SHIELD SMALL SOURCE EXPERIMENTS

Summary Report

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1. SUMMARY

A shielding experiment has been performed which provided a proof test of the shield calculation methods used in the MCR shield design and provides a partial confirmation of cross-section data. Measurements were made with a gamma ray spectrometer behind a configuration consisting of a layer of gamma shielding material followed by a neutron shield, with the configuration exposed to neutrons from a Po-Be source.

The test configuration is shown in Figs. 1 and 2. Secondary gamma rays produced in the assembly were measured with the spectrometer and compared directly with calculations. Calculations were performed with United Nuclear's ADONIS three-dimensional rectangular geometry Monte Carlo code, which is derived from the one-dimensional Monte Carlo codes used in the basic MCR shield analysis.

Experiments were performed with samples of iron, lead, and stainless steel, and calculations were completed for the no-sample configuration and for an iron sample. Comparison of experimental and calculated detector response for these two cases is shown in Figs. 3 and 4. Agreement is within a factor of two at all energies, and the calculated energy deposited in the crystal is within 25% of that observed for the no-sample case and within 15% for the iron case. This agreement demonstrates the validity of the experiment and of the basic calculation techniques used, and gives support to the secondary gamma production cross

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Fig. 1 — Small source experiment

Fig. 2 — Small source experiment, schematic, vertical cross section

Fig. 3 — Comparison of measured and calculated response with no sample

Fig. 4 — Comparison of measured and calculated response with 2-in. iron sample

sections used for iron. Modification of the inelastic gamma spectrum used for iron is suggested by the results.

Samples of tungsten alloy, depleted uranium, zirconium hydride, and titanium hydride are on hand, but have not been run.

2. BACKGROUND DISCUSSION

2.1 OBJECTIVE

The objective of the small source experiments is to provide a simple and inexpensive proof test of shield calculation methods and cross-section data used in the MCR shield design. The experimental configuration has been designed to permit direct calculation of the experimentally measured results by the same basic methods used in MCR shield design. Direct comparison of experimental and calculated results is expected either to confirm the calculation methods and the cross-section data used in shield design or to suggest the need for changes in the data.

2.2 NEED FOR EXPERIMENTS

In principle, the Monte Carlo methods used in MCR shield design are well suited for calculation of radiation penetrating thick composite shields and for calculation of interactions such as secondary gamma production occurring within the shield. The practical engineering application of these techniques to a shield design, however, involves the use of both physical and mathematical approximations and requires extensive and detailed cross-section data for all materials involved. These cross sections often must be interpolated or extrapolated from the best available data. Cross sections for secondary gamma production are among the most important and are considered the least accurately known of all cross sections used in the MCR shield design. For these reasons, an experimental confirmation of the Monte Carlo calculation methods, with emphasis on

the ability to predict secondary gamma production in composite shields, was required.

Available experimental data from previous lid tank or bulk shielding tests were not suitable for analysis for this purpose because (1) the complexity of the geometrical situation could not be represented accurately by an appropriate existing Monte Carlo program; (2) gamma ray spectra were not generally measured; (3) the ratio of primary gammas to secondary gammas in the experiments performed was often too high to permit satisfactory evaluation of the secondary gammas; (4) precise information on material location and density was not easily available; and (5) some materials of interest in the present program had not been included in previous tests.

Therefore, a simple and inexpensive experiment was set up and performed which permitted direct comparison of experimental shield penetration and secondary gamma production effects with those calculated using a Monte Carlo code and available cross-section data.

2.3 EXPERIMENTAL APPROACH

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

1. The geometry should be well treated by an available Monte Carlo code. (This requirement was met by establishing a rectangular geometry compatible with United Nuclear's ADONIS code. The experimental configuration is described in detail in the next section.)
2. 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. (This requirement was met by placing shield samples immediately adjacent to a Po-Be neutron source.)
3. The effect of secondary gamma production in materials other than the sample being investigated should be minimized, as well as detector

response to gamma rays directly from the neutron source. (This requirement was met by placing a LiH shield behind the sample to attenuate the neutron flux leaving the sample with minimum secondary gamma production, and by using a lead shadow shield between the source and the detector. Additional neutron and gamma shielding further reduces extraneous secondary gamma response.)

4. 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. (This requirement was met by using a NaI scintillation spectrometer with a 256-channel gain stabilized analyzer.)

The output of each experiment is a determination of the number of pulses recorded by the detector in each pulse height channel. Each pulse height channel corresponds to energy deposition in the crystal within a specific narrow energy interval.

2.4 APPROACH TO EXPERIMENT ANALYSIS

To obtain a direct comparison between experiment and calculation, the expected instrument response was calculated and compared with the actual measured response. This gives a more direct comparison than can be achieved 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 analytical evaluation is carried out by performing a Monte Carlo calculation of the expected pulse height distribution in the same rectangular geometry (ADONIS program) as is used for the experiment. The calculation 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
2. Inelastic scattering gamma rays with $E \geq 4$ Mev

3. Capture and inelastic scattering gamma rays with $E < 4 \text{ Mev}$
4. Source gammas

Further breakdowns of the gamma ray signal by region of origin and by initial angle are 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.

3. EXPERIMENTAL MEASUREMENTS

The experimental 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.

3.1 DESCRIPTION OF EXPERIMENT

The experimental configuration uses a rectangular geometry compatible with the ADONIS code. A vertical section through the center of the experimental configuration is shown in Fig. 2. A photograph of this system, located in United Nuclear's Pawling critical facility, is shown in Fig. 1. Fig. 5 is a photograph of the multichannel analyzer used in the measurements.

Basic to the experiment are the Po-Be neutron source at the top, the sample under investigation located adjacent to the source, a thick neutron shield below the sample, and the 3 in. $\times$ 3 in. NaI (thallium activated) crystal scintillation counter located at the bottom of the assembly. The neutron shield of LiH and borated hydrocarbon provides a simulation of an actual shield arrangement, and also serves to reduce activation of the detector crystal.

The rest of the assembly serves to reduce extraneous detector response. The lead shadow shield between the source and the detector reduces the detector response from gammas emitted directly by the source. The additional lead shield-

Fig. 5 — Multichannel analyzer used with the sodium iodide (thallium activated) crystal scintillation counter

ing is provided as a collimator to eliminate gamma rays produced in external material, such as the walls of the room, while permitting gamma rays produced in the sample to penetrate to the detector. The neutron shielding outside the lead serves to reduce the effect of neutrons scattered back from the walls of the room. The boron carbide around the detector further reduces troublesome crystal activation.

Detector gain stability is achieved by using a Cosmic Radiation Laboratories, Inc., Spectrastat. The Spectrastat is a special power supply for the photomultiplier tube. Its output voltage is adjusted automatically to maintain the gain of the crystal-photomultiplier-amplifier system constant. The Spectrastat requires a well-defined peak in the output pulse height distribution to maintain this overall stability. In the small source experiments, such a peak is introduced by placing an Na^{24} source adjacent to the detector. The known response of the detector to the Na^{24} is subtracted later from the measured spectra.

Descriptions of Source Configurations

Several arrangements of the neutron source and sample shielding material have been used. These source configurations are shown schematically in Fig. 6.

Configuration A: A $1\frac{1}{2} \times 1\frac{1}{2}$ in. square hole is left in the center of the sample. The source is positioned on a 2-in. thick Pb block in this hole and the source location is independent of sample thickness.

Configuration B: A $\frac{3}{4}$ in. diameter by $\frac{3}{4}$ in. long hole is left in the sample for the source; source position is again independent of sample thickness.

Configuration C: As sample thickness is changed, source remains at the top of the sample.

This flexibility of the source sample shield material arrangement permitted the use of sample shielding materials on hand, as well as shapes of sample shielding

Fig. 6 — Source-sample configurations

materials which were most easily and inexpensively fabricated. Also, the flexible arrangement permitted varying the ratio of gamma rays produced by neutron inelastic scatter to those from neutron capture.

3.2 EXPERIMENTAL PROGRAM

The experimental shield configurations for which measurements have been made are outlined in Table 1. Samples of TiH_2 , ZrH_2 , $\text{TiH}_2 + \text{B}_4\text{C}$, Mallory 2000 alloy (95% W), and depleted uranium were available for the experiment but were not run.

3.3 EXPERIMENTAL PROCEDURE

Each measurement includes the following steps.

1. Measurement of the residual activity and background radiation prior to the introduction of the Po-Be source.
2. Stabilization of the system gain to center the 2.76 Mev peak of Na^{24} in channel 61. The Na^{24} is introduced in an NaF capsule adjacent to the detector. This source is activated by the Po-Be source in its storage pig each night.
3. Insertion of the Po-Be source and verification of stability.
4. Accumulation of the detector response for several intervals of approximately 1 hr.
5. Measurement of the crystal response as a function of time following the removal of the Po-Be source to obtain an indication of the I^{128} activation during the measurements.

The detector response measured in step 4 for the 2-in. thick stainless steel sample is shown in Fig. 7. This curve is representative of the uncorrected experimental output. The 2.76 Mev and 1.38 Mev total absorption peaks from Na^{24} are evident. Above 3 Mev, where there is little structure in the pulse height distribution, only every third channel has been plotted.

Table 1 — Experimental Shield Configurations for Which
Measurements Were Made

Shield Config- uration	Sample Material	Sample Dimension, in.	Source Config- uration (see Fig. 6)
1	None		
2	None		*
3	2 in. Fe (mild carbon steel)	$12 \times 12 \times 2$	A
4	2 in. Fe (mild carbon steel)	$12 \times 12 \times 2$	A*
5	4 in. Fe (mild carbon steel)	$12 \times 12 \times 4$	A
6	6 in. Fe (mild carbon steel)	$12 \times 12 \times 6$	A
7	1 in. Pb	$12 \times 12 \times 1$	A
8	2 in. Pb	$12 \times 12 \times 2$	A
9	2 in. Pb	$12 \times 12 \times 2$	A*
10	4 in. Pb	$12 \times 12 \times 4$	
11	2 in. type 316 SS	$14 \times 14 \times 2$	B
12	2 in. type 316 SS	$14 \times 14 \times 2$	B*
13	4 in. type 316 SS	$14 \times 14 \times 4$	B
14	4 in. type 316 SS	$14 \times 14 \times 4$	C
15	6 in. type 316 SS	$14 \times 14 \times 6$	B
16	6 in. type 316 SS	$14 \times 14 \times 6$	C

*In these configurations a $14 \times 14 \times 2$ in. region of B_4C powder contained in 0.020 in. polyvinyl chloride was located between the LiH and the sample-source region.

Fig. 7 — Detector response with 4-in. stainless steel sample — source-sample configuration B

3.4 DATA REDUCTION PROCEDURE

The data are modified in the following manner to yield the experimental results which can be compared directly with the output of the ADONIS calculations.

1. The contribution from the Na^{24} source is subtracted. The spectrum of this contribution is measured without the Po-Be source inserted and the correction is normalized to the 2.76 and 1.38 Mev peaks in the responses obtained in Section 3.3, step 4. The Na^{24} source contribution is shown in Fig. 7.
2. The background which is not associated with the Po-Be or Na^{24} sources is subtracted. This background contribution is obtained from the data taken before the introduction of the Po-Be source (Section 3.3, step 1).
3. An approximation of the response resulting from fast neutron interaction in the detector and its immediate environs is subtracted from the data. The spectrum of this response was obtained from an auxiliary measurement with the detector close to the source and shielded from it with lead to eliminate the contribution from the source gammas. The correction is normalized to the I^{128} activation in Section 3.3, step 4. Since the experiment has been set up to minimize this contribution, a precise measurement is not required. The estimate employed here results in a correction of a few percent at energies below 7 Mev and a negligible correction at higher energies for samples such as iron and steel, with strong secondary gamma production. In shield configuration 1, with no sample present, it is about a 20% correction.
4. The data are normalized to one source neutron, using the value of the source strength as given by the vendor, Monsanto Research Laboratories, at the time of manufacture, a 138-day half-life for Po, and the known data accumulation time.
5. The data are combined to provide the response in 120-kev intervals corresponding to the energy intervals being employed in the ADONIS calculation.

3.5 EXPERIMENTAL RESULTS

The experimental results as reduced for comparison with ADONIS calculations are shown in Figs. 8 through 16. Reference should be made to Table 1 for specification of the experimental configuration.

Fig. 8 shows the results obtained for configuration 1, with no sample present.

Fig. 9 contains the results for configuration 2, which differs from the basic no-sample case in that 2 in. of B_4C is introduced between the source and the LiH. The results of configuration 1 are also included for comparison. This B_4C layer is introduced in several experiments between the sample and the LiH region to reduce captures in the sample by absorbing moderated neutrons reflected back into the sample from the LiH. The purpose of configuration 2 is to determine the secondary gamma production in the B_4C layer itself. Introduction of the B_4C produces a substantial increase in the high-energy radiation. The 4.43 Mev gamma from inelastic scattering in carbon, for example, is observed.

Fig. 10 presents the results for configurations 3, 5, and 6 in which the sample consisted of 2 in., 4 in., and 6 in. of mild carbon steel (iron). The large increase in detector response which is produced when the iron samples are introduced indicates that the bulk of the radiation observed with the iron samples is from secondary gamma rays produced in the sample. A direct comparison with configuration 1, however, does not yield a quantitative determination of the secondary production in the sample because the introduction of the sample alters the neutron distribution and the gamma ray production in the remainder of the configuration, as well as modifying the signal produced by direct source gammas.

The expected hardening of the gamma ray spectrum from the sample as the sample thickness is increased is apparent in Fig. 10. Also, as expected, the change in detector response when the sample thickness is increased from 4 in. to 6 in. is small.

Fig. 8 — Pulse height distribution with no sample — configuration 1

Fig. 9 — Pulse height distribution for 2-in. B₄C sample — configuration 2

Fig. 10 — Pulse height distribution for 2-in., 4-in., and 6-in. mild carbon steel (iron) samples — configurations 3, 5, and 6

Fig. 11 — Pulse height distribution for configurations 2, 3, and 4 - addition of 2-in. B_4C between 2-in. iron sample and LiH

Fig. 12 — Pulse height distribution for configurations 7, 8, and 10 — 1 in., 2 in., and 4 in. of lead

Fig. 13 — Pulse height distribution for configurations 2, 8, and 9 — addition of 2-in. B₄C between 2-in. lead sample and LiH

Fig. 14 — Pulse height distribution for configurations 11, 13, and 15 — 2-in., 4-in., and 6-in. type 316 stainless steel — source configuration B

Fig. 15 — Pulse height distribution for configurations 11, 14, and 16 — 2 in., 4 in., and 6 in. of stainless steel — source configuration C

Fig. 16 — Pulse height distribution for configurations 2, 11, and 12 — addition of 2-in. B₄C between 2-in. stainless steel sample and LiH

Fig. 11 shows the results from configurations 2, 3, and 4, illustrating the change in detector response upon adding the 2-in. B_4C layer between the 2-in. iron sample and the LiH. Comparison of configuration 4 with configuration 3 shows a significant reduction in the high-energy radiation with little change in the low-energy radiation. The expectation is thus confirmed that the B_4C reduces iron captures by attenuating the low energy neutrons reflected back from the LiH. Comparison of configurations 2 and 4 indicates that the presence of the iron sample significantly reduces gamma production in the B_4C .

Fig. 12 contains the results for configurations 7, 8, and 10 for which the samples were, respectively, 1 in., 2 in., and 4 in. of lead. Very little difference is observed between these curves, indicating that for more than 1 in. of lead the attenuation of the radiation by the lead nearly compensates for the increased secondary production. Comparison with configuration 1 shows significant response from the lead in the intermediate energy region, presumably from inelastic scattering.

Fig. 13 compares the results of configurations 2, 8, and 9, showing the effect of introducing the 2-in. B_4C layer between the 2-in. lead sample and the LiH. Comparison of configurations 8 and 9 confirms the expectation that very little capture radiation is produced in the lead sample. Comparison of configurations 2 and 9 shows that presence of the lead sample reduces secondary gamma production in the B_4C .

Fig. 14 shows the results for configurations 11, 13, and 15 containing, respectively, 2 in., 4 in., and 6 in. of type 316 stainless steel with the source 2 in. above the bottom of the sample. The same comments apply to this material as the mild carbon steel sample. Fig. 15 compares configurations 11, 14, and 16, again containing 2 in., 4 in., and 6 in. of stainless steel, but with the source immediately above the sample in all cases. These measurements were performed to permit additional checks of the ability to calculate the behavior of the system.

Fig. 17 gives an intercomparison of the results for configurations 3, 8, and 11, containing 2 in. of iron, lead, and stainless steel, respectively. The significantly greater secondary production in the steels as compared with lead, particularly at the highest energies, is noteworthy. The higher secondary gamma production in stainless steel than in iron is significant also. This increase in the high-energy region is attributed to neutron capture in nickel. The close agreement of the curves in the 5 to 7 Mev region indicates that most of the secondary gamma response in this region is from inelastic gammas, which is borne out by the calculations given in the next section.

Fig. 17 — Pulse height distribution — comparison of several sample shielding materials

4. DESCRIPTION AND RESULTS OF CALCULATIONS

4.1 METHOD OF CALCULATION

The calculations were performed using the ADONIS program on the United Nuclear CDC-1604-A computer.* ADONIS calculates the solution of the Boltzmann transport equation by the Monte Carlo method for either neutrons or gamma rays in three-dimensional rectangular geometry. The ADONIS code is a direct descendent of the SANE and SAGE one-dimensional spherical geometry Monte Carlo codes, differing only in geometry. The SANE and SAGE codes are the fundamental methods currently being used for MCR shield analysis.

4.2 CALCULATION OF THE NO-SAMPLE CONFIGURATION

4.2.1 Description

Calculations of the gamma ray response of the sodium iodide crystal detector were carried out for the no-sample configuration to provide a better basis for interpreting later calculations with samples present. The geometric configuration used in the calculations is presented in Fig. 18, and the material compositions for the various regions are tabulated in Table 2.

*B. Eisenman and E. Hennessy, ADONIS - An IBM-7090 Monte Carlo Shielding Code Which Solves For The Transport of Neutrons or Gamma Rays in Three-Dimensional Rectangular Geometry, UNUCOR-635 (Mar. 1963).

Table 2 — Region Compositions Used in 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 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 mockup)	H	1.00×10^{-20}
	C	3.94×10^{-2}			
	B	5.12×10^{-4}	9	Fe	8.41×10^{-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 = 3.33×10^{21} . For the gamma ray problem, C was substituted for B in compositions 3, 4, and 5.

LEGEND

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Fig. 18 — Small source experiment — configuration for calculation

The calculations were performed in several steps. A neutron ADONIS problem was run using a Po-Be neutron spectrum source.* Due to restrictions imposed by the ADONIS code, it was necessary to divide the configuration into two parts. The first part contained the upper half through the lithium-hydride region, region 44. 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 was then used as the source for the lower part of the configuration.

The output of each neutron ADONIS problem was an interaction tape listing all gamma ray producing interactions (neutron capture and inelastic scatter). This output was processed by a separate routine, GASP†, to provide gamma ray source tapes giving photon production for each of up to 10 specified gamma ray energy groups as a function of up to 30 neutron energy groups. These gamma ray source tapes were used as input for several gamma ray ADONIS problems in which the response of the crystal detector to various incident gamma photons was recorded. The latter calculation was divided into several steps to increase the efficiency of the ADONIS calculation. The source generation and subsequent ADONIS gamma problem was run separately for high-energy (≥ 4 Mev) capture, and high-energy (≥ 4 Mev) inelastic gamma rays. Low-energy gamma ray sources (< 4 Mev) from both inelastic and capture were combined in a third problem. In all cases, the discrete photon energies were selected to match best the gamma ray spectra. A fourth gamma ray ADONIS problem was run for the gamma rays resulting from captures in hydrogen. These problems were run for both the upper and lower parts of the configuration. Finally, the contribution from primary gamma rays from the Po-Be source was calculated.

*The no-sample calculation was performed using a previously chosen Po-Be source spectrum rather than the presently recommended spectrum as given in Section 7.2. The difference between the two spectra is not believed to make a significant difference in the response calculated for the no-sample case.

†For a listing of the responses used see Section 7.1.

To increase further the efficiency of the ADONIS calculation, neutrons were picked from an isotropic distribution for the 2π solid angle facing the shield, and the gamma rays were chosen for the upper half with direction confined to a relatively narrow cone directed toward the detector. The sources of both neutrons and gamma rays were considered to be isotropic and the results were normalized to allow for the above handling of the problem.

4.2.2 Results

The calculated results for the no-sample configuration are given in Fig. 19, and tabulated in Table 3. Several of the separate components and the total calculated detector response are shown. To avoid confusion, the calculated data are shown as continuous curves, rather than as histograms, which would more properly indicate the energy group nature of the calculation.

As shown in Fig. 19, primary source gammas and hydrogen capture gammas dominate the calculated response at low energies, while inelastic scattering gammas predominate at intermediate and high energies. The pronounced structure apparent in several of the curves at high energies shows the standard 3-peaked response of the 3 in. $\times$ 3 in. detector crystal at several gamma ray energies. This structure is emphasized by the somewhat oversimplified spectra used in the calculations for inelastic and capture gamma ray sources.

The primary source gammas originate from the excited state of C^{12} , resulting from the $Be^9(\alpha,n)C^{12}$ reaction. For the calculation, 0.6 of a 4.43 Mev photon per source neutron was used.

A discussion of the calculation in comparison with the experiment is given in Section 5.

Table 3 — Summary of No-Sample Calculations

Group Upper Energy, Mev	Primary Gammas				Soft Secondary Gammas, Upper Half				Soft Secondary Gammas, Lower Half				Hard Inelastics			Hard Captures			Grand Total
	0-20°	20-30°	30-45°	Sub- total	H Cap- tures, 0-30°	Other Soft Gammas, 0-30°	All, 30-45°	Sub- total	H Cap- tures, 0-45°	H Cap- tures, 45-60°	Other, 0-90°	Sub- total	Upper Half, 0-30°	Lower Half, 0-90°	Sub- total	Upper Half, 0-30°	Lower Half, 0-90°	Sub- total	
0.60	1.06-7	4.93-8	6.6-8	2.22-7	2.48-8	3.65-8	1.73-8	7.86-8	3.05-8	4.70-8	6.37-9	8.39-8	9.08-9	5.01-11	9.13-9	1.18-9	2.41-10	1.42-9	3.96-7
0.72	8.35-8	5.90-8	4.95-8	1.92-7	2.11-8	3.10-8	1.27-8	6.48-8	2.52-8	3.12-8	3.69-9	6.01-8	7.98-9	3.94-11	8.02-9	6.98-10	1.50-10	8.48-10	3.26-7
0.84	7.02-8	3.83-8	4.16-8	1.50-7	1.74-8	2.58-8	4.39-9	4.76-8	2.17-8	1.63-8	1.79-9	3.98-8	6.27-9	2.49-11	6.30-9	7.20-10	1.79-10	8.99-10	2.45-7
0.96	6.37-8	5.20-8	4.40-8	1.60-7	1.72-8	2.37-8	4.87-9	4.58-8	2.06-8	0.82-8	2.05-9	3.08-8	5.76-9	3.04-11	5.79-9	6.48-10	1.71-10	8.19-10	2.44-7
1.08	5.08-8	2.66-8	2.71-8	1.04-7	1.68-8	1.86-8	6.19-9	4.16-8	1.52-8	0.58-8	1.74-9	2.27-8	5.15-9	2.55-11	5.18-9	6.25-10	2.71-10	8.96-10	1.75-7
1.20	5.04-8	4.52-8	3.35-8	1.29-7	1.62-8	1.53-8	1.80-9	3.33-8	1.69-8	0.19-8	1.95-9	2.08-8	4.72-9	2.34-11	4.74-9	5.69-10	8.25-11	6.51-10	1.88-7
1.32	4.64-8	2.83-8	2.42-8	0.990-7	1.54-8	1.12-8	1.03-9	2.76-8	1.47-8	0.02-8	6.60-10	1.56-8	4.14-9	1.87-11	4.16-9	5.18-10	7.00-11	5.88-10	1.48-7
1.44	4.25-8	2.44-8	1.86-8	0.854-7	1.35-8	8.20-9	6.31-10	2.23-8	1.47-8	0.02-8	8.14-10	1.57-8	4.34-9	1.67-11	4.36-9	4.57-10	1.17-10	5.74-10	1.28-7
1.56	5.15-8	2.65-8	1.03-8	0.883-7	1.27-8	1.07-8	4.51-9	2.79-8	1.20-8	0.02-8	6.32-10	1.28-8	3.83-9	1.83-11	3.85-9	4.32-10	7.35-11	5.06-10	1.33-7
1.68	3.53-8	1.82-8	1.42-8	0.677-7	1.38-8	5.97-9	9.86-11	1.99-8	1.28-8	0.02-8	1.12-9	1.41-8	3.88-9	1.60-11	3.90-9	4.44-10	6.55-11	5.10-10	1.06-7
1.80	2.90-8	2.80-8	0.82-8	0.652-7	1.41-8	5.28-9	0	1.94-8	1.52-8	0.03-8	5.60-10	1.61-8	2.96-9	1.35-11	2.97-9	3.81-10	1.46-11	3.96-10	1.04-7
1.92	2.14-8	1.62-8	0.14-8	0.389-7	1.22-8	3.48-9		1.57-8	1.37-8	0.02-8	6.42-10	1.45-8	3.12-9	1.61-11	3.14-9	3.32-10	1.52-11	3.47-10	7.26-8
2.04	2.21-8	7.37-9	0.106-8	0.316-7	9.38-9	3.41-9		1.28-8	1.05-8	0.02-8	3.70-10	1.11-8	3.13-9	1.17-11	3.14-9	3.82-10	1.80-11	4.00-10	5.90-8
2.16	1.36-8	1.67-9	0.205-8	0.173-7	9.32-9	3.48-9		1.28-8	1.07-8	0.02-8	3.78-10	1.13-8	2.69-9	1.13-11	2.70-9	3.67-10	1.32-11	3.80-10	4.45-8
2.28	1.51-8	1.09-9	0.174-8	0.179-7	2.05-8	2.89-9		2.34-8	2.61-8	0.06-8	3.33-10	2.70-8	2.39-9	1.17-11	2.40-9	3.33-10	1.22-11	3.45-10	6.83-8
2.40	1.22-8	1.05-10	3.50-10	0.126-7	2.13-9	2.61-9		4.74-9	2.74-9	0	3.47-10	3.09-9	2.30-9	1.37-11	2.31-9	2.91-10	2.10-11	3.12-10	2.31-8
2.52	9.54-9	0	1.80-11	0.096-7	0	2.86-9		2.86-9	0		4.64-10	4.64-10	1.98-9	1.48-11	1.99-9	4.24-10	1.89-11	4.43-10	1.54-8
2.64	1.11-8		0	0.111-7		3.20-9		3.20-9			5.51-10	5.51-10	2.19-9	1.42-11	2.20-9	2.80-10	7.10-12	2.87-10	1.74-8
2.76	7.99-9			0.080-7		2.52-9		2.52-9			4.17-10	4.17-10	2.04-9	1.42-11	2.05-9	2.53-10	4.81-12	2.58-10	1.32-8
2.88	6.98-9			0.070-7		1.93-9		1.93-9			3.87-10	3.87-10	2.11-9	2.09-11	2.13-9	1.69-10	4.95-12	1.74-10	1.16-8
3.00	5.90-9			0.059-7		2.07-9		2.07-9			4.66-10	4.66-10	2.16-9	2.04-11	2.18-9	1.55-10	5.10-12	1.60-10	1.07-8
3.12	3.42-9			0.034-7		3.41-9		3.41-9			5.92-10	5.92-10	2.50-9	2.28-11	2.52-9	1.61-10	5.25-12	1.66-10	1.01-8
3.24	4.07-9			0.0407-7		8.54-10		8.54-10			3.74-10	3.74-10	2.13-9	1.79-11	2.15-9	1.57-10	5.40-12	1.62-10	7.60-9
3.36	2.18-9			0.0218-7		0		0			2.96-10	2.96-10	2.16-9	1.85-11	2.18-9	1.48-10	5.60-12	1.54-10	4.91-9
3.48	3.29-9			0.0329-7							3.69-10	3.69-10	2.23-9	2.42-11	2.25-9	1.40-10	5.80-12	1.46-10	6.06-9
3.60	8.28-10			0.828-9							3.33-10	3.33-10	2.53-9	2.86-11	2.56-9	1.79-10	6.00-12	1.85-10	3.91-9
3.72	5.29-11			0.0529-9							3.55-10	3.55-10	2.16-9	2.11-11	2.18-9	1.73-10	6.15-12	1.79-10	2.77-9
3.84	0			0							4.55-10	4.55-10	1.94-9	1.53-11	1.96-9	1.36-10	6.45-12	1.42-10	2.56-9
3.96											5.37-10	5.37-10	2.24-9	1.83-11	2.26-9	1.16-10	6.65-12	1.23-10	2.92-9
4.08											4.30-10	4.30-10	2.78-9	2.47-11	2.80-9	1.31-10	7.10-12	1.38-10	3.37-9
4.20											2.87-10	2.87-10	2.27-9	1.63-11	2.29-9	1.46-10	7.35-12	1.53-10	2.73-9
4.32											2.53-10	2.53-10	1.68-9	6.85-12	1.69-9	9.83-11	7.75-12	1.06-10	2.05-9
4.44											3.70-10	3.70-10	1.61-9	3.17-12	1.61-9	5.75-11	8.15-12	6.56-11	2.05-9
4.56											2.41-10	2.41-10	1.78-9	3.58-12	1.78-9	4.85-11	8.55-12	5.70-11	2.08-9
4.68											8.01-11	8.01-11	1.35-9	2.47-12	1.35-9	4.62-11	9.02-12	5.52-11	1.49-9
4.80											0	0	1.03-9	1.67-12	1.03-9	4.48-11	9.60-12	5.44-11	1.08-9
4.92													1.24-9	2.19-12	1.24-9	4.77-11	1.03-11	5.80-11	1.30-9
5.04													1.02-9	1.33-12	1.02-9	4.52-11	1.12-11	5.64-11	1.08-9
5.16													5.91-10	1.90-13	5.91-10	4.33-11	1.23-11	5.56-11	6.47-10
5.28													5.07-10	0	5.07-10	4.19-11	1.40-11	5.59-11	5.63-10

Note: Values are detector counts per source neutron in each 120-kev wide pulse height channel. The second number in each entry is the exponent (power of 10). For example, 1.06-7 = 1.06×10^{-7} .

Table 3 — (Continued)

Group Upper Energy, Mev	Primary Gammas				Soft Secondary Gammas, Upper Half				Soft Secondary Gammas, Lower Half				Hard Inelastics			Hard Captures			Grand total
					H-Cap- tures, 0-30°	Other Soft Gammas, 0-30°	All, 30-45°	Sub- total	H Cap- tures, 0-45°	H Cap- tures, 45-60°	Other, 0-90°	Sub- total	Upper Half, 0-30°	Lower Half, 0-90°	Sub- total	Upper Half, 0-30°	Lower Half, 0-90°	Sub- total	
	0-20°	20-30°	30-45°	Sub- total															
5.40	0	0	0	0	0	0	0	0	0	0	0	0	5.61-10	0	5.61-10	4.11-11	1.58-11	5.69-11	6.18-10
5.52													6.09-10		6.09-10	4.43-11	1.76-11	6.19-11	6.71-10
5.64													6.64-10		6.64-10	5.05-11	2.02-11	7.07-11	7.35-10
5.76													6.05-10		6.05-10	5.19-11	2.37-11	7.56-11	6.81-10
5.88													5.62-10		5.62-10	5.69-11	2.75-11	8.44-11	6.46-10
6.00													5.94-10		5.94-10	6.59-11	3.19-11	9.78-11	6.92-10
6.12													5.90-10		5.90-10	7.98-11	4.04-11	1.20-10	7.10-10
6.24													5.36-10		5.36-10	9.88-11	5.28-11	1.52-10	6.88-10
6.36													5.18-10		5.18-10	1.25-10	6.85-11	1.94-10	7.12-10
6.48													5.57-10		5.57-10	1.43-10	7.35-11	2.16-10	7.73-10
6.60													4.23-10		4.23-10	1.45-10	6.30-11	2.08-10	6.31-10
6.72													2.83-10		2.83-10	1.60-10	7.26-11	2.33-10	5.16-10
6.84													3.20-10		3.20-10	1.77-10	8.60-11	2.63-10	5.83-10
6.96													3.86-10		3.86-10	1.78-10	9.11-11	2.69-10	6.55-10
7.08													2.74-10		2.74-10	1.55-10	6.22-11	2.17-10	4.91-10
7.20													1.47-10		1.47-10	1.27-10	3.39-11	1.61-10	3.08-10
7.32													1.60-10		1.60-10	1.23-10	3.48-11	1.58-10	3.18-10
7.44													1.96-10		1.96-10	1.05-10	3.95-11	1.45-10	3.41-10
7.56													2.06-10		2.06-10	4.58-11	1.03-11	5.61-11	2.62-10
7.68													1.35-10		1.35-10	3.12-11	2.71-13	3.15-11	1.67-10
7.80													6.58-11		6.58-11	3.19-11	0	3.19-11	9.77-11
7.92													7.07-11		7.07-11	4.28-12		4.28-12	7.50-11
8.04													7.98-11		7.98-11	0		0	7.98-11
8.16													9.08-12		9.08-12				9.08-12
8.28													0		0				0

Note: Values are detector counts per source neutron in each 120-kev wide pulse height channel. The second number in each entry is the exponent (power of 10). For example, 1.06-7 = 1.06×10^{-7} .

Fig. 19 — Breakdown of calculated response with no sample

4.3 CALCULATION OF THE 2-IN. THICK IRON SAMPLE

4.3.1 Description

The gamma ray response of the detector was also calculated for the case of the 2-in. thick iron sample. The geometric configuration and material compositions for the various regions were the same as for the no-sample configuration, except that iron was substituted for void in regions 31 through 34.

The calculations were performed in several steps. A neutron ADONIS problem was run using the "recommended" Po-Be neutron spectrum source described in Section 7.2. The resultant gamma-producing interaction tape was processed in several steps. The source generation routine, GASP, and corresponding ADONIS gamma problem were run separately for the following components:

- Soft inelastic and capture gammas (<4 Mev)

- Hard capture gammas (≥ 4 Mev) from iron

- Hard inelastic gammas (≥ 4 Mev) from iron

- Hard capture gammas (≥ 4 Mev) from lead and other elements (Al, O)

- Hard inelastic gammas (≥ 4 Mev) from lead and other elements (Al, O).

In addition, the primary gamma contribution was calculated. For this calculation, 0.6 of a 4.43 Mev photon per source neutron was used.

As before, the neutrons were chosen from an isotropic distribution for the 2π solid angle facing the shield, and the gammas were picked with direction confined to a relatively narrow cone. The results were normalized to allow for the above treatment.

The contribution from gamma rays produced in the shielding material below the lithium hydride was determined as follows. It was first shown that the spectral shape of the neutron flux being transmitted below the lithium-hydride region (No. 44) was approximately the same for both the iron and no-sample configura-

tions. The flux in the lowest lithium-hydride region (No. 44) was found to be about 1.3 times greater for the no-sample case than for the iron case. Thus, an approximate value of the contribution from the lower-half gammas for the iron case was obtained, dividing the total contribution from gammas born in the lower half of the no-sample case by 1.3. On this basis, the contribution from the lower-half gammas was less than 10% of the total detector response over most of the energy range.

4.3.2 Results

The calculated crystal response for the 2-in. thick iron sample is plotted in Fig. 20 and are tabulated in Table 4. The results for the separate components, subtotals and grand totals are given.

Fig. 20 — Breakdown of calculated response with 2-in. iron sample

Table 4 — Summary of 2-in. Iron Sample Calculations

Group Upper Energy, Mev	Primary Gammas			Soft Secondary Gammas		Iron Hard Captures			Iron Hard Inelasticities			Lead Hard Capture			Lead Hard Inelasticities			Grand Total
	0-20°	20-45°	Sub- total	Upper Half	Lower Half*	0-20°	20-45°	Sub- total	0-20°	20-45°	Sub- total	0-20°	20-45°	Sub- total	0-20°	20-45°	Sub- total	
0.60	1.14-7	1.30-7	2.44-7	3.38-7	6.59-8	1.09-8	3.39-9	1.43-8	5.35-8	1.92-8	7.27-8	2.94-10	0.58-10	3.52-10	3.25-10	2.18-10	5.43-10	7.45-7
0.72	1.03-7	1.08-7	2.11-7	2.62-7	4.70-8	7.97-9	3.22-9	1.119-8	4.22-8	1.88-8	6.10-8	2.70-10	0.22-10	2.92-10	2.88-10	1.05-10	3.93-10	6.14-7
0.84	8.81-8	6.61-8	1.54-7	2.06-7	3.12-8	6.42-9	2.24-9	8.66-9	3.56-8	1.66-8	5.22-8	2.49-10	0.19-10	2.68-10	2.39-10	1.30-10	3.67-10	4.58-7
0.96	8.06-8	8.60-8	1.67-7	3.20-7	2.42-8	7.43-9	1.99-9	9.42-9	3.15-8	1.89-8	5.04-8	2.05-10	0.21-10	2.26-10	2.12-10	5.27-11	2.65-10	5.90-7
1.08	5.94-8	8.60-8	1.45-7	9.60-8	1.80-8	6.03-9	2.06-9	8.09-9	2.84-8	1.81-8	4.65-8	1.68-10	0.18-10	1.86-10	1.93-10	8.99-11	2.83-10	3.26-7
1.20	6.20-8	9.65-8	1.58-7	9.54-8	1.63-8	5.85-9	2.11-9	7.96-9	2.90-8	1.19-8	4.09-8	1.77-10	0.19-10	1.96-10	1.92-10	9.83-11	2.90-10	3.33-7
1.32	5.61-8	4.95-8	1.06-7	5.60-8	1.23-8	5.07-9	2.12-9	7.19-9	2.36-8	0.74-8	3.10-8	1.47-10	0.19-10	1.66-10	1.83-10	3.56-11	2.19-10	2.73-7
1.44	4.06-8	6.25-8	1.03-7	4.83-8	1.23-8	4.58-9	0.85-9	5.43-9	2.32-8	3.20-8	5.52-8	1.54-10	0.18-10	1.72-10	1.56-10	6.29-11	2.19-10	2.43-7
1.56	3.00-8	4.65-8	7.65-8	5.40-8	1.01-8	4.41-9	1.37-9	5.78-9	2.33-8	0.63-8	2.96-8	1.37-10	0.10-10	1.47-10	1.51-10	2.35-11	1.75-10	1.83-7
1.68	2.84-8	4.10-8	6.94-8	4.95-8	1.11-8	4.27-9	0.92-9	5.19-9	2.21-8	0.15-8	2.36-8	1.50-10	0.21-10	1.71-10	1.47-10	3.01-11	1.77-10	1.67-7
1.80	2.76-8	6.90-8	9.66-8	4.88-8	1.26-8	4.27-9	1.06-9	5.33-9	2.10-8	0.12-8	2.22-8	1.19-10	0.12-10	1.31-10	1.26-10	2.21-11	1.48-10	2.10-7
1.92	2.33-8	2.50-8	4.83-8	3.89-8	1.13-8	4.13-9	0.81-9	4.94-9	1.97-8	0.18-8	2.15-8	1.03-10	0.08-10	1.11-10	1.08-10	3.77-11	1.46-10	1.24-7
2.04	3.58-8	2.88-8	6.46-8	4.37-8	8.70-9	3.85-9	0.70-9	4.55-9	2.02-8	0.20-8	2.22-8	1.16-10	0.01-10	1.17-10	8.01-11	2.20-11	1.02-10	1.42-7
2.16	2.30-8	1.68-8	3.98-8	2.90-8	8.80-9	3.50-9	0.82-9	4.32-9	1.87-8	0.16-8	2.03-8	8.32-11	0.02-11	8.34-11	8.01-11	3.52-11	1.15-10	1.02-7
2.28	2.16-8	3.40-9	2.50-8	3.49-8	2.11-8	3.82-9	0.85-9	4.67-9	1.94-8	0	1.96-8	7.55-11	0.08-11	7.63-11	6.19-11	6.99-12	6.89-11	8.76-8
2.40	1.89-8	2.46-9	2.14-8	2.22-8	2.16-8	3.62-9	1.13-9	4.75-9	1.86-8		1.86-8	7.63-11	0.01-11	7.64-11	6.23-11	4.26-12	6.66-11	7.17-8
2.52	1.41-8	7.13-9	2.12-8	2.40-8	3.89-10	3.30-9	1.19-9	4.49-9	1.86-8		1.86-8	5.64-11	0	5.64-11	5.52-11	1.23-11	6.75-11	5.88-8
2.64	8.30-9	1.29-9	8.43-9	3.42-8	4.46-10	3.30-9	1.22-9	4.52-9	1.91-8		1.91-8	5.96-11		5.96-11	6.08-11	2.23-12	6.30-11	6.67-8
2.76	1.46-8	0	1.46-8	2.22-8	3.41-10	3.04-9	0.95-9	3.99-9	1.99-8		1.99-8	4.25-11		4.25-11	5.25-11	0	5.25-11	6.11-8
2.88	6.55-9		6.55-9	2.16-8	3.22-10	2.92-9	1.26-9	4.18-9	2.13-8		2.13-8	3.37-11		3.37-11	5.40-11		5.40-11	5.41-8
3.00	4.40-9		4.40-9	1.60-8	3.76-10	3.06-9	1.48-9	4.54-9	2.70-8		2.70-8	3.36-11		3.36-11	4.30-11		4.30-11	5.24-8
3.12	3.64-9		3.64-9	2.65-8	4.85-10	3.25-9	1.17-9	4.42-9	2.80-8		2.80-8	3.37-11		3.37-11	3.67-11		3.67-11	6.31-8
3.24	1.70-9		1.70-9	6.80-9	3.18-10	3.15-9	0.76-9	3.91-9	2.59-8		2.59-8	2.68-11		2.68-11	3.34-11		3.34-11	3.87-8
3.36	9.40-10		9.40-10	0	2.50-10	3.06-9	0.75-9	3.81-9	2.76-8		2.76-8	2.63-11		2.63-11	2.25-11		2.25-11	3.26-8
3.48	1.07-9		1.07-9		3.12-10	3.13-9	1.42-9	4.55-9	3.18-8		3.18-8	2.90-11		2.90-11	2.16-11		2.16-11	3.78-8
3.60	1.29-9		1.29-9		2.87-10	3.38-9	0.36-9	3.74-9	3.68-8		3.68-8	2.81-11		2.81-11	1.91-11		1.91-11	4.22-8
3.72	1.26-9		1.26-9		2.98-10	3.09-9	0.02-9	3.11-9	3.16-8		3.16-8	2.72-11		2.72-11	2.29-11		2.29-11	3.63-8
3.84	7.63-10		7.63-10		3.72-10	2.93-9	0	2.93-9	2.80-8		2.80-8	2.65-11		2.65-11	1.33-11		1.33-11	3.21-8
3.96	5.86-10		5.86-10		4.39-10	3.04-9		3.04-9	3.00-8		3.00-8	2.48-11		2.48-11	5.48-12		5.48-12	3.41-8
4.08	8.75-10		8.75-10		3.61-10	3.33-9		3.33-9	3.58-8		3.58-8	2.46-11		2.46-11	7.92-12		7.92-12	4.14-8
4.20	5.75-10		5.75-10		2.43-10	3.08-9		3.08-9	2.80-8		2.80-8	2.23-11		2.23-11	5.20-12		5.20-12	3.19-8
4.32	1.77-10		1.77-10		2.09-10	2.82-9		2.82-9	2.06-8		2.06-8	1.96-11		1.96-11	1.60-12		1.60-12	2.38-8
4.44	0		0		2.98-10	2.85-9		2.85-9	1.68-8		1.68-9	1.75-11		1.75-11	0		0	2.00-8
4.56					1.98-10	3.04-9		3.04-9	1.65-8		1.65-8	1.95-11		1.95-11				1.98-8
4.68					7.15-11	3.38-9		3.38-9	1.26-8		1.26-8	1.58-11		1.58-11				1.61-8
4.80					8.84-12	3.92-9		3.92-9	1.08-8		1.08-8	1.41-11		1.41-11				1.47-8
4.92						4.98-9		4.98-9	1.58-8		1.58-8	1.47-11		1.47-11				2.08-8
5.04						5.43-9		5.43-9	1.62-8		1.62-8	1.33-11		1.33-11				2.16-8
5.16						5.13-9		5.13-9	1.32-8		1.32-8	1.38-11		1.38-11				1.83-8
5.28						5.62-9		5.62-9	1.18-8		1.18-8	1.42-11		1.42-11				1.74-8

*Lower-half soft gammas estimated by sealing no-sample calculation.

Table 4 — (Continued)

Group Upper Energy, Mev	Primary Gammas			Soft Secondary Gammas		Iron Hard Captures			Iron Hard Inelastics			Lead Hard Capture			Lead Hard Inelastics			Grand Total
	0-20°	20-45°	Sub- total	Upper Half	Lower Half	0-20°	20-45°	Sub- total	0-20°	20-45°	Sub- total	0-20°	20-45°	Sub- total	0-20°	20-45°	Sub- total	
5.40	0	0	0	0	0	6.28-9	0	6.28-9	1.31-8	0	1.31-8	1.29-11	0	1.29-11	0	0	0	1.94-8
5.52	↓	↓	↓	↓	↓	6.33-9	↓	6.33-9	1.49-8	↓	1.49-8	1.16-11	↓	1.16-11	↓	↓	↓	2.12-8
5.64	↓	↓	↓	↓	↓	4.83-9	↓	4.83-9	1.56-8	↓	1.56-8	1.00-11	↓	1.00-11	↓	↓	↓	2.05-8
5.76	↓	↓	↓	↓	↓	3.70-9	↓	3.70-9	1.09-8	↓	1.09-8	1.04-11	↓	1.04-11	↓	↓	↓	1.46-8
5.88	↓	↓	↓	↓	↓	4.02-9	↓	4.02-9	6.36-9	↓	6.36-9	1.08-11	↓	1.08-11	↓	↓	↓	1.04-8
6.00	↓	↓	↓	↓	↓	4.55-9	↓	4.55-9	6.39-9	↓	6.39-9	1.00-11	↓	1.00-11	↓	↓	↓	1.09-8
6.12	↓	↓	↓	↓	↓	3.24-9	↓	3.24-9	7.71-9	↓	7.71-9	7.86-12	↓	7.86-12	↓	↓	↓	1.09-8
6.24	↓	↓	↓	↓	↓	2.88-9	↓	2.88-9	3.30-9	↓	3.30-9	6.61-12	↓	6.61-12	↓	↓	↓	6.18-9
6.36	↓	↓	↓	↓	↓	3.43-9	↓	3.43-9	8.5-11	↓	8.5-11	7.86-12	↓	7.86-12	↓	↓	↓	3.52-9
6.48	↓	↓	↓	↓	↓	4.15-9	↓	4.15-9	0	↓	0	8.79-12	↓	8.79-12	↓	↓	↓	4.15-9
6.60	↓	↓	↓	↓	↓	4.82-9	↓	4.82-9	↓	↓	↓	6.06-12	↓	6.06-12	↓	↓	↓	4.82-9
6.72	↓	↓	↓	↓	↓	5.03-9	↓	5.03-9	↓	↓	↓	5.57-12	↓	5.57-12	↓	↓	↓	5.03-9
6.84	↓	↓	↓	↓	↓	4.33-9	↓	4.33-9	↓	↓	↓	6.55-12	↓	6.55-12	↓	↓	↓	4.33-9
6.96	↓	↓	↓	↓	↓	4.82-9	↓	4.82-9	↓	↓	↓	6.88-12	↓	6.88-12	↓	↓	↓	4.82-9
7.08	↓	↓	↓	↓	↓	5.48-9	↓	5.48-9	↓	↓	↓	4.71-12	↓	4.71-12	↓	↓	↓	5.48-9
7.20	↓	↓	↓	↓	↓	5.80-9	↓	5.80-9	↓	↓	↓	2.58-12	↓	2.58-12	↓	↓	↓	5.80-9
7.32	↓	↓	↓	↓	↓	4.18-9	↓	4.18-9	↓	↓	↓	2.65-12	↓	2.65-12	↓	↓	↓	4.18-9
7.44	↓	↓	↓	↓	↓	2.37-9	↓	2.37-9	↓	↓	↓	2.99-12	↓	2.99-12	↓	↓	↓	2.37-9
7.56	↓	↓	↓	↓	↓	2.12-9	↓	2.12-9	↓	↓	↓	7.81-13	↓	7.81-13	↓	↓	↓	2.12-9
7.68	↓	↓	↓	↓	↓	2.28-9	↓	2.28-9	↓	↓	↓	2.05-14	↓	2.05-14	↓	↓	↓	2.28-9
7.80	↓	↓	↓	↓	↓	9.67-10	↓	9.67-10	↓	↓	↓	0	↓	0	↓	↓	↓	9.67-10
7.92	↓	↓	↓	↓	↓	1.74-11	↓	1.74-11	↓	↓	↓	↓	↓	↓	↓	↓	↓	1.74-11
8.04	↓	↓	↓	↓	↓	0	↓	0	↓	↓	↓	↓	↓	↓	↓	↓	↓	0

*Lower-half soft gammas estimated by sealing no-sample calculation.

5. COMPARISON AND EVALUATION OF CALCULATION AND EXPERIMENT

Fig. 3 compares the results of the calculation and experiment for the no-sample case, and Fig. 4 gives a similar comparison for the 2-in. iron sample. The figures show the detector response in counts per 120 kev-wide channel per source neutron as a function of pulse energy, as determined directly by calculation and experiment. The calculated and measured points are plotted as they were independently determined, with no normalization or other arbitrary adjustment. The agreement in each case is good, both in terms of the magnitude of the detector response and its energy dependence. The total energy deposited in the crystal in each case (integral of the curve above 0.72 Mev) is summarized in Table 5. This total crystal dose agrees within 25% for the no-sample case and within 15% for the 2-in. iron sample. Individual channel responses agree within about a factor of two over the entire range.

Table 5 — Total Energy Deposited
in Detector Crystal

Case	Energy Deposited, ev/Source Neutron	
	Experiment	Calculation
No sample	3.60	2.74
2-in. iron sample	9.7	8.3

The following comments can be made about details of the comparison.

1. The calculated results are generally lower than experimental, especially over the energy range $0.6 < E < 2.4$ Mev. This is due partially to exclusion of large angles from the primary gamma source calculation for the no-sample case, and to exclusion of large angles for secondary gammas for the iron case.

Normalization of the gamma emission from the Po-Be source also may contribute slightly to this discrepancy (see Section 7.2).

2. The calculated values exceed the experimental values in the intermediate energy range for the iron sample. From Fig. 20, this can be seen to be attributable to hard inelastic gammas in the iron. This suggests that an excessively hard spectrum is used for the gammas emitted from inelastic scatter of high-energy neutrons in iron. The spectrum of these gammas is not well known. A change of the spectrum to emit more of the energy as softer gammas would greatly improve agreement between calculation and experiment in this area. This diagnostic feature of the experiment, pointing out specific weaknesses in the cross-section data, is one of its principal attributes. However, because of the statistical nature of the calculation, it is desirable to confirm the suggested spectrum modification by a second evaluation before introducing such a change.
3. There is much more "structure" apparent in the calculated results than in the experimental data. This is mainly because discreet spectra were used both for capture and inelastic scattering gammas, in keeping with the approach used in MCR shield analysis, whereas in reality the spectra are mainly continuous. The fact that some of the actual discreet lines, such as hydrogen (2.23 Mev), lead (7.38 Mev), and iron (5.9, 7.64 Mev) capture lines, show up less strongly in the measurements than in the

calculations suggests that the detector system smears out the response to these lines more than indicated by the response functions used in the calculations. This may well be due to slight gain drifts in the amplifier system over the course of a measurement. The response functions used for the 3 in. $\times$ 3 in. NaI crystal are based on values calculated and checked experimentally by ORNL.* Above 6 Mev, however, the ORNL detailed response data had to be extrapolated, using the known total crystal response. Some uncertainty in the detailed response to the high-energy gamma lines is introduced by this extrapolation.

4. The good agreement of measurement and experiment at the very highest energies, where capture gammas predominate, suggests that the capture cross sections and high-energy spectra are adequately handled.

*C. D. Zerby and H. S. Moran, Calculation of the Pulse Height Response of NaI (Tl) Scintillation Counters, ORNL-3169 (Jan. 3, 1962).

6. CONCLUSIONS

The experiment has provided a useful and valuable proof check of the Monte Carlo calculation techniques used in shield design.

The experiment achieved its objective of providing a configuration in which secondary gamma rays, produced in sample shielding materials, were the principal radiation measured.

Discrepancies between measurement and calculation can be used as a basis for cross section modification.

The experiments support most of the cross-section data used in the analysis. In the case of gamma rays from inelastic scatter of high-energy neutrons, the experiments suggest softening of the gamma ray spectrum used in the calculation.

Comparison of calculation and experiment for other shielding materials can provide a further proof test of calculation methods and cross-section data used in the design of the MCR shield.

7. APPENDIX

7.1 SECONDARY GAMMA DATA

Table 6 summarizes the capture 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 fractional probability of each gamma ray.

It may be noted that carbon is treated in an unusual manner. For historical reasons, carbon inelastic scatter was treated as absorption in the basic Monte Carlo cross-section data used in this analysis. Therefore, it was necessary to introduce the 4.43 Mev inelastic scatter gamma as a capture gamma response. This treatment neglects actual capture in carbon (expected to be very small). Perhaps, more significantly, neutrons undergoing inelastic scatter in carbon are no longer available for capture. This may result in an underestimate of hydrogen capture gammas from the unborated hydrocarbon region.

7.2 POLONIUM BERYLLIUM 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 general shape, there is enough difference among measurements

Table 6 — Secondary Gamma Data
(All Energies in Mev, Unless Indicated Otherwise)

HYDROGEN

Neutron Energy	Capture Gammas
	2.23
0.03 ev	1.0
18.0	1.0

No inelastic gammas.

LITHIUM

No capture or inelastic gammas.

BERYLLIUM

No capture or inelastic gammas.

BORON-10 AND BORON-11

No capture or inelastic gammas.

CARBON

Neutron Energy	Capture Gammas
	4.43
0.037 ev	0
4.5	1.0
18.0	1.0

Inelastic treated as absorption (see text).

OXYGEN

Neutron Energy	Inelastic Gammas						
	1	2	3	4	5	6.1	7
6.63	0	0	0	0	0	0	0
8.51	0	0	0	0	0	1.0	0
10.9	0.07	0.03	0.01	0	0	0.72	0.20
14.0	0.09	0.10	0.16	0.18	0.05	0.31	0.16
18.0	0.04	0.10	0.17	0.27	0.12	0.27	0.13

No capture gammas.

Table 6 — (Continued)

ALUMINUM

Neutron Energy	Capture Gammas					
	1	1.8	2.5	4	6	7.72
0.03 ev	0	0	0	0	0	0
2.83	1.0	1.5	0.7	.0.6	0.2	0.2
3.5	0.2	0.3	0.14	.0.12	0.04	0.04
18.0	0	0	0	0	0	0

	Inelastic Gammas							
	1	2	3	4	5	6	7	8
0.897	0	0	0	0	0	0	0	0
1.15	1.0	0	0	0	0	0	0	0
1.48	1.0	0	0	0	0	0	0	0
1.90	1.0	0	0	0	0	0	0	0
2.44	0.97	0.035	0	0	0	0	0	0
3.13	0.78	0.15	0.07	0	0	0	0	0
4.02	0.50	0.16	0.34	0	0	0	0	0
5.16	0.33	0.17	0.39	0.11	0	0	0	0
6.63	0.15	0.21	0.29	0.26	0.10	0	0	0
8.51	0.15	0.18	0.18	0.22	0.23	0.14	0.04	0
10.9	0.21	0.20	0.18	0.17	0.16	0.17	0.15	0.13
14.0	0.32	0.19	0.18	0.20	0.17	0.14	0.12	0.21
18.0	0.33	0.19	0.18	0.21	0.18	0.17	0.14	0.20

IRON

Neutron Energy	Capture Gammas					
	0.75	1.5	2.5	4	5.9	7.64
0.03 ev	0	0	0	0	0	0
4.9	0.70	0.60	0.27	0.23	0.25	0.40
5.5	0.21	0.18	0.08	0.07	0.08	0.12
18.0	0	0	0	0	0	0

Table 6 — (Continued)

IRON (Continued)

Inelastic Gammas										
	0.85	1	2	3	4	5	6	7	8	9
0.699	0	0	0	0	0	0	0	0	0	0
0.897	1.0	↓	↓	↓	↓	↓	↓	↓	↓	↓
1.15	1.0	↓	↓	↓	↓	↓	↓	↓	↓	↓
1.48	1.0	↓	↓	↓	↓	↓	↓	↓	↓	↓
1.90	1.0	↓	↓	↓	↓	↓	↓	↓	↓	↓
2.44	1.0	↓	↓	↓	↓	↓	↓	↓	↓	↓
3.13	1.0	0.08	0.14	↓	↓	↓	↓	↓	↓	↓
4.02	1.0	0.19	0.36	↓	↓	↓	↓	↓	↓	↓
5.16	1.01	0.17	0.36	0.15	↓	↓	↓	↓	↓	↓
6.63	1.16	0.08	0.18	0.45	0.21	0.02	↓	↓	↓	↓
8.51	1.36	0.14	0.24	0.39	0.18	0.22	0.11	↓	↓	↓
10.9	1.73	0.21	0.30	0.33	0.20	0.18	0.16	0.13	0.07	0.01
14.0	2.13	0.29	0.37	0.27	0.17	0.16	0.13	0.10	0.07	0.06
18.0	2.22	0.34	0.43	0.25	0.11	0.10	0.07	0.06	0.04	0.02

LEAD

Capture Gammas	
Neutron Energy	
	6.73 7.38
0.03 ev	0 0
18.0	0.077 1.0

Inelastic Gammas					
	1	2	3	4	5
0.699	1.0	0	0	0	0
0.897	1.0	↓	↓	↓	↓
1.15	1.0	↓	↓	↓	↓
1.48	1.0	↓	↓	↓	↓
1.90	1.16	↓	↓	↓	↓
2.44	1.34	↓	↓	↓	↓
3.13	1.23	0.030	↓	↓	↓
4.02	1.08	0.045	0.36	↓	↓
5.16	1.16	0.065	0.46	↓	↓
6.63	1.17	0.097	0.55	0.016	↓
8.51	1.15	0.13	0.57	0.050	0.011
10.9	1.43	0.15	0.66	0.068	0.030
14.0	1.82	0.18	0.72	0.049	0.015
18.0	1.93	0.17	0.66	0.039	0

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 can now 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.

7.2.1 Mound Laboratory Po-Be Sources⁴

The source used in the small source experiments is a 40-curie standard 0.7-in. diameter by 0.7-in. long Mound Laboratory source. Since 1950, Mound Laboratory sources have been made by diffusion of polonium at high temperature into a porous beryllium plug. This process produces a stable Po-Be intermetallic compound, a few molecules thick, on the surface of beryllium particles which are large compared with the range of the polonium alpha particle. Sources made before 1950, or at other places, may have been made as alternate layers of beryllium with polonium-bearing material or as powder mixtures. The layered sources may give a slightly different spectrum since some slowing down of the alpha particles occurs in the polonium layer before the particles enter the beryllium. The intermetallic compound developed in the Mound Laboratory source is stable up to about 700°C, preventing any crystal growth which might cause an aging effect on the spectrum.

7.2.2 Spectrum Measurements

It is generally considered that the measurement of proton recoil tracks in photo-emulsions 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. 21, the Po-Be neutron spectrum is reproducible when a consistent measurement technique is used. In Fig. 21, 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 Henry⁵ at ORNL. The plotted points are values taken from plots prepared by averaging a number of measurements made on two different sources by Breen, Hertz, and Wright⁶ at Mound Laboratory. In each case, the data have been renormalized to a basis of one neutron. These measurements agree within a few percent 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. 22 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 differences 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

Fig. 21 — Counter measurements of neutron spectra of three Mound Laboratory Po-Be sources. Curve — Cochran and Henry (Reference 5); o source PN-47 (Reference 6); ● source RB-2 (Reference 6).

Fig. 22 — Comparison of Po-Be source measurements. Histogram — Notarrigo et al. (Reference 2); curve — average of three curves of Fig. 21.

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. 22 are in good agreement with photographic emulsion measurements of the Po-Be neutron spectrum by Medveczky⁷ (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.⁸ Agreement is good with a recent photographic emulsion measurement of the spectrum of a Pu-Be source by Stewart.⁹ 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. 23 shows a smoothed spectrum derived from the measurements of Notarrigo et al. This spectrum has been used in analysis of the small source experiments.

7.2.3 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 work¹⁰ has shown that the 4.43 Mev level returns to the ground state by emission of a 4.43 Mev gamma. The 7.65 level normally decays by internal pair

Fig. 23 — Po-Be source spectrum used for small source experiment analysis

production, but occasionally 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, 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 — Gamma Rays from Po-Be Neutron Source

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

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