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SHIELD SMALL SOURCE EXPERIMENTS

Report No. 2

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Shield Small Source Experiments

ABSTRACT

MCR shield design analytical techniques and cross sections were tested by comparing calculations with measurements for experiments emphasizing the secondary gamma-ray production in specific test materials.

Measurements were made with a gamma-ray spectrometer behind a configuration consisting of a layer of gamma-ray shielding material (the test sample) adjacent to a small Po-Be neutron source, followed by neutron and gamma-ray shielding material to reduce extraneous detector response. Calculations of spectrometer response for samples of lead, tungsten, and titanium hydride were performed with a three-dimensional Monte Carlo code, derived from the one-dimensional Monte Carlo codes used in the MCR preliminary design shield analysis.

The calculations generally gave very satisfactory predictions of the measurements, the agreement being within twice the uncertainty in the experiment.

Differences between the measured and calculated responses indicated that some of the cross sections used, particularly for lead, were low and should be revised. It is estimated that these revisions would lead to a 10% increase in predicted total dose rate external to the MCR shield.

TABLE OF CONTENTS

1.	SUMMARY	1
	1.1 Shield Small Source Experiments	1
	1.2 Conclusions	8
	1.3 References for Section 1	9
2.	INTRODUCTION	10
	2.1 Description of the Military Compact Reactor (MCR) and its Shield	10
	2.2 Requirements for MCR Shield Design Analysis . . .	12
	2.3 Description of the Shield Small Source Experi- ments	13
	2.3.1 Experimental Approach	13
	2.3.2 Analytical Approach	14
	2.4 Shield Small Source Experiments – Phase I	15
	2.5 References for Section 2	16
3.	EXPERIMENTAL MEASUREMENTS AND RESULTS . . .	20
	3.1 Description of Experiment	20
	3.2 Experimental Program	22
	3.3 Experimental Procedure	23
	3.4 Data Reduction Procedure	26
	3.5 Experimental Results	28
4.	DESCRIPTION AND RESULTS OF CALCULATIONS . . .	39
	4.1 Description of Computational Methods	39
	4.2 Samples and Configurations Calculated	45
	4.3 Calculated Results	46
	4.3.1 Two-Inch Lead Sample	46
	4.3.2 One-Inch Tungsten Alloy Sample	56

4.3.3	Two-Inch $\text{TiH}_2\text{-B}_4\text{C}$ Sample	56
4.4	Machine Computation Time	66
4.5	References for Section 4	66
5.	COMPARISON AND EVALUATION OF CALCULATION AND EXPERIMENT	68
5.1	The Lead Sample Crystal Response	70
5.2	The Tungsten Sample Crystal Response	75
5.3	The $\text{TiH}_2\text{-B}_4\text{C}$ Sample Crystal Response	76
5.4	References for Section 5	78
6.	CONCLUSIONS	79
7.	APPENDIX	81
7.1	Po-Be Source Spectrum	81
7.2	Gamma Emission from Po-Be Neutron Source	85
7.3	Secondary Gamma-Ray Data	88
7.4	Composition of Materials	88
7.5	References for Section 7	88

TABLES

3.1	Experimental Shield Configurations for which Measurements Were Made	24
4.1	Region Compositions Used in Lead Calculation	48
4.2	Region Compositions Used in Tungsten Calculation	50
4.3	Region Compositions Used in Titanium Hydride Calculation	52
4.4	Summary of Calculated Results – Two-Inch Lead Sample	55
4.5	Summary of Calculated Results – Tungsten Sample	59
4.6	Summary of Calculated Results – Titanium Hydride Sample	64
4.7	CDC-1604-A Computer Time for UNC A-15 (Three- Dimensional Monte Carlo) Analyses	67
5.1	Total Energy Deposited in Detector Crystal	69

7.1	Gamma Rays from Po-Be Neutron Source	86
7.2	Secondary Gamma-Ray Production Parameters (Hydrogen, Lithium, Boron, and Carbon)	89
7.3	Secondary Gamma-Ray Production Parameters (Beryllium)	90
7.4	Secondary Gamma-Ray Production Parameters (Oxygen)	91
7.5	Secondary Gamma-Ray Production Parameters (Aluminum)	92
7.6	Secondary Gamma-Ray Production Parameters (Titanium)	93
7.7	Secondary Gamma-Ray Production Parameters (Iron)	94
7.8	Secondary Gamma-Ray Production Parameters (Tungsten)	95
7.9	Secondary Gamma-Ray Production Parameters (Lead)	96
7.10	Composition of Sample Materials	98

FIGURES

1.1	Small Source Experiment	3
1.2	Small Source Experiment, Schematic, Vertical Cross Section	4
1.3	Comparison of Measured and Calculated Response with 2-in. Lead Sample	5
1.4	Comparison of Measured and Calculated Response with 1-in. Tungsten Sample	6
1.5	Comparison of Measured and Calculated Response with 2-in. TiH_2 -1% B_4C Sample	7
2.1	FY-64 Military Compact Reactor (MCR) System	11
2.2	Comparison of Measured and Calculated Response with No Sample -- Phase I	17
2.3	Comparison of Measured and Calculated Response with 2-in. Iron Sample	18

2.4	Breakdown of Calculated Response with 2-in. Iron Sample	19
3.1	Multichannel Analyzer Used with the Sodium Iodide (Thallium Activated) Crystal Scintillation Counter . . .	21
3.2	Source-Sample Configurations	25
3.3	Raw Data before Reduction – 1-in. $\text{TiH}_2\text{-B}_4\text{C}$ Sample	27
3.4	Pulse Height Distribution with No Sample	29
3.5	Pulse Height Distribution for Configurations 1, 2, and 4 – 1/2 in., 1 in., and 2 in. of Depleted Uranium	30
3.6	Pulse Height Distribution for Configurations 2 and 3 – 1 in. Depleted Uranium and 1 in. Depleted Uranium on a 2-in. B_4C Pillow	31
3.7	Pulse Height Distribution for Configurations 5, 6, and 8 – 1/2 in., 1 in., and 2 in. of Tungsten	32
3.8	Pulse Height Distribution for Configurations 6 and 7 – 1 in. Tungsten and 1 in. Tungsten on a 2-in. B_4C Pillow	33
3.9	Pulse Height Distribution for Configurations 9, 10, and 12 – 1 in., 2 in., and 4 in. of $\text{TiH}_2\text{-B}_4\text{C}$	34
3.10	Pulse Height Distribution for Configurations 10 and 11 – 2 in. $\text{TiH}_2\text{-B}_4\text{C}$ and 2 in. $\text{TiH}_2\text{-B}_4\text{C}$ on a 2-in. B_4C Pillow	35
3.11	Comparison of Pulse Height Distributions for the No Sample, 2 in. Depleted Uranium, 2 in. Tungsten, and 2 in. $\text{TiH}_2\text{-B}_4\text{C}$ Samples	36
4.1	Macroscopic Flow Chart for UNC Code A-15 (UNC-SAM)	42
4.2	Small Source Experiment – Configuration for 2-in. Lead Sample Calculation	47
4.3	Small Source Experiment – Configuration for 1-in. Tungsten Sample Calculation	49
4.4	Small Source Experiment – Configuration for 2-in. $\text{TiH-B}_4\text{C}$ Sample Calculation	51
4.5	Comparison of Measured and Calculated Response with 2-in. Lead Sample	53

4.6	Breakdown of Calculated Response with 2-in. Lead Sample	54
4.7	Comparison of Measured and Calculated Response with 1-in. Tungsten Sample	57
4.8	Breakdown of Calculated Response with 1-in. Tungsten Sample	58
4.9	Comparison of Measured and Calculated Response with 2-in. TiH_2 -1% B_4C Sample	62
4.10	Breakdown of Calculated Response with 2-in. TiH_2 - B_4C Sample	63
5.1	Comparison of Measured and Calculated Response with 2-in. Lead Sample, Including Calculated Response with Revised Lead Inelastic GASP Values	73
5.2	Comparison of Measured and Calculated Response with No Sample – Phases I and II	74
7.1	Counter Measurements of Neutron Spectra of Three Mound Laboratory Po-Be Sources	83
7.2	Comparison of Po-Be Source Measurements	84
7.3	Po-Be Source Spectrum Used for Small Source Experiment Analysis	87

1. SUMMARY

A series of shielding experiments was performed by United Nuclear Corporation as part of the Military Compact Reactor (MCR) shield design and development effort. These Shield Small Source Experiments were performed to assess the analytical shield design techniques and cross sections for individual materials of interest in the MCR shield, with emphasis on the secondary gamma-ray production process in the specific material highlighted in each test. Other experiments, reported in Reference 1, used an MCR critical assembly as a radiation source with shield configurations closely simulating portions of the MCR preliminary design shield.

The MCR is a highly mobile, self-sufficient electric power generating plant, with a nonthermal, compact, liquid-metal cooled reactor as heat source. The shield design is based on the requirements of light weight and small size needed for MCR mobility.

To achieve the size and weight requirements imposed on the MCR system, it is essential that the shield design be based on an analysis which assures adequate attenuation without including a large safety factor. As part of the program to establish the validity of MCR shield design techniques, the experiments described herein were undertaken.

1.1 SHIELD SMALL SOURCE EXPERIMENTS

In the Shield Small Source Experiments, measurements were made with a gamma-ray spectrometer behind a configuration consisting of a layer of gamma-ray shielding material (the test sample) followed by a neutron shield, with the configuration exposed to neutrons from a small Po-Be source adjacent to the test sample. Additional material is provided to

reduce extraneous detector response. A photograph of the experimental assembly, located at United Nuclear's Pawling laboratories, is shown in Fig. 1.1, and a cross-sectional elevation is illustrated in Fig. 1.2. The entire configuration is built in rectangular geometry to facilitate direct comparison with calculations, and is designed so that secondary gamma rays from the test material constitute the principal radiation reaching the detector.

Calculations were performed with United Nuclear Corporation's three-dimensional Monte Carlo code UNC A-15, which is derived from the one-dimensional Monte Carlo codes used in the MCR preliminary design shield analysis.

The Shield Small Source Experiments were performed in two phases. The first phase was completed in 1963.² That phase of the program demonstrated the validity of the experiment and of the basic calculation techniques used. Analysis of an iron sample gave very good agreement with experimental measurements of the spectrometer response except in one region, which suggested that too hard a spectrum was used for gamma rays produced by inelastic scatter of high energy neutrons in iron. This conclusion was subsequently verified by new experimental data. Experiments were also conducted for samples of stainless steel and lead, but no analyses were performed.

The second phase, initiated in November 1964, is presented in this report. During this phase, measurements were performed for several configurations of three types of material (depleted uranium, Malloy 2000 tungsten alloy, and titanium hydride containing 1.3 w/o B₄C) as well as for a no sample configuration. Calculations of the spectrometer response for one experimental configuration each of lead (measurements performed during Phase I), tungsten, and titanium hydride were made using the UNC A-15 code.

The calculated and experimental spectrometer responses for the lead, tungsten, and titanium hydride samples are presented in Figs. 1.3, 1.4, and 1.5, respectively.

Fig. 1.1 — Small Source Experiment

Fig. 1.2 — Small Source Experiment, Schematic, Vertical Cross Section

Fig. 1.3 — Comparison of Measured and Calculated Response with 2-in. Lead Sample

Fig. 1.4 — Comparison of Measured and Calculated Response with 1-in. Tungsten Sample

Fig. 1.5 — Comparison of Measured and Calculated Response with 2-in. TiH_2 -1% B_4C Sample

1.2 CONCLUSIONS

These experiments provided a useful and valuable check of the accuracy of the MCR analytical shield design techniques and cross sections. In addition, the objective of providing a configuration in which the principal radiation measured was due to secondary gamma rays produced in the sample shielding materials was achieved.

The calculations yielded spectrometer responses ranging from 50 to 80% of the measured values. These results are generally very satisfactory, since the combined experimental and analytical uncertainty is about 25%.

Where the differences between calculated and measured results exceeded 25%, cross-section adjustments were indicated. Thus, the following cross sections used in the calculations require upward revision:

1. Lead

- a. Neutron capture cross sections
- b. Cross sections for secondary gamma-ray production by inelastic scattering

2. Mallory 2000 Tungsten Alloy

Cross sections for hard (>4 Mev) gamma-ray production by inelastic scattering in tungsten

3. Titanium Hydride-B₄C

Cross sections for gamma-ray production above 2.5 Mev by Ti inelastic scattering.

Only the cross-section improvements indicated for lead have any significant influence on previous predictions of dose rate from the MCR shield. The net effect of all these changes in cross section is estimated to increase the predicted total neutron and gamma-ray dose rate external to the MCR shield by approximately 10%.

1.3 REFERENCES FOR SECTION 1

1. Schamberger, R., Grochowski, G., and Hanfling, R.: Military Compact Reactor Critical Assembly Shield Experiments, UNC-5115 (Apr. 1965). (Confidential).
2. Schamberger, R., Celnik, J., and Ross, R.: Shield Small Source Experiments, UNC-5072 (Nov. 8, 1963).

2. INTRODUCTION

A shield experiments program, described in this report, was performed by United Nuclear Corporation as part of the Military Compact Reactor (MCR) shield design and development effort. The experiments were performed to assess the analytical shield design techniques and cross sections for individual materials of interest in the MCR shield.

2.1 DESCRIPTION OF THE MILITARY COMPACT REACTOR (MCR) AND ITS SHIELD

The Military Compact Reactor is a highly mobile, self-sufficient, electric power generating plant of 3000-kw(e) capacity being developed for the U. S. Army under AEC Contract AT(30-1)-3078 with Allison Division of General Motors Corporation. United Nuclear Corporation, as subcontractor to GM, is responsible for the design and development of the reactor, shields, and primary coolant loop.

The plant designed in FY-64, to be built and transported in five packages, is shown in Fig. 2.1 assembled for field operation. The shield is the shaded region at the forward end of the power generation unit (reactor, primary coolant system, and heat exchangers) which is directly coupled during operation with the power conversion unit (turbine and alternators). These coupled units constitute the large assembly at the center of the drawing.

The heat source for the MCR is a nonthermal, compact, liquid-metal cooled reactor. The MCR shield design is based on the requirements of light weight and small size needed for MCR mobility. The complete design of the reactor and shield, including the analyses of their performance is fully described in Volumes I, II, and III of the MCR Preliminary Design Report.¹

Taken from:
EDR 3950, Vol. II

Fig. 2.1 — FY-64 Military Compact Reactor (MCR) System

2.2 REQUIREMENTS FOR MCR SHIELD DESIGN ANALYSIS

To achieve the stringent size and weight requirements imposed on the MCR system, it is essential that the shield design be based on an analysis which assures that attenuation requirements are satisfied without including a large safety factor. Such an analysis, which requires the accurate determination of neutron and gamma-ray production and transport through a complex geometry of many materials, has three major aspects:

1. Calculation

The ability to perform the solution of the transport equation for neutron and gamma rays in complex geometries.

2. Cross Sections

Knowledge of cross-section data for all interactions of importance (particularly secondary gamma-ray production), including an accurate representation of the data when used in the solution of the transport equation.

3. Engineering Application

The determination of the effects of engineered shield features, such as ducts, gaps, and other inhomogeneities.

Uncertainties in each of these areas existed for the calculation of MCR shields. Therefore, verification of and/or improvement in the MCR shield design analytical techniques were of prime importance to the program. The MCR shield small source experiments described in this report were performed as part of an initial effort toward this end.

In addition to the small source experiments, a Critical Assembly Shield Experiments program was performed in conjunction with the MCR-C1 critical assembly radiation source.² The latter experiments, reported separately,³ were directed toward an over-all assessment of the applicability and accuracy of analytical techniques and cross sections used for MCR shield design. This was achieved by using these techniques and cross sections to calculate the shielding characteristics (e.g., neutron and gamma-ray dose rates, material activations, and gamma heating) of several partial representations of an MCR-type shield.

The experiments performed to date have not been addressed toward engineering application. An additional experimental program, utilizing a Shield Measurements Reactor or a Shield Spectrum Generator⁴ as a radiation source had been planned as the basis for investigating the engineering aspects. These additional experiments have not been initiated.

2.3 DESCRIPTION OF THE SHIELD SMALL SOURCE EXPERIMENTS

The purpose of the MCR Shield Small Source Experiments was to assess the accuracy of the MCR analytical shield design techniques and cross sections for individual materials of interest in the MCR shield, with emphasis on the secondary gamma-ray production process in the specific material highlighted in each test. These experiments were performed in two phases. The first phase, completed in 1963 and reported in Reference 5, is described briefly in Section 2.4. The second phase, which was initiated in November 1964, is presented in this report. The experimental and analytical approaches used for the current program are summarized below, with the details and results presented in Section 3 (experimental) and Section 4 (analytical). Comparison and evaluation of the results are presented in Section 5, with the conclusions stated in Section 6.

2.3.1 Experimental Approach

The experiment was designed on the basis of the following criteria:

1. Requirement: The geometry should be well-treated by an available Monte Carlo code.

Solution: The test assembly was constructed in rectangular geometry compatible with the UNC A-15 (UNC-SAM) three-dimensional Monte Carlo code. UNC A-15 is the direct descendant of the UNC A-1 (ADONIS) code used during the first phase of the program.

2. Requirement: Shield materials of interest should be exposed to a high-energy neutron flux of sufficient intensity to produce readily measured secondary gamma radiation. The source neutron spectrum should be known.

Solution: The shield material samples were placed immediately adjacent to a Po-Be neutron source.

3. Requirement: The effect of secondary gamma-ray production in materials other than the sample being investigated should be minimized. Detector response to gamma rays directly from the neutron source should also be minimized.

Solution: An LiH shield was located behind the sample to attenuate the neutron flux leaving the sample with a minimum of secondary gamma-ray production. A lead shadow shield was used between the source and the detector. Additional neutron and gamma shielding of the detector further reduced extraneous secondary gamma response.

4. Requirement: The experiment should give as much information as feasible on gamma-ray spectra to permit distinguishing between gammas from different sources and to give an indication of inelastic gamma-ray energies.

Solution: An NaI scintillation spectrometer was used with a 256-channel analyzer.

The output of each experiment was the number of pulses per source neutron detected by the scintillation spectrometer. These pulses were divided into groups according to the amplitude of the pulse. Each group corresponded to an interval of 120 kev in the energy deposited by the gamma ray in the spectrometer.

2.3.2 Analytical Approach

To meet the objectives of this program, the analysis of the experiment was performed with the UNC A-15 (UNC-SAM) three-dimensional Monte Carlo code, which uses the same analytical methods as the one-dimensional UNC A-2 (SANE) and UNC A-3 (SAGE) Monte Carlo codes used in the MCR design. It is also capable of accurately describing the experimental geometry. The analyses predicted the pulse height distribution measured by the gamma-ray spectrometer, for direct comparison with the output of the measurements. This is a simpler and more direct procedure for obtaining a comparison than would be en-

countered by converting the measured instrument response to the incident gamma-ray spectrum, since gamma rays entering the detector at one energy generate counts in channels at other energies.

The calculation of the spectrometer response determines separately the contribution of each of the following components to the expected pulse height distribution.

1. Capture gamma rays with $E \geq 4$ Mev from the sample
2. Capture gamma rays with $E < 4$ Mev from the sample (for the lead calculation all capture gamma rays were run in one problem)
3. Inelastic scattering gamma rays with $E \geq 4$ Mev from the sample
4. Inelastic scattering gamma rays with $E < 4$ Mev from the sample
5. Source gammas
6. Other gammas.

Further breakdowns of the gamma-ray signal by region of origin were sometimes made to facilitate the calculations. The composite curve resulting from the sum of these components for all materials used in the experimental configuration is compared directly to the experimental curve.

2.4 SHIELD SMALL SOURCE EXPERIMENTS – PHASE I

The initial phase of the Shield Small Source Experiments was performed during 1962-1963. Measurements were obtained for samples of iron, lead, stainless steel, and for the no-sample case. The experimental configuration and detector used were essentially identical to that employed in the present measurements (see Fig. 1.2). The two small differences were: (1) the Po-Be neutron source used in the initial measurements was of a slightly smaller geometric size, and (2) gain stabilization was achieved with a Cosmic Radiation Laboratories, Inc., Spectrastat and a small Na^{24} source. The Spectrastat is a special power supply for the photomultiplier tube with an output voltage to the tube which is adjusted automatically to maintain a constant gain for the crystal-photomultiplier-amplifier system. (During the second phase, this

gain stabilization system was eliminated, which resulted in less spectral distortion in the low energy region. See Section 3.1.)

Calculations were performed for the no-sample configuration and for a 2-in. thick iron sample using the UNC A-1 (ADONIS) three-dimensional Monte Carlo code (the immediate antecedent of UNC A-15). The rectangular geometry of the experiment was adopted to conform with the requirement of this code, which was the only three-dimensional Monte Carlo program available at that time. The calculated and experimental spectrometer responses for the no-sample and iron sample cases are presented in Figs. 2.2 and 2.3 respectively, with the breakdown of the calculated response shown in Fig. 2.4.

The results of this program are reported in UNC-5072.⁵ The agreement obtained between the calculated and measured results demonstrated the validity of the experiment and the basic analytical technique used. The iron sample data (Figs. 2.3 and 2.4) showed excellent agreement except in one region, which suggested too large a secondary gamma-ray production cross section due to inelastic scattering for iron at energies in the range of 4 to 6 Mev. This conclusion was subsequently verified by new experimental data.

2.5 REFERENCES FOR SECTION 2

1. Allison Division, General Motors Corporation: MCR Preliminary Design Report, Vols. I, II, and III, EDR-3950 (July 31, 1964). (Confidential).
2. Brooks, W., Hanfling, R., and Marotta, C.: Military Compact Reactor Critical Experiments Program, MCR-C1, UNC-5109 (Mar. 15, 1965). (Confidential).
3. Schamberger, R., Grochowski, G., and Hanfling, R.: Military Compact Reactor Critical Assembly Shield Experiment, UNC-5115 (Apr. 1965). (Confidential).
4. Loeb, W. A.: Shield Spectrum Generator Experiment Design Report, UNC-5103 (Dec. 1964). (Confidential).
5. Schamberger, R., Celnik, J., and Ross, R.: Shield Small Source Experiments, UNC-5072 (Nov. 8, 1963).

Fig. 2.2 — Comparison of Measured and Calculated Response with No Sample — Phase I

Fig. 2.3 — Comparison of Measured and Calculated Response with 2-in. Iron Sample

Fig. 2.4 — Breakdown of Calculated Response with 2-in. Iron Sample

3. EXPERIMENTAL MEASUREMENTS AND RESULTS

The experimental measurements portion of the small source program was designed to provide reliable experimental observation of gamma radiation which could be compared directly with the results of calculations. The gamma-ray source of most interest is the secondary source produced by neutron interactions in materials considered in the MCR shield. These measurements were performed at United Nuclear's Critical Experiments Laboratory, located in Pawling, New York.

3.1 DESCRIPTION OF EXPERIMENT

The experimental configuration is a rectangular geometry compatible with the UNC A-15 (or UNC A-1) codes. A vertical section through the center of the experimental configuration is shown in Fig. 1.2. A photograph of this system, located in United Nuclear's Pawling critical facility, is shown in Fig. 1.1. Fig. 3.1 is a photograph of the multichannel analyzer used in the measurements.

Sample materials were placed immediately on top of the lithium hydride as shown in Fig. 1.2. A nominally 40-curie Po-Be neutron source (0.76 in. in diameter and 0.94-in. high) was located so that the bottom of the source was 2 in. above the bottom of the sample. For samples of thicknesses less than 2 in., the source was placed on a $1\frac{1}{2} \times 1\frac{1}{2}$ -in. Pb block of appropriate height. The lithium hydride below the sample provided a simulation of a high performance shield with little secondary gamma-ray production. In conjunction with the borated hydrocarbon below the sample, it also served to reduce neutron interactions in the NaI crystal.

Fig. 3.1 — Multichannel Analyzer Used with the Sodium Iodide (Thallium Activated) Crystal Scintillation Counter

The rest of the assembly served to reduce extraneous detector response. The lead shadow shield between the source and the detector reduced detector response from gammas emitted directly by the source. The additional lead shielding was provided as a collimator to attenuate secondary gamma rays produced in external material, such as the walls of the room, while it permitted gamma rays produced in the sample to penetrate to the detector. The neutron shielding outside the lead served to reduce the effect of neutrons scattered back from the walls of the room. The boron carbide around the detector further reduced neutron absorption in the crystal.

For each configuration, measurements were taken after the detector system gain had come to a reasonably steady value before the neutron source was put in place. Data were accumulated at several intervals of approximately 1-hr duration. The gain was decreased slightly each time to compensate for the changes in the photomultiplier tube gain which occurs in the subsequent measurement. A sufficient number of measurements were taken at 1-hr intervals to allow for selection of data with no more than 1 to 2% shift in gain. This procedure was adopted in preference to gain stabilization using the Spectrastat and an external gamma source (Phase I) because measurements had indicated that it was nearly as fast and resulted in less spectral distortion in the low energy region.

3.2 EXPERIMENTAL PROGRAM

Measurements were made with samples of Mallory 2000 tungsten alloy (95 w/o W, 3.5 w/o Ni, 1.5 w/o Cu), titanium hydride containing 1.3 w/o boron carbide, and depleted uranium (0.28% U^{235}). In addition, measurements were made for the no-sample configuration. For each particular material, measurements were made on three sample thicknesses, one thinner and one thicker than the sample analyzed. The effort and costs involved in the analytical prediction of any one configuration are much larger than those involved in obtaining experimental results, once the experimental apparatus has been set up. Consequently, these additional measurements were performed to add supporting evidence that the experimental results for the one configuration for each material being analyzed (the intermediate thickness in each case) are reasonable.

A measurement with the intermediate thickness of each material was also made with a 2-in. B_4C layer between the sample and the LiH region. This additional shielding served to reduce the number of low energy captures resulting from neutrons backscattered from the LiH. These measurements were performed to obtain a qualitative indication of the significance of low energy neutron captures resulting from neutrons moderated in the LiH and returned to the sample, and to observe the effect of a heavily absorbing layer on them.

The particular samples for which measurements were obtained are outlined in Table 3.1. Fig. 3.2 contains a sketch of each source-sample configuration, identified by number.

3.3 EXPERIMENTAL PROCEDURE

Each measurement included the following steps:

1. Measuring the residual activity and background radiation prior to the introduction of the Po-Be source.
2. Adjusting the amplifier gain so that the 2.76-Mev peak of Na^{24} was in channel 61. The Na^{24} was contained in a capsule of NaF, on the end of a short wand, which could be positioned adjacent to the detector. The source was activated by the Po-Be neutron source in its storage pig each night.
3. Removing the Na^{24} source and introducing the Po-Be neutron source.
4. Accumulating the detector response at intervals of approximately 1 hr until sufficient counts were accumulated to insure good statistical accuracy with the desired small gain shift. After each interval, the gain of the system was checked and adjusted in small increments.
5. Measuring the crystal response as a function of time following the removal of the Po-Be source to determine the 25.0-minute half-life I^{128} activation of the crystal during the measurements.

TABLE 3.1 — EXPERIMENTAL SHIELD CONFIGURATIONS FOR WHICH MEASUREMENTS WERE MADE

<u>Source-Sample Configuration</u>	<u>Sample Material</u>	<u>Dimensions, in.</u>
1	1/2-in. depleted U (0.28% U ²³⁵)	14 × 14 × 1/2
2	1-in. depleted U	14 × 14 × 1
3*	1-in. depleted U	14 × 14 × 1
4	2-in. depleted U	14 × 14 × 2
5	1/2-in. Mallory 2000 (95 w/o W), 3.5 w/o Ni, 1.5 w/o Cu)	12 × 12 × 1/2
6	1-in. Mallory 2000	12 × 12 × 1
7*	1-in. Mallory 2000	12 × 12 × 1
8	2-in. Mallory 2000	12 × 12 × 2
9	1-in. TiH ₂ -B ₄ C (1.3 w/o B ₄ C)	14 ¹ / ₄ × 12 ³ / ₈ × 1
10	2-in. TiH ₂ -B ₄ C	14 ¹ / ₄ × 12 ³ / ₈ × 2
11*	2-in. TiH ₂ -B ₄ C	14 ¹ / ₄ × 12 ³ / ₈ × 2
12	4-in. TiH ₂ -B ₄ C	14 ¹ / ₄ × 12 ³ / ₈ × 4
13	No sample	

*In these measurements, a 14 × 14 × 2-in. region of B₄C powder contained in 0.020-in. polyvinyl chloride was located between the LiH and the sample source regions.

Fig. 3.2 — Source-Sample Configurations

The I^{128} activity was measured to gain an indication of the neutron interactions in the crystal (see Section 3.4).

The detector response measured in step 4 for the 1-in. thick TiH_2-B_4C sample is shown in Fig. 3.3 as representative of the uncorrected experimental data. The 2.23-Mev hydrogen capture gamma-ray peak is observed in channel 48.

3.4 DATA REDUCTION PROCEDURE

The data were processed in the following manner to yield experimental results which can be compared directly with the output of the UNC A-15 Monte Carlo calculations.

1. A sufficient number of measurements (taken at ~1-hr intervals) which vary in gain by less than 1 to 2% to insure adequate statistics were combined.
2. The background recorded before the introduction of the Po-Be source was subtracted.
3. Data were combined to provide the response in 120-kev intervals corresponding to the energy intervals being employed in the Monte Carlo calculations.
4. An approximation of the response resulting from fast neutron interaction in the detector and its immediate environs was subtracted from the data. The magnitude of this response was determined by normalization to the measured I^{128} activation of the crystal. Since the small source experiment had been set up to minimize neutron interaction in the crystal, a precise evaluation of the neutron response was not required. The estimate employed resulted in a correction of a few per cent at energies below 7 Mev, and a negligible correction at higher energies for samples with strong secondary gamma production. In shield configuration 13, with no sample present, it was about a 20% correction above 7 Mev.
5. The data were normalized to one source neutron using the value of the source strength as given by the vendor (Nuclear

Fig. 3.3 — Raw Data before Reduction — 1-in. $\text{TiH}_2\text{-B}_4\text{C}$ Sample

Materials and Equipment Corporation) at the time of manufacture, a 138-day half life for Po, and the known data accumulation time.*

3.5 EXPERIMENTAL RESULTS

The experimental results, reduced for comparison with UNC A-15 calculations, are shown in Figs. 3.4 through 3.11. They consist of the pulse height distributions plotted as histograms of counts per 120-kev channel per source neutron vs gamma energy absorbed in the spectrometer. Reference should be made to Fig. 3.2 for specification of the particular experimental source-sample configuration.

Fig. 3.4 shows the results with no sample present (configuration 13).

Fig. 3.5 presents the results for 1/2-in., 1-in., and 2-in. depleted uranium samples. No large increase in detector response is observed by the introduction of a depleted uranium sample as can be seen in Fig. 3.11 where the no sample and 2-in. depleted uranium sample are plotted on the same graph. This is an indication of low secondary gamma production.

Increasing the depleted uranium sample thickness from 1 in. to 2 in. decreases the measured response as would be expected, since uranium is a good gamma-ray shield as well as an effective neutron shield. Increasing the sample thickness from 1 in. to 2 in. adds a gamma-ray source which is effectively shielded from the spectrometer by the lower half of the sample while reducing the neutron fluxes which produce gamma rays in other parts of the assembly.

Fig. 3.6 shows the result of adding a 2-in. thick region of B_4C between the 1-in. depleted uranium sample and the LiH region. The increase in the detector response is due to secondary gamma-ray production in the B_4C . The magnitude of the response to gamma rays produced in the B_4C is large enough to obscure any effect of a reduction in low energy capture in the uranium.

*An independent check of the source strength agreed within 2% of the value given by the vendor.

Fig. 3.4 — Pulse Height Distribution with No Sample

Fig. 3.5 — Pulse Height Distribution for Configurations 1, 2, and 4 — 1/2 in., 1 in., and 2 in. of Depleted Uranium

Fig. 3.6 — Pulse Height Distribution for Configurations 2 and 3 — 1 in. Depleted Uranium and 1 in. Depleted Uranium on a 2-in. B_4C Pillow

Fig. 3.7 — Pulse Height Distribution for Configurations 5, 6, and 8 — 1/2 in., 1 in., and 2 in. of Tungsten

Fig. 3.8 — Pulse Height Distribution for Configurations 6 and 7 —
1 in. Tungsten and 1 in. Tungsten on a 2-in. B₄C Pillow

Fig. 3.9 — Pulse Height Distribution for Configurations 9, 10, and 12 — 1 in., 2 in., and 4 in. of $\text{TiH}_2\text{-B}_4\text{C}$

Fig. 3.10 — Pulse Height Distribution for Configurations 10 and 11 —
2 in. $\text{TiH}_2\text{-B}_4\text{C}$ and 2 in. $\text{TiH}_2\text{-B}_4\text{C}$ on a 2-in. B_4C Pillow

Fig. 3.11 — Comparison of Pulse Height Distributions for the No Sample, 2 in. Depleted Uranium, 2 in. Tungsten, and 2 in. $\text{TiH}_2\text{-B}_4\text{C}$ Samples

Fig. 3.7 contains the results of measurements with Mallory 2000 of thicknesses 1/2 in., 1.0 in., and 2.0 in., which correspond to configurations 5, 6, and 8, respectively. The no sample case has also been replotted in this figure. Comparison of the W sample measurements with the no sample situation indicates substantial secondary gamma-ray production below 7 Mev. In addition, increasing the W sample thickness reduces the detector response in the lower energy portion of the distribution. The magnitude of this effect with the W sample is less than that observed with depleted uranium. This can be attributed partly to the increased secondary production in the W.

The result of placing the 2.0-in. B_4C pillow between the 1.0-in. W sample and the LiH region is presented in Fig. 3.8. Comparing this measurement with the 1.0-in. W sample alone suggests that a significant portion of the secondary production in W is a result of low energy neutron captures.

Fig. 3.9 presents the results of measurements made with 1-in., 2-in., and 4-in. samples of TiH_2-B_4C . The large increase in detector response over the no sample measurement establishes that the bulk of the radiation observed with the TiH_2-B_4C samples is from secondary gamma rays produced in them. The broad peak at about 6 Mev is indicative of capture gamma rays. It is observed that the secondary gamma-ray production increases with increasing sample thickness. Since the TiH_2 is only a moderately effective gamma-ray shield, the secondary gamma-ray sources in the upper parts of the sample contribute significantly to the detector response. In addition, the addition of samples above the source reflects source neutrons into the lower portion of the sample, thus increasing the secondary gamma-ray production there.

The effect of introducing the 2-in. B_4C pillow between the 2-in. TiH_2-B_4C sample and LiH region is shown in Fig. 3.10. Introduction of 2 in. of B_4C between the sample and the LiH should reduce the detector response to high energy gamma rays produced in the TiH_2 by approximately 35%, as a result of the change in geometry and the gamma-ray attenuation in the B_4C . An increase of a few per cent would be expected for secondary gamma rays produced in the B_4C . Therefore, the observed data are consistent with the conclusion that the introduction of the B_4C layer does not effectively alter whatever low energy neutron

absorption occurs in the sample. Since the $\text{TiH}_2\text{-B}_4\text{C}$ sample contains only 1 w/o B, moderation in the sample should determine the low energy neutron spectrum.

Fig. 3.11 gives an intercomparison of the results for 2-in. samples of depleted U, Mallory 2000 (W), and $\text{TiH}_2\text{-B}_4\text{C}$. Significantly greater secondary gamma-ray production in the $\text{TiH}_2\text{-B}_4\text{C}$ relative to depleted U and W can be observed.

4. DESCRIPTION AND RESULTS OF CALCULATIONS

The spectrometer responses for experimental configurations, one each of lead, tungsten, and titanium hydride, were calculated using the UNC A-15 (UNC-SAM) code, on the CDC-1604-A computer at United Nuclear Corporation. Each analysis consisted of two major parts:

1. Solution of the neutron transport problem to generate gamma-ray sources
2. Solution of the subsequent gamma-ray transport problem, which is divided into several categories of gamma rays, with the contribution of each category determined independently.

The neutron transport calculation was solved using normal Monte Carlo tracking techniques. Gamma-ray transport was evaluated by two scoring methods; normal Monte Carlo tracking to the detector (lead analysis), and "flux-at-a-point" estimation (tungsten and titanium hydride analyses).

A general description of the methods employed is presented in Section 4.1. Section 4.2 contains the details of the specific configuration analyzed. The calculated results for each sample are presented in Section 4.3.

4.1 DESCRIPTION OF CALCULATIONAL METHODS

Calculations of the Shield Small Source experiment were performed using the UNC A-15 (UNC-SAM) shielding code.¹ UNC A-15 calculates the solution of the Boltzmann transport equation, by the Monte Carlo method, for either neutrons or gamma rays in three-dimensional geometry. This program is an elaboration and combination of several rou-

tines used for the initial small source calculations.² The UNC A-15 code differs in three main aspects from the UNC A-1 (ADONIS) code previously used:³ additional geometric capabilities, "flux-at-a-point," and single operating sequence.

1. Additional Geometric Capabilities

In addition to rectangular geometry, routines for tracking in spherical and cylindrical configurations were added. However, though the source and detector regions are cylindrical, these regions were mocked up as rectangular for consistency with the Phase I calculations. Furthermore, it was felt that this approximation would have a very small effect on the calculated results.

2. "Flux-at-a-Point"

The "flux-at-a-point" scoring was incorporated. This routine, developed by M. H. Kalos at United Nuclear, permits the calculation of the flux for a point detector with finite variance, as described in Reference 4. The technique assumes that the calculation of the uncollided flux is determined analytically. For each collision (and source) point generated in the Monte Carlo tracking process, the flux-at-a-point routine determines the "once-more" collided flux contribution. This permits a particle to "score," i.e., to contribute to the detector flux, without requiring it actually to reach the detector.

This routine was used for the tungsten and titanium hydride sample calculations. A point detector was placed at the center of the crystal and the flux was determined at that point.

3. Single Operating Sequence

Previously, the various supplemental routines (the generation of the macroscopic cross sections for each of the regions, the generation of the complex surfaces required by the geometry routines, and the primary and secondary source generation routines) all were run as separate computer jobs. UNC A-15 combines all these routines into a single operating sequence.

Beginning with the microscopic cross-section data tape, the following routines can be performed as one job, as illustrated in Fig. 4.1:

- a. Given the material concentrations, an organized data tape bearing the total macroscopic cross sections, as well as other necessary probability data, is created by DATORG.
- b. EZGEOM then takes the simplified geometrical description of the physical system, as provided by the problem originator, and produces the rather complex set of data required by the transport program ADONIS.
- c. For a primary source problem, VANGEN creates the required neutron or gamma source tape. For a secondary problem, the GASP routine creates the secondary gamma source tape from a previously generated neutron interaction tape using the appropriate probability numbers. The set of GASP input, describing the probable number and energy of gammas created in an absorption or inelastic event for each nuclide of interest as a function of neutron energy, is given in Appendix 7.3.
- d. Given the appropriate output tapes of DATORG, EZGEOM, VANGEN, or GASP, the neutron or gamma-ray transport program, ADONIS, can be performed. The ADONIS tracking process is completely independent of the group structure used to describe neutron and gamma-ray input source spectra. Each neutron or gamma ray is followed at its specific energy as determined by Monte Carlo sampling.

Using the UNC A-15 code sequence enumerated above, the calculations were performed in several steps.

A neutron VANGEN problem was run using the best available Po-Be neutron spectrum (described in Appendix 7.1). For continuity with the no sample and iron sample Phase I configurations previously calculated,² and to save computer time, the configuration was divided into two parts for the neutron ADONIS program. The first part contained the upper half through the lowest lithium-hydride region (region 44 in Fig. 4.2).

Fig. 4.1 — Macroscopic Flow Chart for UNC Code A-15 (UNC-SAM)

A transmitting surface was placed at the lower boundary of region 44 to record on tape the energy, spatial coordinates and direction of all neutrons crossing the boundary. This transmission tape then could be used as the neutron source for the lower part of the configuration.

The output of the neutron ADONIS problem was an interaction tape listing all gamma-ray producing interactions (absorption and inelastic scatter). This output was then processed by GASP to provide the secondary gamma source tapes required by the gamma-ray ADONIS problems.

A gamma organized data tape was created by DATORG. Because the physical configurations for the neutron and gamma-ray ADONIS problems were identical, the same EZGEOM input was used.

For the gamma-ray ADONIS problems, the transmission plane was removed so the gamma rays could be tracked to the detector region. For the lead sample calculation the basic ADONIS scoring routine was used (i.e., not "flux-at-a-point"). A transmission tape was generated listing the energies of all gamma rays striking the detector surface, which were then collected into 75 energy groups. The pulse-height distribution response of the detector (in 120-keV bins) was calculated at the mean energy of each of these groups, using the RESPON program. The 75 separate pulse-height distributions were then summed to yield the total detector output distribution.

For the W and $\text{TiH}_2\text{-B}_4\text{C}$ sample calculations the crystal response was determined using flux-at-a-point scoring. At each source and collision point both the weighted flux contribution and the present photon energy E are recorded on the transmission tape. The RESPON program was modified so that the crystal response contribution from each source and collision point is the actual response for a photon of energy E , multiplied by the weighted flux contribution. Since the flux-at-a-point routine calculates the crystal response per unit area, the final response was multiplied by the crystal cross-sectional area that the incident photon sees. Since most of the photons are incident from above, the cross-sectional area was taken to be the area of the upper face of the crystal (45.6 cm^2). This assumption results in $\sim 7\%$ uncertainty in the final crystal response.

The contribution from the various secondary gamma-ray sources was determined separately to provide more detailed diagnostic capability. The GASP source generation and the subsequent gamma ADONIS problem was run separately for

1. The high-energy (≥ 4 Mev) sample inelastic gamma rays.
2. The low-energy (< 4 Mev) sample inelastic gamma rays.
3. The high-energy (≥ 4 Mev) sample capture gamma rays.
4. The low-energy (< 4 Mev) sample capture gamma rays (these were combined with the high-energy capture gamma rays for the lead analysis).
5. The high-energy (≥ 4 Mev) inelastic and capture gamma rays from the other elements in the upper part of the configuration. (These "other" elements include the impurities found in the sample plus iron, beryllium, aluminum, and lithium. Lead was included as an "other" element for the tungsten and titanium-hydride samples.)
6. The low-energy (< 4 Mev) inelastic and capture gammas from the "other" elements described above. In addition, the hydrogen capture gammas from the lithium-hydride regions were included in this group, except for the $\text{TiH}_2\text{-B}_4\text{C}$ sample in which they were lumped together with the hydrogen capture and carbon inelastic (4.43 Mev) gammas from the sample as a separate problem.
7. The crystal response from the primary source gammas, which was calculated for each sample configuration. These primary gammas originate from the excited state of C^{12} , resulting from the $\text{Be}^9(\alpha, n)\text{C}^{12}$ reaction. For the calculation, 0.6 of a 4.43-Mev photon per source neutron was used.
8. The contribution from secondary gammas originating below the lithium-hydride regions (region 44), which was determined in the following manner. The neutron flux in region 44 for each of the three samples calculated was compared to that for the no-sample configuration previously calculated.² The ratio of

the flux for the sample cases to that for the no-sample configuration (for region 44) was relatively independent of the neutron energy (within 10 to 15%) except above 8 Mev where the ratio was considerably smaller. As the neutron flux above 8 Mev is a small fraction of the total flux, it is meaningful to consider the crystal response from the lower part of the configuration to be equal to that of the no-sample case normalized to the average ratio of the fluxes (below 8 Mev) in region 44. This flux ratio was found to be approximately 1 for the lead sample, 0.8 for the tungsten, and 0.67 for the titanium-hydride sample configuration.

The contribution to the crystal response from the secondary gammas originating in the lower part of the configuration is generally negligible above 2 Mev. Below 2 Mev the contribution is approximately 10%, for both the sample and no-sample cases.

To increase the efficiency of the ADONIS calculations, neutrons were picked from an isotropic distribution for the 2π solid angle facing the shield. For the lead sample calculations, the gamma rays were chosen with a distribution confined to a narrow cone ($\pm 45^\circ$ from vertical axis) directed toward the detector. The contributions from the primary gamma rays and secondary gamma rays (originating in the upper part) fall off rapidly, except for the lowest energy groups, as the angle is increased. As mentioned earlier, the flux-at-a-point scoring routine was used in the gamma-ray ADONIS problems for the tungsten and titanium hydride samples. This required that the gamma rays be picked from an isotropic distribution.

4.2 SAMPLES AND CONFIGURATIONS CALCULATED

The spectrometer response was determined using the UNC A-15 code for the following:

- 2-in. lead sample (experimental measurement performed during Phase I – Reference 2)
- 1-in. tungsten alloy sample (configuration 6 – Table 3.1)
- 2-in. $\text{TiH}_2\text{-B}_4\text{C}$ sample (configuration 10 – Table 3.1).

The analytical representations of the physical configurations and the material concentrations of each region are presented in Fig. 4.2 and Table 4.1 for lead, Fig. 4.3 and Table 4.2 for tungsten, and Fig. 4.4 and Table 4.3 for titanium hydride.

The cylindrical source and detector are mocked up as rectangular regions (see Section 4.1), and the paraffin layer outside the lead regions in the lower part of the configuration is not included in the analysis. These approximations should have a negligible effect on the calculated results.

Impurities in the sample materials (see Appendix 7.4) are not included when their concentration is low or when their cross sections are not available on the element data tape. Some substitutions were made, based on equivalent electron densities. These substitutions introduce minor uncertainties since the major method of gamma-ray attenuation for moderate energy gamma rays is by Compton scattering, which is a function of electron density. The tungsten sample is an exception, however, since relatively large amounts of Ni(3.5 w/o) and Cu(1.5 w/o) contained in the sample were treated as iron. The effect of this substitution on the total crystal response below 6 Mev is generally less than 20% (see Section 5.2).

4.3 CALCULATED RESULTS

4.3.1 Two-Inch Lead Sample

The calculated total crystal response for the 2-in. lead sample configuration is shown in Fig. 4.5. The partial contributions from the various components are plotted in Fig. 4.6 and tabulated in Table 4.4. To avoid confusion, the calculated data are shown as continuous curves, rather than as histograms, which would indicate more properly the energy group nature of the output of the calculation.

As shown in Fig. 4.6, primary source gammas and low energy (<4 Mev) lead inelastic gammas dominate the calculated response at low energies, while lead capture gammas predominate at high energies. At intermediate energies the high-energy (≥ 4 Mev) lead inelastic gammas are the most important source. The contribution from secondary gamma sources in the lower part of the configuration is generally negligible.

LEGEND

Fig. 4.2 — Small Source Experiment — Configuration for 2-in. Lead Sample Calculation

TABLE 4.1 — REGION COMPOSITIONS USED IN LEAD CALCULATION

Region Number	Composition Number	Region Number	Composition Number	Region Number	Composition Number
1	8	28	8	55	7
2	8	29	1	56	7
3	8	30	8	57	7
4	8	31	8	58	4
5	8	32	8	59	7
6	8	33	8	60	7
7	8	34	8	61	7
8	8	35	8	62	7
9	8	36	8	63	4
10	8	37	8	64	7
11	8	38	7	65	7
12	8	39	8	66	7
13	7	40	1	67	7
14	8	41	1	68	8
15	8	42	1	69	7
16	8	43	1	70	8
17	8	44	1	71	8
18	1	45	7	72	8
19	8	46	7	73	5
20	8	47	7	74	8
21	8	48	2	75	5
22	8	49	7	76	5
23	6	50	7	77	5
24	8	51	7	78	8
25	7	52	7	79	5
26	8	53	3	80	8
27	8	54	7		

Notes for Table 4.1:

Composition Number	Element	Concentration ($\times 10^{24}$), atoms/cm ³	Composition Number	Element	Concentration ($\times 10^{24}$), atoms/cm ³
1	H	5.22×10^{-2}	5	C	1.46×10^{-2}
	Li	5.22×10^{-2}		B	6.03×10^{-2}
	O	7.73×10^{-4}	6	Be	2.73×10^{-2}
	Al	2.82×10^{-3}		Fe	5.44×10^{-2}
2	H	7.48×10^{-2}	7	Pb	3.20×10^{-2}
	C	3.78×10^{-2}	8 (void mock-up)	H	1.00×10^{-20}
3	H	7.84×10^{-2}			
	C	3.94×10^{-2}			
	B	5.12×10^{-4}	9	Pb	3.20×10^{-2}
4	H	7.24×10^{-2}			
	C	3.64×10^{-2}			
	B	1.18×10^{-2}			

Po on the source was mocked up by Fe. Na was substituted for Al in the gamma-ray problem using the correct electron density. Na atom concentration $\sim 3.33 \times 10^{21}$. For the gamma-ray problem, C was substituted for B in compositions 3, 4, and 5.

Fig. 4.3 — Small Source Experiment — Configuration for 1-in. Tungsten Sample Calculation

TABLE 4.2 — REGION COMPOSITIONS USED IN TUNGSTEN CALCULATION

Region Number	Composition Number	Region Number	Composition Number	Region Number	Composition Number
1	8	29	1	57	7
2	8	30	8	58	4
3	8	31	9	59	7
4	8	32	10	60	7
5	8	33	10	61	7
6	8	34	10	62	7
7	8	35	8	63	4
8	8	36	8	64	7
9	8	37	8	65	7
10	8	38	7	66	7
11	8	39	8	67	7
12	8	40	1	68	8
13	7	41	1	69	7
14	8	42	1	70	8
15	8	43	1	71	8
16	8	44	1	72	8
17	8	45	7	73	5
18	1	46	7	74	8
19	8	47	7	75	5
20	8	48	2	76	5
21	8	49	7	77	5
22	8	50	7	78	8
23	6	51	7	79	5
24	8	52	7	80	10
25	7	53	3	81	8
26	8	54	7	82	1
27	8	55	7		
28	8	56	7		

Notes for Table 4.2:

Composition Number	Element	Concentration ($\times 10^{24}$), atoms/cm ³	Composition Number	Element	Concentration ($\times 10^{24}$), atoms/cm ³
1	H	5.22×10^{-2}	5	C	1.46×10^{-2}
	Li	5.22×10^{-2}		B	6.03×10^{-2}
	O	7.73×10^{-4}	6	Be	2.73×10^{-2}
	Al	2.82×10^{-3}		Fe	5.44×10^{-2}
2	H	7.48×10^{-2}	7	Pb	3.20×10^{-2}
	C	3.78×10^{-2}			
3	H	7.84×10^{-2}	8 (void mock-up)	H	1.00×10^{-20}
	C	3.94×10^{-2}			
	B	5.12×10^{-4}	9	W	5.63×10^{-2}
4	H	7.24×10^{-2}		Fe	9.17×10^{-3}
	C	3.64×10^{-2}	10	H	7.30×10^{-2}
	B	1.18×10^{-2}		C	3.65×10^{-2}

Po on the source was mocked up by Fe. Na was substituted for Al in the gamma-ray problem using the correct electron density. Na atom concentration $\sim 3.33 \times 10^{21}$. For the gamma-ray problem, C was substituted for B in compositions 3, 4, and 5.

Fig. 4.4 — Small Source Experiment — Configuration for 2-in. $\text{TiH-B}_4\text{C}$ Sample Calculation

TABLE 4.3 — REGION COMPOSITIONS USED IN TITANIUM HYDRIDE CALCULATION

Region Number	Composition Number	Region Number	Composition Number	Region Number	Composition Number
1	8	28	8	55	7
2	8	29	1	56	7
3	8	30	8	57	7
4	8	31	8	58	4
5	8	32	10	59	7
6	8	33	10	60	7
7	8	34	10	61	7
8	8	35	8	62	7
9	8	36	8	63	4
10	8	37	8	64	7
11	8	38	7	65	7
12	8	39	8	66	7
13	8	40	1	67	7
14	8	41	1	68	8
15	8	42	1	69	7
16	8	43	1	70	8
17	6	44	1	71	8
18	8	45	7	72	8
19	9	46	7	73	5
20	8	47	7	74	8
21	8	48	2	75	5
22	7	49	7	76	5
23	1	50	7	77	5
24	1	51	7	78	8
25	10	52	7	79	5
26	8	53	3		
27	8	54	7		

Notes for Table 4.3:

Composition Number	Element	Concentration ($\times 10^{24}$), atoms/cm ³	Composition Number	Element	Concentration ($\times 10^{24}$), atoms/cm ³
1	H	5.22×10^{-2}	6	Be	2.73×10^{-2}
	Li	5.22×10^{-2}		Fe	5.44×10^{-2}
	O	7.73×10^{-4}	7	Pb	3.20×10^{-2}
	Al	2.82×10^{-3}			
2	H	7.48×10^{-2}	8 (void mock-up)	H	1.00×10^{-20}
	C	3.78×10^{-2}			
3	H	7.84×10^{-2}	9	Ti	3.975×10^{-2}
	C	3.94×10^{-2}		H	7.887×10^{-2}
	B	5.12×10^{-4}		B ¹⁰	3.383×10^{-4}
4	H	7.24×10^{-2}		C	3.881×10^{-3}
	C	3.64×10^{-2}		Fe	1.095×10^{-4}
	B	1.18×10^{-2}	10	H	7.30×10^{-2}
5	C	1.46×10^{-2}		C	3.65×10^{-2}
	B	6.03×10^{-2}			

Po on the source was mocked up by Fe. Na was substituted for Al in the gamma-ray problem using the correct electron density. Na atom concentration $\rightarrow 3.33 \times 10^{21}$. For the gamma-ray problem, C was substituted for B in compositions 3, 4, 5, and 9.

Fig. 4.5 — Comparison of Measured and Calculated Response with 2-in. Lead Sample

Fig. 4.6 — Breakdown of Calculated Response with 2-in. Lead Sample

TABLE 4 4 — SUMMARY OF CALCULATED RESULTS TWO INCH LEAD SAMPLE

E, Mev	Hard Pb Inelastic			Pb Capture			Soft Pb Inelastic			Other Hard			Other Soft			Primary Gamma			Total			Grand Total
	0 to 30°	30 to 45°	Total, 0 to 45°	0 to 30°	30 to 45°	Total, 0 to 45°	0 to 30°	30 to 45°	Total, 0 to 45°	0 to 30°	30 to 45°	Total, 0 to 45°	0 to 30°	30 to 45°	Total, 0 to 45°	0 to 30°	30 to 45°	Total, 0 to 45°	Total			
																			First Half	Second Half		
0 12	2 48 × 10 ⁻⁸	2 77 × 10 ⁻⁹	2 76 × 10 ⁻⁸	1 53 × 10 ⁻⁹	1 52 × 10 ⁻¹⁰	1 68 × 10 ⁻⁹	1 23 × 10 ⁻⁷	1 62 × 10 ⁻⁸	1 39 × 10 ⁻⁷	1 89 × 10 ⁻⁹	2 18 × 10 ⁻¹⁰	2 11 × 10 ⁻⁹	4 45 × 10 ⁻⁸	2 14 × 10 ⁻⁸	6 59 × 10 ⁻⁸	1 35 × 10 ⁻⁷	4 07 × 10 ⁻⁸	1 76 × 10 ⁻⁷	4 13 × 10 ⁻⁷	8 42 × 10 ⁻⁸	4 97 × 10 ⁻⁷	
0 24	5 52 × 10 ⁻⁸	6 96 × 10 ⁻⁹	6 22 × 10 ⁻⁸	3 31 × 10 ⁻⁹	3 98 × 10 ⁻¹⁰	3 71 × 10 ⁻⁹	3 33 × 10 ⁻⁷	5 73 × 10 ⁻⁸	3 90 × 10 ⁻⁷	5 36 × 10 ⁻⁹	8 64 × 10 ⁻¹⁰	6 22 × 10 ⁻⁹	1 22 × 10 ⁻⁷	9 47 × 10 ⁻⁸	2 17 × 10 ⁻⁸	3 43 × 10 ⁻⁷	1 02 × 10 ⁻⁷	4 45 × 10 ⁻⁷	1 12 × 10 ⁻⁶	6 03 × 10 ⁻⁸	1 18 × 10 ⁻⁶	
0 36	3 89 × 10 ⁻⁸	4 11 × 10 ⁻⁹	4 30 × 10 ⁻⁸	2 53 × 10 ⁻⁹	2 84 × 10 ⁻¹⁰	2 81 × 10 ⁻⁹	2 32 × 10 ⁻⁷	4 04 × 10 ⁻⁸	2 72 × 10 ⁻⁷	3 45 × 10 ⁻⁹	3 03 × 10 ⁻¹⁰	3 75 × 10 ⁻⁹	8 26 × 10 ⁻⁸	4 09 × 10 ⁻⁸	1 24 × 10 ⁻⁸	2 21 × 10 ⁻⁷	6 91 × 10 ⁻⁸	2 90 × 10 ⁻⁷	7 36 × 10 ⁻⁷	4 00 × 10 ⁻⁸	7 76 × 10 ⁻⁷	
0 48	2 57 × 10 ⁻⁸	2 73 × 10 ⁻⁹	2 84 × 10 ⁻⁸	1 60 × 10 ⁻⁹	2 21 × 10 ⁻¹⁰	1 82 × 10 ⁻⁹	1 62 × 10 ⁻⁷	2 34 × 10 ⁻⁸	1 85 × 10 ⁻⁷	2 11 × 10 ⁻⁹	3 23 × 10 ⁻¹⁰	2 43 × 10 ⁻⁹	4 77 × 10 ⁻⁸	2 90 × 10 ⁻⁸	7 67 × 10 ⁻⁸	1 54 × 10 ⁻⁷	3 72 × 10 ⁻⁸	1 91 × 10 ⁻⁷	4 85 × 10 ⁻⁷	3 10 × 10 ⁻⁸	5 16 × 10 ⁻⁷	
0 60	1 85 × 10 ⁻⁸	2 84 × 10 ⁻⁹	2 13 × 10 ⁻⁸	1 20 × 10 ⁻⁹	1 68 × 10 ⁻¹⁰	1 37 × 10 ⁻⁹	9 20 × 10 ⁻⁸	2 18 × 10 ⁻⁸	1 14 × 10 ⁻⁷	1 98 × 10 ⁻⁹	1 88 × 10 ⁻¹⁰	2 17 × 10 ⁻⁹	3 10 × 10 ⁻⁸	1 89 × 10 ⁻⁹	3 29 × 10 ⁻⁸	1 26 × 10 ⁻⁷	3 42 × 10 ⁻⁸	1 60 × 10 ⁻⁷	3 31 × 10 ⁻⁷	2 30 × 10 ⁻⁸	3 54 × 10 ⁻⁷	
0 72	1 74 × 10 ⁻⁸	1 99 × 10 ⁻⁹	1 94 × 10 ⁻⁸	9 58 × 10 ⁻¹⁰	6 34 × 10 ⁻¹¹	1 02 × 10 ⁻⁹	9 12 × 10 ⁻⁸	5 76 × 10 ⁻⁹	9 70 × 10 ⁻⁸	1 27 × 10 ⁻⁹	1 45 × 10 ⁻¹⁰	1 42 × 10 ⁻⁹	1 67 × 10 ⁻⁸	7 15 × 10 ⁻⁹	2 39 × 10 ⁻⁸	1 02 × 10 ⁻⁷	1 73 × 10 ⁻⁸	1 19 × 10 ⁻⁷	2 61 × 10 ⁻⁷	2 09 × 10 ⁻⁸	2 82 × 10 ⁻⁷	
0 84	1 32 × 10 ⁻⁸	1 52 × 10 ⁻⁹	1 47 × 10 ⁻⁸	8 84 × 10 ⁻¹⁰	8 93 × 10 ⁻¹¹	9 73 × 10 ⁻⁹	7 92 × 10 ⁻⁸	7 69 × 10 ⁻⁹	8 69 × 10 ⁻⁸	1 28 × 10 ⁻⁹	1 63 × 10 ⁻¹⁰	1 44 × 10 ⁻⁹	1 37 × 10 ⁻⁸	4 04 × 10 ⁻⁹	1 77 × 10 ⁻⁸	7 56 × 10 ⁻⁸	1 74 × 10 ⁻⁸	9 30 × 10 ⁻⁸	2 15 × 10 ⁻⁷	1 57 × 10 ⁻⁸	2 31 × 10 ⁻⁷	
0 96	1 14 × 10 ⁻⁸	1 92 × 10 ⁻⁹	1 33 × 10 ⁻⁸	6 96 × 10 ⁻¹⁰	9 72 × 10 ⁻¹¹	7 93 × 10 ⁻¹⁰	6 11 × 10 ⁻⁸	2 86 × 10 ⁻⁹	6 40 × 10 ⁻⁸	1 06 × 10 ⁻⁹	1 42 × 10 ⁻¹⁰	1 20 × 10 ⁻⁹	1 04 × 10 ⁻⁸	4 87 × 10 ⁻¹⁰	1 09 × 10 ⁻⁸	8 34 × 10 ⁻⁸	1 14 × 10 ⁻⁸	9 48 × 10 ⁻⁸	1 85 × 10 ⁻⁷	1 58 × 10 ⁻⁸	2 01 × 10 ⁻⁷	
1 08	1 16 × 10 ⁻⁸	1 41 × 10 ⁻⁹	1 30 × 10 ⁻⁸	6 22 × 10 ⁻¹⁰	6 58 × 10 ⁻¹¹	6 88 × 10 ⁻¹⁰	4 89 × 10 ⁻⁸	7 18 × 10 ⁻⁹	5 61 × 10 ⁻⁸	8 62 × 10 ⁻¹⁰	4 04 × 10 ⁻¹¹	9 02 × 10 ⁻¹⁰	1 14 × 10 ⁻⁸	2 72 × 10 ⁻⁹	1 41 × 10 ⁻⁸	6 96 × 10 ⁻⁸	1 38 × 10 ⁻⁸	8 34 × 10 ⁻⁸	1 68 × 10 ⁻⁷	1 29 × 10 ⁻⁸	1 81 × 10 ⁻⁷	
1 20	9 83 × 10 ⁻⁹	1 69 × 10 ⁻⁹	1 15 × 10 ⁻⁸	6 11 × 10 ⁻¹⁰	7 19 × 10 ⁻¹¹	6 83 × 10 ⁻¹⁰	5 44 × 10 ⁻⁸	8 11 × 10 ⁻⁹	6 25 × 10 ⁻⁸	7 73 × 10 ⁻¹⁰	2 32 × 10 ⁻¹¹	7 96 × 10 ⁻¹⁰	7 76 × 10 ⁻⁹	6 79 × 10 ⁻¹¹	7 83 × 10 ⁻⁹	7 92 × 10 ⁻⁸	1 38 × 10 ⁻⁸	9 30 × 10 ⁻⁸	1 76 × 10 ⁻⁷	1 42 × 10 ⁻⁸	1 90 × 10 ⁻⁷	
1 32	8 20 × 10 ⁻⁹	9 86 × 10 ⁻¹⁰	9 19 × 10 ⁻⁹	5 22 × 10 ⁻¹⁰	5 87 × 10 ⁻¹¹	5 81 × 10 ⁻¹⁰	5 86 × 10 ⁻⁸	2 62 × 10 ⁻¹⁰	5 89 × 10 ⁻⁸	7 19 × 10 ⁻¹⁰	1 01 × 10 ⁻¹¹	7 29 × 10 ⁻¹⁰	1 58 × 10 ⁻⁹	0	1 58 × 10 ⁻⁸	6 48 × 10 ⁻⁸	9 03 × 10 ⁻⁹	7 38 × 10 ⁻⁸	1 59 × 10 ⁻⁷	1 61 × 10 ⁻⁸	1 75 × 10 ⁻⁷	
1 44	7 55 × 10 ⁻⁹	6 20 × 10 ⁻¹⁰	8 17 × 10 ⁻⁹	5 29 × 10 ⁻¹⁰	3 70 × 10 ⁻¹¹	5 66 × 10 ⁻¹⁰	4 08 × 10 ⁻⁸	0	4 08 × 10 ⁻⁸	6 66 × 10 ⁻¹⁰	1 18 × 10 ⁻¹¹	6 78 × 10 ⁻¹⁰	4 27 × 10 ⁻⁹		4 27 × 10 ⁻⁹	6 06 × 10 ⁻⁸	1 56 × 10 ⁻⁸	7 62 × 10 ⁻⁸	1 32 × 10 ⁻⁷	1 45 × 10 ⁻⁸	1 47 × 10 ⁻⁷	
1 56	6 83 × 10 ⁻⁹	8 14 × 10 ⁻¹⁰	7 64 × 10 ⁻⁹	5 38 × 10 ⁻¹⁰	2 83 × 10 ⁻¹¹	5 66 × 10 ⁻¹⁰	3 82 × 10 ⁻⁸		3 82 × 10 ⁻⁸	5 05 × 10 ⁻¹⁰	5 57 × 10 ⁻¹²	5 11 × 10 ⁻¹⁰	4 34 × 10 ⁻⁹		4 34 × 10 ⁻⁹	6 48 × 10 ⁻⁸	7 24 × 10 ⁻⁹	7 20 × 10 ⁻⁸	1 24 × 10 ⁻⁷	1 11 × 10 ⁻⁸	1 35 × 10 ⁻⁷	
1 68	5 96 × 10 ⁻⁹	1 35 × 10 ⁻⁹	7 31 × 10 ⁻⁹	5 67 × 10 ⁻¹⁰	2 79 × 10 ⁻¹¹	5 95 × 10 ⁻¹⁰	3 67 × 10 ⁻⁸		3 67 × 10 ⁻⁸	4 82 × 10 ⁻¹⁰	3 38 × 10 ⁻¹¹	4 85 × 10 ⁻¹⁰	3 15 × 10 ⁻⁹		3 15 × 10 ⁻⁹	5 73 × 10 ⁻⁸	1 21 × 10 ⁻⁸	6 94 × 10 ⁻⁸	1 18 × 10 ⁻⁷	1 13 × 10 ⁻⁸	1 29 × 10 ⁻⁷	
1 80	5 45 × 10 ⁻⁹	3 51 × 10 ⁻¹⁰	5 80 × 10 ⁻⁹	3 94 × 10 ⁻¹⁰	4 95 × 10 ⁻¹¹	4 44 × 10 ⁻¹⁰	4 58 × 10 ⁻⁸		4 58 × 10 ⁻⁸	5 09 × 10 ⁻¹⁰	1 21 × 10 ⁻¹²	5 10 × 10 ⁻¹⁰	4 77 × 10 ⁻⁹		4 77 × 10 ⁻⁹	4 15 × 10 ⁻⁸	1 37 × 10 ⁻⁸	5 52 × 10 ⁻⁸	1 13 × 10 ⁻⁷	2 70 × 10 ⁻⁸	1 40 × 10 ⁻⁷	
1 92	4 98 × 10 ⁻⁹	3 43 × 10 ⁻¹⁰	5 32 × 10 ⁻⁹	4 10 × 10 ⁻¹⁰	2 33 × 10 ⁻¹¹	4 33 × 10 ⁻¹⁰	2 12 × 10 ⁻⁸		2 12 × 10 ⁻⁸	4 05 × 10 ⁻¹⁰	0	4 05 × 10 ⁻¹⁰	2 70 × 10 ⁻⁹		2 70 × 10 ⁻⁹	2 98 × 10 ⁻⁸	9 93 × 10 ⁻⁹	3 17 × 10 ⁻⁸	3 17 × 10 ⁻⁸	6 17 × 10 ⁻⁸	6 48 × 10 ⁻⁸	
2 04	5 67 × 10 ⁻⁹	2 00 × 10 ⁻¹⁰	5 87 × 10 ⁻⁹	3 73 × 10 ⁻¹⁰	1 45 × 10 ⁻¹¹	3 88 × 10 ⁻¹⁰	2 15 × 10 ⁻⁸		2 15 × 10 ⁻⁸	3 98 × 10 ⁻¹⁰		3 98 × 10 ⁻¹⁰	2 98 × 10 ⁻⁹		2 98 × 10 ⁻⁹	2 81 × 10 ⁻⁸	2 07 × 10 ⁻¹⁰	2 83 × 10 ⁻⁸	5 95 × 10 ⁻⁸	4 98 × 10 ⁻¹⁰	6 00 × 10 ⁻⁸	
2 16	4 85 × 10 ⁻⁹	2 18 × 10 ⁻¹⁰	5 07 × 10 ⁻⁹	3 85 × 10 ⁻¹⁰	2 83 × 10 ⁻¹¹	4 13 × 10 ⁻¹⁰	2 91 × 10 ⁻⁸		2 91 × 10 ⁻⁸	3 56 × 10 ⁻¹⁰		3 56 × 10 ⁻¹⁰	2 50 × 10 ⁻⁹		2 50 × 10 ⁻⁹	2 54 × 10 ⁻⁸	0	2 54 × 10 ⁻⁸	6 29 × 10 ⁻⁸	5 72 × 10 ⁻¹⁰	6 35 × 10 ⁻⁸	
2 28	4 20 × 10 ⁻⁹	6 20 × 10 ⁻¹⁰	4 82 × 10 ⁻⁹	3 58 × 10 ⁻¹⁰	1 03 × 10 ⁻¹¹	3 68 × 10 ⁻¹⁰	2 84 × 10 ⁻⁸		2 84 × 10 ⁻⁸	3 68 × 10 ⁻¹⁰		3 68 × 10 ⁻¹⁰	3 75 × 10 ⁻⁹		3 75 × 10 ⁻⁹	2 00 × 10 ⁻⁸		2 00 × 10 ⁻⁸	5 78 × 10 ⁻⁸	4 36 × 10 ⁻¹⁰	5 82 × 10 ⁻⁸	
2 40	4 51 × 10 ⁻⁹	8 13 × 10 ⁻¹⁰	4 59 × 10 ⁻⁹	3 54 × 10 ⁻¹⁰	3 31 × 10 ⁻¹²	3 57 × 10 ⁻¹⁰	1 99 × 10 ⁻⁸		1 99 × 10 ⁻⁸	3 48 × 10 ⁻¹⁰		3 48 × 10 ⁻¹⁰	1 78 × 10 ⁻⁹		1 78 × 10 ⁻⁹	1 51 × 10 ⁻⁸		1 51 × 10 ⁻⁸	4 21 × 10 ⁻⁸	4 13 × 10 ⁻¹⁰	4 25 × 10 ⁻⁸	
2 52	3 89 × 10 ⁻⁹	5 00 × 10 ⁻¹⁰	3 94 × 10 ⁻⁹	2 96 × 10 ⁻¹⁰	2 14 × 10 ⁻¹²	2 98 × 10 ⁻¹⁰	1 59 × 10 ⁻⁸		1 59 × 10 ⁻⁸	3 64 × 10 ⁻¹⁰		3 64 × 10 ⁻¹⁰	1 09 × 10 ⁻⁹		1 09 × 10 ⁻⁹	1 25 × 10 ⁻⁸		1 25 × 10 ⁻⁸	3 41 × 10 ⁻⁸	4 91 × 10 ⁻¹⁰	3 46 × 10 ⁻⁸	
2 64	3 86 × 10 ⁻⁹	9 06 × 10 ⁻¹²	3 87 × 10 ⁻⁹	3 22 × 10 ⁻¹⁰	1 86 × 10 ⁻¹²	3 24 × 10 ⁻¹⁰	1 98 × 10 ⁻⁸		1 98 × 10 ⁻⁸	3 48 × 10 ⁻¹⁰		3 48 × 10 ⁻¹⁰	8 10 × 10 ⁻¹⁰		8 10 × 10 ⁻¹⁰	9 06 × 10 ⁻⁹		9 06 × 10 ⁻⁹	3 43 × 10 ⁻⁸	5 72 × 10 ⁻¹⁰	3 49 × 10 ⁻⁸	
2 76	3 7																					

The pronounced structure apparent in the curves in Fig. 4.6 (and also for the other samples) shows the standard three-peaked response of the 3-in. $\times$ 3-in. detector crystal. Some of the peaks are washed out by interference from neighboring ones, and by statistics. This structure is emphasized in the calculations by the fact that the continuous gamma spectra resulting from inelastic and capture events, for several of the elements, are represented as being discrete in energy for purposes of GASP.

4.3.2 One-Inch Tungsten Alloy Sample

The calculated total crystal response for the 1-in. Mallory 2000 (W) sample is shown in Fig. 4.7. The partial contributions are plotted in Fig. 4.8 and are tabulated in Table 4.5.

As seen from Fig. 4.8, it is difficult to select one source as being the major contributor except at the high energy end (>5 Mev). Above about 6.5 Mev the calculated crystal response is due to "other" hard gammas. These are due to the inelastic and capture events in the lead, iron impurities in the sample (copper and nickel were lumped together with the iron), and aluminum in the LiH regions. The calculated detector response over the energy range 4.5 to 6.5 Mev is due to the high energy (≥ 4 Mev) gammas resulting from tungsten capture events.

The tungsten capture and inelastic gammas, both high and low energy, dominate the energy range from approximately 2.5 to 4.5 Mev. Below 2.5 Mev the crystal response comes from the low energy (<4 Mev) W inelastic gammas, the primary gammas from the source and "other" low energy gammas. The latter contribution is dominated by the 2.23-Mev hydrogen capture gamma whose peak is clearly shown.

4.3.3 Two-Inch $\text{TiH}_2\text{-B}_4\text{C}$ Sample

The calculated total crystal response for the 2-in. $\text{TiH}_2\text{-B}_4\text{C}$ sample is shown in Fig. 4.9. The partial contributions are plotted in Fig. 4.10 and tabulated, together with the grand total, in Table 4.6.

The dominant feature in the response for the $\text{TiH}_2\text{-B}_4\text{C}$ sample is the peak* at about 6 Mev, due to the titanium capture events. The titanium

*The three-peaked response is clearly discernible.

Fig. 4.7 — Comparison of Measured and Calculated Response with 1-in. Tungsten Sample

Fig. 4.8 — Breakdown of Calculated Response with 1-in. Tungsten Sample

TABLE 4.5 — SUMMARY OF CALCULATED RESULTS — TUNGSTEN SAMPLE

E, Mev	Soft W Captures	Hard W Captures	Soft W Inelastic	Hard W Inelastic	Other Soft	Other Hard	Primary Gammas	Total First Half	Total Second Half	Grand Total
0.12	1.75 (-8)	3.44 (-9)	1.32 (-7)	4.42 (-9)	4.65 (-8)	1.89 (-9)	5.79 (-8)			
0.24	1.71 (-8)	3.37 (-9)	1.29 (-7)	4.34 (-9)	4.56 (-8)	1.85 (-9)	5.65 (-8)			
0.36	1.72 (-8)	3.37 (-9)	1.29 (-7)	4.35 (-9)	4.61 (-8)	1.86 (-9)	5.65 (-8)			
0.48	1.86 (-8)	3.64 (-9)	1.43 (-7)	4.65 (-9)	4.88 (-8)	2.01 (-9)	6.11 (-8)			
0.60	1.71 (-8)	3.67 (-9)	1.24 (-7)	4.41 (-9)	4.42 (-8)	1.85 (-9)	5.43 (-8)	2.46 (-7)	6.73 (-8)	3.13 (-7)
0.72	3.15 (-8)	6.44 (-9)	2.69 (-7)	7.11 (-9)	8.21 (-8)	3.38 (-9)	1.10 (-7)	5.11 (-7)	4.82 (-8)	5.59 (-7)
0.84	2.46 (-8)	4.82 (-9)	2.12 (-7)	6.93 (-9)	6.25 (-8)	2.74 (-9)	9.94 (-8)	4.13 (-7)	3.20 (-8)	4.45 (-7)
0.96	2.79 (-8)	4.35 (-9)	1.90 (-7)	5.52 (-9)	5.61 (-8)	2.42 (-9)	5.38 (-8)	3.40 (-7)	2.48 (-8)	3.65 (-7)
1.08	2.09 (-8)	3.66 (-9)	1.75 (-7)	6.61 (-9)	5.29 (-8)	3.25 (-9)	9.26 (-8)	3.54 (-7)	1.84 (-8)	3.72 (-7)
1.20	2.14 (-8)	3.94 (-9)	8.53 (-8)	4.74 (-9)	6.02 (-8)	1.82 (-9)	4.70 (-8)	2.24 (-7)	1.67 (-8)	2.41 (-7)
1.32	1.63 (-8)	3.16 (-9)	7.84 (-8)	3.86 (-9)	6.98 (-8)	1.30 (-9)	5.65 (-8)	2.29 (-7)	1.26 (-8)	2.42 (-7)
1.44	1.46 (-8)	2.85 (-9)	7.11 (-8)	3.58 (-9)	5.29 (-8)	1.55 (-9)	4.46 (-8)	1.91 (-7)	1.26 (-8)	2.04 (-7)
1.56	1.55 (-8)	2.73 (-9)	5.93 (-8)	3.99 (-9)	5.56 (-8)	1.20 (-9)	3.85 (-8)	1.77 (-7)	1.03 (-8)	1.87 (-7)
1.68	1.30 (-8)	2.78 (-9)	6.89 (-8)	3.62 (-9)	4.65 (-8)	1.43 (-9)	3.40 (-8)	1.70 (-7)	1.14 (-8)	1.81 (-7)
1.80	9.39 (-9)	2.22 (-9)	8.76 (-8)	3.14 (-9)	6.52 (-8)	1.17 (-9)	4.02 (-8)	2.09 (-7)	1.29 (-8)	2.22 (-7)
1.92	9.85 (-9)	2.49 (-9)	2.42 (-8)	3.18 (-9)	5.02 (-8)	1.04 (-9)	2.11 (-8)	1.12 (-7)	1.16 (-8)	1.24 (-7)
2.04	7.52 (-9)	2.43 (-9)	2.10 (-8)	3.17 (-9)	4.55 (-8)	9.35 (-10)	2.37 (-8)	1.04 (-7)	8.88 (-9)	1.13 (-7)
2.16	7.80 (-9)	2.32 (-9)	1.44 (-8)	2.96 (-9)	4.16 (-8)	7.20 (-10)	2.26 (-8)	9.25 (-8)	9.04 (-9)	1.02 (-7)
2.28	5.97 (-9)	2.28 (-9)	2.19 (-8)	3.44 (-9)	9.71 (-8)	7.52 (-10)	1.61 (-8)	1.48 (-7)	2.16 (-8)	1.70 (-7)
2.40	7.52 (-9)	2.42 (-9)	7.48 (-9)	3.32 (-9)	1.16 (-8)	8.85 (-10)	1.70 (-8)	5.01 (-8)	2.50 (-9)	5.26 (-8)
2.52	5.02 (-9)	2.70 (-9)	4.53 (-9)	2.82 (-9)	1.43 (-9)	8.80 (-10)	4.10 (-8)	5.84 (-8)	3.98 (-10)	5.88 (-8)
2.64	7.71 (-9)	1.93 (-9)	6.16 (-9)	2.51 (-9)	1.32 (-9)	9.03 (-10)	1.09 (-8)	3.15 (-8)	4.58 (-10)	3.20 (-8)
2.76	4.97 (-9)	1.77 (-9)	9.71 (-9)	2.76 (-9)	2.95 (-9)	9.35 (-10)	3.86 (-9)	2.69 (-8)	3.49 (-10)	2.72 (-8)
2.88	5.06 (-9)	1.91 (-9)	5.56 (-9)	2.96 (-9)	1.89 (-9)	1.02 (-9)	3.54 (-9)	2.19 (-8)	3.30 (-10)	2.22 (-8)
3.00	3.86 (-9)	1.88 (-9)	3.12 (-9)	2.54 (-9)	1.12 (-9)	1.16 (-9)	1.87 (-9)	1.55 (-8)	3.94 (-10)	1.59 (-8)
3.12	4.34 (-9)	2.05 (-9)	2.90 (-9)	2.69 (-9)	1.03 (-9)	1.20 (-9)	1.35 (-9)	1.55 (-8)	4.96 (-10)	1.60 (-8)
3.24	4.15 (-9)	2.14 (-9)	3.41 (-9)	2.62 (-9)	1.09 (-9)	1.13 (-9)	1.40 (-9)	1.60 (-8)	3.18 (-10)	1.63 (-8)
3.36	2.12 (-9)	2.19 (-9)	5.47 (-9)	2.91 (-9)	1.80 (-9)	1.25 (-9)	2.60 (-9)	1.83 (-8)	2.57 (-10)	1.86 (-8)

Note: Values are detector counts per source neutron in each 120-keV wide pulse height channel. The number in parentheses in each entry is the exponent (power of 10). For example, 1.30 (-9) = 1.30×10^{-9} .

TABLE 4.5 — (CONTINUED)

E, Mev	Soft W Captures	Hard W Captures	Soft W Inelastic	Hard W Inelastic	Other Soft	Other Hard	Primary Gammas	Total First Half	Total Second Half	Grand Total
3.48	2.25 (-9)	1.99 (-9)	2.22 (-9)	2.91 (-9)	6.75 (-10)	1.30 (-9)	5.93 (-10)	1.19 (-8)	3.19 (-10)	1.22 (-8)
3.60	1.25 (-9)	2.36 (-9)	1.23 (-9)	3.31 (-9)	4.13 (-10)	1.48 (-9)	1.38 (-10)	1.02 (-8)	2.94 (-10)	1.05 (-8)
3.72	1.48 (-9)	2.40 (-9)	1.28 (-9)	3.48 (-9)	3.07 (-10)	1.49 (-9)	6.61 (-11)	1.05 (-8)	3.06 (-10)	1.08 (-8)
3.84	1.03 (-9)	2.30 (-9)	2.24 (-9)	3.61 (-9)	4.88 (-10)	1.47 (-9)	7.02 (-11)	1.12 (-8)	3.82 (-10)	1.16 (-8)
3.96	6.11 (-10)	2.26 (-9)	1.28 (-9)	3.10 (-9)	2.78 (-10)	1.37 (-9)	7.93 (-11)	8.98 (-9)	4.50 (-10)	9.43 (-9)
4.08	3.59 (-10)	2.34 (-9)	3.45 (-11)	3.08 (-9)	7.52 (-12)	1.45 (-9)	6.34 (-11)	7.34 (-9)	3.70 (-10)	7.71 (-9)
4.20	0	2.69 (-9)	0	2.98 (-9)	0	1.27 (-9)	4.10 (-11)	6.98 (-9)	2.49 (-10)	7.23 (-9)
4.36		2.55 (-9)		3.04 (-9)		1.14 (-9)	3.55 (-11)	6.79 (-9)	2.04 (-10)	6.99 (-9)
4.44		2.38 (-9)		2.02 (-9)		8.53 (-10)	4.55 (-11)	5.29 (-9)	2.97 (-10)	5.59 (-9)
4.56		2.28 (-9)		1.11 (-9)		5.20 (-10)	2.84 (-11)	3.93 (-9)	1.94 (-10)	4.12 (-9)
4.68		2.28 (-9)		1.03 (-9)		4.74 (-10)	9.21 (-12)	3.80 (-9)	7.33 (-11)	3.87 (-9)
4.80		2.06 (-9)		1.27 (-9)		5.43 (-10)	0	3.87 (-9)	9.04 (-12)	3.87 (-9)
4.92		1.92 (-9)		8.66 (-10)		4.38 (-10)		3.22 (-9)	1.00 (-11)	3.23 (-9)
5.04		1.94 (-9)		4.65 (-10)		3.44 (-10)		2.74 (-9)	1.00 (-11)	2.75 (-9)
5.16		2.12 (-9)		6.98 (-10)		4.74 (-10)		3.29 (-9)	1.00 (-11)	3.30 (-9)
5.28		2.03 (-9)		8.07 (-10)		5.47 (-10)		3.39 (-9)	1.12 (-11)	3.40 (-9)
5.40		1.74 (-9)		4.29 (-10)		4.16 (-10)		2.58 (-9)	1.26 (-11)	2.59 (-9)
5.52		1.64 (-9)		1.77 (-10)		3.23 (-10)		2.14 (-9)	1.41 (-11)	2.15 (-9)
5.64		1.74 (-9)		1.60 (-10)		3.55 (-10)		2.26 (-9)	1.62 (-11)	2.28 (-9)
5.76		1.81 (-9)		2.02 (-10)		4.93 (-10)		2.40 (-9)	1.90 (-11)	2.42 (-9)
5.88		1.36 (-9)		7.98 (-11)		3.20 (-10)		1.76 (-9)	2.20 (-11)	1.78 (-9)
6.00		8.12 (-10)		1.95 (-12)		2.71 (-10)		1.09 (-9)	2.55 (-11)	1.12 (-9)
6.12		1.24 (-9)		0		2.68 (-10)		1.50 (-9)	3.23 (-11)	1.53 (-9)
6.24		1.33 (-9)				2.85 (-10)		1.61 (-9)	4.22 (-11)	1.65 (-9)
6.36		4.61 (-10)				2.72 (-10)		7.34 (-10)	5.48 (-11)	7.89 (-10)
6.48		1.46 (-10)				2.19 (-10)		3.65 (-10)	5.88 (-11)	4.24 (-10)
6.60		8.84 (-11)				2.18 (-10)		3.06 (-10)	5.04 (-11)	3.56 (-10)
6.72		6.79 (-11)				2.23 (-10)		2.91 (-10)	5.81 (-11)	3.49 (-10)

Note: Values are detector counts per source neutron in each 120-kev wide pulse height channel. The number in parentheses in each entry is the exponent (power of 10). For example, $1.30 (-9) = 1.30 \times 10^{-9}$.

TABLE 4.5 — (CONTINUED)

<u>E, Mev</u>	<u>Soft W Captures</u>	<u>Hard W Captures</u>	<u>Soft W Inelastic</u>	<u>Hard W Inelastic</u>	<u>Other Soft</u>	<u>Other Hard</u>	<u>Primary Gammas</u>	<u>Total First Half</u>	<u>Total Second Half</u>	<u>Grand Total</u>
6.84	0	6.75 (-11)	0	0	0	2.00 (-10)	0	2.68 (-10)	6.88 (-11)	3.37 (-10)
6.96		6.73 (-11)				2.05 (-10)		2.69 (-10)	7.29 (-11)	3.42 (-10)
7.08		4.45 (-11)				2.07 (-10)		2.52 (-10)	4.98 (-11)	3.02 (-10)
7.20		2.35 (-11)				2.01 (-10)		2.24 (-10)	2.71 (-11)	2.51 (-10)
7.32		2.26 (-11)				1.93 (-10)		2.16 (-10)	2.78 (-11)	2.44 (-10)
7.44		2.49 (-11)				2.08 (-10)		2.34 (-10)	3.16 (-11)	2.66 (-10)
7.56		6.75 (-12)				1.96 (-10)		2.02 (-10)	8.24 (-12)	2.10 (-10)
7.68		1.77 (-13)				1.41 (-10)		1.41 (-10)	2.17 (-13)	1.41 (-10)
7.80		0				1.47 (-10)		1.47 (-10)	0	1.47 (-10)
7.92						1.66 (-10)		1.66 (-10)		1.66 (-10)
8.04						1.83 (-10)		1.83 (-10)		1.83 (-10)
8.16						1.28 (-10)		1.28 (-10)		1.28 (-10)
8.28						5.97 (-11)		5.97 (-11)		5.97 (-11)
8.40						6.38 (-11)		6.38 (-11)		6.38 (-11)
8.52						7.11 (-11)		7.11 (-11)		7.11 (-11)
8.64						9.30 (-12)		9.30 (-12)		9.30 (-12)
8.76						2.60 (-12)		2.60 (-12)		2.60 (-12)
8.88						5.43 (-13)		5.43 (-13)		5.43 (-13)
9.00						2.55 (-16)		2.55 (-16)		2.55 (-16)
9.12						1.14 (-16)		1.14 (-16)		1.14 (-16)
9.24						1.33 (-16)		1.33 (-16)		1.33 (-16)
9.36						1.58 (-16)		1.58 (-16)		1.58 (-16)
9.48						7.30 (-17)		7.29 (-17)		7.29 (-17)
9.60						1.88 (-18)		1.88 (-18)		1.88 (-18)
9.72						0		0		0

Note: Values are detector counts per source neutron in each 120-kev wide pulse height channel. The number in parentheses in each entry is the exponent (power of 10). For example, $1.30 (-9) = 1.30 \times 10^{-9}$.

Fig. 4.9 — Comparison of Measured and Calculated Response with 2-in. TiH_2 -1% B_4C Sample

Fig. 4.10 — Breakdown of Calculated Response with 2-in. $\text{TiH}_2\text{-B}_4\text{C}$ Sample

TABLE 4.6 — SUMMARY OF CALCULATED RESULTS — TITANIUM HYDRIDE SAMPLE

E, Mev	Hard Ti Captures	Soft Ti Captures	Hard Ti Inelastic	Soft Ti Inelastic	Carbon and Hydrogen	Other Hard	Other Soft	Primary Gammas	Total First Half	Total Second Half	Grand Total
0.12	9.85 (-9)	1.67 (-8)	2.22 (-8)	6.81 (-8)	5.33 (-8)	1.55 (-9)	9.82 (-9)	2.77 (-7)	4.57 (-7)	0	0
0.24	1.30 (-8)	1.62 (-8)	2.18 (-8)	6.62 (-8)	5.23 (-8)	1.51 (-9)	9.49 (-9)	2.71 (-7)	4.52 (-7)	↓	↓
0.36	1.31 (-8)	1.63 (-8)	2.19 (-8)	6.62 (-8)	5.27 (-8)	1.51 (-9)	9.47 (-9)	2.72 (-7)	4.53 (-7)	↓	↓
0.48	1.41 (-8)	1.77 (-8)	2.35 (-8)	7.27 (-8)	5.63 (-8)	1.64 (-9)	1.05 (-8)	2.94 (-7)	4.90 (-7)	↓	↓
0.60	1.28 (-8)	1.64 (-8)	2.21 (-8)	6.42 (-8)	5.32 (-8)	1.49 (-9)	8.75 (-9)	2.73 (-7)	4.52 (-7)	5.61 (-8)	5.08 (-7)
0.72	2.39 (-8)	3.07 (-8)	3.64 (-8)	1.37 (-7)	8.54 (-8)	2.60 (-9)	2.17 (-8)	4.67 (-7)	8.05 (-7)	4.02 (-8)	8.45 (-7)
0.84	1.91 (-8)	2.46 (-8)	3.33 (-8)	9.65 (-8)	7.66 (-8)	2.80 (-9)	1.38 (-8)	4.61 (-7)	7.28 (-7)	2.67 (-8)	7.55 (-7)
0.96	1.63 (-8)	2.63 (-8)	3.17 (-8)	9.17 (-8)	7.90 (-8)	1.89 (-9)	1.41 (-8)	3.57 (-7)	6.18 (-7)	2.07 (-8)	6.39 (-7)
1.08	1.77 (-8)	2.25 (-8)	2.65 (-8)	1.08 (-7)	6.29 (-8)	1.19 (-9)	8.83 (-9)	3.52 (-7)	6.00 (-7)	1.53 (-8)	6.15 (-7)
1.20	1.46 (-8)	2.15 (-8)	2.61 (-8)	5.22 (-8)	6.52 (-8)	1.73 (-9)	6.13 (-9)	3.25 (-7)	5.12 (-7)	1.39 (-8)	5.26 (-7)
1.32	1.32 (-8)	1.29 (-8)	2.33 (-8)	6.15 (-8)	6.71 (-8)	1.34 (-9)	1.14 (-8)	2.84 (-7)	4.75 (-7)	1.05 (-8)	4.86 (-7)
1.44	1.27 (-8)	2.10 (-8)	2.30 (-8)	3.83 (-8)	5.69 (-8)	9.60 (-10)	4.42 (-9)	2.74 (-7)	4.31 (-7)	1.06 (-8)	4.42 (-7)
1.56	1.16 (-8)	2.91 (-9)	2.09 (-8)	5.20 (-8)	5.27 (-8)	9.59 (-10)	4.59 (-9)	2.27 (-7)	3.73 (-7)	8.59 (-9)	3.82 (-7)
1.68	1.17 (-8)	2.52 (-9)	1.87 (-8)	4.24 (-8)	5.43 (-8)	1.28 (-9)	3.52 (-9)	2.45 (-7)	3.79 (-7)	9.45 (-9)	3.88 (-7)
1.80	1.07 (-8)	2.42 (-9)	1.60 (-8)	1.86 (-8)	6.40 (-8)	8.40 (-10)	4.29 (-9)	2.12 (-7)	3.29 (-7)	1.08 (-8)	3.40 (-7)
1.92	1.11 (-8)	3.21 (-9)	1.40 (-8)	1.62 (-8)	5.43 (-8)	7.46 (-10)	2.79 (-9)	1.71 (-7)	2.73 (-7)	9.69 (-9)	2.83 (-7)
2.04	9.48 (-9)	3.58 (-9)	1.45 (-8)	1.67 (-8)	4.40 (-8)	8.70 (-10)	3.42 (-9)	1.21 (-7)	2.14 (-7)	7.42 (-9)	2.21 (-7)
2.16	8.57 (-9)	2.69 (-9)	1.39 (-8)	1.98 (-8)	4.29 (-8)	9.30 (-10)	3.36 (-9)	1.25 (-7)	2.17 (-7)	7.55 (-9)	2.25 (-7)
2.28	9.16 (-9)	2.92 (-9)	1.40 (-8)	2.37 (-8)	1.04 (-7)	9.16 (-10)	5.19 (-9)	1.60 (-7)	3.20 (-7)	1.80 (-8)	3.38 (-7)
2.40	6.93 (-9)	2.00 (-9)	1.17 (-8)	1.85 (-8)	1.10 (-8)	8.10 (-10)	2.10 (-9)	8.08 (-8)	1.34 (-7)	2.08 (-9)	1.36 (-7)
2.52	7.29 (-9)	5.11 (-9)	1.23 (-8)	1.64 (-8)	6.03 (-11)	1.04 (-9)	1.34 (-9)	3.83 (-8)	8.18 (-8)	3.32 (-10)	8.21 (-8)
2.64	7.29 (-9)	9.25 (-10)	1.15 (-8)	1.83 (-8)	4.95 (-11)	1.35 (-9)	1.20 (-9)	5.72 (-8)	9.78 (-8)	3.81 (-10)	9.82 (-8)
2.76	6.61 (-9)	0	1.17 (-8)	1.74 (-8)	5.40 (-11)	1.02 (-9)	3.12 (-9)	3.38 (-8)	7.37 (-8)	2.90 (-10)	7.40 (-8)
2.88	6.52 (-9)	↓	1.01 (-8)	1.72 (-8)	5.97 (-10)	9.54 (-10)	1.39 (-9)	2.40 (-8)	6.08 (-8)	2.75 (-10)	6.11 (-8)
3.00	5.88 (-9)	↓	9.79 (-9)	1.12 (-8)	7.01 (-10)	1.00 (-9)	7.23 (-10)	2.38 (-8)	5.31 (-8)	3.28 (-10)	5.34 (-8)
3.12	6.29 (-9)	↓	1.11 (-8)	1.24 (-8)	8.26 (-10)	1.27 (-9)	7.27 (-10)	2.94 (-8)	6.20 (-8)	4.13 (-10)	6.24 (-8)
3.24	5.59 (-9)	↓	1.10 (-8)	8.48 (-9)	1.11 (-10)	1.41 (-9)	7.33 (-10)	1.96 (-8)	4.69 (-8)	2.65 (-10)	4.72 (-8)
3.36	5.56 (-9)	↓	1.03 (-8)	1.41 (-8)	1.69 (-10)	1.16 (-9)	1.07 (-9)	9.18 (-9)	4.15 (-8)	2.13 (-10)	4.17 (-8)
3.48	6.29 (-9)	↓	1.36 (-8)	2.82 (-9)	2.23 (-10)	1.15 (-9)	5.20 (-10)	9.81 (-9)	3.44 (-8)	2.66 (-10)	3.47 (-8)
3.60	6.43 (-9)	↓	1.52 (-8)	2.45 (-10)	2.00 (-10)	1.18 (-9)	4.31 (-10)	1.39 (-8)	3.76 (-8)	2.45 (-10)	3.78 (-8)
3.72	5.70 (-9)	↓	1.32 (-8)	0	2.14 (-10)	1.61 (-9)	2.66 (-10)	3.80 (-9)	2.48 (-8)	2.55 (-10)	2.51 (-8)
3.84	5.93 (-9)	↓	1.36 (-8)	↓	2.79 (-10)	1.29 (-9)	3.84 (-10)	1.27 (-9)	2.28 (-8)	3.18 (-10)	2.31 (-8)
3.96	6.70 (-9)	↓	1.54 (-8)	↓	3.30 (-10)	7.63 (-10)	2.18 (-10)	1.33 (-9)	2.47 (-8)	3.75 (-10)	2.51 (-8)
4.08	7.07 (-9)	↓	1.64 (-8)	↓	2.62 (-10)	7.54 (-10)	5.91 (-12)	1.03 (-9)	2.55 (-8)	3.08 (-10)	2.58 (-8)
4.20	6.02 (-9)	↓	1.26 (-8)	↓	1.73 (-10)	6.83 (-10)	0	6.49 (-10)	2.01 (-8)	2.07 (-10)	2.03 (-8)
4.32	5.20 (-9)	↓	8.46 (-9)	↓	1.55 (-10)	7.06 (-10)	↓	5.36 (-10)	1.51 (-8)	1.78 (-10)	1.53 (-8)
4.44	4.74 (-9)	↓	5.60 (-9)	↓	2.28 (-10)	5.03 (-10)	↓	6.51 (-10)	1.17 (-8)	2.54 (-10)	1.20 (-8)
4.56	6.15 (-9)	↓	1.06 (-8)	↓	1.49 (-10)	3.23 (-10)	↓	4.04 (-10)	1.76 (-8)	1.69 (-10)	1.78 (-8)
4.68	5.47 (-9)	↓	6.82 (-9)	↓	4.94 (-11)	3.24 (-10)	↓	1.31 (-10)	1.28 (-8)	6.11 (-11)	1.29 (-8)
4.80	4.74 (-9)	↓	2.43 (-9)	↓	0	3.88 (-10)	↓	0	7.56 (-9)	7.51 (-12)	7.57 (-9)
4.92	5.06 (-9)	↓	1.47 (-9)	↓	↓	3.07 (-10)	↓	↓	6.84 (-9)	8.33 (-12)	6.85 (-9)
5.04	5.38 (-9)	↓	1.93 (-9)	↓	↓	2.42 (-10)	↓	↓	7.55 (-9)	8.35 (-12)	7.56 (-9)

Note: Values are detector counts per source neutron in each 120-kev wide pulse height channel. The number in parentheses in each entry is the exponent (power of 10). For example, 130 (-8) = 1.30×10^{-8} .

TABLE 4.6 — (CONTINUED)

E, Mev	Hard Ti Captures	Soft Ti Captures	Hard Ti Inelastic	Soft Ti Inelastic	Carbon and Hydrogen	Other Hard	Other Soft	Primary Gammas	Total First Half	Total Second Half	Grand Total
5.16	5.84 (-9)	0	1.89 (-9)	0	0	3.27 (-10)	0	0	8.06 (-9)	8.33 (-12)	8.07 (-9)
5.28	6.52 (-9)		1.77 (-9)			3.81 (-10)			8.67 (-9)	9.33 (-12)	8.68 (-9)
5.40	7.34 (-9)		2.00 (-9)			2.90 (-10)			9.63 (-9)	1.05 (-11)	9.64 (-9)
5.52	8.71 (-9)		2.28 (-9)			2.07 (-10)			1.12 (-8)	1.17 (-11)	1.12 (-8)
5.64	1.11 (-8)		2.39 (-9)			2.20 (-10)			1.37 (-8)	1.35 (-11)	1.37 (-8)
5.76	1.22 (-8)		1.88 (-9)			2.60 (-10)			1.43 (-8)	1.58 (-11)	1.43 (-8)
5.88	1.11 (-8)		1.41 (-9)			2.10 (-10)			1.27 (-8)	1.83 (-11)	1.27 (-8)
6.00	1.11 (-8)		1.44 (-9)			1.52 (-10)			1.27 (-8)	2.13 (-11)	1.27 (-8)
6.12	1.33 (-8)		1.55 (-9)			1.28 (-10)			1.50 (-8)	2.69 (-11)	1.50 (-8)
6.24	1.43 (-8)		1.07 (-9)			1.46 (-10)			1.55 (-8)	3.52 (-11)	1.55 (-8)
6.36	1.06 (-8)		7.39 (-10)			1.70 (-10)			1.15 (-8)	4.57 (-11)	1.15 (-8)
6.48	5.84 (-9)		7.74 (-10)			1.44 (-10)			6.76 (-9)	4.90 (-11)	6.81 (-9)
6.60	5.61 (-9)		5.36 (-10)			1.28 (-10)			6.27 (-9)	4.20 (-11)	6.31 (-9)
6.72	6.52 (-9)		2.78 (-10)			1.25 (-10)			6.92 (-9)	4.84 (-11)	6.97 (-9)
6.84	2.81 (-9)		2.92 (-10)			1.37 (-10)			3.24 (-9)	5.73 (-11)	3.30 (-9)
6.96	4.24 (-10)		3.44 (-10)			1.58 (-10)			9.26 (-10)	6.07 (-11)	9.87 (-10)
7.08	3.01 (-10)		1.63 (-10)			1.49 (-10)			6.13 (-10)	4.15 (-11)	6.54 (-10)
7.20	1.87 (-10)		6.33 (-12)			1.25 (-10)			3.18 (-10)	2.26 (-11)	3.41 (-10)
7.32	1.02 (-10)		0			1.18 (-10)			2.20 (-10)	2.32 (-11)	2.43 (-10)
7.44	1.18 (-10)					1.44 (-10)			2.62 (-10)	2.63 (-11)	2.88 (-10)
7.56	1.07 (-10)					1.31 (-10)			2.38 (-10)	6.86 (-12)	2.45 (-10)
7.68	3.40 (-11)					8.46 (-11)			1.19 (-10)	1.81 (-13)	1.19 (-10)
7.80	9.03 (-13)					8.94 (-11)			9.03 (-11)	0	9.03 (-11)
7.92	0					1.12 (-10)			1.12 (-10)		1.12 (-10)
8.04						1.24 (-10)			1.24 (-10)		1.24 (-10)
8.16						8.47 (-11)			8.47 (-11)		8.47 (-11)
8.28						3.68 (-11)			3.68 (-11)		3.68 (-11)
8.40						4.09 (-11)			4.09 (-11)		4.09 (-11)
8.52						4.79 (-11)			4.79 (-11)		4.79 (-11)
8.64						4.96 (-12)			4.96 (-12)		4.96 (-12)
8.76						2.58 (-14)			2.58 (-14)		2.58 (-14)
8.88						2.17 (-14)			2.17 (-14)		2.17 (-14)
9.00						1.81 (-14)			1.81 (-14)		1.81 (-14)
9.12						1.72 (-14)			1.72 (-14)		1.72 (-14)
9.24						1.91 (-14)			1.91 (-14)		1.91 (-14)
9.36						1.38 (-14)			1.38 (-14)		1.38 (-14)
9.48						6.21 (-15)			6.21 (-15)		6.21 (-15)
9.60						4.76 (-15)			1.76 (-15)		4.76 (-15)
9.72						5.47 (-15)			5.47 (-15)		5.47 (-15)
9.84						1.75 (-15)			1.75 (-15)		1.75 (-15)
9.96						6.72 (-17)			6.72 (-17)		6.72 (-17)
10.08						0			0		0

Note: Values are detector counts per source neutron in each 120-kev wide pulse height channel. The number in parentheses in each entry is the exponent (power of 10). For example, 130 (-8) = 1.30×10^{-8} .

capture spectrum consists of three very strong lines at 1.38, 6.41, and 6.75 Mev superimposed on a generally continuous spectrum. The 6-Mev capture gammas dominate the crystal response from 7 Mev down to almost 4.5 Mev. Above 7 Mev, the calculated results show very few gammas, resulting in a sharp decline in the total crystal response above 6.8 Mev.

From about 3.5 to 4.5 Mev, the high energy (≥ 4 Mev) Ti inelastic gammas dominate the response. Below 3.5 Mev the major contributors are primary source gammas. These gammas are more important in the $\text{TiH}_2\text{-B}_4\text{C}$ sample, relative to the other sample configurations, because there is no intervening layer of lead between the source and sample, as in the W sample. This, plus the fact that TiH_2 is only a moderate gamma attenuator, have the effect of significantly decreasing the total number of mean-free-paths between source and detector for the source gammas.

The strong 2.23-Mev hydrogen capture gamma is very evident in the calculated response curve.

4.4 MACHINE COMPUTATION TIME

The CDC-1604-A computer hours required for the UNC A-15 (three-dimensional Monte Carlo) analyses are summarized in Table 4.7. For each sample, the neutron calculation required about 2 hr of machine time. The gamma-ray transport calculations performed with the normal Monte Carlo scoring took 1.6 hr per problem, while "flux-at-a-point" solutions required ~ 1 hr per problem.

4.5 REFERENCES FOR SECTION 4

1. Eisenman, B. and Nakache, F. R.: UNC-SAM: A FORTRAN Monte Carlo System For the Evaluation of Neutron or Gamma-Ray Transport in Three-Dimensional Geometry, UNC-5093 (Aug. 1964).
2. Schamberger, R., Celnik, J., and Ross, R.: Shield Small Source Experiments, UNC-5072 (Nov. 1963).
3. ADONIS - Advanced Shield Computational Techniques, Vol. V., UNUCOR-635.
4. Kalos, M.H.: On the Estimate of Flux at a Point Using Monte Carlo Techniques, Nuclear Sci. and Eng., Vol. 16 (May 1963).

TABLE 4.7 — CDC-1604-A COMPUTER TIME FOR UNC A-15
(THREE-DIMENSIONAL MONTE CARLO) ANALYSES

<u>Sample</u>	<u>Neutron Calculation, hr</u>	<u>Gamma-Ray Calculation</u>		<u>Total Hours</u>
		<u>No. of Problems</u>	<u>Hours</u>	
Lead	2	12*	20	22
Tungsten	2	7†	7	9
Titanium hydride	2	8†	8	10

*Direct tracking, two source angular ranges, six categories of gamma rays.

†Flux-at-a-point, one source angular range, seven or eight categories of gamma rays.

5. COMPARISON AND EVALUATION OF CALCULATION AND EXPERIMENT

The calculated and experimental results for the 2-in. lead sample, the 1-in. tungsten alloy sample, and the 2-in. $\text{TiH}_2\text{-B}_4\text{C}$ sample are compared and evaluated in Sections 5.1, 5.2, and 5.3, respectively. The total energy deposited in the crystal for each case (integral of the "curve" above 0.72 Mev) is summarized in Table 5.1 (the iron and no sample data from Phase I are included for completeness). The total crystal dose agrees within ~20% for the no sample, iron, and tungsten cases, and within 35% and 50% for the $\text{TiH}_2\text{-B}_4\text{C}$ and lead cases, respectively. In general, individual channel responses agree within a factor of two over the entire energy range.

The differences between the calculated and measured results may be attributed to a combination of several factors:

1. Uncertainties in the measurement
2. Errors in the analytical representation of the experiment
3. Statistical uncertainty in the Monte Carlo solution of the transport equation.

The uncertainties in the measurements are generally small (~15%). At high energies, where the response is very small, the uncertainty naturally is larger. As evidenced by the two no sample measurements, reproducibility of the configuration and the associated measurements is approximately 10%. Differences between the experiment and its analytical representation are also small (introducing an error of ~10% in calculated results), except for the substitution of iron for the Ni and Cu constituents of the tungsten alloy sample (discussed in Section 5.2).

TABLE 5.1 — TOTAL ENERGY DEPOSITED IN
DETECTOR CRYSTAL*

<u>Sample</u>	Experimental, <u>ev/n</u>	Calculated, <u>ev/n</u>	Calculated/ <u>Expt.</u>
No sample	3.48†	2.74‡	0.79
1-in. W	6.25	5.17	0.83
2-in. TiH ₂ -B ₄ C	17.2	11.4	0.66
2-in. Pb	6.24	3.37	0.54
2-in. Fe	9.7‡	8.3‡	0.85

*Integral of the curve above 0.72 Mev.

†Average of Phase I and Phase II measurements.

‡Data obtained during Phase I.

The statistical uncertainties in the Monte Carlo analysis are normally 10 to 15%. In combination, these three factors constitute an uncertainty of about 25%, which is essentially independent of the sample material and its associated cross sections. Analyses of results of all of the experiments in the present series, as well as other experiments on this program¹ in which the same basic calculational techniques were used to calculate neutron and gamma-ray dose rates outside a shield surrounding a critical assembly source, suggest that the uncertainties usually result in an underestimate of the experimental data. The nature of the analysis is such as to favor a deviation in this direction, as some of the approximations are of a type that tends to neglect possible additional sources of gamma radiation in order to keep the problem description within economic bounds.

Differences between the measured and analytical results in excess of the 25% associated with the analytical uncertainties are reasonably attributed to the remaining source of error – the cross sections. Therefore, the following discussions are directed toward highlighting these areas of potential error in cross sections, which could account for the differences that exceed ~25%.

5.1 THE LEAD SAMPLE CRYSTAL RESPONSE

Fig. 4.5 compares the results of the calculation and experiment for the 2-in. lead sample configuration. The figure shows the detector response in counts per 120-kev wide channel per source neutron as a function energy loss in the detector (pulse energy), as determined directly by calculation and experiment with no other normalization of data.

In general, the calculated response follows the experimental curve, but it is low by approximately a factor of two except at the high-energy end (>6.5 Mev). The partial response curves of Fig. 4.6 show that in this high energy range the calculated crystal response is due almost exclusively to the lead capture gammas. The good agreement between the calculated and measured results in this energy region suggests the validity of the neutron lead capture cross sections. However, comparison with more recently developed data has indicated that the lead capture cross sections used in this calculation were too large at high neutron energies. In addition, the capture gamma-ray spectrum appropriate

for thermal neutrons has been used for all neutron captures. The use of more accurate cross sections with the appropriate high energy capture spectrum would reduce the calculated high energy response by a factor of approximately two. This would tend to make the calculated crystal response equal to half of the measured value over the entire energy range.

At the low energy end, a small part of the discrepancy is due to the exclusion of large angles ($>45^\circ$) from the primary gamma (4.43 Mev) and low energy (<4 Mev) lead inelastic calculations. The inclusion of the 3.22-Mev primary gamma (as indicated in Appendix 7.2) would also raise the calculated response slightly in the low energy region. Neither of these effects should substantially alter the calculated results.

Another possible source of discrepancy below 3.5 Mev is the uncertainty in the production of low energy (<4 Mev) gammas from inelastic scattering in lead. This uncertainty exists in the excitation cross sections for the various levels in the main lead isotopes and in some of the level decay schemes.

At intermediate energies the high energy lead inelastic gammas dominate the calculated response. Part of the discrepancy between the calculated and experimental results in this energy range may be due to the treatment of the $(n,2n)$ reaction in the calculations, in which the $(n,2n)$ reaction was treated as an inelastic event. The composite neutron spectrum for both neutrons emerging from the $(n,2n)$ reaction is combined and renormalized to one neutron. A single neutron is then selected from the composite energy spectrum, thus throwing away the second neutron. The loss of these second neutrons results in a loss of secondary gammas they might have produced. In addition, the competition between the (n,n') and $(n,2n)$ reactions, as a function of neutron energy, is not well established.

Preliminary calculations indicate that the treatment of the $(n,2n)$ reaction in the Monte Carlo code and possible errors in the magnitude of the $(n,2n)$ competition with the purely inelastic (n,n') reaction could increase the total calculated crystal response in the intermediate region by 10 to 20%.

At the present time a great deal of uncertainty exists in the gamma production cross sections for lead, especially as the neutron energy increases beyond 4 Mev. This is due to the fact that there are three important lead isotopes, having different level and decay schemes. Also not much good experimental data are available on the gamma production values for $E_n > 4$ Mev, especially for the production of high energy (≥ 4 Mev) gammas. To fill the experimental gap, statistical model calculations were performed, supplemented by Perkin's data.²

As the calculated crystal response was too low, the maximum credible gamma production cross sections from inelastic scatter events with lead were put into GASP, the routine that converts neutron interaction events into gamma-ray sources, and the effect was determined. These values assume that practically all of the excess energy in the compound nucleus is removed by gamma emission rather than by the emerging neutron. These also assume that there is not much (n,2n) competition.

A direct UNC A-15 calculation of the contribution to the crystal response from the revised lead inelastic gamma production cross sections was not made. Instead, the GASP tapes generated from the revised and original lead data were compared, and their ratio was computed. The original contribution to the crystal response from the lead inelastic gammas was multiplied by this ratio. The revised lead inelastic contribution to the response was then added to the contribution from the other sources to arrive at the total revised calculated crystal response.

As can be seen from Fig. 5.1, the use of the altered GASP lead inelastic values removes most of the discrepancy between the calculated and measured detector response for pulse energies below about 3.5 Mev. For pulse energies above 3.5 Mev, the revisions in the GASP lead inelastic data were insufficient to reduce the discrepancy significantly. This suggests that the neutron inelastic cross sections used in the calculations may be low, particularly in the high energy range.

Some of the remaining discrepancy in the response energy range from 3.5 to 6 Mev may be due, not to the treatment of the lead sample, but to the remainder of the configuration. It is noted that the calculated crystal response for the no sample configuration (Fig. 5.2) is low by a factor of two at

Fig. 5.1 — Comparison of Measured and Calculated Response with 2-in. Lead Sample, Including Calculated Response with Revised Lead Inelastic GASP Values

Fig. 5.2 — Comparison of Measured and Calculated Response with No Sample — Phases I and II

pulse energies near 3.8 and 5 Mev. However, since the factor-of-two discrepancy occurs over a much more limited energy range for the no sample case, and since the relative magnitude of these sources is reduced with the lead sample present, the bulk of the discrepancy for the lead sample configuration appears to be due to deficiencies in the lead cross sections.

5.2 THE TUNGSTEN SAMPLE CRYSTAL RESPONSE

Fig. 4.7 compares the calculated and measured detector response for the 1-in. Mallory 2000 tungsten alloy (95% W) sample configuration.

As shown in the figure, the calculated and experimental results agree very well for pulse energies below 2.4 Mev. The crystal response in this energy range comes from the low energy (<4 Mev) tungsten inelastic gammas, primary and "other" low energy gammas (mainly hydrogen captures).

The hard (≥ 4 Mev) tungsten capture gammas are the primary contributors to the detector response over the energy range from 5 to 6 Mev, with the calculated and measured response for this energy range agreeing within 25 to 30% (i.e., within the over-all uncertainty). This indicates that the neutron capture cross sections for tungsten are reasonable, and also suggests that the calculated low energy tungsten capture gamma rays are reasonable.

From 6.2 to 7.8 Mev, the calculated response is less than half the measured response. Fig. 4.8 shows that the response in this energy range is due solely to "other" hard gammas. Most of the crystal response from "other" hard gammas comes from the tungsten alloying elements (Ni, Cu) and from aluminum in the LiH regions. Since nickel and copper cross sections were not available on the neutron element data tape, they were treated as iron. However, the capture spectra for both nickel and copper contain more high energy (>4 Mev) gammas and they both have significantly larger neutron capture cross sections. As the "iron" captures constituted more than one-half of the total captures contributing to the "other hard gammas," the effect of properly including nickel and copper as impurities in the sample would raise the calculated

crystal response above 6.2 Mev by up to 50%. This, by itself, is insufficient to remove the discrepancy between the calculated and experimental results in this energy region. The remaining discrepancy in the 6.2 to 7.8-Mev response range must be associated with the remaining "other" sources. In any case, it is not attributable to the tungsten. Below 6 Mev, the effect of using iron, rather than nickel and copper, on the total crystal response is generally less than 20%.

The tungsten capture and inelastic gammas, both high and low energy, are the major contributors to the response from 2.5 to 4.5 Mev. As stated earlier, the tungsten captures and low energy inelastic gamma rays are treated reasonably well. The discrepancy of 25 to 40% between the calculated and measured response over this energy range suggests, therefore, that the hard (≥ 4 Mev) tungsten inelastic gamma production is underestimated. As seen from the gamma production data listed in Appendix 7.3 and from Fig. 4.8, there are very few 5-Mev photons produced in tungsten inelastic scattering events, and none above 6 Mev. The tungsten inelastic gamma production data were based on some statistical model calculations modified to include some experimental data by Perkin.² These inelastic gamma production cross sections could be low by as much as a factor of two for the high energy gammas. Increasing the response contribution from the hard tungsten inelastic gammas by about 50% would reduce the discrepancy between calculation and measurement to less than the over-all uncertainty in the experiment.

5.3 THE $\text{TiH}_2\text{-B}_4\text{C}$ SAMPLE CRYSTAL RESPONSE

The calculated and measured crystal response for the 2-in. $\text{TiH}_2\text{-B}_4\text{C}$ sample configuration is shown in Fig. 4.9.

For pulse energies below 2.4 Mev, the calculated results are only about 20% lower than the measured response within the over-all uncertainty of the experiment. As seen from Fig. 4.10, this crystal response is due almost entirely to the primary gamma contribution.

Between 2.4 and 3.6 Mev, the calculated response is approximately 40 to 50% lower than the measured results. Fig. 4.10 shows that the dominant source in this energy range is the primary gammas with the

titanium inelastic gammas becoming important for $E_\gamma > 3.2$ Mev. For pulse energies between 3.6 and about 4.5 Mev, the calculated detector response comes mainly from the hard (≥ 4 Mev) titanium inelastic gammas. Since the difference in this energy range is roughly 50%, the production of hard (4 to 5 Mev) Ti inelastic gammas may be too low by a factor of 1.5. In the response energy range of 2.4 to 3.6 Mev, this implies that the production cross sections for soft (2.5 to 4 Mev) Ti inelastic gammas may also be too low by a factor of 1.5. A 50% increase in the production of 2.5 to 5-Mev photons from inelastic scattering events in Ti would require a significant reduction in the production of low energy (< 2.5 Mev) gammas. The appropriateness of making this change cannot be checked by the response curves, as these low energy (< 2.5 Mev) Ti inelastic gammas do not contribute significantly to the total crystal response.

The response from 5 to 7 Mev is governed by the hard (~ 6.5 Mev) Ti capture gammas. The calculated response in this region is low by a factor of approximately two.

It is difficult to assess how much of this discrepancy is due to incorrect Ti neutron capture cross sections and/or incorrect high energy (> 5 Mev) Ti inelastic gamma production data, or (to a lesser degree) to incorrect Ti neutron inelastic cross sections.

Above 7.2 Mev, the calculated response drops sharply, whereas the experimental results show significant high energy (> 7 Mev) pulses. Fig. 3.9 indicates that these high energy gamma rays are produced in the $\text{TiH}_2\text{-B}_4\text{C}$ sample.

The titanium itself does not contribute significantly to this response. The difference between calculation and experiment is attributed in large measure to approximations made whereby several different elements (other than Ti) were grouped as single elements for simplicity in the code calculation. Since the cross sections of these other elements were not of intrinsic interest, revision of this portion of the calculation to reflect their contributions more accurately was not made.

5.4 REFERENCES FOR SECTION 5

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2. Private Communication from Mr. Perkin to R. Schamberger.

6. CONCLUSIONS

The Shield Small Source Experiments have provided a useful and valuable check of the accuracy of the MCR analytical shield design techniques and cross sections.

The experiments with tungsten, titanium hydride, and uranium (like those performed earlier on iron, stainless steel, and lead) achieved the objective of providing a configuration in which the principal radiation measured was due to secondary gamma rays produced in the sample shielding materials.

Three-dimensional Monte Carlo calculations gave predictions of the experimental responses ranging from 50 to 80% of the measured values. These results are generally very satisfactory, since the combined experimental and analytical uncertainty is about 25%.

Differences between calculated and measured results greater than 25% can be used to indicate areas of cross-section inaccuracy. On this basis, cross-section data appear to be satisfactory except as follows:

1. Lead

The neutron capture cross sections used were too low. Replacement with more recent cross sections, now available, will effect the necessary correction.

The cross sections for secondary gamma-ray production by inelastic scattering need revision. The present cross sections may be low by as much as a factor of two.

2. Mallory 2000 Tungsten Alloy

The hard (>4 Mev) gamma-ray production by inelastic scattering in tungsten is probably underestimated.

3. Titanium Hydride-B₄C

The cross sections for the production of gamma photons above 2.5 Mev by Ti inelastic scattering may be low by a factor of 1.5.

Some combination of the following cross sections for titanium may be too low (to explain a discrepancy in the 5 to 7 Mev spectrometer response), but there is insufficient data to determine which:

- a. Inelastic scattering production of gammas > 5 Mev
- b. Neutron capture
- c. Neutron inelastic scatter.

Of all the cross-section improvements indicated as being required, only those for lead have any significant influence on previous predictions of dose rate from the MCR shield. It is estimated that when all of the cross-section changes have been properly made the net effect will be an increase of approximately 10% in the total neutron and gamma-ray dose rate external to the MCR shield.

7. APPENDIX

7.1 Po-Be SOURCE SPECTRUM

A number of measurements of the neutron spectra of polonium-beryllium sources have been reported. Most of the early work is summarized in reports by Hess,¹ Notarrigo, et al.,² and Hanson.³ Although these spectra are all in qualitative agreement as to the general shape, there is enough difference among measurements to be significant in interpreting the results of the small source experiments. Until recently, the differences were not readily reconcilable for several reasons:

1. Different experimenters used different measuring techniques, some of which were not fully evaluated.
2. Many experimenters did not report the nature and construction of the neutron source used.
3. None of the measurements could be correlated with theoretical calculations of the Po-Be spectrum.

Most recent measurements and theoretical calculations of the spectrum have served to confirm some of the earlier work, with the result that the Po-Be spectrum now can be considered established with good precision. It is the purpose of this Appendix to summarize some of the most recent data and recommend a spectrum to be used in analysis of the small source experiments.

It is generally considered that the measurement of proton recoil tracks in photoemulsions is the most reliable technique for neutron spectra in the Mev range, and it is capable of the best precision and energy resolution. Since this method is time-consuming and tedious, however,

measurements have been made with various types of proton recoil counters with pulse height discriminators. The use of different measurement techniques probably accounts for most of the discrepancy in early measurements.

As indicated by the data of Fig. 7.1, the Po-Be neutron spectrum is reproducible when a consistent measurement technique is used. In Fig. 7.1, spectra of three different Mound Laboratory sources are shown, all measured by the same type proton recoil proportional counter. The solid curve is a measurement made by Cochran and Henry at ORNL.⁴ The plotted points are values taken from plots prepared by averaging a number of measurements made on two different sources by Breen, Hertz, and Wright at Mound Laboratory.⁵ In each case, the data have been renormalized to a basis of one neutron. These measurements agree within a few per cent over the entire range.

The recent report by S. Notarrigo et al.² presents the results of a careful measurement of the spectrum of a Po-Be source made of mixed powders using the photographic emulsion track technique. Fig. 7.2 compares a histogram plot of their results, normalized to one neutron emitted, with the average of the three Mound Laboratory sources discussed above. These spectra are in good general agreement, the difference being accounted for largely, if not entirely, by the poorer energy resolution of the counter method and by a slight difference in the effective normalization of the two curves caused by different extrapolations to zero at the ends.

Notarrigo et al. also compare their measured spectrum with a theoretically calculated spectrum, and again get very satisfactory agreement. Their calculated spectrum is significantly different from the previous calculation by Hess¹ owing to the use of more recent and more consistent cross-section data in the new calculation.

Finally, as pointed out by Notarrigo et al., the spectra of Fig. 7.2 are in good agreement with photographic emulsion measurements of the Po-Be neutron spectrum by Medveczky⁶ (except for small differences thought to be attributable to the fact that he used a source constructed in layers) and with those of Whitmore and Baker.⁷ Agreement is good with a recent photographic emulsion measurement of the spectrum of

Fig. 7.1 — Counter Measurements of Neutron Spectra of Three Mound Laboratory Po-Be Sources

Fig. 7.2 — Comparison of Po-Be Source Measurements

a Pu-Be source by Stewart.⁸ The two sources are expected to give similar spectra since the energy of the alpha particles emitted by Po and by Pu are nearly the same.

In view of the convincing agreement of the most careful measurements and calculations, the Po-Be neutron spectrum now seems well established. Earlier discrepancies in reported spectra are believed to have been caused by inexactness of the counter techniques used in some of the investigations.

Fig. 7.3 shows a smoothed spectrum derived from the measurements of Notarrigo et al. This spectrum has been used in analysis of the small source experiments.

7.2 GAMMA EMISSION FROM Po-Be NEUTRON SOURCE

The report of Notarrigo et al. develops the neutron spectrum of the Po-Be source in terms of separate spectra for three different processes involved in the $\text{Be}^9(\alpha, n)\text{C}^{12}$ reaction. These processes correspond to transitions to the 7.65 and 4.43-Mev levels and to the ground state of C^{12} . Integration of the curves given in this reference indicates that 17, 57, and 26% of the neutrons emitted correspond, respectively, to the above levels.

Previous work⁹ has shown that the 4.43-Mev level returns to the ground state by emission of a 4.43-Mev gamma. The 7.65-Mev level normally decays by internal pair production, but occasionally it emits gammas of 3.22 and 4.43 Mev in cascade. The 3.22-Mev gammas have been observed, and their intensity has been reported as 3 to 7% of the intensity of the 4.43-Mev gamma. No gammas of 7.65 Mev have been reported.

These data are summarized in Table 7.1, indicating that from 0.59 to 0.61 of a 4.43-Mev gamma is emitted per neutron, together with 0.02 to 0.04 of a 3.22-Mev gamma. The calculations given in Section 4 used 0.6 of a 4.43-Mev gamma per neutron and neglected the 3.22-Mev gamma. Inclusion of the 3.22-Mev gamma would raise the calculated curves slightly in the low energy region without affecting the high energy results.

TABLE 7.1 — GAMMA RAYS FROM Po-Be NEUTRON SOURCE

Relative Intensity of 3.22-Mev Gamma	Direct, 4.43 Mev	Gammas per Neutron		Total 4.43 Mev
		Cascade, 4.43 Mev	Cascade, 3.22 Mev	
0.03	0.57	0.02	0.02	0.59
0.07	0.57	0.04	0.04	0.61

Fig. 7.3 — Po-Be Source Spectrum Used for Small Source Experiment Analysis

7.3 SECONDARY GAMMA-RAY DATA

Tables 7.2 through 7.9 summarize the absorption and inelastic gamma-ray data used in the calculations. The neutron energies given in the first column are the upper energies of neutron energy groups over which the secondary gamma spectrum is taken to be constant. The gamma energies are discrete gammas assumed to be emitted in the designated event, and the tabulated numbers are the number of gammas emitted for each gamma energy.

7.4 COMPOSITION OF MATERIALS

Table 7.10 lists the compositions of the sample materials for which results are given in this report.

7.5 REFERENCES FOR SECTION 7

1. Hess, W. H.: Neutrons from (α ,n) Sources, UCRL-3839.
2. Notarrigo, S. et al.: Nuc. Phys., 29:507-514 (1962).
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TABLE 7.2 — SECONDARY GAMMA-RAY
PRODUCTION PARAMETERS

(Hydrogen, Lithium, Boron, and Carbon)

Hydrogen

<u>E_n, ev</u>	<u>Capture Gammas, $E_\gamma = 2.23$ Mev</u>
0.03 + 0	1.0
18.0 + 6	1.0

No inelastic gammas.

Lithium

No capture or inelastic gammas.

Boron-10 and Boron-11

No capture or inelastic gammas.

Carbon

<u>E_n, ev</u>	<u>Capture Gammas, $E_\gamma = 4.43$ Mev</u>
0.03 + 0	0
4.5 + 6	1.0
18.0 + 6	1.0

Note: Inelastic scattering treated as an
absorption.

TABLE 7.3 — SECONDARY GAMMA-RAY PRODUCTION PARAMETERS
(Beryllium)

<u>E_n, ev</u>	Capture Gammas					
	<u>E_γ = 0.855 Mev</u>	<u>E_γ = 2.59 Mev</u>	<u>E_γ = 3.365 Mev</u>	<u>E_γ = 3.441 Mev</u>	<u>E_γ = 5.956 Mev</u>	<u>E_γ = 6.807 Mev</u>
0.001 + 0	0.24	0.21	0.28	0.11	0.20	0.65
1.0 + 3	0.24	0.21	0.28	0.11	0.20	0.65
2.0 + 7	0	0	0	0	0	0

No inelastic gammas.

TABLE 7.4 — SECONDARY GAMMA-RAY PRODUCTION PARAMETERS
(Oxygen)

	Capture Gammas, $E_\gamma =$ 3.5 Mev		Capture Gammas, $E_\gamma =$ 3.5 Mev
E_n , ev		E_n , ev	
0.0001 + 0	0	11.5 + 6	0.367
7.33 + 6	0	12.1 + 6	0.356
7.70 + 6	0.1	12.7 + 6	0.399
8.10 + 6	0.40	13.3 + 6	0.393
8.51 + 6	0.45	14.0 + 6	0.407
8.95 + 6	0.50	14.75 + 6	0.422
9.41 + 6	0.50	15.5 + 6	0.431
9.89 + 6	0.50	16.3 + 6	0.434
10.4 + 6	0.497	17.1 + 6	0.433
10.9 + 6	0.462	20.0 + 6	0.423

	Inelastic Gammas, E_γ (in Mev) =																
E_n , ev	0.25	0.75	1.25	1.75	2.25	2.75	3.25	3.75	4.25	4.75	5.25	5.75	6.25	6.75	7.25	7.75	9
0.0001 + 0	0																
6.0 + 6	0																
6.3 + 6	0												1.0				
7.33 + 6	0												1.0				
7.40	0	0.0006											0.987	0.013			
7.70	0	0.0049											0.900	0.077	0.0225		
7.87	0	0.0046											0.905	0.073	0.0205		
8.1	0	0.0070											0.857	0.108	0.0340		
8.35	0	0.0060											0.878	0.093	0.0280		
8.51	0	0.0073											0.852	0.114	0.0332		
8.70	0	0.0065											0.867	0.103	0.0290		
8.84	0	0.0067											0.863	0.105	0.0310		
8.95	0	0.0064											0.869	0.101	0.0284		
9.1	0	0.0074											0.849	0.116	0.0345		
9.2	0	0.0062											0.870	0.098	0.0279		
9.34	0	0.0077											0.824	0.121	0.0353		
9.41	0	0.0092	0	0.0002	0.0008	0.0006	0.0006						0.773	0.146	0.0421	0	0.042
9.89	0	0.0083	0	0.0003	0.0026	0.0023	0.0034	0.0008					0.697	0.130	0.0380	0	0.141
10.4	0	0.0109	0	0.0004	0.0037	0.0048	0.0064	0.0033	0.0008	0	0	0	0.377	0.161	0.0580	0	0.233
10.9	0	0.0113	0.0001	0.0005	0.0041	0.0061	0.0094	0.0062	0.0031	0.0007	0	0	0.270	0.158	0.0665	0	0.272
11.5	0	0.0118	0.0003	0.0007	0.0042	0.0067	0.0112	0.0098	0.0065	0.0032	0.0008	0	0.234	0.153	0.0794	0	0.286
12.1	0	0.0122	0.0005	0.0010	0.0045	0.0069	0.0117	0.0116	0.0099	0.0068	0.0033	0.0012	0.204	0.148	0.0889	0	0.282
12.7	0	0.0120	0.0006	0.0014	0.0048	0.0073	0.0120	0.0123	0.0116	0.0098	0.0068	0.0034	0.192	0.140	0.0918	0	0.272
13.3	0	0.0116	0.0008	0.0016	0.0052	0.0079	0.0124	0.0128	0.0125	0.0115	0.0096	0.0067	0.180	0.129	0.0941	0	0.260
14.0	0	0.0112	0.0009	0.0019	0.0055	0.0083	0.0130	0.0136	0.0134	0.0126	0.0114	0.0095	0.172	0.122	0.0968	0.0004	0.246
14.75	0	0.0108	0.0009	0.0020	0.0057	0.0086	0.0135	0.0145	0.0146	0.0139	0.0127	0.0112	0.164	0.116	0.0997	0.0023	0.234
15.5	0	0.0106	0.0010	0.0022	0.0058	0.0089	0.0138	0.0152	0.0156	0.0153	0.0142	0.0127	0.158	0.113	0.1035	0.0052	0.226
16.3	0	0.0104	0.0010	0.0022	0.0059	0.0091	0.0141	0.0156	0.0166	0.0166	0.0158	0.0145	0.157	0.112	0.1044	0.0074	0.223
17.1	0	0.0101	0.0010	0.0023	0.0060	0.0099	0.0143	0.0162	0.0173	0.0176	0.0172	0.0161	0.155	0.111	0.1031	0.0089	0.222
18.1 + 6	0	0.0099	0.0010	0.0023	0.0060	0.0093	0.0144	0.0164	0.0178	0.0185	0.0184	0.0177	0.154	0.111	0.1029	0.0112	0.222

TABLE 7.5 — SECONDARY GAMMA-RAY PRODUCTION PARAMETERS
(Aluminum)

	Capture Gammas, E_γ in Mev =						
E_n , ev	<u>0.2</u>	<u>1.5</u>	<u>1.78</u>	<u>2.5</u>	<u>4.0</u>	<u>6.0</u>	<u>7.73</u>
0.001 + 0	2.5	0.95	1.0	0.7	0.8	0.2	0.25
2.8 + 6	2.5	0.95	1.0	0.7	0.8	0.2	0.25
2.0 + 7	0	0	0	0	0	0	0

	Inelastic Gammas, E_γ in Mev =														
E_n , ev	<u>0.75</u>	<u>1.25</u>	<u>1.75</u>	<u>2.25</u>	<u>2.75</u>	<u>3.25</u>	<u>3.75</u>	<u>4.25</u>	<u>4.75</u>	<u>5.25</u>	<u>5.75</u>	<u>6.25</u>	<u>6.75</u>	<u>7.5</u>	<u>8.5</u>
1.15 + 6	1	0													
1.48 + 6	0.36	0.64	0												
1.90 + 6	0.31	0.69	0												
2.44 + 6	0.29	0.71	0												
3.13 + 6	0.17	0.47	0.04	0.33	0										
4.02 + 6	0.22	0.40	0.18	0.38	0.07	0.01									
5.16 + 6	0.29	0.38	0.24	0.37	0.13	0.05	0.01	0.01	0						
6.63 + 6	0.28	0.34	0.28	0.27	0.14	0.11	0.04	0.06	0.05	0.02	0.005	0			
8.51 + 6	0.27	0.33	0.30	0.20	0.13	0.12	0.07	0.09	0.08	0.06	0.04	0.03	0.01	0	
10.9 + 6	0.26	0.31	0.29	0.18	0.10	0.10	0.09	0.09	0.09	0.08	0.07	0.06	0.04	0.06	0.06
14.0 + 6	0.26	0.31	0.18	0.15	0.10	0.10	0.10	0.09	0.10	0.09	0.08	0.07	0.06	0.12	0.23
18.02 + 6	0.25	0.30	0.12	0.16	0.11	0.11	0.11	0.09	0.11	0.10	0.09	0.09	0.08	0.15	0.44

TABLE 7.6 — SECONDARY GAMMA-RAY PRODUCTION PARAMETERS
(Titanium)

E_n , ev	Capture Gammas, E_γ in Mev =				
	<u>1.4</u>	<u>2.5</u>	<u>4.5</u>	<u>6.6</u>	<u>7.5</u>
0.01 + 0	1.6	0.16	0.24	0.78	0.015
2.3 + 6	1.6	0.16	0.24	0.78	0.015
20.0 + 6	0	0	0	0	0

E_n , ev	Inelastic Gammas, E_γ in Mev =						
	<u>1.0</u>	<u>1.5</u>	<u>2.25</u>	<u>3.25</u>	<u>4.5</u>	<u>6.0</u>	<u>7.0</u>
0.01 + 0	0						
0.897 + 6	0						
1.152 + 6	1.0						
1.479 + 6	1.0						
1.899 + 6	1.0						
2.438 + 6	1.0						
3.131 + 6	1.0	0.19	0				
4.020 + 6	1.0	0.37	0				
5.162 + 6	0.83	0.42	0.063	0.17	0.11	0	0
6.628 + 6	0.50	0.25	0.23	0.61	0.46	0.005	0
8.511 + 6	0.53	0.31	0.48	0.80	0.54	0.10	0.031
10.93 + 6	0.56	0.36	0.69	1.02	0.68	0.14	0.075
14.03 + 6	0.52	0.37	0.79	1.18	0.81	0.17	0.10
18.02 + 6	0.45	0.32	0.68	1.01	0.70	0.15	0.08

TABLE 7.7 — SECONDARY GAMMA-RAY PRODUCTION PARAMETERS

(Iron)

E_n , ev	Capture Gammas, E_γ in Mev =														
	0.50	1.4	1.63	1.74	2.143	2.72	3.5	4.5	5.5	6.0	6.5	7.285	7.639	8.5	9.3
0.001 + 0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
4.02 + 6	0.8	0.112	0.201	0.287	0.0582	0.1118	0.115	0.115	0.06	0.13	0.06	0.0313	0.2994	0.0093	0.021
5.162 + 6	0.53	0.0743	0.133	0.19	0.0386	0.074	0.0762	0.0762	0.0398	0.0862	0.0398	0.0208	0.198	0.0062	0.0139
6.628 + 6	0.047	0.0066	0.0118	0.0168	0.0034	0.0066	0.0068	0.0068	0.0035	0.0076	0.0035	0.0018	0.0176	0.0005	0.0012
20. + 6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

E_n , ev	Inelastic Gammas, E_γ in Mev =															
	0.845	2.0	2.75	3.25	3.75	4.25	4.75	5.25	5.75	6.25	6.75	7.25	7.75	8.25	8.75	9.25
0.001 + 0	0															
1.152 + 6	1.0															
1.479 + 6	1.0															
1.899 + 6	1.0															
2.438 + 6	1.0															
3.131 + 6	1.17	0.13														
4.02 + 6	1.2	0.25	0.06													
5.162 + 6	1.34	0.31	0.11	0.056	0.04	0.01	0.002									
6.628 + 6	1.38	0.35	0.185	0.11	0.102	0.042	0.026	0.018	0.010							
8.511 + 6	1.40	0.37	0.26	0.17	0.15	0.103	0.078	0.063	0.04	0.025	0.0085	0.003				
10.93 + 6	1.43	0.465	0.31	0.23	0.195	0.163	0.135	0.105	0.085	0.064	0.043	0.026	0.0144	0.01	0.0052	0.003
14.03 + 6	1.32	0.52	0.315	0.222	0.175	0.160	0.1351	0.108	0.082	0.063	0.046	0.033	0.0215	0.016	0.0103	0.0063
18.02 + 6	0.88	0.46	0.145	0.11	0.085	0.068	0.055	0.038	0.028	0.020	0.014	0.008	0.007	0.004	0.0025	0.0016

TABLE 7.8 — SECONDARY GAMMA-RAY PRODUCTION PARAMETERS

(Tungsten)

E_n , ev	Capture Gammas, E_γ in Mev =														
	0.125	0.375	0.625	0.875	1.125	1.375	1.625	1.875	2.125	2.375	2.625	2.875	3.125	3.375	3.625
0.0001 + 0	0.3	0.3	0.3	0.353	0.219	0.209	0.202	0.193	0.172	0.158	0.139	0.121	0.116	0.114	0.075
2.0 + 7	0.3	0.3	0.3	0.353	0.219	0.209	0.202	0.193	0.172	0.158	0.139	0.121	0.116	0.114	0.075

E_n , ev	Capture Gammas, E_γ in Mev =														
	3.875	4.125	4.375	4.625	4.875	5.125	5.375	5.625	5.875	6.125	6.375	6.625	6.875	7.125	7.375
0.0001 + 0	0.064	0.059	0.028	0.041	0.031	0.060	0.018	0.011	0.013	0.056	0.007	0.002	0.001	0.002	0.004
2.0 + 7	0.064	0.059	0.028	0.041	0.031	0.060	0.018	0.011	0.013	0.056	0.007	0.002	0.001	0.002	0.004

E_n , ev	Inelastic Gammas, E_γ in Mev =										
	0.25	1.0	1.75	2.25	2.75	3.25	3.75	4.25	4.75	5.25	5.75
0.001 + 0	0										
0.12 + 6	0										
0.70 + 6	1										
0.90 + 6	1	0.1	0								
1.15 + 6	0.85	0.2	0								
1.48 + 6	0.65	0.4	0								
1.90 + 6	0.42	1.0	0.15	0							
2.44 + 6	0.42	1.4	0.46	0							
3.13 + 6	0.38	1.7	0.69	0.08	0						
4.02 + 6	0.38	1.9	0.86	0.23	0						
5.16 + 6	0.36	2.1	0.95	0.38	0.08	0.08	0.04	0			
6.63 + 6	0.32	2.2	1.0	0.40	0.20	0.18	0.14	0.07	0.04	0.02	0
8.51 + 6	0.24	2.1	0.84	0.36	0.26	0.14	0.12	0.08	0.05	0.04	0.02
10.9 + 6	0.20	2.1	0.60	0.30	0.20	0.08	0.07	0.04	0.03	0.03	0.02
14.0 + 6	0.24	2.2	0.45	0.22	0.10	0.06	0.04	0.02	0.02	0.02	0.01
18.0 + 6	0.45	2.6	0.41	0.18	0.09	0.05	0.03	0.02	0.01	0	0

TABLE 7.9 — SECONDARY GAMMA-RAY PRODUCTION PARAMETERS
(Lead)

E_n , ev		Capture Gammas, E_γ in Mev =		Inelastic Gammas, E_γ in Mev =														
		6.734	7.380															
0.02 + 0		0	0															
1.802 + 7		0.077	1.0															
E_n , ev		0.25	0.75	1.25	1.75	2.25	2.75	3.25	3.75	4.25	4.75	5.25	5.75	6.25	6.75	7.25	8	
0.01 + 0		0																
0.5 + 6		0																
0.75 + 6		0	1.0	0														
1.0 + 6		0	1.0	0														
1.25 + 6		0	1.0	0														
1.5 + 6		0	1.037	0														
1.75 + 6		0	1.062	0.015	0													
2.0 + 6		0	1.095	0.027	0.013	0												
2.25 + 6		0	1.125	0.043	0.037	0												
2.5 + 6		0	1.126	0.057	0.068	0												
2.75 + 6		0	1.06	0.08	0.1	0	0.08	0										
3.0 + 6		0	0.87	0.092	0.104	0	0.185	0										
3.25 + 6		0.006	0.761	0.108	0.115	0	0.231	0										
3.5 + 6		0.013	0.745	0.12	0.13	0	0.281	0										
4.0 + 6		0.018	0.692	0.135	0.162	0.006	0.337	0.012	0.018	0								
4.5 + 6		0.026	0.623	0.137	0.182	0.018	0.376	0.026	0.069	0.021	0							
5.0 + 6		0.03	0.575	0.135	0.2	0.03	0.395	0.035	0.112	0.06	0.01	0						
5.5 + 6		0.029	0.553	0.133	0.213	0.046	0.402	0.043	0.131	0.087	0.038	0						
6.0 + 6		0.027	0.523	0.127	0.208	0.062	0.393	0.05	0.134	0.101	0.069	0.007	0					
6.5 + 6		0.022	0.491	0.12	0.196	0.073	0.37	0.055	0.133	0.111	0.093	0.026	0.015	0				
7.0 + 6		0.021	0.467	0.117	0.183	0.078	0.331	0.061	0.131	0.113	0.1	0.043	0.028	0.013	0			
7.5 + 6		0.025	0.429	0.113	0.158	0.072	0.291	0.064	0.124	0.113	0.098	0.053	0.034	0.034	0.015	0.008	0	
8.0 + 6		0.034	0.391	0.112	0.14	0.063	0.246	0.065	0.119	0.11	0.089	0.057	0.04	0.038	0.031	0.025	0.002	
8.5 + 6		0.054	0.351	0.11	0.112	0.05	0.2	0.062	0.106	0.098	0.075	0.054	0.039	0.035	0.035	0.027	0.005	

TABLE 7.9 — (CONTINUED)

E_n , ev	Inelastic Gammas, E_γ in Mev =															
	0.25	0.75	1.25	1.75	2.25	2.75	3.25	3.75	4.25	4.75	5.25	5.75	6.25	6.75	7.25	8
9.0 + 6	0.103	0.319	0.109	0.087	0.039	0.153	0.058	0.089	0.078	0.058	0.045	0.037	0.033	0.033	0.025	0.005
9.5 + 6	0.205	0.292	0.111	0.065	0.026	0.111	0.045	0.069	0.059	0.045	0.037	0.028	0.026	0.026	0.022	0.005
10.0 + 6	0.285	0.277	0.112	0.053	0.018	0.081	0.032	0.048	0.04	0.032	0.026	0.02	0.02	0.02	0.016	0.004
10.5 + 6	0.404	0.259	0.117	0.04	0.012	0.052	0.02	0.034	0.028	0.02	0.018	0.014	0.014	0.014	0.012	0.004
11.0 + 6	0.463	0.254	0.125	0.044	0.014	0.032	0.014	0.022	0.018	0.012	0.012	0.008	0.01	0.01	0.008	0.002
11.5 + 6	0.522	0.248	0.134	0.056	0.02	0.02	0.01	0.014	0.01	0.008	0.006	0.004	0.006	0.006	0.004	0.001
12.0 + 6	0.540	0.252	0.142	0.072	0.032	0.02	0.008	0.01	0.006	0.004	0.004	0.002	0.004	0.004	0.003	0.0008
12.5 + 6	0.559	0.261	0.1547	0.087	0.045	0.023	0.009	0.007	0.003	0.003	0.001	0.001	0.001	0.001	0.001	0.0007
13.0 + 6	0.571	0.277	0.164	0.103	0.059	0.027	0.011	0.009	0.003	0.001	0.001	0.0003	0.001	0.001	0.001	0.0003
13.5 + 6	0.583	0.289	0.174	0.119	0.069	0.035	0.019	0.011	0.005	0.003	0.001	0.0003	0.0003	0.0007	0.0003	0.0003
14.0 + 6	0.587	0.292	0.180	0.123	0.079	0.047	0.027	0.015	0.007	0.003	0.001	0.0003	0.0003	0.0003	0.0003	0.0003
14.5 + 6	0.587	0.305	0.186	0.123	0.087	0.057	0.039	0.023	0.011	0.005	0.003	0.0007	0.0003	0.0003	0.0003	0.0003
15.0 + 6	0.587	0.305	0.188	0.132	0.095	0.063	0.047	0.031	0.017	0.009	0.005	0.001	0.0007	0.0003	0.0003	0.0003
16.0 + 6	0.579	0.301	0.188	0.134	0.099	0.071	0.053	0.035	0.025	0.015	0.009	0.004	0.003	0.001	0.0007	0.0003
17.0 + 6	0.567	0.293	0.182	0.13	0.099	0.075	0.057	0.043	0.033	0.023	0.015	0.011	0.009	0.005	0.003	0.001
18.02 + 6	0.539	0.273	0.168	0.119	0.095	0.069	0.059	0.047	0.035	0.031	0.023	0.021	0.019	0.011	0.007	0.004