg)	Purity (wt. %)	Isotopic Abundance (%)
		Diameter (inch)	Thickness (mils)			
<u>CAPTURE</u>						
$\text{Au}^{197}(\text{n}, \gamma)\text{Au}^{198}$	Gold	0.125	10	36.85	99.99	100
$\text{Mo}^{98}(\text{n}, \gamma)\text{Mo}^{99}$	Molybdenum	0.125	25	54.10	99.97	9.45
$\text{U}^{238}(\text{n}, \gamma)$	Depleted uranium	0.250	5	67.50	18 ppm U-235	—
$\text{Na}^{23}(\text{n}, \gamma)\text{Na}^{24}$	Sodium chloride	Powder form		131.01	99.7	100
$\text{Cr}^{50}(\text{n}, \gamma)\text{Cr}^{51}$	Chromium	Powder form		454.09	99.98	4.31
<u>THRESHOLD</u>						
$\text{In}^{115}(\text{n}, \text{n}')\text{In}^{115\text{m}}$	Indium	0.250	10	61.50	99.999	95.77
$\text{U}^{238}(\text{n}, \text{f})$	Depleted uranium	0.250	5	67.50	18 ppm U-235	—
<u>FISSION</u>						
$\text{U}^{235}(\text{n}, \text{f})$	Enriched uranium	0.250	5	62.30	93.17% U^{235}	—
$\text{Pu}^{239}(\text{n}, \text{f})$	Plutonium	0.250	10	154.00	97.9% Pu^{239}	—

TABLE 4.2. Nuclear Properties of Foil Materials (H2, E4, C1, P1, T1, N1, W4)

Reaction	Product Nuclide	$T_{\frac{1}{2}}$	E_{γ} (MeV)	Percent γ
<u>CAPTURE</u>				
$\text{Au}^{197}(\text{n}, \gamma)\text{Au}^{198}$	Au^{198}	2.8 days	0.412	99.82
$\text{Mo}^{98}(\text{n}, \gamma)\text{Mo}^{99}$	Mo^{99}	66 hours	0.78	14.0
$\text{U}^{238}(\text{n}, \gamma)$	Np^{239}	2.33 days	0.103	—
$\text{Na}^{23}(\text{n}, \gamma)\text{Na}^{24}$	Na^{24}	14.9 hours	1.38 2.75	100 100
$\text{Cr}^{50}(\text{n}, \gamma)\text{Cr}^{51}$	Cr^{51}	27.8 days	0.32	9.0
<u>THRESHOLD</u>				
$\text{In}^{115}(\text{n}, \text{n}')\text{In}^{115\text{m}}$	$\text{In}^{115\text{m}}$	4.5 hours	0.335	94.5
$\text{U}^{238}(\text{n}, \text{f})$	Gross fission products	≈ 2.5 hours	≥ 0.5	—
<u>FISSION</u>				
$\text{U}^{235}(\text{n}, \text{f})$	Gross fission products	≈ 2.5 hours	≥ 0.5	—
$\text{Pu}^{239}(\text{n}, \text{f})$	Gross fission products	≈ 2.5 hours	≥ 0.5	—

FIG. 4.1 TYPICAL FOIL HOLDER RODS.

the foil-holder rods were modified to have a hollowed-out section (2 inches long, 0.250 inch deep) for holding powdered foil materials contained in polyethylene capsules, again as shown in Fig. 4.1.

Prior to irradiation, all the foils or foil capsules were loaded in the foil-holder rods which were then inserted into the 3/8-inch-I.D., steel holder tubes in the BTF Blanket Mock-Up No. 2 subassembly (Fig. 3.1). The foil holders were lined up in the axial Z-direction, positioned at $Z = 0.92, 2.96, 5.00, 6.84, 8.88, 10.92, 12.76, 14.80$ and 16.84 inches, respectively, from the converter-blanket interface. All the foil holders were oriented with the foil or foil capsules facing the converter assembly.

The reaction rate measurements in the reflector region were made by placing foils in recessed spots milled in a 2-inch-diameter, foil-holder rod. The latter could be inserted into the 2-1/8-inch-diameter hole drilled horizontally through the 18-inch steel reflector at a distance of three inches below the mid-height of the blanket assembly (Fig. 3.1).

The irradiation time for the foils varied from 4 to 24 hours, depending on the properties of the detector foils. Table 4.3 lists the typical irradiation time of each reaction rate traverse.

4.2.3 Foil Counting Techniques

The counting techniques for each of the foil materials used in the reaction rate measurements are discussed in this section, which concludes with a summary table (Table 4.3) showing the gamma energies detected, discriminator settings employed, typical counts accumulated and typical irradiation time for each of the foil materials.

The counting system described in Section 3.2.2 was also used to measure the irradiated foil gamma activities for the axial reaction rate traverses.

4.2.3.1 Capture Reaction Rate Measurements

Gold Foil Counting. The gold foil counting technique was the same as that described in Section 3.2.3, except that the well-type crystal counting system was used instead of the MIT MARK I automatic counter.

Molybdenum Foil Counting. The same procedure was used as described in Section 3.2.3.

Uranium-238 Capture. The U^{238} neutron capture rate data were obtained by counting the Np^{239} decay activities of the irradiated depleted uranium foils.

The "103-keV" peak in the γ -ray and X-ray spectrum of the Np^{239} was counted. This characteristic composite peak is due to the 106-keV γ -ray of Np^{239} , and the 99-keV and 100-keV X-rays of Pu^{239} from the internal conversion of higher energy γ -rays of Np^{239} . The lower and upper limits chosen for the analyzer window were 84 keV and 122 keV, respectively. The Np^{239} counting was commenced about 12 hours after irradiation to allow for the decay of fission product activity to the point where the Compton background would not obscure the low energy Np^{239} gammas.

Sodium and Chromium Counting. The 1.38-MeV gamma ray from Na^{24} was detected; the channel width used to straddle the photopeak was 180 keV. Similarly, the 0.32-MeV energy gamma ray of Cr^{51} was counted with a channel width setting of 100 keV.

The powdered foil materials, sodium carbonate and chromium, were contained in polyethylene capsules during irradiation and counting. The background activities contributed by the foil capsules were measured by

irradiating empty foil capsules in the blanket assembly. The activities of the polyethylene foil capsules were found to be negligibly small in comparison with the activities of the detector foil materials.

4.2.3.2 Threshold Reaction Rate Measurement

Indium Foil Counting. The same procedure was used as described in Section 3.2.3.

U²³⁸ Fission Product Counting. The U²³⁸ fission reaction counting rate was measured by taking an integral count of the γ -ray activity of the fission products above 0.72 MeV. This base line setting was chosen to eliminate the maximum β -ray bremsstrahlung energy of 0.72 MeV from Np²³⁹, which decays to Pu²³⁹ with a 2.33-day half-life.

It was necessary to have about four hours cooling time (after the completion of blanket irradiation) before fission product counting began. The time delay was necessary to let the 23-minute half-life U²³⁹ decay almost completely to Np²³⁹. This was important because, in its decay, U²³⁹ emits a 1.2-MeV β -ray; the cooling period prevented the possible inclusion of the 1.2-MeV β -ray bremsstrahlung radiation in the fission product counting. The fission product counting was accomplished as soon as possible after this cooling period. The approximately 2.5-hour mean half-life of the fission product activity limited the fission product counting to about 8 hours after irradiation, after which the activity became too small for obtaining statistically meaningful results.

4.2.3.3 Fission Reaction Rate Measurements

U²³⁵ and Pu²³⁹ Fission Product Counting. The U²³⁵ and Pu²³⁹ fission reaction rates were determined by counting the irradiated U²³⁵ and Pu²³⁹ foils using gross gamma counts in the window between 0.5 MeV

and 1.0 MeV. (The discriminator settings of the counter are given in Table 4.3). Because of the extremely high residual background of the Pu^{239} foils (in the form of low energy gammas), a 3/8 inch thick lead filter was placed between Pu^{239} foils and the detector in order to reduce the count rate to levels which would not cause saturation of the counting system ($\leq 200,000$ cpm).

The foils were also counted for the fission product pair Ba^{140} - La^{140} to confirm the results of the gross fission rate measurements. The 1.60 MeV. gamma of La^{140} was detected, using a 120 KeV window to straddle the peak. (The discriminator settings of the analyzer were: $E_{\min} = 3.80$ volts, $E_{\max} = 4.10$ volts.)

Table 4.3 summarizes the counting characteristics of the foil materials used in the reaction rate measurements; the gamma energies detected, discriminator settings of the analyzer, typical counts accumulated, counting time and irradiation time.

4.2.4 Corrected Activities

The "raw" counts were corrected for counter dead-time, background activities, weights and sample decay time as described in Section 3.2.4. The corrected activities of the axial (Z-direction) reaction rate measurements as functions of position (distance from the converter-blanket interface) are given in Table 4.4 and 4.5. Typically, over 10,000 counts were accumulated to ensure a statistical precision of better than $\pm 1\%$. At deep penetrations, this error was sometimes exceeded. The errors shown in Table 4.4 were attributed to uncertainties in both counting statistics and foil weights (Appendix E).

TABLE 4.3. Typical Data Pertinent to Foil Counting

Reaction	Gamma Energy Detected (MeV)	Discriminator Settings (volts)*		Typical Counts Accumulated (Background subtracted)	Counting Time (minutes)	Typical Irradiation Time (hours)
		E _{min}	E _{max}			
<u>CAPTURE</u>						
Au ¹⁹⁷ (n, γ)Au ¹⁹⁸	0.412	0.90	1.25	24,700	10	6
Mo ⁹⁸ (n, γ)Mo ⁹⁹	0.78	1.75	2.15	14,413	2	4
U ²³⁸ (n, γ)	0.103	0.23	0.33	43,703	1	5½
Na ²³ (n, γ)Na ²⁴	1.38	3.15	3.65	10,749	3	16
Cr ⁵⁰ (n γ)Cr ⁵¹	0.32	0.70	0.95	10,394	10	19
<u>THRESHOLD</u>						
In ¹¹⁵ (n, n')In ^{115m}	0.335	0.70	1.10	44,690	20	4
U ²³⁸ (n, f)	0.72 to ∞	1.92	∞	37,880	20	5½
<u>FISSION</u>						
U ²³⁵ (n, f)	0.5 to 1.00	1.14	2.30	258,094	1	5
Pu ²³⁹ (n, f)	0.5 to 1.00	1.14	2.30	301,683	1	5

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

TABLE 4.4. Results of Axial (Z) Reaction Rate Traverses

	POSITION (from blanket- converter interface) cm	REACTION RATES			
		$\text{Au}^{197}(\text{n}, \gamma)\text{Au}^{198}$ cpm/mg	$\text{In}^{115}(\text{n}, \text{n}')\text{In}^{115\text{m}}$ cpm/mg	$\text{U}^{238}(\text{n}, \gamma)$ cpm/mg	$\text{U}^{238}(\text{n}, \text{f})$ cpm/mg
Blanket Region	2.34	1508.5 ± 10.4	834.6 ± 4.9	22446 ± 73	315.1 ± 3.8
	7.50	1297.0 ± 9.2	569.6 ± 4.1	18709 ± 67	208.3 ± 3.3
	12.70	1174.2 ± 9.2	384.3 ± 3.5	16308 ± 63	134.7 ± 2.8
	17.40	978.6 ± 8.4	268.7 ± 2.9	12932 ± 56	94.6 ± 2.6
	22.60	791.7 ± 7.6	191.6 ± 2.4	10283 ± 51	65.3 ± 2.4
	27.80	673.7 ± 6.8	140.4 ± 2.0	8541 ± 47	42.0 ± 2.2
	32.60	538.9 ± 6.0	101.2 ± 1.8	7126 ± 43	20.4 ± 2.1
	37.60	458.4 ± 5.6	76.5 ± 1.5	5509 ± 38	17.0 ± 2.1
	42.80	414.5 ± 5.2	61.5 ± 1.4	5103 ± 37	15.2 ± 2.0
Reflector Region	48.2	508.8 ± 5.6	45.46 ± 0.84	8979 ± 26	12.9 ± 1.8
	50.8	520.2 ± 5.6	45.10 ± 0.84	8337 ± 25	—
	52.3	508.3 ± 5.6	41.04 ± 0.72	8624 ± 25	9.15 ± 1.6
	58.4	426.5 ± 5.2	28.82 ± 0.48	6870 ± 23	7.2 ± 1.3
	63.5	342.4 ± 4.8	23.69 ± 0.40	5180 ± 19	6.9 ± 1.3
	68.5	263.9 ± 4.0	19.02 ± 0.36	3808 ± 17	2.88 ± 0.52
	73.6	197.7 ± 3.6	14.57 ± 0.32	2816 ± 14	1.76 ± 0.54
	78.7	149.1 ± 3.2	11.00 ± 0.28	2215 ± 13	0.72 ± 0.6
	83.8	113.7 ± 2.8	8.19 ± 0.24	1649 ± 11	—
	88.9	81.9 ± 2.4	6.50 ± 0.24	1178 ± 10	—

TABLE 4.5. Additional Axial (Z) Reaction Rate Traverses

POSITION (from blanket- converter interface) cm	REACTION RATES				
	Mo ⁹⁸ (n, γ) cpm/mg	Cr ⁵⁰ (n, γ) cpm/mg	Na ²³ (n, γ) cpm/mg	U ²³⁵ (n, f) cpm/mg	Pu ²³⁹ (n, f) cpm/mg
2.34	133.1 $\pm$ 2.5	149.8 $\pm$ 1.1	131.9 $\pm$ 0.8	8547.8 $\pm$ 92.4	378.6 $\pm$ 5.3
7.50	120.0 $\pm$ 2.6	125.6 $\pm$ 1.0	111.3 $\pm$ 0.8	7394.9 $\pm$ 86.0	327.8 $\pm$ 4.9
12.70	108.4 $\pm$ 2.3	105.3 $\pm$ 1.0	87.7 $\pm$ 0.7	6311.4 $\pm$ 79.4	263.9 $\pm$ 4.0
17.40	86.8 $\pm$ 2.1	86.7 $\pm$ 0.8	74.8 $\pm$ 0.6	5216.3 $\pm$ 72.2	200.3 $\pm$ 3.2
22.60	74.1 $\pm$ 2.1	67.2 $\pm$ 0.8	58.9 $\pm$ 0.6	4298.0 $\pm$ 65.5	175.8 $\pm$ 2.8
27.80	59.3 $\pm$ 1.8	57.1 $\pm$ 0.7	48.2 $\pm$ 0.5	3461.0 $\pm$ 58.8	143.0 $\pm$ 2.4
32.60	46.2 $\pm$ 1.8	44.3 $\pm$ 0.6	38.5 $\pm$ 0.4	2820.9 $\pm$ 53.1	112.9 $\pm$ 1.9
37.60	34.5 $\pm$ 1.7	36.7 $\pm$ 0.6	29.8 $\pm$ 0.4	2280.8 $\pm$ 47.7	83.7 $\pm$ 1.5
42.80	30.8 $\pm$ 1.5	32.2 $\pm$ 0.5	26.5 $\pm$ 0.4	1999.1 $\pm$ 44.7	72.0 $\pm$ 1.3

4.2.5 Sources of Errors in Foil Activation

Errors in the measured activity of foils such as those used in this work may arise from the following effects.

- 1) Gamma attenuation: attenuation of the photons from the foil due to absorption in the foil material.
- 2) Flux depression: depression of the neutron flux outside the detector due to absorption.
- 3) Self-shielding: reduction of flux inside the detector due to neutron absorption at the surface.

Calculations for the self-attenuation of the gamma rays in the foil materials (E3) showed that the extent of these effects was negligible: the errors introduced were less than 0.1%. In addition, the error should also be systematic and should thus cancel out for relative reaction rate traverses.

In the energy range of interest, the neutron absorption mean free paths (typically a few centimeters) were much longer than the foil thickness (0.051 cm) and the diameter of the foil capsules (0.625 cm) in the case of the powdered foils. Thus flux depression and self-shielding could also be neglected. This was confirmed by the following experiment. Varying thicknesses of several foil materials (gold, molybdenum and depleted uranium) were irradiated simultaneously in adjacent, equivalent positions in the blanket and the activities per gram were plotted as a function of the foil thickness (Fig. 4.2). The foil self-shielding correction factor for foil material of thickness, t , would be the ratio of its foil activity to that of "zero" thickness (obtained by extrapolation). It was found that in the worst case, the measured activity per gram was only decreased by 1.5%, even when the foil thickness was increased to 50 mils.

FIG. 4.2 SENSITIVITY OF GAMMA ACTIVITY TO FOIL THICKNESS.

This error lies within the statistical errors of most foil activities. Hence, it was concluded that the self-absorption correction was not necessary. Note, however, that this refers only to U^{238} foils irradiated on steel foil-holder rods: the activity of U^{238} foils in fuel rods is appreciably depressed by self-shielding, as described in Section 4.4.

The perturbation of the foil activities caused by the effect of the iron rod foil holders was studied. This effect was checked by measuring the activities of thin gold and indium foils (each 5 mils thick) placed on the surfaces of two different foil holders: a 5/16-inch-O.D., iron rod and a uranium fuel rod. The differences between these activities using different foil holders were observed to be less than 0.7%. The use of iron foil holders, therefore, did not cause significant errors in the measurement of foil activities.

Another source of error came from the perturbation caused by the Mylar tape or polyethylene vials which were employed to hold the foil materials. The effect of Mylar tape on foil activities of several materials — e.g., gold, indium, molybdenum, and uranium-238 — was studied. It was found, in general, that the foil activities were not affected when the Mylar tape used to hold the foil material was varied from 1 mil to 10 mils in thickness. However, the effect of the polyethylene vials on foil activity was measurable. The activities of Na^{24} and Cr^{51} decreased by about 2% when aluminum capsules were used instead of polyethylene vials. This showed that the presence of the polyethylene vials did actually soften the neutron spectrum in the neighborhood of the foil detector. The activity data shown in Table 4.5 and Fig. 4.4 have been corrected for the softening effect caused by the polyethylene vials.

4.3 EXPERIMENTAL RESULTS

Results of the axial traverses of reaction rate measurements in the BTF Blanket Mock-Up No. 2 have been given in Tables 4.4 and 4.5. Figures 4.3 to 4.7 show the relative measured reaction rate distributions compared with theoretical predictions. The theoretical calculations were made with the one-dimensional transport code, ANISN, S_8 option, using the 26-group Russian ABBN cross-section set. The effect of U^{238} spatial self-shielding in the converter plate and blanket fuel rods has been taken into account. Broad group cross sections for U^{238} , which account for resonance self-shielding, have been generated with the IDIOT program (I1) using the resonance data from Schmidt (S3). These cross sections were then incorporated into the 26-group ABBN set.

For the sake of comparison, it is found convenient to normalize both the experimental and calculated results to unity at the center of the Blanket Mock-Up, i.e., at $Z = 8.88$ inches (22.6 cm). The general features of each reaction rate traverse will first be discussed. This discussion will be followed by an attempt to explain the discrepancies between experimental and theoretical results.

Figure 4.3 shows the axial capture rate distribution of $U^{238}(n, \gamma)$ and $Au^{197}(n, \gamma)$ in the blanket and reflector regions. In general, the shape of the two capture rate traverses agrees qualitatively with that predicted by theory. However, the theory predicts a higher reaction rate deep within the blanket, near the blanket-reflector interface, than is found experimentally. Some of the general disagreement between experimental and theoretical results near and in the reflector may, perhaps, be due to the well-known difficulties which come up in few-group descriptions of keV-range neutron diffusion in iron; this and other contributions to the

FIG. 4.3 AXIAL CAPTURE RATE DISTRIBUTIONS.

discrepancies will be discussed in more detail subsequently.

Figure 4.4 shows the relative capture rate of chromium, molybdenum and sodium in the blanket region. One common feature in all three traverses (plus the two in the preceding figure) is that the slope of the measured activity profile is, in general, not as steep as that of the calculated profile.

Figures 4.5 and 4.6 show the relative axial threshold reaction rate distributions of $\text{In}^{115}(\text{n}, \text{n}')\text{In}^{115\text{m}}$ and $\text{U}^{238}(\text{n}, \text{f})$, respectively. The $\text{In}^{115}(\text{n}, \text{n}')\text{In}^{115\text{m}}$ reaction has a threshold energy of about 300 keV, while the $\text{U}^{238}(\text{n}, \text{f})$ reaction has a threshold energy of about 1 MeV. In general, the experimental threshold reaction rate traverses for both the $\text{U}^{238}(\text{n}, \text{f})$ and $\text{In}^{115}(\text{n}, \text{n}')\text{In}^{115\text{m}}$ reactions are in fair agreement with the theoretical predictions for almost the entire region of the blanket. However, in the reflector region, the measured activities of the $\text{In}^{115}(\text{n}, \text{n}')\text{In}^{115\text{m}}$ and $\text{U}^{238}(\text{n}, \text{f})$ reactions become much higher than the theoretical values. In the $\text{In}^{115}(\text{n}, \text{n}')\text{In}^{115\text{m}}$ traverse, the possibility of In^{116} interference (see Section 3.2.3) is not sufficient to explain the large discrepancies between experimental and theoretical results in the reflector region.

In order to investigate the large discrepancies between calculated and measured threshold traverses in the reflector region, sensitivity studies of the effect on the threshold reaction rate due to iron density uncertainties and outer boundary conditions have been performed. The change in the ANISN-calculated threshold reaction rates is less than $\pm 1\%$ even when the iron density in the reflector is varied from $+5\%$ to -5% . Also, the effect of the presence of the BTF's concrete shield doors upon reflector activation is found to be negligible, except in the rear 15 to

FIG.4.4 ADDITIONAL AXIAL CAPTURE RATE DISTRIBUTIONS.

FIG. 4.5 AXIAL THRESHOLD REACTION RATE DISTRIBUTION,
 $^{115}\text{In}(N,N')$.

FIG. 4.6 AXIAL THRESHOLD REACTION RATE DISTRIBUTION,
238U(N,F).

20 cm of the steel reflector. Thus, the poor agreement between experimental and theoretical reaction rate distributions in the reflector is not likely due to the iron density uncertainties or to the choice of outer boundary conditions. The most likely explanation is that the first-flight neutrons which contribute most of the threshold foil activation, are just not well described by a one-dimensional computational model employing an axially uniform buckling to account for transverse leakage.

A comparison of the calculated and measured U^{235} and Pu^{239} axial fission distributions in the Blanket Mock-Up is shown in Fig. 4.7. The data on the fission rate traverses are consistent with those of the capture and threshold reaction rates in that the slope from the measured activity traverses are, in general, not as steep as calculated; although systematic, the differences are not great.

There are at least two major factors which affect significantly the calculations of theoretical reaction rate distributions in the BTF Blanket Mock-Up No. 2. The first is the self-shielding effect of U^{238} capture in both the converter and blanket assemblies. Self-shielded cross sections for U^{238} have been used in the above theoretical calculations. It will be shown in Section 4.4 that the use of unshielded U^{238} cross sections can affect the reaction rate calculations substantially. Furthermore, the effect of using different cross-section weighting spectra can also be significant in the calculation of reaction rate distributions. Throughout our calculations so far, the 26-group Russian cross-section set, which has a $1/E$ weighting spectrum in the keV and eV regions, has been used. The influence of weighting spectra on reaction rate calculations is discussed further in Section 4.5.

FIG. 4.7 AXIAL FISSION RATE DISTRIBUTIONS.

4.4 EFFECT OF SELF-SHIELDED U^{238} CROSS SECTIONS ON REACTION RATE CALCULATIONS

A study has been made of the effect of U^{238} spatial self-shielding in the converter plate and blanket fuel rods. Table 4.6 shows the comparison between the U^{238} self-shielded and unshielded cross sections. In the previous section, all the calculations were made with the shielded cross sections. In order to see the effect of self-shielded U^{238} cross sections on reaction rate calculations, the unshielded U^{238} cross-section set was used in the one-dimensional ANISN code, S_8 option, to recalculate some typical reaction rate traverses. Figures 4.8 and 4.9 show, respectively, the $U^{238}(n, \gamma)$, $Au^{197}(n, \gamma)$ and $Pu^{239}(n, f)$ reaction rate distributions calculated by using both the self-shielded and unshielded U^{238} cross sections. Experimental data are also shown in the figures for comparison. It can be seen that the agreement between experimental and theoretical results is considerably improved when self-shielded U^{238} cross sections are used, especially in the case of $Pu^{239}(n, f)$.

It is important to note that the corrections discussed so far are for self-shielding in the uranium fuel rods used in both the converter and blanket assemblies. The experiments cited in Section 4.3 have shown that U^{238} foils on the steel holders do not shield themselves. Thus, in calculating the activities shown in Fig. 4.8, shielded cross sections are used for flux calculations but unshielded cross sections are used for the calculation of foil activities. In order to illuminate these differences, out-of-rod vs. in-rod U^{238} capture traverses were performed in Blanket Mock-Up No. 2. Figure 4.10 shows both the experimental and theoretical results for the two cases. The unshielded data (foils outside fuel rods)

TABLE 4.6
Comparison of U^{238} Unshielded and Shielded Cross Sections

Group No.	E_L	UNSHIELDED			SELF-SHIELDED		
		σ_a	σ_{tr}	$\sigma_{i \rightarrow i}$	σ_a	σ_{tr}	$\sigma_{i \rightarrow i}$
11	10.0 keV	0.75	13.882	12.989	0.485	12.402	11.726
12	4.65 "	0.78	15.35	14.412	0.766	15.234	14.318
13	2.15 "	1.2	16.50	15.133	0.605	12.809	12.043
14	1.0 "	2.1	18.00	15.726	0.651	11.835	11.017
15	465 eV	3.6	23.00	19.188	0.710	11.815	10.931
16	215.0 "	4.5	18.50	13.847	0.771	10.732	9.749
17	100.0 "	17.0	80.00	62.313	1.273	14.328	12.402
18	46.5 "	15.0	40.00	24.727	3.498	16.777	12.592
19	21.5 "	58.0	140.00	81.105	3.136	16.215	12.806
20	10.0 "	82.0	120.00	37.585	7.904	20.466	11.667
21	4.65 "	171.0	190.00	18.793	8.745	18.568	9.408

FIG. 4.8 EFFECT OF ^{238}U SELF-SHIELDED CROSS SECTIONS
ON REACTION RATE CALCULATIONS.

FIG. 4.9 EFFECT OF USING 238-U SELF-SHIELDED CROSS-SECTIONS ON 239-PU FISSION RATE CALCULATIONS.

FIG. 4.10 COMPARISON OF SHIELDED (IN-ROD) AND UNSHIELDED (EX-ROD) ^{238}U FOIL TRAVERSES.

are normalized to 1.0 at the center of the blanket. In general, the slope of the shielded traverse (foils inside fuel rods) is steeper than that of the unshielded case as predicted by theory. Moreover, the U^{238} capture rates of the shielded foils are a factor of approximately 0.6 less than the activities of the unshielded foils. Hence, the unshielded foil results will overestimate the breeding rates in the blanket, and only the shielded U^{238} foil activation results give the "actual" breeding rates.

4.5 THE INFLUENCE OF THE WEIGHTING SPECTRUM

In fast reactor calculations, it is well known that there can be large errors in the low energy flux tail due to errors in elastic group removal cross sections caused by the use of a weighting spectrum for generation of the cross sections which is not sufficiently similar to the actual local spectrum (K1). Possible methods for alleviating these difficulties are discussed later. A prescription suggested for further investigation is discussed in Appendix C, Section C.1.

In order to assess the magnitude of this contribution to the discrepancies between the experimental and theoretical traverses, two modified weighting schemes for the elastic downscattering cross sections of the Russian set were tested. The two weighting schemes are:

- a) weighting over a $\phi(u) \propto E$ spectrum,
- b) weighting over the calculated spectrum at the center of Blanket Mock-Up No. 2.

These weightings are both harder than the built-in Russian set weighting ($\phi(u) = \text{constant}$) which is much too soft even for an LMFBR blanket.

The formulae for correcting the downscatter cross sections for the two weighting schemes are given in Appendix C, Section C.2. Two sets of

modified cross sections were generated for the two different weighting spectra. ANISN runs for the BTF Blanket No. 2 were then made with these two modified cross-section sets. In addition, self-shielded cross sections for U^{238} were employed. Figure 4.11 shows the $Au^{197}(n, \gamma)$ and $U^{238}(n, \gamma)$ reaction rate distributions calculated with the two different spectrum-weighted cross-section sets, and with the original Russian set which has a $\phi(u) = \text{constant}$ weighting.

The following general conclusions may be drawn:

- a) Modifying the elastic downscatter weighting produces significant changes in the calculated spectra, reflected here in the actual traverses.
- b) The changes are somewhat smaller than those effected by changing from unshielded to shielded U^{238} cross sections.
- c) No one weighting scheme best fits the experimental data over the entire blanket and reflector. However, as expected, the harder than $1/E$ spectra tend to show a trend toward improved agreement, particularly in the case of gold.

Of course, even better agreement between experimental and theoretical results can be achieved by employing more sophisticated calculation methods, such as using more energy groups and more blanket zones, each having its own cross-section set, but these are beyond the scope of the present work and will not be pursued further here.

FIG.4.11 EFFECT OF DIFFERENT WEIGHTING SPECTRA ON REACTION RATE CALCULATIONS.

4.6 NEUTRON BALANCE

4.6.1 Introduction

In previous sections, reaction rate distributions in the BTF Blanket Mock-Up No. 2 for a number of different materials have been discussed. In economic studies, one is more interested in the total absorption, fission and leakage rates integrated over the whole blanket, or over a subregion. It is therefore of considerable interest to establish a neutron balance for the entire blanket mock-up to account for the fraction of neutrons lost through leakage and absorption, and the contribution produced by fission, and also to assess the accuracy to which the dominant reaction rates can be calculated and measured.

The calculations establishing such a neutron balance in Blanket Mock-Up No. 2 are discussed in Section 4.6.2. Wherever possible, the predicted values in the blanket neutron balance table are checked against experimental measurements. The experimental method of verification is given in Section 4.6.3. Finally, comparisons of the experimental and theoretical results of the blanket neutron balance are discussed in Section 4.6.4.

4.6.2 Theoretical Calculations

The total reaction rate, $R_{\alpha,\beta}$, incurred in the whole blanket by an isotope α , undergoing reaction type β , is given by the volume integral:

$$R_{\alpha,\beta} = \int_{\text{vol}} \int_0^{\infty} N_{\alpha}(x, y, z) \sigma_{\alpha,\beta}(E) \phi(E, x, y, z) dV dE \quad (4.1)$$

where $N_{\alpha}(x, y, z)$ = atom density of material at point x, y, z

$\sigma_{\alpha,\beta}(E)$ = reaction cross section of nuclide α for reaction type β

$\phi(E, x, y, z)$ = flux per unit energy at point x, y, z .

It is assumed that the nuclide concentrations are homogenized,

$N_\alpha(x, y, z) = N_\alpha$, and the analysis of the blanket is one-dimensional in the z-direction, $dV = dz A$, A being the cross-sectional area of the blanket.

Then,

$$R_{\alpha, \beta} = N_\alpha A \int_{z=0}^L \int_0^\infty \sigma_{\alpha, \beta}(E) \phi(E, z) dE dz \quad (4.2)$$

or, in summation form,

$$R_{\alpha, \beta} = N_\alpha A \sum_{k=0}^M \sum_{i=1}^g \sigma_{\alpha, \beta, i} \phi_{k, i} \Delta Z_k \Delta E_i \quad (4.3)$$

where L = thickness of blanket

M = number of mesh points

g = number of energy groups.

Since U^{238} capture is the major reaction occurring in the blanket, it is convenient to normalize all the other reactions to U^{238} capture (UC).

$$\frac{R_{\alpha, \beta}}{R_{UC}} = \frac{N_\alpha}{N_{U^{238}}} \frac{\sum_{k=0}^M \sum_{i=1}^g \sigma_{\alpha, \beta, i} \phi_{k, i} \Delta Z_k \Delta E_i}{\sum_{k=0}^M \sum_{i=1}^g \sigma_{UC, i} \phi_{k, i} \Delta Z_k \Delta E_i} \quad (4.4)$$

The total reaction rates of $U^{238}(n, \gamma)$, $Fe(n, \gamma)$, $Na(n, \gamma)$, $Cr(n, \gamma)$, $U^{235}(n, \gamma)$, $U^{238}(n, f)$ and $U^{235}(n, f)$ were calculated by using the one-dimensional transport code, ANISN, S_8 option, and the 26-group ABBN Russian cross-section set. In addition, the in-leakage from the converter and the out-leakage to the reflector were calculated by means of the same code. The transverse leakage from the blanket was obtained by calculating the term $\bar{D} B^2 \phi$ using an average $\bar{D}$ value of 1.33 cm

($\bar{D} = (\sum_{i=1}^g \frac{1}{3(\Sigma_{tr})_i} \phi_i) / \sum_{i=1}^g \phi_i$), $B^2 = 0.000704 \text{ cm}^{-2}$ (result from Chap. 3), and the total blanket flux $\phi = 1.218 \times 10^{10} \text{ n/cm}^2 \text{ sec}$ (corresponding to an ANISN normalization of a $1.22 \times 10^9 \text{ n/cm}^2 \text{ sec}$ thermal neutron flux incident on the converter assembly). All the calculations were then normalized to an integral U^{238} capture reaction rate set equal to unity, Eq. 4.4. Table 4.8 shows the summary of the blanket neutron balance results.

4.6.3 Experimental Verification

The calculated values for some of the total reaction rates in the blanket neutron balance have been checked in measurements made with the following method.

The reaction rate occurring in the entire blanket $R_{\alpha, \beta}$ is given by:

$$R_{\alpha, \beta} = R_{o, \alpha, \beta} \int_0^L f_{\alpha, \beta}(z) dz \quad (4.5)$$

where $R_{o, \alpha, \beta}$ = reaction rate at the center of the blanket

$f_{\alpha, \beta}(z)$ = reaction rate distribution function

L = thickness of blanket.

The integral $\int_0^L f_{\alpha, \beta}(z) dz$ can be obtained by numerical integration of the reaction rate distribution function $f_{\alpha, \beta}(z)$ as shown in Figs. 4.3 to 4.7, or by calculation with the ANISN code. The results are tabulated in Table 4.7. The reaction rate at the center of the blanket $R_{o, \alpha, \beta}$ can be obtained by measuring the corresponding activities of irradiated detector foils in the center of the blanket.

$$R_{o, \alpha, \beta} = N_{\alpha} \frac{A_{o, \alpha, \beta}^{bkt}}{(F\epsilon)} \quad (4.6)$$

where N_α = atom density of isotope α

$A_{o,\alpha,\beta}^{bkt}$ = activity of foil α , β the reaction type, at center of blanket

F = fraction of the total activation detector used for activity determination

ϵ = overall efficiency of the counting system.

The quantity $(1/F\epsilon)$ can be evaluated using calibration experiments.

This was done by irradiating the detector foils in the hohlraum of the MITR (2CH1 facility), which has a pure thermal spectrum. Then,

$$\frac{A_{\alpha,\beta}^{th}}{F\epsilon} = \sigma_{\alpha,\beta}^{th} \phi_{HOL}^{th} \quad (4.7)$$

where $A_{\alpha,\beta}^{th}$ = activity of foil produced by the thermal spectrum, measured using procedures identical for the BTF-irradiated foils

$\sigma_{\alpha,\beta}^{th}$ = thermal cross section of foil α , of the reaction type β

ϕ_{HOL}^{th} = thermal flux in the hohlraum.

Combining Eqs. 4.5, 4.6 and 4.7,

$$R_{\alpha,\beta} = \frac{A_{o,\alpha,\beta}^{bkt}}{A_{\alpha,\beta}^{th}} N_\alpha \sigma_{\alpha,\beta}^{th} \phi_{HOL}^{th} \int_0^L f_{\alpha,\beta}(z) dz \quad (4.8)$$

For the U^{238} capture, in particular,

$$R_{UC} = \frac{A_{o,UC}^{bkt}}{A_{UC}^{th}} N_{U^{238}} \sigma_{UC}^{th} \phi_{HOL}^{th} \int_0^L f_{UC}(z) dz \quad (4.9)$$

The ratio of reaction rates with respect to U^{238} capture, $R_{\alpha,\beta}/R_{UC}$, is then obtained by combining Eqs. 4.8 and 4.9.

$$\frac{R_{\alpha, \beta}}{R_{UC}} = \frac{N_{\alpha}}{N_{U^{238}}} \frac{A_{o, \alpha, \beta}^{bkt}}{A_{o, UC}^{bkt}} \frac{\sigma_{\alpha, \beta}^{th}}{\sigma_{UC}^{th}} \frac{A_{UC}^{th}}{A_{\alpha, \beta}^{th}} \frac{\int_0^L f_{\alpha, \beta}(z) dz}{\int_0^L f_{UC}(z) dz} \quad (4.10)$$

Table 4.7 lists all the measured parameters $A_{o, \alpha, \beta}^{bkt}$, $A_{\alpha, \beta}^{th}$ and other quantities necessary to obtain $R_{\alpha, \beta}/R_{UC}$ by Eq. 4.10.

4.6.4 Results of Blanket Neutron Balance

Table 4.8 summarizes both the theoretical and experimental results for the neutron balance in BTF Blanket Mock-Up No. 2. The reaction rate of U^{238} capture is again normalized to unity. The theoretical predictions of sodium and chromium capture reaction rates and of U^{235} , U^{238} fission rates can be compared with experimental measurements. The experimental results for the neutron balance (Table 4.8) are subject to large errors ($> 5\%$ in all cases), due to the uncertainties in $\sigma_{\alpha, \beta}^{th}$ and the statistical uncertainties in the measured activities of $A_{o, \alpha, \beta}^{bkt}$ and $A_{\alpha, \beta}^{th}$ (see Appendix E). However, the experimental and theoretical results agree with one another within these experimental uncertainties.

The equation for the neutron balance in the blanket has the following form:

Production of neutrons = Loss of neutrons

In-leakage + Neutrons produced by fission = Neutrons lost by absorption + Out-leakage + Transverse leakage

Substitution of the theoretical values from Table 4.7 gives:

$$\underbrace{1.212 + 0.267}_{1.479} = \underbrace{1.1796 + 0.0513 + 0.220}_{1.451}$$

TABLE 4.7. Listing of Parameters Used to Obtain $R_{\alpha, \beta}/R_{UC}$

Reaction Rate, $R_{\alpha, \beta}$	N_{α} (10^{24} nuclei/cc)	$\frac{\int_0^L f_{\alpha, \beta}(z) dz}{\int_0^L f_{UC}(z) dz}$	$A_{o, \alpha, \beta}^{bkt}$ (cpm/mg)	$A_{\alpha, \beta}^{th}$ (cpm/mg)	$\sigma_{\alpha, \beta}^{th}$ (At 2200 m/sec)	$\frac{R_{\alpha, \beta}}{R_{UC}}$
$U^{238}(n, \gamma)$ (UC)	8.108×10^{-3}	1.00	10282.5 ± 50.7	7377 ± 80	2.76×0.09	1.000
$Na(n, \gamma)$	8.128×10^{-3}	1.065	58.91 ± 0.57	3804 ± 40	0.505 ± 0.01	0.0022 ± 0.0002
$Cr(n, \gamma)$	4.064×10^{-3}	1.035	67.2 ± 0.8	1985 ± 30	16.3 ± 1.3	0.074 ± 0.009
$U^{235}(n, f)$	8.8×10^{-5}	1.050	4298.0 ± 5	$112,419 \pm 100$	582 ± 10	0.066 ± 0.007
$U^{238}(n, f)$	8.108×10^{-3}	1.659	65.3 ± 2.4	0.336 ± 0.008	$<0.5 \times 10^{-3}$	0.042 ± 0.008

*See reference (E2).

TABLE 4.8. Neutron Balance for BTF Blanket Mock-Up No. 2

Reaction/Leakage Rate	Theoretical	Experimental
$U^{238}(n, \gamma)$	1.0000	1.00
$Fe(n, \gamma)$	0.0036	---
$Na(n, \gamma)$	0.0025	0.0022 ± 0.0002
$Cr(n, \gamma)$	0.0650	0.074 ± 0.006
$U^{235}(n, \gamma)$	0.0016	---
Total capture (c)	1.0727	
$U^{238}(n, f)$	0.0485	0.042 ± 0.003
$U^{235}(n, f)$	0.0584	0.066 ± 0.004
Total fission (f)	<u>0.1069</u>	
Total absorption (c+f)	1.1796	
In-leakage	1.2120	
Out-leakage	0.05134	
Transverse leakage ($\bar{D}B^2\phi$)	0.220	

The fission rate term has been calculated by assuming an average value of $\bar{\nu} = 2.50$, and the calculated fission rate of 0.1069 listed in Table 4.8. All other terms of the neutron balance equation are obtained directly from Table 4.8. The numerical values on the two sides of the neutron balance equation agree to within 2.0%, which confirms the calculated neutron balance in BTF Blanket Mock-Up No. 2. The error of 2.0% can be attributed mainly to the uncertainty in the estimation of the effective one-group values for the transverse leakage and the value of $\bar{\nu}$. It is also interesting to note that the number of neutrons produced by fission in the blanket is approximately equal to that lost by transverse leakage, and out-leakage. The in-leakage neutrons from the converter equal, approximately, the neutrons lost by absorption in the blanket.

4.7 COMPARISON OF BTF BLANKET NO. 2 SPECTRAL INDICES TO ZPPR

Since spectral indices can provide a useful characterization of ambient spectra, the comparison of spectral indices between BTF and ZPPR blankets (D1) is an important step in assessing whether there are any major spectral differences between blankets of the present BTF mock-up and of the realistic benchmark critical assemblies. The fission ratios of U^{235}/Pu^{239} , U^{238}/Pu^{239} , and U^{238}/U^{235} are important spectral indices available for comparison.

The reaction ratio at Z cm into the BTF Blanket can be measured in the following way. Equation 4.8 in Section 4.6 can be written as

$$\frac{R_{\alpha, \beta}(z)}{N_{\alpha}} = \frac{A_{o, \beta}^{bkt} f_{\alpha, \beta}(z)}{A_{\alpha, \beta}^{th}} \sigma_{\alpha, \beta}^{th} \phi_{HOL}^{th} \quad (4.11)$$

Let R_1 and R_2 be two different reactions. The reaction ratio R_1/R_2 can be written as follows:

$$\frac{R_1}{R_2} = \frac{N_1}{N_2} \frac{\left[A_o^{bkt} f(z) \right]_1}{\left[A_o^{bkt} f(z) \right]_2} \frac{A_1^{th} \sigma_2^{th}}{A_2^{th} \sigma_1^{th}}$$

The measured values of A_o^{bkt} , A^{th} , and σ^{th} for U^{238} , U^{235} , and Pu^{239} fission are listed in Table 4.7. $f(z)$ for U^{238} , U^{235} , and Pu^{239} fissions are obtained from Figs. 4.3 and 4.7, respectively.

The fission ratios at different positions of the BTF Blanket Mock-Up are to be compared with the result obtained from the blanket of a typical fast critical assembly, ZPPR-2. First, however, some comment is in order on how representative fission ratios change in going from core to blanket. Table 4.9 shows the measured fission ratios of Blanket Mock-Up No. 2 at 7.5 cm from the blanket-converter interface compared with those of several critical assemblies at core center (P3). It is seen that the fission ratio of U^{235} and Pu^{239} in the blanket mock-up does not differ too much from that obtained from the core center of typical critical assemblies. However, the fission ratio of U^{238} to Pu^{239} in the blanket mock-up is a factor of two or three times less than in the critical assembly cores: in this respect, it should be considered a far more sensitive criterion of similitude.

In Table 4.10, the experimental and theoretical fission ratios of U^{238}/U^{235} , U^{238}/Pu^{239} , and Pu^{239}/U^{235} at two different positions in the BTF Blanket Mock-Up No. 2 are compared with the corresponding ratios in the ZPPR blanket. The ZPPR blanket measurements have been reported in Table VII of reference (Z2). The theoretical results for the BTF Blanket were obtained from the one-dimensional ANISN, S_8

TABLE 4.9. Comparison of Fission Ratios Between BTF Blanket Mock-Up No. 2 and Fast Critical Assemblies

Fission Ratio	ZPPR-2	ZPR-3-48	ZPR-6-7	BTF Blanket Mock-Up No. 2
	CENTER OF CORE			7.5 cm from blanket- converter interface
U^{235}/Pu^{239}	1.067	1.025	1.047	1.016
U^{238}/Pu^{239}	0.0215	0.0315	0.0216	0.0083

TABLE 4.10. Comparison of Fission Ratios Between ZPPR and BTF Blankets

Fission Ratio	Distance from Core-(Converter)-Blanket Interface (cm)	ZPPR-2 (Z2) Experimental	BLANKET MOCK-UP NO. 2		
			Experimental	Theoretical	
				Shielded U ²³⁸ Cross Section	Unshielded U ²³⁸ Cross Section
U ²³⁸ (n, f)	7.5	0.0093 ± 2.1%	0.0081 ± 6%	0.0070	0.0091
U ²³⁵ (n, f)	18.9	0.0047 ± 2.1%	0.0046 ± 6%	0.0042	0.0054
U ²³⁸ (n, f)	7.5	0.0102 ± 4.2%	0.0083 ± 6%	0.0082	0.0103
Pu ²³⁹ (n, f)	18.9	0.0052 ± 4.2%	0.0053 ± 6%	0.0051	0.0065
Pu ²³⁹ (n, f)	7.5	0.9088 ± 2.1%	0.984 ± 6%	0.854	0.883
U ²³⁵ (n, f)	18.9	0.8969 ± 2.1%	0.867 ± 6%	0.821	0.831

calculations, using the 26-group ABBN Russian set. Results using both shielded and unshielded U^{238} cross sections are shown in the last two columns of Table 4.10. It is seen that the measured fission ratios of U^{238}/U^{235} and U^{238}/Pu^{239} in the BTF blanket are in good agreement. Table 4.10 also shows that the calculated blanket fission ratios of U^{238}/U^{235} and U^{238}/Pu^{239} obtained using shielded U^{238} cross section are less than calculated using unshielded U^{238} cross section. In other words, a slightly harder spectrum was obtained when unshielded U^{238} cross sections were used. The effect of shielded vs. unshielded U^{238} cross sections on the blanket spectrum is a very important question and will be discussed further in Section 5.4.

The final two rows in the table show the expected good agreement between the Pu^{239} and U^{235} fission ratios. Although it would be desirable to measure and compare more fission ratios in ZPPR -2, these measurements and the calculations discussed in Chapter II provide considerable assurance of the similitude of BTF Blanket No. 2 and the ZPPR-2 assembly. As noted in Chapter II, the compositions of the ZPPR and BTF blankets are not identical; hence one should not expect exact agreement. The BTF blanket has a relatively higher concentration of low mass diluents and should therefore exhibit a softer ambient spectrum; any such trend is obscured by the stronger effect of the driving spectrum and the statistical uncertainty in the results.

4.8 SUMMARY

The capture reaction rates of $\text{Au}^{197}(n, \gamma)$, $\text{U}^{238}(n, \gamma)$, $\text{Mo}^{98}(n, \gamma)$, $\text{Cr}^{50}(n, \gamma)$, $\text{Na}^{23}(n, \gamma)$, as well as threshold reaction rates of $\text{U}^{238}(n, f)$, $\text{In}^{115}(n, n')$ and fission rates of $\text{U}^{235}(n, f)$, $\text{Pu}^{239}(n, f)$ have been measured in the axial (simulating radial) direction of BTF Blanket Mock-Up No. 2. In general, the agreement between experimental and theoretical results is good, except that the theory predicts a steeper slope than observed. The agreement becomes much poorer in the reflector region where the measured threshold reaction rates in particular are higher than those predicted theoretically. Some of the reflector discrepancies are undoubtedly due to the well-known difficulties regarding adequate few-group description of keV-range neutron diffusion in the iron reflector (M2); other contributions due to inadequate calculation of first-flight neutron propagation are suspected.

Studies were carried out on the effect of self-shielded U^{238} cross sections and the influence of the downscatter cross-section weighting spectrum on reaction rate calculations. It was found that the agreement between experimental and theoretical results was considerably improved when self-shielded U^{238} cross sections were used. The influence of the weighting spectrum was investigated by comparing typical reaction rate distributions, using different weighting spectra. Although no one weighting scheme best fitted the experimental data, the modification of the elastic downscatter weighting did produce significant changes in the calculated blanket spectra and hence in the reaction rate distributions.

An overall neutron balance in BTF Blanket Mock-Up No. 2 has also been made. The theoretical one-group balance was closed to within 2.0%.

The theoretical predictions of sodium and chromium capture reaction rates and U^{235} , U^{238} fission rates in the neutron balance table were checked against experimental measurements. The agreement between the experimental and theoretical results were within the experimental uncertainties.

Comparison of the limited number of spectral indices measured in common in the BTF mock-up and in ZPPR Assembly No. 2 confirmed the predicted spectral similitude in the blanket region.

Chapter V

NEUTRON SPECTRUM MEASUREMENTS

5.1 INTRODUCTION

One major objective of the LMFBR Blanket Physics Research Project at M.I.T. is to develop accurate and reliable methods for neutron spectrum measurements in the Blanket Facility. To this end, several different methods have been used. In general, these methods fall into the following categories. Details of each method are discussed in the references indicated.

1) Instrumental Neutron Spectrum Measurements. He^3 , Li^6 , and proton recoil spectrometers are used to measure neutron spectra in the energy range of several keV to a few MeV (O1).

2) Neutron Spectrum Measurements by Prompt Gamma Spectrometry. This method is based on determination of the spectrum of incident neutrons by analyzing the spectrum of the prompt gammas emitted by the target material (F3).

3) Germanium Line Broadening. In this method, the spectrum of incident neutrons is obtained by analyzing the broadening of the 691.4-keV internal conversion line of Ge^{72} in the $\text{Ge}(\text{Li})$ detector (K2).

4) Foil Activation Methods. Multiple foil packets containing resonance and threshold absorbers are irradiated within the blanket, and the neutron spectra are unfolded from the activity measurements. This method is applicable over a large range of neutron energies (eV to MeV), while other methods are usually restricted to certain energy ranges.

The second and third methods above are developmental in nature and are not being pursued with high priority, nor are they relied upon for benchmark measurements. The instrumental methods are being assigned the role of priority benchmark measurements. Thus, foil techniques are being relied upon for routine flux measurements and are the subject of the present chapter.

In the foil activation category, two different techniques have been employed at M.I.T. One technique, used by S. L. Ho and N. A. Passmann (H2, P2), is to irradiate a large number of foil materials in mixed powder form, all contained in one capsule, and then to count simultaneously all the gamma activities using high resolution Ge(Li) spectrometers. Neutron spectra are unfolded from the measured activities using the SAND II code (M1). Like many other iterative unfolding codes, the SAND II code requires an initial flux guess and a large amount of foil activity data as input. Since the unfolding process is purely mathematical in nature, such iterative codes may give unreasonable results (e.g., negative fluxes) for the unfolded spectra. The second approach, which is the one employed in the present work, is to employ only a few carefully chosen activities but to unfold spectra by making more use of reactor theory in the unfolding process.

The present concern is thus twofold: first to develop an improved unfolding method having general utility in LMFBR applications, and second, to apply the method to measure spectra in BTF Blanket Mock-Up No. 2. A complete description of the unfolding method is given in Section 5.2. Standard procedures for making the foil activity measurements are discussed in the next section. Section 5.4 presents the neutron spectra in BTF Blanket Mock-Up No. 2 as obtained by this particular foil

activation technique. Results are compared with calculated spectra and with N. A. Passmann's measurements. Comparisons with instrumental measurements will be reported by N. R. Ortiz in Reference (O1).

5.2 THE UNFOLDING METHOD

5.2.1 Description of the Unfolding Process

Although multiple-foil activation methods are a well-developed tool for neutron spectrometry, considerable simplification can be made in the unfolding of blanket spectra from activity measurements by making more use of reactor theory.

Dunn and Becker (D3), Stacey (S4) and others have demonstrated the applicability of slowing-down theory to fast reactors. The slowing-down density is found to be a very smoothly varying function of neutron energy, and it can therefore be characterized by a few-parameter correlation. The results reported herein are based upon selection of a specific functional relation between the flux and the slowing-down density. The following form (D4) was found adequate for present purposes, which involved characterization of LMFBR core, blanket and reflector spectra:

$$\phi(E) = \frac{C_0}{E\xi(E)\Sigma_{TR}(E)} \exp\{-C_1E - C_2E^{-C_3}\} \quad (5.1)$$

where $\xi(E)\Sigma_{TR}(E)$ = slowing-down power of medium in which spectrum is to be determined, including both inelastic and elastic moderation

C_0 = normalization constant

C_1, C_2, C_3 = parameters characteristic of the spectrum.

Typical values of C_1 , C_2 and C_3 are 0.78, 0.32 and 0.33, respectively.

The constants C_0, C_1, C_2, C_3 can be determined by minimizing the discrepancy between measured and calculated foil activities. For instance, the parameters characterizing the BTF Blanket Mock-Up No. 2 spectra, $\phi^{bkt}(E)$, can be determined in the following way.

By assuming that $\phi(u)$ is constant within an energy group, the blanket flux $\phi^{bkt}(E)$ can be written in multigroup form,

$$\phi_i^{bkt}(E) = \frac{\Delta u_i C_0}{(\xi \Sigma_{TR})_i} \exp \{-C_1 \bar{E}_i - C_2 \bar{E}_i - C_3\} \quad (5.2)$$

where i denotes the i -th energy group, and

$$\bar{E}_i = \frac{\int_{E_i}^{E_{i-1}} E \phi(E) dE}{\int_{E_i}^{E_{i-1}} \phi(E) dE} \quad (5.3)$$

The calculated foil activity of the j -th foil in the blanket, a_j^{bkt} , is given by

$$a_j^{bkt} = \sum_{i=1}^N \sigma_j^{act} \phi_i^{bkt} \quad (5.4)$$

where N = the number of energy groups (in our case, $N = 26$)

σ_j^{act} = the activation cross section of the j -th foil.

Also, let A_j^{bkt} be the experimentally measured activity of the j -th foil in the BTF Blanket Mock-Up No. 2, and let N_f be the total number of detector foil materials.

Define the error measure,

$$\epsilon^{bkt} = \sum_{j=1}^{N_f} \left(\frac{A_j^{bkt} - a_j^{bkt}}{A_j^{bkt}} \right)^2 \quad (5.5)$$

The normalization constant C_0 can be calculated by setting $\frac{\partial \epsilon^{bkt}}{\partial C_0} = 0$. Then, ϵ^{bkt} can be minimized by varying the three parameters C_1, C_2 and C_3 . In the present work, the three parameters C_1, C_2 and C_3 for the blanket spectra, ϕ_i^{bkt} , are solved for by a table-look-up method to find the entry where ϵ^{bkt} is minimum. A computer code, DRISTAN (Appendix D), has been written for this purpose. The corresponding values for C_1, C_2 and C_3 at which ϵ^{bkt} is minimum, denoted by C_1^*, C_2^* and C_3^* , are then used to calculate the group fluxes in the BTF Blanket Mock-Up by Eq. 5.2.

The strengths and weaknesses of this approach are evident from examination of Eq. 5.1. For example, although the characteristic fine structure in $\phi(E)$ due to scattering resonances is recoverable in great detail through introduction of the slowing-down power, considerably more input data are needed than for other unfolding schemes. While the functional dependence upon E in Eq. 5.1 is so formulated that physically unrealistic solutions such as negative values of the flux are excluded, one is also limited in the variety of environments for which a given functional form is suitable. The most attractive feature, however, is that the spectrum can be unfolded from a small set of activities.

5.2.2 Testing the Unfolding Method – Computer Experiments

Before this method was applied to unfold neutron spectra in BTF Blanket Mock-Up No. 2, computer experiments were set up to test the method.

A reference reactor having the ZPPR two-zone core coupled with a blanket whose composition is the same as that of the BTF Blanket Mock-Up No. 2 was chosen to test the unfolding method (see Table 2.5 in Chapter II). Group fluxes, ϕ_i , and the activities $a_j = \sum_{i=1}^N \sigma_{ji}^{act} \phi_i$ for five

($j=5$) activation reactions – namely, $\text{Au}^{197}(n, \gamma)$, $\text{In}^{115}(n, n')$, $\text{Mo}^{98}(n, \gamma)$, $\text{U}^{238}(n, f)$ and $\text{U}^{238}(n, \gamma)$ – at different points in each region of the reference reactor were calculated by means of ANISN in the S_8 option, using the 26-group ABBN cross-section set. The calculated activities, a_j , were then fed into the DRISTAN code to solve for the parameters C_1^* , C_2^* and C_3^* . The spectrum generated from C_1^* , C_2^* and C_3^* , denoted by ϕ_i^{DRIS} , was compared with the original spectrum, ϕ_i^{ANISN} , from the ANISN calculations. Figures 5.1 and 5.2 show typical comparisons between the two spectra, ϕ_i^{DRIS} and ϕ_i^{ANISN} , in the blanket and reflector regions. The agreement between the two spectra, ϕ_i^{DRIS} and ϕ_i^{ANISN} , is in general good, particularly in the region in which most of the neutrons lie.

Table 5.1 shows the parameters C_1^* , C_2^* and C_3^* characterizing various spectra, ϕ_i , in different regions of the reactor. The change in the parameters C_1^* , C_2^* from region to region can be readily explained from physical arguments. In Eq. 5.1, the parameter C_2 controls the shape of the low energy end of the spectrum. As C_2 increases, the low energy tail is depleted and the spectrum is hardened. This is indeed the case, as is evident from the calculation results in Table 5.1. In Core Zone 2, $C_2^* = 1.2175$ (the highest value listed) which suggests a relatively hard spectrum; and in the reflector region, $C_2^* = 0.025$ (the lowest value) which suggests a much softer spectrum. On the other hand, the parameter C_1 controls the shape of the high energy part of the spectrum. As C_1 is increased, the high energy tail is depleted and the spectrum softens. Again, the calculated results show that C_1^* increases from 0.1505 in the core to 1.7675 in the reflector region, which clearly indicates the softening of the spectra as one proceeds from the core to the reflector.

FIG.5.1 COMPARISON OF Φ_1^{ANISN} AND Φ_1^{DRIS} SPECTRA
 IN THE BLANKET REGION, $R = 124.75$ CM.

FIG.5.2 COMPARISON OF ϕ_1^{ANISN} AND ϕ_1^{DRIS} SPECTRA
 IN THE REFLECTOR, $R = 155.6$ cm.

TABLE 5.1
The Variation of Flux Characterization
Parameters with Position

Region	Parameters		
	C_1^*	C_2^*	C_3^*
Core, Zone 1	0.1505	0.7675	0.230
Core, Zone 2	0.1505	1.2175	0.250
Blanket	0.2750	0.220	0.333
Reflector	1.7675	0.025	0.333

5.2.3 Flux Sensitivity to Input Activities

It has been shown that the above method is able to unravel both hard and soft LMFBR spectra. However, it may still not be a good method if the output spectrum is too sensitive to the input activities, i.e., if a small error in the measured activities can cause a large change in the calculated fluxes. Thus, some sensitivity test runs were performed by varying the input activities $\pm 5\%$, one at a time. The changes in the output group fluxes were found to be less than 0.5% in the 10-keV to 100-keV region and, in general, higher than 1% only at the high and low energy ends. The results of the sensitivity tests are encouraging because even when the measured activities are $\pm 5\%$ in error, one can still determine the group fluxes to within $\pm 5\%$ in the 10-keV to 100-keV region.

5.3 FOIL ACTIVATION

5.3.1 Choice of Detector Foils

Since the present method has the ability to unfold the blanket spectra from a small number of foil activity measurements, the minimum (three) foil materials were used in the experiments. However, the selected foil materials had to provide sufficient energy coverage of the blanket spectra. Indium, molybdenum and gold were chosen to sample neutrons of high energy, low energy and neutrons of both high and low energies, respectively (also refer to Chap. III, Section 3.1). In order to see the effect of using different detector foil materials, the experiment was repeated with manganese instead of molybdenum to sample low energy neutrons.

Table 5.2 lists the foil materials selected for use in this study along with certain pertinent properties of the initial and product nuclides; $T_{\frac{1}{2}}$ is the half-life of the product nuclide; the E_{γ} are the energies of gamma rays of the product nuclide along with their corresponding percent yield per disintegration (% γ). The last column gives the percent abundance of the target isotope as it occurs in nature.

5.3.2 Foil Irradiation and Counting Procedure

Details of the foil irradiation in the BTF Blanket Mock-Up No. 2 and of the counting procedures have been given in Section 4.2 and will not be repeated here.

A set of the three detector foils (gold, indium and molybdenum) was irradiated for about six hours at three foil-holder positions, namely, at axial (radial) depths of $Z = 7.5$ cm, 22.6 cm and 37.6 cm, at about mid-assembly height. The experiment was repeated with another set of

TABLE 5.2. Properties of Foil Materials Used in Blanket Spectrum Measurements

Reaction	Foil Materials	Weight (mg)	$T_{\frac{1}{2}}$	E_{γ} (MeV)	% γ	% Abundance in Nature
$\text{Au}^{197}(\text{n}, \gamma)\text{Au}^{198}$	Gold	38.65	2.7 days	0.412	99.82	100
$\text{In}^{115}(\text{n}, \text{n}')\text{In}^{115\text{m}}$	Indium	61.50	4.5 hours	0.335	94.5	95.77
$\text{Mo}^{98}(\text{n}, \gamma)\text{Mo}^{99}$	Molybdenum	54.10	66.0 hours	0.78	14.0	9.45
$\text{Mn}^{55}(\text{n}, \gamma)\text{Mn}^{56}$	Manganese (powder form)	577.70	2.58 hours	0.845	98.07	100

detector foils (gold, indium and manganese) to test the effect on the resulting neutron spectra of replacing the detector foil molybdenum with manganese, at 7.5 cm into the blanket.

The measurement of the foil activities followed standard procedures. Each of the irradiated foils was counted separately for the corresponding emitted gamma (Table 5.2) with the counting system described in Chapter III, Section 3.2.2. The raw data of the foil count rate were corrected for background, counter dead-time and decay time, and the final measured activities were expressed in counts/min-mg. The self-absorption and self-shielding corrections were measured and found to be very small and were therefore neglected, as discussed in Section 4.2.5. Table 5.3 lists the measured activities at different positions in the BTF Blanket Mock-Up No. 2. The $\pm 2.5\%$ experimental errors quoted are due to statistical uncertainties in foil counts and foil weight uncertainties.

5.3.3 Corrected Gamma Activities

In order to calculate the activities of the foil materials on the same basis, the efficiency of the counter has to be determined. Three standard gamma sources, Ba^{133} , Na^{22} and Cs^{137} , were used to determine the energy-dependent efficiency of the counter. The initial activities of the sources given in μc were corrected for radioactive decay and the branching ratio of the gamma rays.

It is convenient to have the input foil activities fed into the unfolding code, DRISTAN, in the form of disintegrations per second per nucleus; thus, the measured activities of the j -th detector foil material A_m^j expressed in counts/min-mg must be further corrected according to the following equation:

TABLE 5.3. Measured Spectrum-Packet Foil Activity in Blanket Mock-Up No. 2

Reaction	Measured Activities (cpm/mg) in Blanket Mock-Up No. 2		
	Z = 7.5 cm	Z = 22.6 cm	Z = 37.6 cm
$\text{Au}^{197}(\text{n}, \gamma)\text{Au}^{198}$	6.21×10^4	3.38×10^4	1.96×10^4
$\text{In}^{115}(\text{n}, \text{n}')\text{In}^{115\text{m}}$	8.68×10^2	3.19×10^2	1.29×10^2
$\text{Mo}^{98}(\text{n}, \gamma)\text{Mo}^{99}$	1.23×10^2	7.61×10^1	3.54×10^1
<hr/>			
$\text{Au}^{197}(\text{n}, \gamma)\text{Au}^{198}$	6.16×10^4		
$\text{In}^{115}(\text{n}, \text{n}')\text{In}^{115\text{m}}$	8.78×10^2		
$\text{Mn}^{55}(\text{n}, \gamma)\text{Mn}^{56}$	7.90×10^3		
(Probable experimental errors are $\pm 2.5\%$)			

$$A_{\text{dnp}}^j = \left(\frac{1}{f_j \Gamma_j \gamma_j} \right) \left(\frac{M}{A_o} \right) c A_m^j \quad (5.6)$$

where A_{dnp}^j = activity of the j-th detector foil material in disintegrations per second per nucleus

A_m^j = measured activity of the j-th foil in counts/min-mg

c = conversion factor, 1000/60, converting A_m^j in counts/min-mg to counts/sec-gm

M = molecular weight of the foil material

f_j = fraction of isotopic abundance in the j-th foil

Γ_j = fraction of detected gamma from the j-th detector foil material

γ_j = efficiency of the counter in detecting the emitted gamma from the j-th detector foil assembly

A_o = Avogadro's Number.

Table 5.4 summarizes the corrected foil activities in disintegrations per second per nucleus, measured at different depths in the Blanket Mock-Up No. 2.

5.4 RESULTS OF SPECTRUM MEASUREMENTS IN BLANKET MOCK-UP NO. 2

DRISTAN was used to unfold the neutron spectra of BTF Blanket Mock-Up No. 2 from the activity measurements of gold, indium and molybdenum detector foil materials. The corrected saturation activities of these foils were listed in Table 5.4. The unfolded neutron spectra expressed in flux per unit lethargy at the three irradiation positions in Blanket Mock-Up No. 2 are given in Table 5.5 and are also presented graphically in Fig. 5.3. The unfolded spectra are typical of

TABLE 5.4 Corrected Spectrum-Packet Foil Activities in Blanket Mock-Up No. 2

Reaction	Corrected Activities (dis/sec-nucleus) in Blanket Mock-Up No. 2		
	Z = 7.5 cm	Z = 22.6 cm	Z = 37.6 cm
$\text{Au}^{197}(\text{n}, \gamma)\text{Au}^{198}$	9.50×10^{-16}	5.17×10^{-16}	2.99×10^{-16}
$\text{In}^{115}(\text{n}, \text{n}')\text{In}^{115\text{m}}$	5.54×10^{-18}	2.04×10^{-18}	0.822×10^{-18}
$\text{Mo}^{98}(\text{n}, \gamma)\text{Mo}^{99}$	9.36×10^{-17}	5.78×10^{-17}	3.69×10^{-17}
<hr/>			
$\text{Au}^{197}(\text{n}, \gamma)\text{Au}^{198}$	9.43×10^{-16}		
$\text{In}^{115}(\text{n}, \text{n}')\text{In}^{115\text{m}}$	5.60×10^{-18}		
$\text{Mn}^{55}(\text{n}, \gamma)\text{Mn}^{56}$	1.36×10^{-16}		
(Probable experimental errors are $\pm 2.5\%$)			

TABLE 5.5 BTF Blanket Mock-Up No. 2 Spectra Unfolded by DRISTAN

ENERGY GROUP			FLUX PER UNIT LETHARGY $\phi_i(u)$			
No.	E_L	Δu_i	Unfolded from Au-In-Mo Foil Data			Unfolded from Au-In-Mn Foil Data
			Z = 7.5 cm	Z = 22.6 cm	Z = 37.6 cm	Z = 7.5 cm
1	6.5 MeV	0.48	0.875×10^{-3}	0.680×10^{-4}	0.257×10^{-4}	0.597×10^{-4}
2	4.0 "	0.48	0.402×10^{-2}	0.849×10^{-3}	0.469×10^{-3}	0.786×10^{-3}
3	2.5 "	0.48	0.975×10^{-2}	0.382×10^{-2}	0.267×10^{-2}	0.365×10^{-2}
4	1.4 "	0.57	0.239×10^{-1}	0.141×10^{-1}	0.115×10^{-1}	1.379×10^{-2}
5	0.8 "	0.57	0.688×10^{-1}	0.527×10^{-1}	0.476×10^{-1}	0.521×10^{-1}
6	0.4 "	0.69	0.197	0.176	0.169	0.176
7	0.2 "	0.69	0.173	0.169	0.168	0.169
8	0.1 "	0.69	0.210	0.215	0.217	0.216
9	46.5 keV	0.77	0.200	0.210	0.213	0.210
10	21.5 "	0.77	0.162	0.172	0.175	0.172
11	10.0 "	↓	0.148	0.158	0.161	0.158
12	4.65 "		0.864×10^{-1}	0.921×10^{-1}	0.939×10^{-1}	0.921×10^{-1}
13	2.15 "		0.144×10^{-1}	0.153×10^{-1}	0.156×10^{-1}	0.153×10^{-1}
14	1.0 "		0.452×10^{-1}	0.480×10^{-1}	0.487×10^{-1}	0.478×10^{-1}
15	465.0 eV		0.267×10^{-1}	0.282×10^{-1}	0.286×10^{-1}	0.280×10^{-1}
16	215.0 "		0.126×10^{-1}	0.132×10^{-1}	0.133×10^{-1}	0.130×10^{-1}
17	100.0 "		0.473×10^{-2}	0.496×10^{-2}	0.496×10^{-2}	0.486×10^{-2}
18	46.5 "		0.134×10^{-2}	0.140×10^{-2}	0.138×10^{-2}	0.135×10^{-2}
19	21.5 "		0.265×10^{-3}	0.274×10^{-3}	0.267×10^{-3}	0.262×10^{-3}
20	10.0 "		0.325×10^{-4}	0.329×10^{-4}	0.319×10^{-4}	0.312×10^{-4}
21	4.65 "	0.77	--	--	--	--

FIG.5.3 NEUTRON SPECTRA IN BLANKET No.2
(3 FOIL DATA UNFOLDED BY BRISTAN)

fast reactor blanket spectra, with a large depression in the flux at 3 keV due to the sodium scattering resonance. The most probable neutron energy of the blanket spectra is about 100 keV. Figure 5.3 shows the unfolded blanket spectra at $Z=7.5$ cm., 22.6 cm. and 37.6 cm. These results indicate, at best, only slight softening of the low energy tail of blanket spectrum as one proceeds deeper into the blanket. However, other measurements, such as reaction rate ratios would suggest that the actual softening is greater, and therefore imply that the present foil technique is not sufficiently sensitive in this region. For example, the measured molybdenum to gold reaction rate ratios (from Table 5.4) are 0.0985, 0.112 and 0.123 at depths, Z , of 7.5, 22.6 and 37.6 cm. respectively.

To see the effect of replacing the detector foil molybdenum with manganese on the resulting blanket spectra, the two sets of activity measurements, Au-In-Mo vs. Au-In-Mn, were again unfolded by DRISTAN separately. Figure 5.4 shows the comparison of the two corresponding unfolded spectra at $Z=7.5$ cm in the blanket. Except in the high energy groups, the two unfolded blanket spectra are very similar. It can be concluded that the result of this unfolding technique does not strongly depend on the detector foil used. In other words, it will still give the same spectrum results even if other detector foils with slightly different nuclear cross sections are used instead.

Figure 5.5 shows the comparison between the blanket spectra at $Z=7.5$ cm unfolded by DRISTAN from the gold, indium and molybdenum activity measurements and that obtained from the one-dimensional S_8 transport code ANISN, using the ABBN Russian set with self-shielded U^{238} cross sections. The measured blanket spectra obtained from N.A. Passmann's 7 foil data and unfolded by SAND II are also given

FIG.5.4 BLANKET No.2 NEUTRON SPECTRA UNFOLDED BY
DRISTAN AT Z = 7.5 CM.

FIG. 5.5 INTERCOMPARISON OF NEUTRON SPECTRA
AT 7.5cm. INTO BLANKET No.2

in Fig. 5.5 for intercomparison (P2). Other comparisons with more accurate measurements, such as the proton recoil and He^3 spectrometers, will be discussed in a subsequent publication by N.R. Ortiz (O1). There is general agreement between the neutron spectra unfolded from the foil activation measurements and that predicted from theory, except for the unexpectedly larger dip from 3 keV to a few tens of keV in Passmann's data (probably due to acknowledged problems involved in matching the resonance capture activity of manganese in his work). The spectrum determined in the present work is in much better agreement with the calculated data than the SAND II unfolded results. The measured spectra are also harder than calculated. It is interesting to note that if unshielded U^{238} cross sections are used, the measured spectrum now becomes softer than calculated (Fig. 5.6). This suggests, if one assumes the measured spectra to be correct, that the self-shielding corrections may be overestimated.

5.5 SUMMARY

An unfolding method based on slowing-down theory has been applied successfully in the experimental work on BTF Blanket Mock-Up No. 2. The resulting neutron spectra showed good agreement with the results of 26-group calculations. The method was also tested by substitution of a second resonance-type detector, manganese, for molybdenum. The spectra unfolded independently, using the two different foil sets (Mn, Au, In vs. Mo, Au, In), were found to differ by less than 5% in the energy groups comprising the low energy flux tail.

While more foils can be used and the resulting overdetermined solutions for C_0 , C_1 , C_2 and C_3 resolved by least squares analysis,

FIG.5.6 EFFECT ON THE NEUTRON SPECTRA CALCULATIONS
IN BLANKET No.2 AT $Z = 7.5$ cm. USING
SHIELDED AND UNSHIELDED ^{238}U CROSS-SECTIONS.

test cases showed that no discernible improvement resulted. This agrees with some of Corcoran's observations (C2) as to the marginal amount of additional information contained in cross sections which do not differ appreciably in energy dependence from others in the collection employed. Apparently, by choosing only one of each of the basic shapes (threshold, continuous, resonance), one can acquire most of the information contained in larger sets of foil data. Further remarks on overdetermined vs. underdetermined solutions are given in Reference (D5).

Chapter VI

CONCLUSIONS AND RECOMMENDATIONS

6.1 SUMMARY

The MIT Blanket Test Facility and Blanket Mock-Up No. 2, which is the first of a series of realistic mock-ups of typical and advanced LMFBR blanket configurations and compositions, have been described in Chapter II of this report. In Chapters III, IV and V, experimental measurements using foil activation techniques made on the subject blanket have been presented. In this final chapter, conclusions drawn from both the experimental and theoretical work on Blanket Mock-Up No. 2 are discussed.

1. Buckling Measurements

(a) Fundamental-mode cosine flux shapes were achieved in the vertical and horizontal directions in the blanket mock-up. Furthermore, the same shapes were observed using detector foils sensitive to different regions of the neutron spectrum. Thus, transverse leakage may be characterized by a simple buckling term and the problem reduced to an effective one-dimensional problem involving traverses through the blanket.

(b) The effective height of the vertical cosine distribution was found to be 152.4 cm and the effective width of the horizontal cosine distribution was found to be 188.0 cm, yielding a total lateral buckling of $B^2 = 0.000704 \pm 0.000028 \text{ cm}^{-2}$. This lateral buckling yields an axial spatial flux distribution in the blanket assembly equivalent to the radial spatial

flux distribution in the radial blanket of an LMFBR having a core volume roughly of 8200 litres (that is, an equivalent core height of 109 cm, given by the 152-cm effective height minus 2×21.5 cm axial reflector savings, and an equivalent core radius of 135 cm, as calculated in Section 3.3.3). However, calculations showed that an increase in the lateral buckling to 0.00089 cm^{-2} (equivalent to an LMFBR core volume of about 3200 litres) had no significant effect on the axial distribution of gold and indium activities. In other words, once the core becomes large enough, blanket neutronics are a very weak function of core size and slab geometry can be a good approximation to cylindrical geometry.

(c) The horizontal and vertical gold-to-indium activation ratios indicated that lateral spectral equilibrium was attained in a large central volume of the blanket assembly. The ratios showed that backscattering perturbed the blanket spectrum only in the outer 30 cm of the blanket assembly.

2. Reaction Rate Measurements and Neutron Balance

(a) The agreement between experimental and theoretical reaction rate distributions in $\text{Au}^{197}(n,\gamma)$, $\text{U}^{238}(n,\gamma)$, $\text{U}^{238}(n,f)$,.....etc. was good in the blanket region, except that the theory predicted a slightly steeper slope. The poorest agreement occurred in the steel reflector region, probably due to the well-known difficulties involved in describing neutron diffusion in iron due to the window in its cross section, near 25 keV. The adequacy of first-flight neutron penetration calculations deep in the blanket is also suspected.

(b) The use of unshielded U^{238} cross sections in the multigroup calculations had a significant effect on the resulting blanket spectra and

reaction rate distributions. Much poorer agreement between experiment and theory would result if unshielded U^{238} cross sections were used.

(c) The use of different weighting spectra, e.g., $\phi(u) \propto E$ or the calculated mid-blanket spectrum, for preparation of the input cross-section set also affected the multigroup calculations of blanket spectra and reaction rate distributions. However, this effect was small compared with the U^{238} shielding effect.

(d) Theoretical calculations on the neutron balance in BTF Blanket Mock-Up No. 2 showed that the net one-group balance between the total production and the total loss of neutrons in the blanket agreed to within 2.0%. Experimental measurements on the integral sodium and chromium capture reaction rates and U^{235} and U^{238} fission rates checked with the theoretical predictions in the neutron balance table.

3. Neutron Spectrum Measurements

(a) An unfolding method based on slowing-down theory has been applied successfully in the experimental work on BTF Blanket Mock-Up No. 2. This unfolding technique was found not strongly dependent on the detector foils used since similar spectra were obtained using two different foil sets (Mn, Au, In and Mo, Au, In).

(b) The unfolded spectrum of Blanket Mock-Up No. 2 was found harder than that of the ANISN calculations using shielded U^{238} cross sections but softer than that calculated using unshielded U^{238} cross sections.

6.2 FUTURE WORK

A number of conclusions pertinent to the conduct of future work are suggested by the experience accumulated during the course of this research and the subsequent evaluation of the results. These recommendations include improvements in experimental technique, suggestions for additional investigations and, most importantly, improvements in the analysis of results. Most stem from the need for resolution of the observed differences between measured and calculated results. The most obvious unresolved discrepancy involves the threshold reaction rates in the reflector.

1. Improved Experimental Methods

(a) In the future, foils rather than powder should be used whenever possible; and if powder is used, metal capsules rather than plastic should be employed.

(b) A more efficient counting system should be used for deep-penetration activation measurements. The use of the new well-counter now available in the MITR activation analysis facility, a 3-inch-diameter NaI detector, should improve achievable counting rates by a factor of around three.

(c) Different detector foil sets (other than Mo/Mn, Au, In) should be tested for spectrum unfolding applications and continuing improvements in accuracy sought. Clearly, in the future, use of multiple-isotope materials such as Mo, Cr and the like should be avoided because of the lack of cross-section data for the separated isotopes.

2. Additional Experiments

(a) In addition to the $U^{238}(n, f)$ and $In^{115}(n, n')$ traverses, other threshold reaction rates such as $Ti^{48}(n, p) Sc^{48}$ and $Rh^{103}(n, n') Rh^{103m}$ should be measured in the reflector region of future blankets.

(b) Epithermal albedo measurements should be made at the converter assembly/hohlraum interface, using the method employed by D. Goebel (G3) in thermal lattice research at M.I.T. It is presently assumed that the epithermal albedo is effectively zero based on the large cadmium ratio measured in the hohlraum during BTF operations, but this should be confirmed by a more direct experiment.

(c) Additional experimental investigation of $U^{238}(n, \gamma)$ self-shielding is called for. Work is under way at M.I.T. using a realistic subassembly constructed of stainless-steel-clad UO_2 fuel and sodium metal coolant (F4), which should provide additional valuable data.

3. Improved Data Analysis

(a) Improvements in the spectrum unfolding method based on slowing-down theory can be made. The group cross-section data used in the present work can be improved by recourse to more recent ENDF data and by collapsing over a more suitable weighting spectrum. Another aspect requiring upgrading involves the observed increasing discrepancy between the unfolded spectrum and ANISN calculations in the flux tail above 1 MeV. This is attributed to the use of a relation between $q(E)$ and $\phi(E)$ which, strictly speaking, is appropriate only for multiply-collided neutrons. Improvement should be sought through the inclusion of a separate first-flight term (I2).

(b) The use of more energy groups (50) and higher order S_N approximation ($N=16$) should be investigated to improve reflector region calculations.

(c) Sensitivity studies relating activation traverses and cross-section changes should be carried out. In particular, key comparisons should be re-run using an up-to-date cross section set based on ENDF/B-II data. Differences in cross section data may well account for the consistent over-prediction of the slope of the activation traverses.

(d) Deep penetration calculations in two or three dimensions should be made to check the prediction of threshold activation traverses in the reflector region. It is likely that the one-dimensional calculations used in the present work are inadequate for description of the transport of neutrons involved in such interactions.

(e) Additional state-of-the-art heterogeneity calculations should be made for comparison with experimental measurements. U^{238} self-shielding is clearly an important factor, and it must be calculated with considerable confidence if BTF results are to serve as reliable bench-mark data.

4. Methods Development

Several less specifically oriented capabilities are also worthy of development over a larger time span and at lower priority:

(a) A simple way to adjust elastic downscatter cross sections to accommodate local spectrum shape should be pursued. One as yet unsuccessful scheme is reported in Appendix C. Other approaches, such as the use of piecewise polynomials in energy space, may deserve investigation.

(b) Systematic methods for handling the severe resonance mismatches at the blanket/reflector interface must be worked out. Due to the softer spectrum and the greater dissimilitude in compositions, this is considerably more difficult than the core/blanket interface problem.

(c) Alternative methods for unfolding neutron spectra from foil activation data would be useful. Some work is currently under way at M.I.T. to systematically adjust multigroup calculations to best agree with experimentally measured foil activities.

6.3 CONCLUSION

Perhaps the most important general observations which can be made as a result of this work are that the BTF concept gives every evidence of being a valid approach for investigation of blanket neutronics without requiring a critical assembly driver, and that, for the most part, state-of-the-art calculation techniques give a fairly good picture of physical reality. The only discrepancy of major potential importance to LMFBR development uncovered in this work is the threshold detector mismatch in the reflector. If first-flight neutrons penetrate deeper than estimated, the potential for accentuated radiation damage to core structure exists. Resolution of this item is thus a high priority task which should be factored into future project objectives.

Appendix A

ADDITIONAL DATA ON BLANKET MOCK-UP No. 2

TABLE A.1. Subassembly Loadings of Sodium Chromate
in Blanket Mock-Up No. 2

Subassembly Number (See Fig. A.1)	Loading of Sodium Chromate (kg)
1	30.735
2	31.442
3	30.911
4	31.044
5	31.136
6	30.845
7	31.800
8	30.879
9	31.314
10	30.984
11	30.976
12	30.735
13	30.913
14	31.150
15	31.496
16	31.013
17	30.978
18	31.495
19	31.177
20	31.143
21	30.997
22	31.230
23	31.555
24	31.184
25	30.511

Average subassembly loading: 31.106 ± 0.294 kg.

FIG. A.1 BLANKET SUBASSEMBLY NUMBERING

TABLE A.2. Calculation of Atom Densities in Blanket Mock-Up No. 2

$$N_i = W_i A / V M_i \quad (\text{Eq. 2.1})$$

Material i	Weight in Unit Volume, V W_i (kg)	Molecular Weight M_i	Atom Densities N_i (10^{24} nuclei/cc)
U^{235}	0.948	235.117	0.000088
U^{238}	88.349	238.07	0.008108
Fe	35.202	55.85	0.013750
C	0.052	12.00	0.000096
Na_2CrO_4	30.133	161.992	0.004064
Cr	—	—	0.004064
Na	—	—	0.008128
O	—	—	0.016256
H_2O	0.030	18.016	0.0000365
O	—	—	0.0000365
H	—	—	0.0000730

$$A = \text{Avogadro's number} = 0.6023 \times 10^{24}$$

$$V = \text{unit volume} = 5.92 \text{ in.} \times 5.92 \text{ in.} \times 48 \text{ in.}$$

$$= 1680.00 \text{ in.}^3 = 27566.76 \text{ cc}$$

Appendix B

LISTING OF FOIL ACTIVITIES FOR THE
BUCKLING DETERMINATIONSTABLE B.1. Data for Gold Vertical Activation Traverses
in Blanket Mock-Up No. 2* (See Fig. 3.4)

Vertical Distance from Center of Blanket, Y (in.)	Au ¹⁹⁸ Activity in cpm/mg-m				
	Z=0.92 in.	Z=5.00 in.	Z=8.88 in.	Z=12.76 in.	Z=16.84 in.
	(Z = depth into blanket)				
Top					
24	82.79	62.09	47.44	36.41	29.98
21	83.73	62.37	45.52	33.55	21.12
18	105.20	82.01	55.47	39.31	24.61
15	130.03	100.92	68.30	46.69	30.69
12	157.16	117.96	78.42	54.76	37.10
9	173.63	132.02	88.79	61.45	43.54
6	186.78	142.42	94.70	67.32	46.01
3	192.14	150.45	98.65	70.05	48.98
0	198.79	152.31	99.81	71.08	52.04
-3	197.51	153.96	100.22	70.64	50.70
-6	194.23	149.25	99.91	68.37	48.31
-9	184.85	143.28	94.82	65.87	46.67
-12	174.29	129.67	87.26	61.38	40.83
-15	154.46	117.57	78.31	54.44	36.03
-18	131.95	97.11	66.23	45.74	29.93
-21	104.34	77.71	53.99	38.51	24.97
-24	78.74	61.55	45.56	33.67	24.58
Bottom					

*Typical counting-statistic errors are less than $\pm 1.5\%$.

TABLE B.2. Data for Indium Vertical Activation Traverses
in Blanket Mock-Up No. 2 (See Fig. 3.5)

Vertical Distance from Center of Blanket, Y (in.)	In ^{115m} Activity in cpm/mg-m		
	Z = 0.92 in.	Z = 8.88 in.	Z = 12.76 in.
	(Z = depth into blanket)		
Top			
24 ↑	251.1	121.6	48.1
21	363.6	122.8	58.8
18	519.1	160.1	81.2
15	620.4	178.8	98.2
12	683.6	199.3	110.3
9	777.5	225.9	121.0
6	821.6	235.9	128.0
3	837.2	250.0	139.9
0	846.7	243.5	143.6
-3	857.6	246.8	145.5
-6	840.7	237.2	135.6
-9	806.7	221.9	135.1
-12	777.6	205.8	123.1
-15	714.7	178.2	109.4
-18	591.1	154.8	95.0
-21	458.6	131.5	74.7
-24 ↓	335.8	103.3	60.9
Bottom			

TABLE B.3. Data for Molybdenum Activity Traverses
in Blanket Mock-Up No. 2 (See Fig. 3.5)

Vertical Distance from Center of Blanket, Y (in.)	Mo ⁹⁹ Activity in cpm/mg-m		
	Z = 0.92 in.	Z = 8.88 in. (Z = depth into blanket)	Z = 12.76 in.
24 ↑ Top	1599	1043	—
21	2380	1534	1062
18	3349	2028	1401
15	4303	2604	1699
12	5141	3117	2020
9	5764	3428	2129
6	6194	3652	2626
3	6490	3861	2608
0	6926	3921	2663
-3	6750	3958	2619
-6	6630	3843	2576
-9	6354	3618	2483
-12	5846	3365	2232
-15	5059	2954	1983
-18	4214	2496	1720
-21	3270	1990	1309
-24 ↓ Bottom	2243	1468	993

TABLE B.4. Data for Horizontal Activation Traverses
in Blanket Mock-Up No. 2 (See Figs. 3.7 and 3.8)

Horizontal Distance from Center of Blanket, X (in.)	Activity in cpm/mg-m			
	*Z=8.88 in.	Z=8.88 in.	Z=8.88 in.	Z=8.88 in.
	*Y=-3.0 in.	Y=-3.0 in.	Y=+3.0 in.	Y=-3.0 in.
	In ^{115m}	Mo ⁹⁹	Au ¹⁹⁸	Au ¹⁹⁸
25.5	102.3	791	36.57	36.91
22.5	134.6	1916	53.25	52.91
19.5	179.8	2453	67.32	67.74
16.5	203.4	2940	80.96	79.65
13.5	216.9	3269	89.39	89.33
10.5	240.5	3534	97.32	96.85
7.5	244.1	3695	101.67	100.61
4.5	257.5	3887	104.40	104.30
1.5	249.7	3907	106.80	106.97
-1.5	250.0	3902	104.69	105.10
-4.5	246.7	3923	105.07	104.22
-7.5	241.4	3778	102.83	101.06
-10.5	231.1	3595	96.09	97.37
-13.5	221.4	3355	90.30	91.78
-16.5	202.9	2941	80.54	82.21
-19.5	170.1	2538	69.96	69.18
-22.5	139.9	1947	52.86	54.60
-25.5	104.1	1298	36.70	35.70

* Z = depth into blanket; Y = vertical distance from center of blanket.

TABLE B.5. Data for Horizontal Activation Traverses in
Blanket Mock-Up No. 2 (See Figs. 3.7, 3.8)

Horizontal Distance from Center of Blanket, X (in.)	Activity in cpm/mg-m at Z = 0.0 in., Y = -3.0 in. (Z = depth into blanket Y = vertical distance from center of blanket)		
	Au ¹⁹⁸	In ^{115m}	Mo ⁹⁹
27.0	62.0	512	1901
24.0	94.6	761	2899
21.0	126.1	970	4074
18.0	149.6	1060	5062
15.0	171.7	1215	5705
12.0	182.9	1267	6329
9.0	191.5	1328	6656
6.0	200.9	1397	6900
3.0	201.5	1358	7077
0.0	203.7	1379	7100
-3.0	203.3	1357	6979
-6.0	198.3	1347	7039
-9.0	189.7	1322	6665
-12.0	182.0	1276	6430
-15.0	172.6	1221	5956
-18.0	151.1	1107	5166
-21.0	126.4	976	4161
-24.0	92.9	788	3039
-27.0	64.0	591	2006

Appendix C

CORRECTION SCHEMES FOR ELASTIC DOWNSCATTERING CROSS SECTIONS

C.1 Correction Schemes for Elastic Downscattering Cross Sections

A prescription to account for the effect of flux shape on the elastic downscattering cross section can be derived in the following way (F4).

The downscatter source from group i is given by

$$\Sigma_{i \rightarrow i+1} \cdot \phi_i \cong \Sigma_s \frac{\xi_i}{\Delta u_i} \cdot \phi(u_i) \Delta u_i, \quad \text{for } \xi_i \ll \Delta u_i \quad (\text{C.1})$$

where $\Sigma_{i \rightarrow i+1}$ = the downscattering cross section from groups i to $i+1$

ϕ_i = flux in group i

Δu_i = lethargy width of group i

$\phi(u_i)$ = flux per unit lethargy at the boundary between groups i and $i+1$.

For a $1/E$ spectrum, the downscattering cross section from group i , denoted by $\Sigma_{1/E}$, equals $\Sigma_s \frac{\xi_i}{\Delta u_i}$. Equation C.1 can then be written as

$$\Sigma_{i \rightarrow i+1} \cdot \phi_i \cong \Sigma_{1/E} \cdot \phi(u_i) \Delta u_i \quad (\text{C.2})$$

If each group is assumed to have an average $\phi(u)$ given by $\frac{\phi_j}{\Delta u_j}$ at the center of the Δu bin, $\phi(u_i)$ can be obtained by Lagrangian interpolation, using as many groups above i and below $i+1$ as we choose to use. In general, inclusion of more than the two adjacent groups appears unwarranted because of the errors inherent in the basic assumption used in the derivation and because the weighting for these remote groups turns

out to be so small in the interpolation formulae.

By applying Lagrange's interpolation formula for groups i and $i+1$,

$$\phi(u_i) = \frac{\frac{\Delta u_i}{2}}{\left(\frac{\Delta u_{i+1}}{2} + \frac{\Delta u_i}{2}\right)} \cdot \frac{\phi_{i+1}}{\Delta u_{i+1}} + \frac{-\frac{\Delta u_{i+1}}{2}}{-\left(\frac{\Delta u_{i+1}}{2} + \frac{\Delta u_i}{2}\right)} \cdot \frac{\phi_i}{\Delta u_i} \quad (C.3)$$

and thus Eq. C.2 becomes

$$\Sigma_{i \rightarrow i+1} \phi_i = \Sigma_{1/E} \left\{ \frac{\Delta u_i}{\Delta u_{i+1}} \left(\frac{\Delta u_i}{\Delta u_{i+1} + \Delta u_i} \right) \phi_{i+1} + \left(\frac{\Delta u_{i+1}}{\Delta u_{i+1} + \Delta u_i} \right) \phi_i \right\} \quad (C.4)$$

Setting

$$\alpha = \frac{\Delta u_{i+1}}{\Delta u_{i+1} + \Delta u_i} \quad (C.5)$$

$$\beta = \frac{\Delta u_i}{\Delta u_{i+1}} \left(\frac{\Delta u_i}{\Delta u_{i+1} + \Delta u_i} \right) \quad (C.6)$$

Equation C.4 becomes

$$\Sigma_{i \rightarrow i+1} \phi_i = \Sigma_{1/E} \{ \alpha \phi_i + \beta \phi_{i+1} \} \quad (C.7)$$

This prescription can be applied directly to the present multigroup code, ANISN. Adjustments on the initial matrix can be made before flux iteration by modification of the microscopic cross-section set. A negative

upscatter term is thereby introduced, and the self-scattering and down-scattering cross sections are modified according to the following equations:

$$\sigma'_{i+1 \rightarrow i} = -\beta_i \sigma_{d,i} \quad (C.8)$$

$$\sigma'_{i-1 \rightarrow i} = \sigma_{i-1 \rightarrow i} - (1 - \alpha_{i-1}) \sigma_{d,i-1} \quad (C.9)$$

$$\sigma'_{i \rightarrow i} = \sigma_{i \rightarrow i} + (1 - \alpha_i) \sigma_{d,i} + \beta_{i-1} \sigma_{d,i-1} \quad (C.10)$$

where σ denotes the unmodified cross section

σ' denotes the modified cross section

σ_d denotes the elastic downscatter cross section.

ANISN runs for the BTF Blanket No. 2 were made using these modified cross sections. However, the code had convergence trouble handling large upscatter terms.

Subsequent work was limited to changing only the downscattering cross sections in the 26-group Russian set:

$$\sigma'_{i-1 \rightarrow i} = \sigma_{i-1 \rightarrow i} - (1 - \alpha_{i-1}) \sigma_{d,i-1} \quad (C.11)$$

$$\sigma'_{i \rightarrow i} = \sigma_{i \rightarrow i} + (1 - \alpha_i) \sigma_{d,i} \quad (C.12)$$

where

$$\alpha_i = \sigma'_{d,i} / \sigma_{d,i} \quad (C.13)$$

C.2 Formulae for α_i for Different Weighting Spectra

The formulae for obtaining the α_i in Eqs. C.11 - C.13 and their values for the two weighting schemes are given below.

C.2.1 α_i for $\phi(u) \propto E$ Weighting

If $\phi(u) = CE = C'e^{-u}$, it can be shown that

$$\alpha_i = \frac{\sigma'_{d,i}}{\sigma_{d,i}} = \frac{\Sigma_{i \rightarrow i+1}}{\Sigma_{1/E}} = \frac{1}{\Sigma_{1/E}} \cdot \left\{ \Sigma_s \cdot \frac{\int_{\Delta u_{i-\xi}}^{\Delta u_i} e^{-u} du}{\int_0^{\Delta u_i} e^{-u} du} \right\} \quad (C.14)$$

$$\cong \frac{\Delta u_i}{e^{\Delta u_i} - 1}$$

For the Russian set, α_i has the values tabulated below.

TABLE C.1 α_i Values for $\phi(u) \propto E$ Weighting Spectrum

Groups	Δu_i	α_i
1 - 3	0.48	1.000*
4 - 5	0.57	0.742
6 - 8	0.69	0.694
9 - 25	0.77	0.664

*No change - fission spectrum weighted.

C.2.2 Mid-Blanket Spectrum Weighting

Rearranging Eq. C.4,

$$\frac{\Sigma_{i \rightarrow i+1}}{\Sigma_{1/E}} = \alpha_i = \frac{\left(\frac{\phi_{i+1}}{\Delta u_{i+1}} \cdot \frac{\Delta u_i}{\phi_i} \right) \Delta u_i + \Delta u_{i+1}}{\Delta u_i + \Delta u_{i+1}} \quad (C.15)$$

Using the calculated spectrum at a distance $Z = 24.43$ cm into the BTF Blanket Mock-Up No. 2, the α_i have been calculated, and the results are listed in Table C.2.

TABLE C.2 α_i Values for Mid-Blanket Weighting Spectrum

Group	Calculated Spectrum	Δu_i	α_i
1	1.432×10^5	0.48	1.0000
2	7.948×10^5	"	1.0000
3	1.534×10^6	"	1.0000
4	3.671×10^6	0.57	1.2815
5	5.738×10^6	"	1.6519
6	1.696×10^7	0.69	1.3109
7	2.750×10^7	"	1.0930
8	3.261×10^7	"	0.9848
9	3.522×10^7	0.77	0.8863
10	2.722×10^7	"	1.0700
11	3.103×10^7	"	0.7843
12	1.764×10^7	"	0.6242
13	4.384×10^6	"	2.3343
14	1.608×10^7	"	0.8925
15	1.263×10^7	"	0.8566
16	9.006×10^6	"	0.8050
17	5.494×10^6	"	0.7226
18	2.446×10^6	"	0.7154
19	1.054×10^6	"	0.6464
20	3.086×10^5	"	0.6360
21	8.392×10^4	"	0.9995
22	8.384×10^4	"	0.8941
23	6.608×10^4	"	0.8148
24	4.161×10^4	"	0.7118
25	1.762×10^4	"	0.6187
26	4.183×10^3	—	1.0000

Appendix D

The DRISTAN Code

D.1 Calculational Method

DRISTAN solves for the three parameters (C_1 , C_2 and C_3) characterizing the blanket spectrum (ϕ_i^{bkt}) by a table look-up method. The fourth parameter, the normalization constant C_0 , is calculated for each set of C_1 , C_2 and C_3 to satisfy the relation $\frac{\partial \epsilon^{bkt}}{\partial C_0} = 0$.

Since the variation of C_3 is very small, C_3 is kept constant while C_1 and C_2 are first varied. In the first iteration, ϵ^{bkt} is calculated while C_1 is varied from 0.2 to 1.7 in steps of 0.15, and C_2 is varied from 0.1 to 1.6 in steps of 0.15. A subroutine MINEP selects the minimum value of ϵ^{bkt} . The corresponding values of C_1 and C_2 at which ϵ^{bkt} is minimum are denoted by C_{11} and C_{21} . In the second iteration, ϵ^{bkt} is again calculated while C_1 is varied from $(C_{11} - 0.075)$ to $(C_{11} + 0.075)$ and C_2 is varied from $(C_{21} - 0.075)$ to $(C_{21} + 0.075)$, both in steps of 0.015. The minimum value of ϵ^{bkt} is again selected to yield C_{12} and C_{22} . Successive iterations are carried out in the same fashion, each time reducing the step size by a factor of 10 to yield final values of C_1 and C_2 . Usually, four to five iterations are sufficient to give the desired accuracy (typically $\pm 0.1\%$ in the convergence of the group fluxes). The above process is repeated while C_3 is varied from 0.2 to 0.4 in arbitrary steps. The set of values C_1 , C_2 and C_3 at which ϵ^{bkt} is minimum is then selected. The corresponding group fluxes ϕ^{bkt} are then calculated by Eq. 5.2. The code gives as printed output all the C_i 's in the last iteration and their corresponding group fluxes.

D.2 Vocabulary of the DRISTAN Code

This section presents a list of all the important symbols of the DRISTAN code and an explanation of their use. Symbols clearly defined in FORTRAN listings or those whose meanings are obvious are not listed. The input and output parameters are denoted by an asterisk.

Symbol in Code	Corresponding Symbol in Text of Report	Meaning	Input Quantity	Output Quantity
NG	N	total number of energy groups	*	*
NF	N_f	total number of foils	*	*
NT	—	total number of iterations	*	*
NI2	—	number of mesh intervals for C_2	*	*
NI3	—	number of mesh intervals for C_3	*	*
DELU(I)	Δu_i	lethargy width	*	*
CSIGMA(I)	$(\xi \Sigma_{tr})_i$	slowing-down power	*	
E(I)	E_i	energy width	*	
ACT(J)	A_j	measured activity	*	
AXS(I, J)	σ_{ij}^{ACT}	activation cross section of the j-th foil in energy group i	*	
PHI(I)	σ_i	group flux		*
A(J)	a_j	calculated activity		
XPSN	ϵ	defined by Eq. 5.5		*
C1	C_0	normalization constant		
C2	C_1	parameter characterizing the spectrum		
C3	C_2			*
C4	C_3			
II	—	number of iteration		*
MINK	—	K-th mesh point for C_2 at min. ϵ		*
MINL	—	L-th mesh point for C_3 at min. ϵ		*
EMIN		value of min. ϵ at the II-th iteration		*

D.3 Discussion of Input Parameters $\bar{E}_i$, $(\bar{\xi}\Sigma_{tr})_i$, σ_{ji}^{ACT}

(a) $\bar{E}_i$: By recalling the definition of E_i ,

$$\bar{E}_i = \frac{\int_{E_i}^{E_{i-1}} E f(E) dE}{\int_{E_i}^{E_{i-1}} f(E) dE} \quad (5.3)$$

it is apparent that $\bar{E}_i$ will change depending on the weighting spectrum. For the present work, the fission spectrum is used as the weighting spectrum ($f(E) = E e^{-E/T}$, where $T = 1.29$ MeV) in the first three groups of the 26 groups in the Russian set; an $f(E) = 1/E$ weighting spectrum is used for the rest of the energy groups.

(b) $(\bar{\xi}\Sigma_{tr})_i$: The slowing-down power $(\bar{\xi}\Sigma_{tr})_i$ in the BTF Blanket Mock-Up No. 2 is given by the following equation.

$$(\bar{\xi}\Sigma_{tr})_i = \sum_{m=1}^M \xi_{m,i} N_m \sigma_{tr,m,i}$$

where

$\xi_{m,i} = \xi$ in the i -th energy group for the m -th blanket material

$\sigma_{tr,m,i} = \sigma_{tr}$ of the i -th energy group for the m -th blanket material

N_m = nuclei concentrations of the m -th blanket material

M = total number of materials in the BTF Blanket Mock-Up No. 2.

$\sigma_{tr,m,i}$ for all the blanket materials — e.g., U^{235} , U^{238} , Na, Cr, etc. — are given by the 26-group ABBN Russian set. $\xi_{m,i}$ for all the blanket materials, including both elastic and inelastic ξ 's, have been calculated by Donohew (D6).

(c) σ_{ji}^{ACT} : The activation cross sections of gold, indium and manganese are obtained by collapsing the 620-group activation cross sections of the SAND II (M1) library over a typical LMFBR core spectrum, while the molybdenum activation cross sections are those of the 26-group ABBN Russian set. The cross sections are listed in Fortran E format in Section D.5 of this appendix.

D.4 Listing of The DRISTAN Code.

C DRISTAN- A COMPUTER CODE TO UNFOLD FAST REACTOR SPECTRUM
 C FROM FOIL ACTIVITIES

```
      DIMENSION DELU(30),CSIGMA(30),E(30),GAMMA(9000),PHI
      1(9000),C2(500),C3(5000),AXS(30,10),A(50),ACT(10),EC4
      2(500)
```

```
      COMMON K,L,EPSN(60,60),EMIN,MINK,MINL,NI2L,NI3L
111  FORMAT (5I10)
112  FORMAT (2(2F8.4,E14.5))
113  FORMAT (4E15.5)
118  FORMAT (5E12.5)
```

```
      READ (5,111) NG,NF,NT,NI2,NI3
      READ (5,112) (DELU(I),CSIGMA(I),E(I),I=1,26)
      READ (5,113) (ACT(J),J=1,NF)
      READ (5,118) ((AXS(I,J),I=1,26),J=1,NF)
      WRITE (6,111) NG,NF,NT,NI2,NI3
      WRITE (6,112) (DELU(I),CSIGMA(I),E(I),I=1,26)
      II=1
```

C C4 IS VARIED FROM -0.2 TO -0.4 AT ARBITRARY STEPS, HOWEVER,
 C C4 IS NORMALLY SET TO EQUAL -0.333

```
      C4=-0.333
      X2=0.95
      X3=0.40
      S2=0.15
      S3=0.15
      C21=X2-6.0*S2
      C31=X3-3.0*S3
      NI3L=NI3+1
      NI2L=NI2+1
      NIX=NI2/2+1
      NIY=NI3/2+1
      ANI2=NIX
      ANI3=NIY
      GO TO 1
10  S2=S2/10.0
      S3=S3/10.0
      C21=X2-ANI2*S2
      C31=X3-ANI3*S3
1  WRITE (6,114) II,X2,S2,X3,S3
114  FORMAT (///,5X,'NO OF ITERATION =',I7,'(X2,S2) =',2E15.5,
      1'(X3,S3)= ',2E15.5)
      DO 7 L=1,NI3L
      XL=L
      C3(L)=C31+XL*S3
      DO 7 K=1,NI2L
      XK=K
      C2(K)=C21+XK*S2
      DO 21 I=1,NG
      GAMMA(I)=DELU(I)/CSIGMA(I)
      EC4(I)=E(I)**C4
```



```

21 PHI(I)=(GAMMA(I)*EXP(-1.0*C2(K)*E(I)-1.0*C3(L)*EC4(I))
   SUMPHI=0.0
   DO 31 I=1,NG
31 SUMPHI=SUMPHI+PHI(I)
   DO 41 I=1,NG
41 PHI(I)=PHI(I)/SUMPHI
   IF (II.LT.3) GO TO 20
   DO 2 I=1,NG
   2 WRITE (6,115) K,L,I,PHI(I)
115 FORMAT (3I15, E15.5)
20 DO 4 J=1,NF
   SIGPHI=0.0
   DO 3 I=1,NG
   3 SIGPHI=SIGPHI+(AXS(I,J))*PHI(I)
   4 A(J)=SIGPHI
   CIU=0.0
   CID=0.0
   DO 5 J=1,NF
   CIU=CIU+A(J)/ACT(J)
   5 CID=CID+(A(J)/ACT(J))**2.0
   CI=CIU/CID
   WRITE (6,113) C1,(A(J),J=1,NF)
   XPSN=0.0
   DO 6 J=1,NF
   6 XPSN=XPSN+((ACT(J)-C1*A(J))/ACT(J))*((ACT(J)-C1*A(J))
     1/ACT(J))
   EPSN(K,L)=XPSN
   7 WRITE (6,117) K,L,EPSN(K,L)
117 FORMAT (2I15, E15.5)
   CALL MINEP
   X2=C2(MINK)
   X3=C3(MINL)
   WRITE (6,121)
   WRITE (6,122) II,MINK,MINL,EMIN,X2,X3
121 FORMAT (//,3X,'II',1X,'MINK',1X,'MINL',11X,'EMIN',13X,
   1'X2',13X,'X3')
122 FORMAT (3I5,3E15.5)
   9 II=II+1
   IF(II.ET.NT) GO TO 10
   WRITE (6,123)
123 FORMAT (///,5X, 'END OF JOB')
   END

   SUBROUTINE MINEP
   COMMON K,L,EPSN(60,60),EMIN,MINK,MINL,NI2L,NI3L
   EMIN=EPSN(1,1)
   L=1
   GO TO 22
21 L=L+1
   IF (L.GT.NI3L) GO TO 27
22 K=1
   GO TO 25
26 K=K+1
25 IF (EMIN.LE.EPSN(K,L)) GO TO 26

```

```

      IF(K.GT.NI2L) GO TO 21
      EMIN=EPSN(K,L)
      MINK=K
      MINL=L
      GO TO 26
27  WRITE (5,131) MINK,MINL,EMIN
131  FORMAT (2I10,E15.5)
      RETURN
      END

```

D.5 Typical Input to The DRISTAN Code

C NG,NF,NT,NI2,NI3

	23	3	4	10	10
C DELU,CSIGMA, E, FOR THE BTF BLANKET MOCK-UP NO.2					
.404	.0494	8.34400E 00	.460	.0294	5.17100E 00
.461	.0207	3.20700E 00	.5697	.0123	1.90600E 00
.5697	.0052	1.07800E 00	.6931	.0020	0.57710E 00
.6931	.0023	0.28860E 00	.6931	.0018	0.14430E 00
.7675	.0017	69.81000E-03	.7675	.0018	32.42000E-03
.7675	.0016	15.04000E-03	.7675	.0021	6.98100E-03
.7675	.0089	3.24200E-03	.7675	.0018	1.50400E-03
.7675	.0017	698.10000E-06	.7675	.0017	324.20000E-06
.7675	.0017	150.40000E-06	.7675	.0017	69.81000E-06
.7675	.0017	33.42000E-06	.7675	.0017	15.04000E-06
.7675	.0017	6.98100E-06	.7675	.0017	3.24200E-06
.7675	.0017	1.50400E-06	.7675	.0017	0.69810E-06
.7675	.0019	0.32420E-06	.2757	.0019	0.02520E-06

C ACT FOR GOLD, INDIUM AND MOLYBDENUM

9.50000E 08	5.54000E 06	9.36000E 07
-------------	-------------	-------------

C AXS FOR GOLD, INDIUM AND MOLYBDENUM

1.41300E-02	1.75100E-02	3.18000E-02	6.31599E-02	1.00050E-01
1.44270E-01	2.37190E-01	3.38500E-01	4.93360E-02	7.60930E-01
1.14317E 00	1.83750E 00	2.85112E 00	6.29101E 00	9.76843E 00
1.25765E 01	1.26000E 01	2.51692E 01	9.76307E 00	6.18696E 00
7.06050E 02	4.70828E 02	2.56700E 01	2.36272E 01	3.24518E 01
8.77411E 01	2.98880E-01	3.46020E-01	3.48850E-01	2.59410E-01
8.01000E-02	1.22000E-02	6.30000E-04	0.0	0.0
0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0
0.0	0.0	2.00000E-02	7.00000E-03	1.50000E-02
2.00000E-02	2.80000E-02	4.20000E-02	5.20000E-02	7.00000E-02
1.00000E-01	1.50000E-01	2.60000E-01	4.50000E-01	5.50000E-01
7.00000E-01	1.70000E 00	1.50000E 00	3.30000E 00	1.10000E 01
1.00000E 01	6.40000E-01	1.60000E-01	2.40000E-01	3.50000E-01
5.20000E-01	7.60000E-01	2.70000E 00		

C AXS FOR MANGANESE (IF MANGANESE WAS USED INSTEAD OF MOLYBDENUM)

D.6 Typical Output of DRISTAN

Only the final results, group fluxes at which the error measure, ϵ^{bkt} , is minimum, are shown.

II	MINK	MINL	EMIN	X2	X3
3	1	3	0.21056E 00	0.28250E 00	0.23050E 00

MINK	MINL	ENERGY GROUP	GROUP FLUXES
1	3	1	0.42015E-03
1	3	2	0.19316E-02
1	3	3	0.46797E-02
1	3	4	0.13646E-01
1	3	5	0.39229E-01
1	3	6	0.13571E 00
1	3	7	0.11915E 00
1	3	8	0.14486E 00
1	3	9	0.15382E 00
1	3	10	0.12477E 00
1	3	11	0.11428E 00
1	3	12	0.66518E-01
1	3	13	0.11428E-01
1	3	14	0.34805E-01
1	3	15	0.20544E-01
1	3	16	0.96638E-01
1	3	17	0.36442E-02
1	3	18	0.10353E-02
1	3	19	0.20410E-03
1	3	20	0.25005E-04
1	3	21	0.16648E-05
1	3	22	0.50492E-07
1	3	23	0.54985E-09

Appendix E

ERROR ANALYSIS

It is important to distinguish between two kinds of "errors" – the probable error and the standard deviation (SDM). They are entirely different quantities. The probable error is the error of measurement associated with the experimental method itself. It will be discussed in Section E.1. The standard deviation is a measure of the reproducibility of a certain experiment; it will be defined later in Section E.2. Since most of the measurements in this work did not involve repetitive determinations of the same quantity, the experimental errors or uncertainties referred to in the text are usually of the probable error category.

E.1 The Probable Error

Consider F as a function of the independent variables $x_1, x_2, x_3, \dots, x_N$.

$$F = F(x_1, x_2, \dots, x_N) \quad (\text{E.1})$$

Taking partial derivatives, we obtain

$$\delta F = \sum_{j=1}^N \left(\frac{\partial F}{\partial x_j} \right) \delta x_j \quad (\text{E.2})$$

Define the variance in F , σ_F^2 , in accord with the usual practice:

$$\sigma_F^2 = \sum_{j=1}^N \left(\frac{\partial F}{\partial x_j} \right)^2 (\delta x_j)^2 \quad (\text{E.3})$$

$$= \sum_{j=1}^N c_j (\delta x_j)^2 \quad (\text{E.4})$$

where $c_j \equiv \left(\frac{\partial F}{\partial x_j} \right)^2$ (E.5)

The probable error in F is defined as

$$\sigma_F = \sqrt{\sum_{j=1}^N c_j (\delta x_j)^2} \quad (\text{E.6})$$

Example 1. The Probable Error of Foil Activities in Typical (Axial) Traverses

The corrected activities from Eq. 3.1 can be written in a simpler form. For $\lambda t_c \ll 1$,

$$A = \frac{N}{w(1 - e^{-\lambda t_i}) (e^{-\lambda t_w}) t_c} \quad (\text{E.7})$$

where A = activity of foil in counts/min-mg

w = weight of foil, mg

t_i = time interval of counting, min

N = counts accumulated in time t_c (after subtraction of background).

Define $k(t) \equiv (1 - e^{-\lambda t_i}) (e^{-\lambda t_w}) t_c$ (E.8)

Then, Eq. E.7 becomes

$$A = \frac{N}{w \cdot k(t)} \quad (\text{E.9})$$

Using Eq. E.6,

$$\frac{\sigma_A^2}{A^2} \cong \frac{\sigma_N^2}{N^2} + \frac{\sigma_w^2}{w^2} \frac{\sigma_k^2}{k^2}$$

Assume the error in measuring time is negligibly small $\delta t = 0$, $\sigma_k = 0$, and use the data for gold activity at $Z = 2.34$ cm.

$$\text{From Table 4.3, } N = 24,700, \quad \sigma_N \cong \sqrt{N} = 157.2, \quad \frac{\sigma_N}{N} = 0.00637.$$

$$\text{From Table 4.1, } w = 36.85, \quad \sigma_w = 0.10, \quad \frac{\sigma_w}{w} = 0.00272.$$

$$\text{From Table 4.4, } A = 1508.5,$$

$$\begin{aligned} \frac{\sigma_A}{A} &= \sqrt{\frac{\sigma_N^2}{N^2} + \frac{\sigma_w^2}{w^2}} = \sqrt{(0.00637)^2 + (0.00272)^2} \\ &= 0.00691, \end{aligned}$$

$$\begin{aligned} \sigma_A &= 0.00691 \times 1508.5 \\ &= 10.4 \quad (\text{as indicated in Table 4.4}). \end{aligned}$$

As can be seen, the counting-statistic contribution is generally dominant.

Example 2. Probable Error in the Integrated Reaction Rate Ratios

From Eq. 4.10, let

$$R = \frac{R_{\alpha,\beta}}{R_{UC}} = \frac{N_{\alpha}}{N_{U^{238}}} \frac{A_{o\alpha,\beta}^{bkt}}{A_{oUC}^{bkt}} \frac{\sigma_{\alpha,\beta}^{th}}{\sigma_{UC}^{th}} \frac{A_{UC}^{th}}{A_{\alpha,\beta}^{th}} \frac{\int_0^L f_{\alpha,\beta}(z) dz}{\int_0^L f_{UC}(z) dz} \quad (\text{E.10})$$

To simplify the notation, let

$$\begin{aligned} x_1 &= N_{\alpha}, & x_5 &= \sigma_{\alpha,\beta}^{th}, & x_8 &= A_{\alpha,\beta}^{th}, \\ x_2 &= N_{U^{238}}, & x_6 &= \sigma_{UC}^{th}, & x_9 &= \int_0^L f_{\alpha,\beta}(z) dz, \\ x_3 &= A_{o\alpha,\beta}^{bkt}, & x_7 &= A_{UC}^{th}, & x_{10} &= \int_0^L f_{UC}(z) dz. \\ x_4 &= A_{oUC}^{bkt}, \end{aligned} \quad (\text{E.11})$$

Equation E.10 becomes

$$R = \frac{R_{\alpha,\beta}}{R_{UC}} = \frac{x_1}{x_2} \frac{x_3}{x_4} \frac{x_5}{x_6} \frac{x_7}{x_8} \frac{x_9}{x_{10}} \quad (\text{E.12})$$

Using Eq. E.3,

$$\frac{\sigma_R^2}{R^2} \cong \sum_{j=1}^{10} \left(\frac{\sigma_{x_j}}{x_j} \right)^2 \quad (\text{E.13})$$

Let us calculate the probable error in the integrated reaction rate ratio of $U^{238}(n, f)$ and $U^{238}(n, \gamma)$. From Table 4.7,

$$\sigma_{x_1} = \sigma_{x_2} = 0$$

$$\frac{\sigma_{x_3}}{x_3} = \frac{2.4}{65.3} = 3.68 \times 10^{-2}$$

$$\frac{\sigma_{x_4}}{x_4} = \frac{50.7}{10282.5} = 0.49 \times 10^{-2}$$

$$\frac{\sigma_{x_5}}{x_5} = \frac{0.01}{0.50} = 2 \times 10^{-2}$$

$$\frac{\sigma_{x_6}}{x_6} = \frac{0.09}{2.76} = 3.26 \times 10^{-2}$$

$$\frac{\sigma_{x_7}}{x_7} = \frac{80}{7377} = 1.08 \times 10^{-2}$$

$$\frac{\sigma_{x_8}}{x_8} = \frac{0.008}{0.336} = 2.38 \times 10^{-2}$$

$$\frac{\sigma_{x_9}}{x_9} = \frac{\sigma_{x_{10}}}{x_{10}} \cong 1\% \quad (\text{rough estimate only}).$$

Substituting these values into Eq. E.13,

$$\begin{aligned}\frac{\sigma_R^2}{R^2} &\cong (10^{-2})^2 \{ (3.68)^2 + (0.49)^2 + (2)^2 + (3.26)^2 + (1.08)^2 + (2.38)^2 + (1)^2 + (1)^2 \} \\ &= (10^{-2})^2 \{ 13.50 + 0.24 + 4.00 + 10.60 + 1.18 + 5.68 + 1.00 + 1.00 \} \\ &= (10^{-2})^2 (37.20)\end{aligned}$$

$$\frac{\sigma_R}{R} = (10^{-2}) \sqrt{37.20} = 6.10 \times 10^{-2} \quad (6.1\%).$$

Similarly, the same analysis can be applied to calculate the probable error of the fission ratios or other spectral indices, except that

$$A_{o\alpha,\beta}^{bkt} \int_0^L f_{\alpha,\beta}(z) dz \text{ is replaced by } A_{o\alpha,\beta}^{bkt} f(z).$$

Example 3. The Probable Error in the Buckling of Blanket Mock-Up No. 2

From Eq. 3.5,

$$B^2 = \left(\frac{\pi}{A}\right)^2 + \left(\frac{\pi}{H}\right)^2 \text{ and since, very approximately } H \approx A,$$

$$B^2 \cong 2\left(\frac{\pi}{A}\right)^2 = \frac{k}{A^2},$$

where $k = 2\pi^2.$

Using Eq. E.4,

$$\sigma_{(B^2)}^2 \cong \left(\frac{2k}{A^3}\right)^2 \sigma_A^2$$

$$\left(\frac{\sigma_{(B^2)}}{B^2}\right)^2 \cong 4 \frac{\sigma_A^2}{A^2}$$

$$\frac{\sigma_{(B^2)}}{(B^2)} \cong 2 \frac{\sigma_A}{A}$$

Numerically, $A = 188 \pm 3.8$, and thus: $\frac{\sigma_{B^2}}{B^2} \cong 2 \left(\frac{3.8}{188}\right) = 4.04\%.$

E.2 The Standard Deviation

The standard deviation is a measure of the reproducibility of a certain experiment. The consistency of the experimental method is measured in terms of the standard deviation. The standard deviation (S.D.) of a measurable quantity F , denoted by ϵ_F , is defined as (E3):

$$\epsilon_F = \sqrt{\frac{1}{N(N-1)} \sum_{i=1}^N (F_i - \bar{F})^2}$$

where $\bar{F} \equiv \frac{1}{N} \sum_{i=1}^N F_i$, and

N is the number of experimental runs.

Example. Ratio of $\text{Au}^{197}(n, \gamma)$ to $\text{In}^{115}(n, n')$ Reaction Rates from Table 5.4

Experimental Run	Gold-to-Indium Ratio	$F_i - \bar{F}$	$(F_i - \bar{F})^2$
1. Au-In-Mo	$\frac{9.50 \times 10^{-16}}{5.54 \times 10^{-18}} = 172$	+2	4
2. Au-In-Mn	$\frac{9.43 \times 10^{-16}}{5.60 \times 10^{-18}} = 168$	-2	4

$$\bar{F} = \frac{1}{2} \sum_{i=1}^2 F_i = \frac{1}{2} (F_1 + F_2) = \frac{1}{2} (172 + 168) = 170$$

$$\epsilon_F = \sqrt{\frac{1}{2(2-1)} \sum_{i=1}^2 (F_i - \bar{F})^2} = \sqrt{\frac{1}{(2)(1)} (4 + 4)} = 2$$

Actually, the preceding analysis is applicable for cases involving many repetitions. For the small number (2) involved here, one should consider corrections such as Student's t -factor if a truly sophisticated error analysis were warranted.

The reproducibility of the axial activity traverses was investigated and was found to be quite good. Table E.1 shows the standard deviation of a repeated $\text{Au}^{197}(n,\gamma)$ traverse. The errors shown in traverses 1 and 2 are the probable errors, i.e. errors assigned to uncertainties in counting statistics and foil weights. The last column shows the average value for the two foil traverses and the corresponding standard deviation from the mean. Similar checks were performed on other foil materials, with comparable results: thus one may assume that all results herein are reproducible within less than $\pm 2\%$.

TABLE E.1 The Standard Deviation of A Repeated
 $\text{Au}^{197}(n,\gamma)$ Traverse.

POSITION (from blanket- converter interface) cm	TRAVERSE 1 (with probable error)	TRAVERSE 2 (with probable error)	AVERAGE (with standard deviations, S.D.)
2.34	$1.905 \pm 1.2\%$	$1.912 \pm 1.2\%$	$1.909 \pm 0.2\%$
7.50	$1.638 \pm 1.2\%$	$1.640 \pm 1.2\%$	$1.639 \pm 0.1\%$
12.70	$1.483 \pm 1.3\%$	$1.491 \pm 1.3\%$	$1.487 \pm 0.3\%$
17.40	$1.236 \pm 1.3\%$	$1.243 \pm 1.3\%$	$1.239 \pm 0.3\%$
22.60	$1.000 \pm 1.3\%$	$1.000 \pm 1.3\%$	$1.000 \pm 0.0\%$
27.80	$0.851 \pm 1.4\%$	$0.860 \pm 1.4\%$	$0.855 \pm 0.7\%$
32.60	$0.681 \pm 1.5\%$	$0.687 \pm 1.5\%$	$0.684 \pm 0.4\%$
37.60	$0.579 \pm 1.6\%$	$0.590 \pm 1.6\%$	$0.584 \pm 1.3\%$
42.80	$0.524 \pm 1.6\%$	$0.530 \pm 1.6\%$	$0.527 \pm 0.6\%$

Appendix F

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