229
5-7-70
C

~~1359~~
1360

ORNL-4475
UC-40 - Radiation Effects on Materials

MASTER

GAMMA-RAY SPECTRA ARISING FROM
FAST-NEUTRON INTERACTIONS IN
ELEMENTS FOUND IN SOILS, CONCRETES,
AND STRUCTURAL MATERIALS

R. E. Maerker
F. J. Muckenthaler

OAK RIDGE NATIONAL LABORATORY

operated by

UNION CARBIDE CORPORATION

for the

U. S. ATOMIC ENERGY COMMISSION

THIS DOCUMENT CONFIRMED AS

UNCLASSIFIED

DIVISION OF CLASSIFICATION

BY

DATE

J. H. Kahn / am
5/15/70

P5223

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

Printed in the United States of America. Available from Clearinghouse for Federal
Scientific and Technical Information, National Bureau of Standards,
U.S. Department of Commerce, Springfield, Virginia 22151
Price: Printed Copy \$3.00; Microfiche \$0.65

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

- A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or
 - B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.
- As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

ORNL-4475

Contract No. W-7405-eng-26

NEUTRON PHYSICS DIVISION

GAMMA-RAY SPECTRA ARISING FROM FAST-NEUTRON
INTERACTIONS IN ELEMENTS FOUND IN SOILS,
CONCRETES, AND STRUCTURAL MATERIALS

R. E. Maerker and F. J. Muckenthaler

NOTE:

This Work Funded by
DEFENSE ATOMIC SUPPORT AGENCY
Under Subtask RRP-A2-11.037

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission nor any person acting on behalf of the Commission

A. Makes any warranty or representation expressed or implied with respect to the accuracy, completeness or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights, or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor

APRIL 1970

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

TABLE OF CONTENTS

	Page No.
ABSTRACT	v
I. INTRODUCTION	1
II. DESCRIPTION OF THE EXPERIMENT.	2
III. ANISN CALCULATIONS OF NEUTRON SPECTRAL FLUXES IN THE SLABS.	18
IV. DETERMINATION OF RESPONSE FUNCTIONS AS A FUNCTION OF GAMMA-RAY SOURCE ENERGY.	19
V. GAMMA-RAY SPECTRA AT 90 DEG FROM FAST-NEUTRON INTERACTIONS USING THE UNALTERED RESPONSE FUNCTIONS. .	20
Oxygen	26
Carbon	32
Iron	35
Aluminum	47
Copper	58
Titanium	69
Zinc	79
Calcium.	90
Potassium.	100
Silicon.	114
Sulfur	127
VI. GAMMA-RAY SPECTRA AT 90 DEG FROM FAST-NEUTRON INTER- ACTIONS USING RESPONSE FUNCTIONS ALTERED TO ACCOUNT FOR EFFECTS OF GAMMA-RAY SCATTERING IN THICK SLABS . .	136

	Page No.
Nickel	138
Stainless Steel.	151
Barium	164
Sodium	176
VII. CONCLUSIONS AND SUMMARY.	191
VIII. ACKNOWLEDGMENTS.	192
IX. APPENDIX	193
REFERENCES AND FOOTNOTES	235

Abstract

Experimental data are presented for the production of secondary gamma rays arising from the interaction of a fission-like spectrum of fast neutrons with oxygen, carbon, iron, aluminum, copper, zinc, titanium, nickel, silicon, calcium, potassium, sodium, barium, sulfur, and a stainless steel. Measurements of the gamma-ray spectra for these elements were made at the Tower Shielding Facility with a carefully calibrated 5-in. by 5-in. NaI(Tl) detector. These data are useful in evaluating the accuracy of $(n, x\gamma)$ cross-section sets. The data are differential in the gamma-ray energy from 1 MeV to approximately 6.5 MeV and are expressed as values of 4π times the average differential gamma-ray production cross section at 90 deg to the incident neutron beam for all neutrons in the beam lying above 1 MeV. These results are estimated to have an accuracy of $\pm 30\%$ and include the contribution from both discrete and continuum gamma rays. Comparisons of these data with results obtained by averaging previously published differential cross sections over the incident spectrum indicate that considerable differences exist between the values obtained at the Tower Shielding Facility and the earlier data. Very approximate experimental data are presented for gamma rays arising from resonance capture of neutrons from a $1/E$ energy spectrum in the energy range 10-200 keV.

GAMMA-RAY SPECTRA ARISING FROM FAST-NEUTRON INTERACTIONS IN ELEMENTS
FOUND IN SOILS, CONCRETES, AND STRUCTURAL MATERIALS

R. E. Maerker and F. J. Muckenthaler

I. Introduction

The importance of secondary gamma-ray production cross sections from fast-neutron interactions in shielding materials has increased in the past few years because of the necessity of predicting dose rates in weapons shielding calculations. Invariably in those situations where the prompt fission gamma rays are strongly attenuated by the bomb materials, the gamma-ray dose rate is dominated by the secondaries produced by fast-neutron interactions. A further increase in the importance of these cross sections is indicated by the need of accurately predicting fast-neutron and gamma-ray fluxes in fast reactors. Shielding of conventional thermal reactors also requires knowledge of these cross sections, although perhaps not to the accuracy demanded by the first two applications since the gamma-ray dose is usually dominated by thermal-neutron captures in the shield.

Existing data on gamma-ray production from fast-neutron interactions in shielding materials is limited almost exclusively to inelastic scattering reactions and may be found in two series of references. The first is BNL-325,¹ which summarizes experimental cross sections obtained for a few gamma rays for each element and is limited to neutron energies less than about 4 MeV, and is therefore very incomplete. The second is the work of United Nuclear²⁻⁵ which supplemented the then existing experimental data with theoretical calculations to arrive at production cross sections as a function of neutron energy from threshold up to 18 MeV for both the discrete and continuum gamma rays. The United Nuclear data are the most complete set of inelastic gamma-ray production cross sections available, although only a few selected elements were analyzed. In this report, a systematic test has been applied to these data by comparing the results of integral experiments with calculated results using these data.

The general purpose of the experiments described in this report is to provide integral checks on the gamma-ray production cross sections from fast-neutron interactions predicted by a given set of cross-section

data. By integrating the cross-section data over a fixed fast-neutron spectrum, comparisons can be made with an experimentally determined cross section for the same spectrum. It is felt that these experiments provide a good integral check on such data sets as those of United Nuclear or others which might be developed in the future.

II. Description of the Experiment

Figure 1 shows the experimental arrangement used at the Tower Shielding Facility. The TSR-II spherical reactor was placed inside a spherical lead-water beam shield to provide a convenient means of collimating the neutron beam and at the same time of reducing the background. The collimator consisted of a cylindrical section $4\frac{1}{4}$ in. in inside diameter and $2\frac{1}{4}$ in. long, attached to a conical section $19\frac{1}{2}$ in. long at the reactor end of the collimator. This arrangement reduced fast-neutron scattering in the collimator. The face of the reactor shield and the inside of the cylindrical section of the collimator were lined with cadmium to eliminate thermal-neutron scattering along the collimator walls and to reduce the number of thermal neutrons diffusing through the shield and reaching the sample. A 2-in.-thick lead disk was inserted in the collimator adjacent to the reactor vessel to reduce the gamma-ray intensity scattered by a sample of material placed in the beam. A 36- by 30-in. lead collar 8 in. thick and containing a 15-in. diameter hole was placed around the collimator opening to prevent capture gamma rays in the cadmium from reaching the detector. The slab samples of material to be investigated, with dimensions varying between 4 ft by 5 ft and 5 ft by 5 ft, were placed vertically in the beam at an angle of 45° to the beam centerline and at a distance of 6 ft from the edge of the reactor shield.

A 5-in.-diameter by 5-in.-long NaI(Tl) crystal was located inside a spherical lead-water shield placed at an angle of 90° to the reactor beam centerline (45° with respect to the slab sample normal) so that the detector viewed the transmitted beam side of the slab. This geometry was chosen to minimize incidence of reactor capture and fission gamma rays on the crystal and to reduce to 0.51 MeV or less the energy of these

Fig. 1. Schematic Diagram of Geometry for Determining Average Secondary Gamma-Ray Production Cross Sections From Fast-Neutron Interactions. In addition, a 6-in. thickness of lithium hydride, not shown, was placed in the detector beam between the slab and the detector collimator.

gamma rays scattered by the sample to the crystal. The detector collimator consisted of a series of four 4-in.-thick lead irises distributed over a distance of 51 in. from the crystal. The inside edges of the irises were beveled to provide a conical-shaped collimator 5 in. in diameter at the crystal surface and 12.3 in. in diameter at the outer edge of the last iris. A 2-in.-thick borated (5.9% by weight natural boron) polyethylene slab was placed against the iris farthest from the crystal to reduce thermal-neutron effects in the crystal.

The axis of the detector collimator intersected the beam centerline at the midpoint of the slab sample. In this geometry, the crystal was 20 ft from the midpoint, and was permitted to view the entire transmitting face of the slab. Two aluminum-walled tanks filled with borated water and covered with a thin layer of borated polyethylene were placed roughly half-way along the slab-crystal axis to further define the volume viewed by the crystal.

The arrangement described in the above paragraphs is identical to that described for the series of capture gamma-ray measurements reported in ORNL-4382.⁶ For the present measurements, however, two changes were made in the previous arrangement. First, a 6-in.-thick lithium hydride slab (density = 0.74 g/cm^3) enclosed in a 1/8-in.-thick aluminum container was placed roughly half-way along the crystal-slab sample axis in the detector beam. The purpose of the lithium hydride shield was to reduce fast-neutron effects in the crystal. It was in place during the entire course of the measurements described in this report. This shield produced negligible gamma-ray scattering effects in the detector collimator and crystal, as can be seen by referring to page 83 of ORNL-4382, where under "Present Results" the capture gamma-ray spectrum of copper is presented with and without the lithium hydride shield in place. Second, one series of measurements described in this report was made with a 2.13-cm-thick boron cover (density = 1.29 g/cm^3 and enriched to 92.3% by weight in the ^{10}B isotope) fitted over the reactor collimator. A second series of measurements was made with a 30-mil thickness of cadmium fitted over the

reactor collimator, with the boron filter removed. The cadmium-covered beam was also employed in the measurements described in ORNL-4382, but for that series of experiments the lithium hydride shield was not necessarily present.

The boron filter was used to selectively reduce the intensity of neutrons of intermediate and thermal energies compared with that of fast neutrons in the MeV range, so that the secondary gamma rays detected were the result of fast-neutron interactions in the slab samples. Figure 2 shows the attenuation of the boron filter as a function of neutron energy. It is seen that to a good approximation the boron filter "cuts off" in the vicinity of 10 keV, and has a constant attenuation of 0.75 to neutrons above 1 MeV. The cadmium filter was assumed to have a cutoff at 0.5 eV, and a constant attenuation of 0.986 to neutrons above 1 MeV. The results of gamma-ray measurements using the boron filter were thus interpretable in terms of incident neutrons above the vicinity of 10 keV, and those using the cadmium filter of all incident neutrons above the vicinity of 0.5 eV.

The spectrum of fast neutrons ($E_n > 1$ MeV) in the beam was assumed to be basically the same as that measured earlier with an NE-213 liquid scintillator in another experiment.⁷ The cited measurement was made with 2 in. of lead placed in the reactor beam rather than with the 2 in. of lead placed next to the reactor core in the reactor collimator as in the present experiment. However, it is felt that both geometries are "good" geometries in the sense that any inelastic scattering in the lead removes the neutron from the beam. A further difference in the two geometries lies in the separation of the reactor collimator from the point of measurement. In the cited measurement it was 100 ft, while in the present measurement it is 6 ft. Thus, the spectrum through 2 in. of lead presented in reference 7 was corrected by dividing by the attenuation through 94 ft of air and then multiplied by the attenuation through the boron filter in the present experiment.

The boron-filtered beam was mapped with a Hurst dosimeter in a plane perpendicular to the beam at a point 6 ft from the edge of the reactor collimator. Only an east-west mapping was performed since previous studies

Fig. 2. Attenuation of the Boron Filter as a Function of Neutron Energy.

have shown similar beams to be reasonably symmetrical.⁸ The results of the beam mapping appear in Fig. 3. Integrating over the surface of a 4.5-ft by 5-ft slab oriented at 45 deg to the plane yields a total incident first-collision dose rate of 1.06×10^7 erg cm²/gr/hr/100 Kw. The factor $2\pi \int_{\text{slab}} D(r) r dr / D(r=0)$ is thus 1800 cm². Applying this normalization to the corrected curve of reference 7 by using the flux-to-dose single-collision factors of Henderson⁹ results in the absolute fast-neutron spectrum incident on the slab samples appearing in Fig. 4. The cumulative spectrum of those neutrons that lie below 14 MeV is presented in Fig. 5 along with a comparison of this spectrum with a fission spectrum normalized to the same number of neutrons above 1 MeV. It can be seen that the incident spectrum is harder than a fission spectrum above 1 MeV, due primarily to the presence of the water in the reactor core.

The absolute spectra of neutrons down to thermal energies that are incident on the slab samples with the boron and cadmium filters in place were inferred from measurements made using the bare beam. A Blosser spectrometer employing various thicknesses of ¹⁰B filters in conjunction with a calibrated BF₃ detector was placed at the centerline of the slab sample but with no slab present and the counting rates measured as a function of filter thickness. The counting rates were consistent with a 1/E spectrum from 5 eV to the vicinity of 10 keV. Below 5 eV the spectrum appeared to be spurious, rising to a factor of ~ 25 higher at 0.5 eV than a 1/E extrapolation of the data above 5 eV. Since the spectra determined previously with this spectrometer of similar beams emerging from a reactor collimator were consistent with a 1/E shape, the data below 5 eV were disregarded and the spectrum was assumed to be 1/E for all energies between 0.5 eV and 10 keV. The centerline flux from the bare beam was

$$\phi(E) = \frac{2.70 \times 10^9}{E} \text{ n/cm}^2/\text{eV}/100 \text{ Kw}/200 \text{ min} .$$

A mapping of the cadmium-covered beam in an east-west direction perpendicular to the beam at points 6 ft from the edge of the reactor collimator was made with a bare and cadmium-covered BF₃ detector. This is shown in Fig. 6 along with a cadmium-covered BF₃ measurement on the same plane from the bare beam.

Fig. 3. Measured Single-Collision Dose Rate Profile Perpendicular to the Boron-Filtered Beam at a Point 6 ft from the Edge of the Reactor Collimator.

Fig. 4. Surface-Integrated Spectrum of Neutrons Above 0.5 MeV Incident on Slab Samples from the Boron-Filtered Beam.

Fig. 5. Cumulative Spectrum of Fast Neutrons Lying Below 14 MeV Incident on the Slab Samples Through the Boron Filter and Comparison with a Fission Spectrum Having the Same Number of Neutrons Above 1 MeV.

Fig. 6. Measured BF_3 Profiles Perpendicular to the Beam for the Several Filtered Beams.

Comparing the cadmium-covered BF_3 counting rates from the bare beam and the cadmium-covered beam, it can be seen that the bare beam averages about 15% more near the beam centerline. This is presumably due to attenuation by the cadmium filter of neutrons in the 0.5 eV - 1 keV energy region. Integrating the cadmium-covered BF_3 counting rates from the cadmium-covered beam over the plane of a 4.5 ft by 5 ft slab sample placed at 45 deg to the beam results in a value 1700 times the centerline value:

$$2\pi \int_{\text{slab}} C(r) r dr / C(r=0) = 1700 \text{ cm}^2 .$$

This number compares favorably with the value 1800 cm^2 obtained for the fast-neutron dose rate.

The spectrum of the surface integrated current incident on the slab sample through the cadmium-covered beam is thus

$$\iint_{\text{slab}} J(E) dA = \frac{2.70 \times 10^9 \times 1700}{1.15 \times E} = \frac{4.0 \times 10^{12}}{E}$$

$$\text{neutrons/eV/100 Kw/200 min, } 0.5 \text{ eV} \leq E \leq 10 \text{ keV.}$$

The corresponding spectrum through the boron-filtered beam is obtained by multiplying the above spectrum by the attenuation of the boron filter shown in Fig. 2. The resultant complete epicadmium spectra incident on the slab samples through the two filters are illustrated in Figs. 7 and 8, where the dotted lines are the estimates for the region not covered by the spectrometers. Table I compares the relative number of neutrons per decade of energy transmitted through the cadmium and boron filters. Over 70% of the neutrons transmitted through the boron filter lie above 1 MeV, and almost 95% of them lie above 0.1 MeV. Almost 15% of the neutrons transmitted through the cadmium filter lie below 1 keV, whereas a negligible amount of these neutrons are passed by the boron filter.

An inspection of Fig. 6 shows that some thermal (i.e., subcadmium) neutrons are incident on the slab samples from both the cadmium- and boron-filtered beams. This rather unexpected occurrence was investigated further

Fig. 7. Surface-Integrated Neutron Spectrum Incident on the Slab Samples
From the Cadmium-Covered Beam.

Fig. 8. Surface-Integrated Neutron Spectrum Incident on the Slab Samples
From the Boron-Filtered Beam.

Table I. Comparison of Epicadmium Neutron Spectra Transmitted Through Cadmium and Boron Filters

Energy Region	Neutrons/100 Kw/200 min		Fraction of Total Neutrons in Transmitted Spectrum	
	Through Cadmium	Through Boron	Through Cadmium	Through Boron
0.5-10 eV	1.2(13) ^a	-	0.054	0
10-100 eV	9.2(12)	-	0.040	0
0.1-1 keV	9.2(12)	8.7(10)	0.040	0.001
1-10 keV	9.2(12)	1.4(12)	0.040	0.011
10-100 keV	1.5(13)	6.7(12)	0.066	0.053
0.1-1 MeV	5.3(13)	2.9(13)	0.233	0.228
1-10 MeV	1.2(14)	9.0(13)	0.527	0.707

^aRead as 1.2×10^{13} .

experimentally. A shadow shield was placed between the reactor collimator and the usual 6-ft measurement point, effectively blocking off all radiation from the reactor collimator, and bare and cadmium-covered profiles were measured in a plane perpendicular to the boron-filtered beam. Fig. 9 shows the results of this measurement, and comparing the (bare-cad) response with the (bare-cad) response for the boron-filtered beam reproduced from Fig. 6 shows the two to be equal. Consequently the thermal neutrons are caused by leakage through the reactor shield of neutrons of less than a few keV in energy. The magnitudes of the surface-integrated subcadmium currents are

$$\int_{\text{slab}} J_{\text{th}} dA = 3.8 \times 10^{11} \text{ neutrons/100 Kw/200 min}$$

for the boron-filtered beam, and

$$\int J_{\text{th}} dA = 8.0 \times 10^{11} \text{ neutrons/100 Kw/200 min}$$

for the cadmium-filtered beam. In the above calculations, the calibration factor 1.32×10^{-2} n/cm²/sec per count/min was used.

The crystal used throughout the experiments possessed a particularly good resolution for sodium iodide, and when used in conjunction with an

Fig. 9. Measured BF_3 Profiles Perpendicular to the Boron-Filtered Beam.

RCA-4525 photomultiplier and associated electronic circuits yielded resolutions which varied from 6.2% at 1.38 MeV (FWHM) to 2.5% at 10.83 MeV.

Pulses from the detection system were fed through a preamplifier to a DD₂ linear amplifier, and from there to a Nuclear Data 512-channel analyzer. Storage in the analyzer of that portion of the gamma-ray spectrum lying below 1 MeV was eliminated to reduce the dead time in the analyzer and thus increase the number of high-energy pulses accepted per unit of time. This was accomplished by adjusting the PHS level so that the output pulses resulted from the deposition of energy within the crystal from gamma rays with energies of 1 MeV or greater. These pulses were shaped to act as a trigger to open the gate in the coincidence circuit of the analyzer. The linear pulse from the amplifier was then delayed for several microseconds to reach the analyzer during the time that the gate was open.

The dead time was always kept less than 20% of the elapsed time, since previous studies with this system had indicated that little, if any, spectral distortion occurred at or below this counting rate level. All counting times referred to in this report should be interpreted as analyzer live times; the actual counting times were somewhat longer. In most cases, the runs were made at power levels somewhat less than 100 Kw because of dead time considerations, but the runs (unless otherwise specified) were normalized to a reactor power of 100 Kw for 200 minutes live time.

The loss of spectral information below 1 MeV was not considered important to the obtaining of the secondary gamma-ray data, since for shielding calculations it is usually unimportant. Furthermore, the presence of reactor gamma radiation scattered by the slab sample that lies below 1 MeV and the increased importance of collimator and borated polyethylene scattering below 1 MeV would complicate any analysis in this region out of all proportion to the usefulness of the results.

III. ANISN Calculations of Neutron Spectral Fluxes in the Slabs

Knowledge of the incident absolute neutron spectra alone is not in general sufficient to predict production cross sections of the secondary gamma rays from the slabs. The fluxes in each of several incident energy groups are perturbed by the slab sample, and these fluxes in turn are dependent on the neutron interaction cross sections of the slab. To a more than sufficient degree of accuracy, however, the neutron interaction cross sections are known well enough to determine the neutron spectral fluxes which give rise to the secondary gamma rays in the relatively thin slabs used in these experiments. Often the approximation has been used for thin slabs that the neutron spectral fluxes suffer no attenuation or buildup, and that spatial integration of these group fluxes over the thickness of the slab, which is the source term for the secondary gamma rays, is adequately represented by the product of the incident flux (assumed constant) and the thickness of the slab. For some of the slab samples used, this approximation was somewhat questionable. In addition, the slowing-down-to-thermal fluxes needed to be known for some of the slab samples that contained an appreciable amount of water. As a consequence, calculations of neutron group fluxes throughout the thickness of many of the slab samples used were performed using the one-dimensional¹¹ S_n code ANISN.¹²

Specifically, ANISN calculations were performed for aluminum, silicon dioxide, calcium oxide, potassium carbonate, sodium carbonate, barium oxide, nickel and stainless steel. These calculations included as part of the slab any aluminum container that was present, and were made for both the boron-filtered and cadmium-filtered beams. For the boron-filtered beam, the incident spectrum from ~ 0.5 keV to 15 MeV was divided into sixteen groups and the group fluxes calculated at up to eighteen points in the slab configuration for twenty-one groups (down to and including a subcadmium group). For the cadmium-filtered beam, an incident $1/E$ spectrum from 0.5 eV to 100 keV was divided into nine groups and the group fluxes calculated at up to eighteen points in the slab configuration for ten groups (including a subcadmium group). Table II presents the common energy groups used in these calculations.

Table II. Energy Groups Used in the ANISN Calculations

Group	Energy Range	Group	Energy Range	Group	Energy Range
1	10-15 MeV	8	0.55-0.90 MeV	15	0.75-3.5 keV
2	6.7-10 MeV	9	0.33-0.55 MeV	16	160-750 eV
3	4.5-6.7 MeV	10	0.20-0.33 MeV	17	37-160 eV
4	3.0-4.5 MeV	11	0.12-0.20 MeV	18	8.3-37 eV
5	2.0-3.0 MeV	12	40-120 keV	19	1.85-8.3 eV
6	1.35-2.0 MeV	13	12-40 keV	20	0.41-1.85 eV
7	0.90-1.35 MeV	14	3.5-12 keV	21	< 0.41 eV

Specific results from these calculations will appear in the sections devoted to analysis of the individual slab samples.

IV. Determination of Response Functions as a Function of Gamma-Ray Source Energy

The experimental determination of the absolute response functions, comparisons with the results of Monte Carlo calculations, description of the interpolation scheme for source energies other than those found experimentally, and a check of these response functions with a relatively well-known complex spectrum are described in Sections IV and V of ref. 6, and will not be repeated here. There it is concluded that the response functions are accurate to $\pm 15\%$ over the important pulse-height range. Figure 10 shows the absolute response functions obtained at eight source energies.

A computer program was written incorporating an interpolation scheme using the absolute response functions appearing in Fig. 10 together with an extension of the 1.38-MeV response down to 0.14-MeV pulse height energy and the addition of a 1.0-MeV response down to 0.62 MeV. These nine absolute response functions, in units of counts/MeV/incident photon, were entered in intervals of 0.02 MeV from 1.0 MeV (0.14 MeV in the case of the 1.38-MeV response and 0.62 MeV in the case of the 1.0-MeV response) to an energy sufficiently beyond the source energy that the response had dropped to less than 0.01 of the total absorption peak response. The

input to the code was a spectrum $\mathcal{J}_\gamma(E_\gamma)$, in units of incident photons/100 Kw/200 min, and the output from the code was the response of the crystal in units of counts/MeV/100 Kw/200 min as a function of pulse-height energy in the geometry of Fig. 1. Thus the code is a "folding" code, but after $\mathcal{J}_\gamma(E_\gamma)$ has been sufficiently well determined by trial and error to reproduce the measured response, the code may be interpreted as an "unfolding" code, and it is in this sense that it is most useful in the work about to be described.

V. Gamma-Ray Spectra at 90 deg from Fast-Neutron Interactions Using the Unaltered Response Functions

This section will describe the spectra obtained from materials thin enough that the effect of scattering of the secondary gamma rays in the material itself was neglected, and hence the unfolding of the measured data used the response functions appearing in Fig. 10, along with the extension of the 1.38-MeV response to lower pulse-height energies and the addition of the 1.0-MeV response, without alteration. Table III lists the elements analyzed under this classification, together with other pertinent information concerning the physical form the element assumed.

Table III. Elements Analyzed Using Unaltered Response Functions

Element	Form	Compound	Density (grams/cm ³)	Thickness (in.)	Main Impurities (% by wt.)	Aluminum Container Side Thickness
						(in.)
Oxygen	Liquid	Borated H ₂ O	1.0	1.0	negligible	0.063
Carbon	Plastic	CH ₂	0.925	1.0	negligible	-
Iron	Metal	-	7.83	0.075	negligible	-
Aluminum	Metal	-	2.7	0.25	0.5% Fe	-
Copper	Metal	-	8.8	0.069	negligible	-
Titanium	Metal	-	4.5	0.266	negligible	-
Zinc	Metal	-	7.1	0.0675	0.5% Cu	-
Calcium	Powder	CaO	1.24*	1.0	1% Fe 1% Mg 1% Si 2.7% H ₂ O	0.073
Potassium	Powder	K ₂ CO ₃	0.819*	1.0	5.4% H ₂ O	0.063
Silicon	Powder	SiO ₂	1.65*	1.0	negligible	0.073
Sulfur	Powder	-	0.583*	1.0	negligible	0.063

Fig. 10. Absolute Responses for a 5- x 5-in. NaI Crystal Preceded by 2-in. Borated Polyethylene in a Simple Collimator.

For those elements that were analyzed in a non-self-supporting form, i.e., liquid and powder, an $\sim 1/16$ -in.-thick aluminum container was built, attached to guy wires, and used for support. This container was of such a size that it could support a 1-in. thickness of material approximately 4 ft by 5 ft in cross sectional area when filled. Aluminum struts were bolted through the container to prevent the structure from buckling when filled. The effective thickness of aluminum was determined by comparing the intensity of the thermal-neutron capture gamma-ray spectra from the empty container with that from a 0.25-in.-thick slab of aluminum (see ref. 6).

The average densities of the powders when packed into the container were measured by comparing the changes in weight when the container was filled with powder and when filled to the same level with water. These densities were considerably lower than handbook values, presumably because of the loose packing and the consequent presence of voids throughout the volume occupied by the powder.

The impurities were determined at the Oak Ridge National Laboratory with the help of the General Analysis Laboratory Group of the Analytical Chemistry Division under the supervision of W. R. Laing. Either or both spectroscopic analysis and chemical analysis techniques were used.

Prior to the presentation of the data and the accompanying analyses, several general remarks should be made. First, the background for all the runs using the boron-filtered beam was assumed to be the same (run 320-B in the appendix), and was obtained for the case of no slab sample in the beam for 100 Kw actual power and 200 min. The background for all the runs using the cadmium-filtered beam was also assumed to be the same (run 319-D in the appendix), and was obtained for the case of no slab sample in the beam, but for 100 Kw and 400 min. In these background determinations, the lithium-hydride shield was present in the detector beam, and the appropriate filter remained in place over the reactor collimator. The channel shift as a function of channel number of each of these background runs was determined with respect to the corresponding foreground runs so that pulse heights corresponding to the same energies deposited in the crystal could be subtracted. This shift was due to a change in the gain of the system from foreground to background runs and not to a zero shift. It was primarily determined by observing the channels in which a 2.22-MeV

hydrogen capture peak, a weak 4.43-MeV peak, and a ~ 6.9 -MeV peak of unknown origin which were always present in the background runs occurred. The gain of the system was always adjusted at the beginning of each run so that the 4.43-MeV total absorption peak from a PoBe source lay within plus or minus one channel of channel 200. The gain shift at the end of each run was observed by again using the PoBe source and noting the channel shift of the absorption peak from its original location. The run was retained if the channel shift was less than 1 channel. Since at times the gain was count rate dependent, the determination of the gain shift between foreground and background runs was primarily obtained from the average location of the known peaks during the run, as noted above.

Second, the contribution to the net response (i.e., foreground less background) from both the boron-filtered and cadmium-filtered beams arising from capture of the incident subcadmium neutrons was subtracted, since these gamma rays represented contamination of the desired response to secondary gamma rays from fast-neutron interactions. The response from gamma rays arising from capture of the incident subcadmium neutrons was calculated for each run by using the spectral intensities in photons/capture obtained in ref. 6, calculating the number of captures per incident neutron using the codes developed in ref. 6, normalizing to the number of subcadmium neutrons incident, attenuating the gamma rays through the slab sample, lithium hydride, air, and borated polyethylene, and folding the resultant spectral intensities with the response functions. These contributions were generally very small compared to the net response, being appreciable only for iron, copper, and aluminum.

Third, the contribution of the aluminum container, when present, to the net response from both the boron-filtered and cadmium-filtered beams was subtracted as accurately as possible, prior to unfolding. Both the contributions from capture of incident subcadmium neutrons and gamma rays from fast-neutron interactions were considered. The former was calculated along with slab material as described in the paragraph above. For the latter, ANISN calculations indicated that the source term was proportional to the total thickness of aluminum present for the thin slab

samples used, so that the contribution from the aluminum container due to gamma rays from other than capture of incident subcadmium neutrons was scaled to the 0.25-in.-thick aluminum net response corrected for capture of incident subcadmium neutrons according to the total thickness of aluminum present (twice the entries in the last column of Table III).

Fourth, only the net boron-filtered responses were unfolded. The experimental net cadmium-filtered responses were used as checks on the analysis of the net boron-filtered responses in the following manner. The unfolded secondary gamma rays from the boron-filtered beam that arise as a result of fast-neutron ($E_n > 1$ MeV) interactions in the slab material should be enhanced by a factor of $\sim 1/0.75 = 1.33$ in the cadmium-filtered beam, because of the almost constant attenuation of the boron filter to neutrons above 1 MeV, while those gamma rays that arise as a result of resonance capture of neutrons in the energy region ~ 10 keV - 1 MeV should be enhanced by a factor of $\sim 1/0.45 = 2.2$ (see Fig. 2). Gamma rays from resonance capture of neutrons at neutron energies less than about 10 keV are probably too weak to be observed through the boron-filtered beam because of the rapidly increasing attenuation of the boron filter to neutrons of these energies. However, any gamma rays arising from these resonance captures at energies less than about 10 keV and from the $1/V$ tail of the thermal spectrum as well as capture at thermal energies from slowing down should appear in the response from the cadmium-filtered beam. Thus the experimental net responses corrected for aluminum contribution and incident thermal-neutron contributions from the cadmium-filtered beam should exceed everywhere the response obtained by multiplying the unfolded intensities from the boron-filtered beam by 1.33, adding in the spectral intensities from thermal-neutron capture normalized to the sum of the $1/V$ captures occurring in the slab from ~ 0.5 eV to 10 keV and the thermal captures from slowing down, and folding. Any significant deviation of this minimum predicted response from the experimental response is interpretable in terms of gamma rays arising from resonance capture, either by failure to recognize a gamma ray in the unfolded boron-filtered response as arising from capture from neutrons in the 10-keV - 1-MeV energy range, in which case the intensity should be multiplied by ~ 2.2 , or in the energy region below 10 keV, in which case an unknown spectrum is contributing to the response.

In practice, the calculation of the cadmium-filtered response was refined only to the point where the procedure was to pick out those few gamma-ray energies above ~ 6 MeV in the unfolded spectra which corresponded to those of strong intensity in the thermal-neutron capture spectra and which were obviously strong in the boron-filtered spectra as well, and multiply their intensities by 2.2, multiply all the other intensities by 1.33, calculate and add in the spectral intensities arising from $1/V$ captures above thermal energies and captures at thermal energies from slowing down, and fold. From a comparison of this response with the experimental response tentative conclusions can usually be reached regarding the order of magnitude of resonance capture gamma rays in the region 10 keV - 200 keV, as well as inferring reasonably accurate production cross sections for secondary gamma rays from fast-neutron interactions in the energy range above 1 MeV. The upper limit of 200 keV rather than ~ 1 MeV is used in the range of resonance captures because it represents a conservative upper limit to the error in the energy of the gamma rays assigned to resonance capture. Thus an intense primary gamma ray appearing at energy E_Y arising from thermal-neutron capture will appear at energy $(E_Y + E_n)$, if at all, when arising from resonance capture at neutron energy E_n . If E_n is greater than about 0.2 MeV, this shift would be easily recognizable.

Fifth and last, the reduction of the unfolded gamma-ray intensities to production cross sections is strictly true only if the gamma rays are isotropic. Since in general they are not, it should be noted that the cross sections that are presented in this report are values of $4\pi \times (d\bar{\sigma}/d\Omega)$ (90 deg). Deviations of this quantity from the true cross section, $4\pi \int_0^1 \frac{d\bar{\sigma}}{d\Omega}(\mu) d\mu$, typically run about 30%, with the true value being the larger. The production cross sections from fast-neutron interactions presented in this report are average cross sections for any spectrum of the same shape as that part of the spectrum above 1 MeV shown in Fig. 4, unless otherwise stated. A partial neutron energy discrimination could be afforded by recognizing the fact that the most probable fast-neutron interaction producing secondary gamma rays, that of inelastic scattering, is energetically possible only for neutrons of energy greater than $[(A+1)/A]E_Y$, where A is the atomic weight of the nucleus and E_Y the energy of the gamma ray.

Often, however, the gamma ray occurs as part of a cascade, and the spectrum of neutrons responsible for its production lies considerably above $[(A+1)/A]E_\gamma$; in fact, the threshold would be $[(A+1)/A]\Gamma$, where Γ is the energy of the lowest lying nuclear level that must be excited to produce E_γ in cascade. Thus, except for cases where the decay scheme involving E_γ is well known, little is gained by dividing by the spectrum of neutrons above threshold rather than 1 MeV. Furthermore, to be consistent the discrete gamma rays would have to be easily separated from the continuum (which are also included as part of the unfolded intensities), because a different range of the neutron spectrum is responsible for the production of the one than the other. For those cases where both the energy threshold for the production of a given gamma ray is well known and the unfolded intensity of the gamma ray is unambiguous (the 1.78-MeV gamma ray in silicon for example) interested readers of this report can multiply the tabulated cross section by the ratio of neutrons above 1.0 MeV to neutrons above the threshold from Fig. 5 to arrive at an average cross section above the threshold for neutron spectra of the same shape as that in Fig. 4.

Oxygen. The general procedure outlined in the preceding presentation for the analysis of the responses from the boron filtered and cadmium filtered beams was altered for the special cases of water and polyethylene, since only one or two gamma rays are of interest.

The foreground runs for 1 in. of borated water were 551-B for the cadmium-filtered beam and 551-C and 552-A for the boron-filtered beam. They were then normalized to 100 Kw and 200 min (see appendix). The normalization to 100 Kw for these and almost all subsequent runs (200 min was the usual analyzer live time) was provided by multiplying the counts in each channel by the ratio of 100 Kw to the actual power during the run. The actual power was in turn calculated by multiplying the nominal operating power by the ratio of gold foil activation counting rates obtained during the run to a standard calculated counting rate. The cadmium-filtered beam counts for all runs were in addition multiplied by the factor 1.014, which is the reciprocal of the average fast-neutron attenuation of the cadmium filter. The calibration $E_c = G_c(C-2)$ was in general used to convert the net responses from channel number to energy in MeV, where for

the borated water G_c varied between 0.0224 MeV/channel at low channels to 0.0228 MeV/channel at high channels for the boron-filtered beam, and between 0.0225 and 0.0229 for the cadmium-filtered beam. The function G_c was determined from recognition of the energies of several of the peaks in the runs. Each net response in counts/channel/100 Kw/200 min was then divided by the local value of G_c and plotted as a function of E . These responses appear as the broken lines in Fig. 11.

The contribution of the aluminum container to each of these curves from gamma rays arising from non-thermal neutron interactions was calculated by scaling the response from a 0.25-in.-thick aluminum slab corrected for captures from incident subcadmium neutrons (see the discussion of aluminum presented further along in this report) by a factor of 0.417. This factor was the ratio obtained from ANISN calculations of the spatially integrated fast-neutron fluxes above 1 MeV in the aluminum container to those in the aluminum slab corrected for the difference in gamma-ray attenuation caused by the presence of the water. (Scaling by the ratio of the total thickness of the aluminum in the two cases and correcting for the photon attenuation of the water would produce the factor 0.49. The additional decrease is caused by the reduction in the fast flux incident on the rear half of the container provided by the water, which more than compensates for the slight increase in the fast flux scattered by the water into the front half of the container.) Subtraction of this non-thermal contribution from the aluminum container results in the solid curves appearing in Fig. 11, where the contribution from thermal-neutron capture of both incident and slowed-down neutrons is apparent in the total absorption peak at 7.72 MeV. The fitted responses to each of these curves is indicated by the circles in Fig. 11, and the spectral intensities obtained from the unfolding of each of these responses are shown in Table IV.

The only gamma rays of interest to us in this report are the three coming from inelastic scattering in oxygen at 6.13 MeV, 6.92 MeV, and 7.12 MeV. These intensities cannot be uniquely extracted from the results presented in Table IV because of the presence of neighboring gamma rays used in the unfolding. However, the intensity for the 6.13-MeV gamma ray and the sum of the intensities for the 6.92-MeV and 7.12-MeV gamma rays almost certainly lie between the limits appearing in Table V. The

Fig. 11. Measured and Fitted Net Responses from 1 inch of Borated Water in an Aluminum Container for the Energy Region 5.3-8.2 MeV.

Table IV. Spectral Intensities Obtained from Unfolding of the Water Responses

Boron-Filtered Beam		Cadmium-Filtered Beam	
E_{γ} (MeV)	I_{γ} (photons/100 Kw/200 min)	E_{γ} (MeV)	I_{γ} (photons/100 Kw/200 min)
8.2	1.4(4) ^a	8.2	3.0(4)
8.1	1.0(4)	8.1	2.5(4)
8.0	7.0(3)	8.0	1.0(4)
7.9	1.7(4)	7.9	2.5(4)
7.8	4.0(4)	7.8	9.0(4)
7.7	2.0(4)	7.7	1.8(5)
7.6	3.1(4)	7.6	6.0(4)
7.5	4.0(3)	7.4	4.5(4)
7.4	3.0(4)	7.2	1.1(5)
7.3	2.1(4)	7.12	1.2(5)
7.2	6.5(4)	7.0	1.3(5)
7.12	1.0(5)	6.9	6.5(4)
7.0	9.0(4)	6.8	1.5(4)
6.9	4.0(4)	6.7	3.5(4)
6.7	1.2(4)	6.5	1.5(4)
6.6	3.5(4)	6.4	4.0(4)
6.5	4.0(4)	6.3	1.0(5)
6.3	1.8(5)	6.25	8.0(4)
6.13	5.2(5)	6.13	8.0(5)
6.0	4.5(4)	6.0	8.0(4)
5.95	1.1(5)	5.95	1.0(5)
5.7	5.0(4)	5.7	7.5(4)
5.5	4.0(4)	5.5	9.0(4)

^aRead as 1.4×10^4 .

Table V. Gamma-Ray Intensities and Average Production Cross Sections Above Threshold for Oxygen

Boron-Filtered Beam		Cadmium-Filtered Beam	
ϕ_γ (photons/100 Kw/200 min)	$\bar{\sigma}_{nn'\gamma}$ (mb)	ϕ_γ (photons/100 Kw/200 min)	$\bar{\sigma}_{nn'\gamma}$ (mb)
<u>6.13 MeV</u>			
$(6.5 \pm 1.5)(5)^a$	85 ± 30	$(9.5 \pm 1.5)(5)$	90 ± 30
<u>6.92 MeV + 7.12 MeV</u>			
$(2.4 \pm 0.6)(5)$	55 ± 20	$(3.3 \pm 0.8)(5)$	55 ± 20

^aRead as $(6.5 \pm 1.5) \times 10^5$.

production cross sections for these gamma rays, $4\pi \frac{d\bar{\sigma}}{d\Omega}$ (90 deg), averaged over that part of the neutron spectrum above threshold that can give rise to them also appears in Table V. These cross sections were calculated by using ANISN results for the group fluxes above threshold integrated over the water thickness for the incident beams shown in Figs. 7 and 8. Thus for the 6.13-MeV gamma ray,

$$\begin{aligned} \phi_\gamma(6.13) \frac{\text{photons}}{200 \text{ min } 100 \text{ Kw}} &= \bar{\sigma}_{nn'\gamma} \frac{x 10^{-27}}{4\pi} \frac{\text{cm}^2 \text{ photons}}{\text{atom steradian neutron}} \\ &\times \frac{0.6023 \times 10^{24}}{18} \frac{\text{atoms}}{\text{gram}} \times 1.0 \frac{\text{grams}}{\text{cm}^3} \times \frac{(\pi/4) \times 25}{144 \times 400} \text{ steradians} \times A_\gamma(6.13) \\ &\times \int_0^{2.54} dx \int_{6.5}^{\infty} \phi(E, x) dE \frac{\text{neutrons cm}}{200 \text{ min } 100 \text{ Kw}}, \end{aligned}$$

where $\bar{\sigma}_{nn'\gamma}$ is the average production cross section in millibarns per neutron above threshold of spectral distribution given by Figs. 7 or 8, $A_\gamma(6.13)$ is the average attenuation of 6.13 MeV photons through the slab, lithium hydride shield, air, and polyethylene = 0.597 for either beam, and the double integral is the total flux of neutrons above threshold integrated over the thickness of the water calculated by ANISN due to the incident surface integrated currents of Figs. 7 or 8:

$$\int_0^{2.54} dx \int_{6.5}^{\infty} \varphi(E, x) dE = \begin{cases} 1.45 \times 10^{13} & \text{for the boron-filtered beam,} \\ 1.93 \times 10^{13} & \text{for the cadmium-filtered beam.} \end{cases}$$

$$\text{Thus, } \bar{\sigma}_{nn'\gamma}(6.13) = 1.85 \times 10^9 \frac{\int_0^{2.54} dx \int_{6.5}^{\infty} \varphi(E, x) dE}{\int_0^{2.54} dx \int_{6.5}^{\infty} \varphi(E, x) dE} \text{ mb.}$$

For the 6.92 MeV + 7.12 MeV gamma rays

$$A_{\gamma}(7.0) = 0.618 ,$$

and

$$\int_0^{2.54} dx \int_{7.4}^{\infty} \varphi(E, x) dE = \begin{cases} 7.93 \times 10^{12} & \text{for the boron-filtered beam} \\ 1.05 \times 10^{13} & \text{for the cadmium-filtered beam,} \end{cases}$$

$$\text{leading to } \bar{\sigma}_{nn'\gamma}(6.92 + 7.12) = 1.79 \times 10^9 \times \frac{\int_0^{2.54} dx \int_{7.4}^{\infty} \varphi(E, x) dE}{\int_0^{2.54} dx \int_{7.4}^{\infty} \varphi(E, x) dE} \text{ mb.}$$

The agreement between the boron-filtered beam results and the cadmium-filtered beam results is quite good, and the factor of ~ 1.33 in the intensities between the two runs is a further indication that the two runs produce consistent results. These results are compared in Table VI with the earlier work of Hall and Bonner¹³ and the recent work of Dickens and Perey,¹⁴ where 4π times the cross section at 90 deg is averaged over the neutron spectrum above threshold appearing in Fig. 4. It can be seen that the present results tend to agree better with the recent work of Perey for the 6.13-MeV gamma ray, although they lie within the limits of error with the results of Hall and Bonner as well. Adequate agreement among all three experiments exists for the sum of the 6.92-MeV and 7.12-MeV gamma rays. Perey also investigated the anisotropy of the 6.13-MeV gamma ray and found $\frac{d\sigma}{d\Omega}(90 \text{ deg}) / \frac{d\sigma}{d\Omega}(55 \text{ deg}) \approx 0.7$, although this ratio varied somewhat as a function of neutron energy. Values of $4\pi (d\sigma/d\Omega)(55 \text{ deg})$ averaged over the incident neutron spectrum are thus $\sim 40\%$ higher than the values presented in Table VI for the 6.13-MeV gamma ray, and these values should closely approximate the actual values of $\bar{\sigma}_{nn'\gamma}$.

Table VI. Comparison of Present Results with Spectrum-Averaged Results of Others for Oxygen

	$4\pi \frac{d\sigma}{d\Omega}$ (90 deg, 6.13 MeV)	$4\pi \frac{d\sigma}{d\Omega}$ (90 deg, 6.92 + 7.12 MeV)
Present Results	90 ± 30 mb	55 ± 20 mb
Hall and Bonner	140 ± 40 mb	45 ± 15 mb
Dickens and Perey	~ 70 mb	~ 50 mb

Carbon. The foreground runs for 1 in. of polyethylene were 365-A for the boron-filtered beam and 364-A for the cadmium-filtered beam. They were then normalized to 100 Kw and 200 min (see appendix). The net responses for the 4.43-MeV gamma ray were found by subtraction of the backgrounds and then corrected for relatively small but noticeable gamma rays at 4.94 MeV and 3.68 MeV arising from thermal-neutron capture in ^{12}C . The conversion from channel number to energy, $E_c = G_c(C-2)$, used a value of $G_c = 0.0223$ MeV/channel for both runs. The measured net response from the boron-filtered beam is shown as the solid curve in Fig. 12. (The net response from the cadmium-filtered beam was not analyzed, because it offered no additional information). The fitted response is shown in Fig. 12 as the open circles, using the intensities appearing in Table VII.

Table VII. Spectral Intensities Obtained from Unfolding of the Polyethylene Response Using the Boron-Filtered Beam

E_γ (MeV)	γ (photons/100 Kw/200 min)
4.42	$3.0(6)^a$
4.22	$1.4(5)$
4.12	$6.0(4)$
4.05	$5.0(4)$
3.95	$5.0(4)$

^aRead as 3.0×10^6 .

Using the value $A_\gamma(4.43) = (3.1 \pm 0.1) \times 10^6$ obtained from Table VII, the average cross section for production of this gamma ray, $4\pi (d\sigma/d\Omega)$ (90 deg), may be calculated:

Fig. 12. Measured and Fitted Net Responses from 1 inch of Polyethylene for the Energy Region 3.7-4.6 MeV for the Boron-Filtered Beam.

$$\phi_{\gamma}(4.43) = \frac{\bar{\sigma}_{nn,\gamma} \times 10^{-27}}{4\pi} \times \frac{0.6023 \times 10^{24}}{14} \times 0.92 \times \frac{(\pi/4) \times 25}{144 \times 400} \\ \times A_{\gamma}(4.43) \times \int_0^{2.54} dx \int_{4.8}^{\infty} \varphi(E,x) dE ,$$

where $A_{\gamma}(4.43) = 0.551$, and the double integral again represents the total flux of neutrons above threshold calculated by ANISN integrated over the thickness of the polyethylene due to the incident surface-integrated current of Fig. 4:

$$\int_0^{2.54} dx \int_{4.8}^{\infty} \varphi(E,x) = 4.05 \times 10^{13} .$$

Thus, $\bar{\sigma}_{nn,\gamma} = 130 \pm 20$ mb, where the error also includes estimated uncertainties in the absolute response function and incident neutron spectrum.

Comparisons of this value with the values obtained by averaging the results of Hall and Bonner¹³ and Dickens and Perey at 90 deg¹⁴ over the incident neutron spectrum of Fig. 8 are shown in Table VIII.

Table VIII. Comparison of Present Results with Spectrum Averaged Results of Others for Carbon

	$4\pi \frac{d\bar{\sigma}}{d\Omega} (90 \text{ deg})$
Present Results	130 ± 20 mb
Hall and Bonner	190 ± 60 mb
Dickens and Perey	~ 110 mb

Perey also investigated the anisotropy of this gamma ray, and tentative results indicate that the ratio $(d\sigma/d\Omega) (90 \text{ deg}) / (d\sigma/d\Omega) (55 \text{ deg})$ has the approximate value of 0.75. Thus the actual value of the average production cross section of this gamma ray is about 35% higher than the values indicated in Table VIII, and these values agree pretty well with the total inelastic scattering cross-section values also determined by Perey.

As with the oxygen results, the present measurement agrees better with that of Perey than with that of Hall and Bonner, and the tentative conclusion is reached that the values of $(d\sigma/d\Omega)$ (90 deg) of Hall and Bonner are about a factor of two high. However, as with the case of the 6.13-MeV gamma ray in oxygen, if the results of Hall and Bonner are interpreted as total production cross sections by multiplication by 4π , the resulting values are probably high by only about 25%.

Iron. The foreground runs using 0.075 in. of iron were 322-E for the boron-filtered beam and 366-A and 365-B for the cadmium-filtered beam. The former was normalized to 100 Kw and 200 min and the latter, through an oversight, to 200 Kw and 200 min (see appendix). After subtraction of background and conversion from channel number to energy, the net responses in units of counts/MeV/100 Kw/200 min are plotted in Figs. 13 and 14. The relation $E_c = G_c (C-2)$ was used, where $0.0216 \leq G_c \leq 0.0223$ MeV/channel for the boron-filtered beam and $0.217 \leq G_c \leq 0.0230$ MeV/channel for the cadmium-filtered beam. (The highest gain is for channel numbers in the vicinity of channel 400, and the lowest gain is for channel numbers in the vicinity of channel 40.)

The contribution to the response from capture of incident thermal neutrons was calculated for each beam filter and subtracted from the net response, and the result of these subtractions also appears in Figs. 13 and 14. This contribution was found by folding the gamma-ray spectrum arising from the capture of thermal neutrons, $I_\gamma(E_\gamma)$, obtained from iron in ref. 6, normalized to the number of captures occurring in the slab. Thus, for the boron-filtered beam,

$$\begin{aligned}
 \mathcal{I}_\gamma(E_\gamma) \text{ photons/100 Kw/200 min} &\approx 3.8 \times 10^{11} \frac{\text{incident thermal neutrons}}{100 \text{ Kw } 200 \text{ min}} \times 0.053 \\
 &\times \frac{\text{captures}}{\text{incident thermal neutron}} \times \frac{I_\gamma(E_\gamma)}{4\pi} \frac{\text{photons/ster}}{\text{capture}} \times \frac{(\pi/4) \times 25}{144 \times 400} \\
 &\times \text{ster} \times A_\gamma(E_\gamma) , \\
 &= 5.5 \times 10^5 I_\gamma(E_\gamma) A_\gamma(E_\gamma) ,
 \end{aligned}$$

Fig. 13. Measured and Calculated Net Responses from 0.075 inches of Iron for the Energy Region 1.0-3.8 MeV.

Fig. 14. Measured and Calculated Net Responses from 0.075 inches of Iron for the Energy Region 3.7-9.5 MeV.

where $A_Y(E_Y)$ is the average attenuation function for photons through the iron at 45 deg, and through the polyethylene, air, and lithium-hydride shield at normal incidence and which is plotted in Fig. 15. The value 3.8×10^{11} is extracted from Section II of this report, and the factor 0.053 is taken from Section IV of ref. 6.

For the cadmium-filtered beam,

$$\Delta_Y(E_Y) = 1.15 \times 10^6 I_Y(E_Y) A_Y(E_Y) \text{ photons/100 Kw/200 min.}$$

A glance at Fig. 14 shows that for iron these captures account for in the vicinity of one-third to one-half of the response for the boron-filtered beam above ~ 5.5 MeV. A check of the calculated response from the capture of these incident thermal neutrons is afforded by observing the net response from the iron to the boron-filtered beam when a cone shaped shadow shield is inserted between the slab and the reactor collimator. This geometry was the same as described in Section II to isolate the source of thermal neutrons incident on the slab using the boron-filtered beam. The runs were 548-A with the iron slab in place, and 548-B with the slab removed, both for 100 Kw nominal power and 200 min (see appendix). The net response from these two runs essentially agreed with that calculated above, verifying that for all future runs the response from capture of the incident thermal neutrons could be adequately calculated without recourse to a measurement.

The corrected response from the boron-filtered beam was unfolded using the fitted response depicted by the open circles in the lower curve of Figs. 13 and 14, and the unfolded intensities are presented in Table IX. Also appearing in Table IX are the gamma-ray production cross sections averaged over the incident spectrum above 1 MeV presented in Fig. 4. These cross sections were calculated from the unfolded intensities in the following manner:

$$\Delta_Y(E_Y) \text{ photons/100 Kw/200 min} = \frac{\bar{\sigma}(E_Y) \times 10^{-3}}{4\pi} \times \frac{0.6023}{55.85} \times 7.83 \times \frac{(\pi/4) \times 25}{400 \times 144} \\ \times \int_0^T dx \int_{1 \text{ MeV}}^{\infty} \phi(E, x) dE \times A_Y(E_Y) ,$$

Fig. 15. Attenuation Function for Secondary Gamma Rays from Iron.

Table IX. Unfolded Intensities and Average Production
Cross Sections for Iron

E_γ (MeV)	γ (photons/ 100 Kw/200 min)	$\bar{\sigma}(E_\gamma)$ mb	E_γ (MeV)	γ (photons/ 100 Kw/200 min)	$\bar{\sigma}(E_\gamma)$ mb
9.50	1.3(4) ^a	capture	6.20	7.5(3)	0.22
9.28	2.5(4)	capture	6.15	1.3(4)	0.38
9.00	2.2(3)	0.05	6.10	1.8(4)	0.52
8.62	6.0(3)	0.17	6.02	3.8(4)	capture
8.55	3.5(3)	0.10	5.92	3.8(4)	capture
8.50	3.5(3)	0.10	5.80	1.4(4)	0.42
8.45	2.5(3)	0.07	5.75	1.0(4)	0.30
8.40	3.0(3)	0.09	5.70	8.0(3)	0.24
8.35	2.5(3)	0.07	5.60	3.0(4)	0.90
8.30	3.0(3)	0.08	5.55	6.0(3)	0.18
8.25	3.0(3)	0.09	5.50	1.3(4)	0.39
8.20	5.0(3)	0.14	5.40	9.0(3)	0.27
8.06	1.6(4)	0.44	5.35	2.0(4)	0.60
7.85	5.0(3)	0.14	5.30	2.5(4)	0.76
7.80	1.0(4)	0.27	5.25	1.6(4)	0.48
7.64	1.3(5)	capture	5.20	1.0(4)	0.31
7.45	5.0(3)	0.14	5.15	2.0(4)	0.62
7.40	2.4(4)	0.68	5.10	2.3(4)	0.71
7.30	4.0(3)	0.11	4.95	9.0(3)	0.28
7.06	2.0(4)	0.57	4.90	5.7(4)	1.8
6.88	7.0(3)	0.20	4.80	2.5(3)	0.08
6.74	5.1(4)	capture	4.70	5.15(4)	1.6
6.62	4.5(3)	0.13	4.60	1.0(4)	0.32
6.55	3.0(3)	0.09	4.50	6.0(4)	3.2
6.50	4.0(3)	0.12	4.40	5.5(4)	1.8
6.45	1.0(4)	0.28	4.35	3.8(3)	0.1
6.40	1.0(4)	0.30	4.30	4.6(4)	1.5
6.35	7.5(3)	0.22	4.20	2.5(4)	0.8
6.30	7.0(3)	0.21	4.15	7.0(3)	0.2

Table IX (cont.)

E_γ (MeV)	$\mathcal{L}_\gamma$ (photons/ 100 Kw/200 min)	$\bar{\sigma}(E_\gamma)$ mb	E_γ (MeV)	$\mathcal{L}_\gamma$ (photons/ 100 Kw/200 min)	$\bar{\sigma}(E_\gamma)$ mb
4.05	1.3(5)	4.3	2.27	4.5(5)	19
3.90	1.25(5)	4.2	2.20	1.8(5)	8
3.80	8.5(4)	2.9	2.12	1.1(6)	47
3.72	9.0(4)	3.1	2.02	2.7(5)	12
3.64	4.2(5)	14.5	1.90	3.7(5)	17
3.54	2.5(5)	8.7	1.82	1.2(6)	56
3.48	2.4(5)	8.4	1.72	2.0(5)	10
3.40	1.5(5)	5.3	1.67	1.4(5)	7
3.30	1.6(5)	5.7	1.61	1.1(5)	5
3.21	2.5(5)	9.0	1.56	1.1(5)	6
3.10	1.25(5)	4.5	1.50	2.0(5)	10
3.05	5.0(4)	1.9	1.45	2.0(5)	10
3.00	2.45(5)	9.1	1.41	5.2(5)	28
2.90	1.4(5)	5	1.35	3.8(5)	21
2.80	3.5(5)	13	1.30	1.8(5)	10
2.70	3.1(5)	12	1.25	1.6(6)	91
2.61	7.6(5)	30	1.18	5.3(5)	31
2.54	4.0(4)	1	1.12	3.5(5)	21
2.50	3.0(5)	12	1.06	5.1(5)	32
2.40	1.0(5)	4	1.00	5.3(5)	35
2.34	3.0(5)	12	0.845	$\sim 8.2(6)$	~ 600

^aRead as 1.3×10^4 .

where $\bar{\sigma}(E_\gamma)$ is the average production cross section above 1 MeV in millibarns, $A_\gamma(E_\gamma)$ is the average attenuation of the gamma rays between point of birth and the crystal (see Fig. 15), and

$$\int_0^T dx \int_{1 \text{ MeV}}^{\infty} \varphi(E, x) dE \approx \frac{T}{\cos 45^\circ} \int_{1 \text{ MeV}}^{\infty} J(E) dE =$$

$$1.414 \times 2.54 \times 0.075 \times 8.9 \times 10^{13} \frac{\text{neutron cm}}{100 \text{ Kw } 200 \text{ min}},$$

where T is the thickness of the slab. The slab is thin enough that the double integral was not evaluated by ANISN, and the approximation used above should be accurate to better than 5%.

Thus, $\bar{\sigma}(E_\gamma) = 1.82 \times 10^{-5} A_\gamma(E_\gamma)/A_\gamma(E_\gamma)$ millibarns.

Several relatively intense gamma rays were observed at energies corresponding to those of intense lines in the thermal-neutron capture spectrum, and these lines were not included as part of the gamma-ray production from fast-neutron interactions. They are undoubtedly caused by capture of neutrons below ~ 200 keV because the gamma-ray energies correspond very closely to those from thermal-neutron capture. These gamma rays are indicated in Table IX by the word "capture."

Multiplying the intensities of the capture gamma rays by 2.2, the intensities of the remaining gamma rays by 1.33, adding in the gamma-ray intensities arising from $1/V$ capture, and folding, results in the upper curves indicated in Figs. 13 and 14 by the open circles. The gamma-ray intensities present in the cadmium-filtered beam response arising from $1/V$ captures were calculated from the following, using $\sigma_{n\gamma}(E) =$

$$\frac{2.53 \sqrt{0.0253}}{\sqrt{E}} = \frac{0.402}{\sqrt{E}} \text{ barns:}$$

$$\int_0^T dx \int_{0.5 \text{ eV}}^{\infty} \sigma_{n\gamma}(E) \varphi(E, x) dE \approx \frac{T}{\cos 45^\circ} \times 1.05 \int_{0.5 \text{ eV}}^{\infty} \sigma_{n\gamma}(E) J(E, x) dE =$$

$$\frac{1.05 T}{\cos 45^\circ} \int_{0.5 \text{ eV}}^{\infty} \frac{0.402}{\sqrt{E}} \times \frac{4.0 \times 10^{12}}{E} dE$$

$$= 1.05 \times 1.414 \times 2.54 \times 0.075 \times 0.402 \times 4.0 \times 10^{12} \times 2 \times 1.414$$

$$= 1.28 \times 10^{12} \text{ barn cm neutrons/200 min 100 Kw}.$$

The number of $1/V$ captures occurring in the slab is thus 1.28×10^{12} $\times \frac{0.6023}{55.85} \times 7.83 = 1.09 \times 10^{11}$ per 100 Kw per 200 min, and the gamma-ray intensities may be calculated from

$$\begin{aligned} \mathcal{I}_\gamma(E_\gamma) &= \frac{1.09 \times 10^{11}}{4\pi} \times \frac{(\pi/4) \times 25}{144 \times 400} A_\gamma(E_\gamma) I_\gamma(E_\gamma) \\ &= 2.95 \times 10^6 A_\gamma(E_\gamma) I_\gamma(E_\gamma), \end{aligned}$$

where $A_\gamma(E_\gamma)$ is taken from Fig. 15 and $I_\gamma(E_\gamma)$, in photons per capture, is taken from ref. 6.

Comparison of the calculated response with the measured response for the cadmium-filtered beam is quite good, the agreement being within $\pm 10\%$ almost everywhere. One important conclusion to be drawn from this agreement is that gamma rays arising from resonance capture below ~ 10 keV are small compared with those arising from $1/V$ capture. This pertains particularly to the large capture resonance of ~ 7 barns at 1.15 keV, which although large is extremely narrow. Indeed, a rough calculation of the resonance integral using the data in BNL-325¹ yields a value of about 5% of the $1/V$ capture integral from 0.5 eV - 1 keV. The other important conclusion to be drawn from the agreement is that the division of gamma rays produced from captures below 200 keV and from fast-neutron interactions above 1 MeV is at least approximately correct.

A comparison of the average production cross sections summed over 0.5 MeV gamma-ray intervals with the data from ref. 5 averaged over the neutron spectrum above 1 MeV in Fig. 4 is presented in Table X.

Remembering that the data from ref. 5 is presumably only inelastic data, and the present results are values of $4\pi (d\bar{\sigma}/d\Omega)$ (90 deg), it is quite remarkable that the results agree so well -- in fact, to within $\pm 30\%$ for all 1-MeV wide intervals below 7 MeV. The conclusions can of course be made that the gamma-ray production data of ref. 5 for gamma rays above 1 MeV is adequate for shielding calculations.¹⁵

Data also exist in ref. 1 for the production cross sections of a few of the more intense lines arising from inelastic scattering as a

Table X. Comparison of Average Production Cross Sections in Millibarns
from Present Results with Data of Celnick and Spielberg for Iron

	Gamma-Ray Energy Range, MeV													Total
	1-1.5	1.5-2	2-2.5	2.5-3	3-3.5	3.5-4	4-4.5	4.5-5	5-5.5	5.5-6	6-6.5	6.5-7	7-7.5	
Present Results	278	110	101	74	44	33	8.7	5.6	3.8	2.4	< 2.1	< 0.6	< 1.5	≤ 665
Celnick and Spielberg ⁵	307		198		78		14		8.5		2.0		0.4	608

function of neutron energy. A comparison of these cross sections averaged over the incident neutron spectrum between 1.0 MeV and E_u , where E_u is the highest energy for which the data are presented in ref. 1, with the present results averaged over the entire range above 1.0 MeV is shown in Table XI. (The isotopic data in ref. 1 were corrected for natural iron abundances.) Considering effects of gamma-ray anisotropy and the fact that the averaging is over a different range of neutron energies, it is concluded that the agreement is adequate, and the data appearing in ref. 1 are consistent with present results.

An inspection of Table IX shows the presence of many lines superimposed on a continuum. The former are recognizable by their relatively strong intensities, such as those in the vicinity of 2.27-, 3.00-, 3.21-, 3.64-, 4.50-, 4.70-, and 4.90-MeV in addition to those appearing in Table XI. The continuum is approximated by the lines of relatively constant intensity that are evenly spaced less than a resolution width of the spectrometer apart, and which arise from fast-neutron interactions at neutron energies above ~ 5 MeV.

A rough but informative check of the capture cross section for iron averaged over the neutron spectrum from ~ 10 keV to 200 keV is afforded by the lines marked "capture" in Table IX. Thus, for intermediate energy capture,

$$\begin{aligned} \phi_Y(E_Y) \text{ photons/200 min/100 Kw} &= A_Y(E_Y) \times \frac{\bar{\sigma}_{n\gamma} \times 10^{-3}}{4\pi} \times \frac{0.6023}{55.85} \times 7.83 \\ &\times \frac{(\pi/4) \times 25}{144 \times 400} \times \int_0^T dx \int_{10 \text{ keV}}^{200 \text{ keV}} \phi(E,x) dE, \end{aligned}$$

where the double integral assumes the approximate value of

$$\frac{1.05 T}{\cos 45^\circ} \int_{10 \text{ keV}}^{200 \text{ keV}} J(E) dE,$$

which with the help of Fig. 8 may be evaluated to be

$$1.05 \times 2.54 \times 0.075 \times 1.414 \times 1.50 \times 10^{13} = 4.25 \times 10^{12} \frac{\text{neutron cm}}{100 \text{ Kw } 200 \text{ min}}.$$

Table XI. Comparison of Average Production Cross Sections in Millibarns from Present Results with BNL Data for Some of the Stronger Iron Gamma Rays

Gamma-Ray Energy	0.845 MeV	1.25 MeV	1.41 MeV	1.82 MeV	2.12 MeV	2.60 MeV
Isotope	^{56}Fe	^{56}Fe	^{54}Fe	^{56}Fe	^{56}Fe	^{56}Fe
Present Results	~ 600	91	28	56	47	30
BNL-325 ¹	~ 800	65	40	45	30	20
E_u , MeV	5.2	5.2	4.0	5.0	5.0	5.0

Thus,

$$\bar{\sigma}_{n\gamma}(E_{\gamma}) = 1.0 \times 10^{-4} \phi_{\gamma}(E_{\gamma})/A_{\gamma}(E_{\gamma}) \text{ millibarns,}$$

and the six capture lines have the cross sections appearing in Table XII. Since these gamma rays probably account for most of the binding energy available, which is in the vicinity of 8 MeV, an approximate value of the average total capture cross section is obtained by evaluating the expres-

sion $\sum_{i=1}^6 E_{\gamma_i} \bar{\sigma}_{n\gamma}(E_{\gamma_i})/8$. The number obtained, 41 mb, is in good agree-

ment with the reasonable assumption that none of these six gamma rays are

in cascade, since $\sum_{i=1}^6 \bar{\sigma}_{n\gamma}(E_{\gamma_i}) = 46 \text{ mb}$. Thus the average capture cross

section for iron in the region ~ 10 -200 keV for the incident spectrum of Fig. 8 is $\approx 40 \text{ mb}$, and this is in fair agreement with a value of $\sim 20 \text{ mb}$ evaluated from data in ref. 1. Very approximate values of $\bar{I}_{\gamma}(E_{\gamma})$ in photons/100 captures averaged over the intermediate neutron spectrum may be obtained by dividing the $\bar{\sigma}_{n\gamma}(E_{\gamma})$ values by 40 mb and multiplying by 100. The resulting values of $\bar{I}_{\gamma}(E_{\gamma})$ that appear in Table XII are compared with the values from the capture of thermal neutrons,⁶ and it is to be noted that the two spectra differ primarily in the magnitude of the ground state transition gamma ray in ^{56}Fe (9.3 MeV) and the magnitude of the 6.74-MeV gamma ray.

Aluminum. The foreground runs for 0.25 in. of aluminum were 307-B for the boron-filtered beam and 550-A for the cadmium-filtered beam, both normalized to 100 Kw and 200 min (see appendix). The counts in run 307-B, however, should be divided by 1.075 since the run was normalized to 100 Kw nominal power. (This is the only run that requires the use of such a factor.) The expression $E_c = G_c(C-2)$ was again used to convert pulse height to energy, where $0.0208 \leq G_c \leq 0.0218 \text{ MeV/channel}$ for the run using the boron-filtered beam and $0.0220 \leq G_c \leq 0.0226 \text{ MeV/channel}$ for the run using the cadmium-filtered beam. The net responses in units of counts/MeV/100 Kw/200 min were formed and appear in Figs. 16 and 17 as the solid lines.

Table XII. Capture Cross Sections for Iron Averaged Over the
Neutron Spectrum from 10-200 keV

E_{γ} (MeV)	$\bar{\sigma}_{n\gamma}(E_{\gamma})$ (Millibarns)	$\bar{\sigma}_{n\gamma}(E_{\gamma}) \times E_{\gamma}$ (MeV·Millibarns)	$\bar{I}_{\gamma}(E_{\gamma})$ (photons/100 captures)	$I_{\gamma}(E_{\gamma})$ (photons/100 thermal-neutron captures)
9.50	2	20	4	
9.28	4	35	9	3.3
7.64	20	150	45	50.6
6.74	8	55	18	
6.02	6	35	14	9.2
5.92	<u>6</u>	<u>35</u>	14	9.0
Total	46	330

Fig. 16. Measured and Calculated Net Responses from 0.25 inches of Aluminum for the Energy Region 1.0-3.3 MeV.

Fig. 17. Measured and Calculated Net Responses from 0.25 inches of Aluminum for the Energy Region 3.3-8.1 MeV.

The contribution to the net response for each beam filter from capture of the incident thermal neutrons was calculated in a manner completely analogous to that described for iron. For the boron-filtered beam the intensities

$$\mathcal{A}_Y(E_Y) = 1.0 \times 10^5 I_Y(E_Y) A_Y(E_Y) \text{ photons/100 Kw/200 min} ,$$

where $I_Y(E_Y)$, in photons/capture, was taken from ref. 6 and $A_Y(E_Y)$ appears in Fig. 18, were folded and the resulting response subtracted from the net response. The result of this subtraction appears as a smooth curve through the open diamonds appearing in Figs. 16 and 17. For the cadmium-filtered beam, intensities were calculated from the expression

$$\mathcal{A}_Y(E_Y) = 2.0 \times 10^5 I_Y(E_Y) A_Y(E_Y) \text{ photons/100 Kw/200 min},$$

then folded, and the resulting response subtracted from the net response. As can be seen from Figs. 16 and 17, the contribution from these captures is small for both beams except in the region above ~ 7 MeV.

The incident thermal-neutron-capture-corrected response from the boron filter was unfolded using the fitted response depicted by the open circles in the lower curve of Figs. 16 and 17, and the unfolded intensities are presented in Table XIII. Also appearing in Table XIII are the gamma-ray production cross sections averaged over the incident spectrum above 1 MeV presented in Fig. 4. These cross sections were calculated from the unfolded intensities in the following manner:

$$\mathcal{A}_Y(E_Y) \text{ photons/100 Kw/200 min} = \frac{\bar{\sigma}(E_Y) \times 10^{-3}}{4\pi} \times \frac{0.6023}{27} \times 2.7 \times \frac{(\pi/4) \times 25}{144 \times 400} \\ \times \int_0^T dx \int_{1 \text{ MeV}}^{\infty} \varphi(E, x) dE \times A_Y(E_Y) ,$$

where $\bar{\sigma}(E_Y)$ is the average production cross section to neutrons above 1 MeV in millibarns, $A_Y(E_Y)$ is taken from Fig. 18, and

Table XIII. Unfolded Intensities and Average Production
Cross Sections for Aluminum

E_γ (MeV)	$\mathcal{L}_\gamma$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb	E_γ (MeV)	$\mathcal{L}_\gamma$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb
7.95	1.0(4) ^a	0.10	4.30	7.0(4)	0.90
7.90	2.2(4)	0.25	4.26	7.0(3)	0.10
7.85	1.0(4)	0.10	4.20	1.0(5)	1.3
7.74	4.0(4)	capture	4.10	7.0(4)	0.95
7.65	5.0(3)	0.06	3.97	2.6(5)	3.5
7.20	1.0(4)	0.10	3.90	4.0(4)	0.55
7.10	1.0(4)	0.10	3.80	1.3(5)	1.8
7.00	1.1(4)	0.15	3.70	6.0(4)	0.85
6.80	3.7(4)	0.40	3.60	4.0(4)	0.65
6.70	1.8(4)	0.20	3.54	3.0(4)	0.40
6.60	4.8(4)	0.55	3.50	8.0(4)	1.1
6.50	2.4(4)	0.30	3.43	1.1(5)	1.6
6.40	4.4(4)	0.50	3.36	6.0(4)	0.9
6.20	4.8(4)	0.55	3.30	8.5(4)	1.2
6.10	1.0(4)	0.10	3.26	8.5(4)	1.2
6.00	4.2(4)	0.50	3.20	2.3(5)	3.4
5.80	6.7(4)	0.80	3.10	1.9(5)	2.8
5.73	8.0(3)	0.10	3.02	2.1(6)	31.3
5.60	9.5(4)	1.15	2.95	3.0(5)	4.6
5.46	1.35(5)	1.65	2.90	2.2(5)	3.3
5.35	5.2(4)	0.60	2.80	2.5(5)	3.9
5.20	9.5(4)	1.2	2.72	4.5(5)	7.1
5.10	6.0(4)	0.75	2.60	2.2(5)	3.5
5.00	5.6(4)	0.70	2.51	2.4(5)	3.9
4.90	2.7(4)	0.35	2.40	3.2(5)	5.3
4.80	1.3(5)	1.65	2.33	3.0(5)	5.1
4.61	2.7(5)	3.45	2.30	7.5(5)	12.7
4.50	1.1(5)	1.4	2.23	4.1(6)	70.4
4.46	8.0(3)	0.10	2.14	1.3(5)	2.3
4.40	1.85(5)	2.4	2.10	2.35(5)	4.0

Table XIII (cont.)

E_γ (MeV)	$\mathcal{N}_\gamma$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb	E_γ (MeV)	$\mathcal{N}_\gamma$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb
2.00	1.5(5)	2.7	1.45	1.0(5)	2.1
1.94	6.0(4)	1.1	1.40	1.4(5)	3.1
1.90	6.0(4)	1.1	1.35	1.6(5)	3.5
1.84	8.0(4)	1.5	1.30	1.0(5)	2.3
1.80	2.2(5)	4.2	1.25	2.4(5)	5.6
1.72	1.0(6)	19.5	1.20	1.7(5)	4.1
1.67	1.4(5)	2.8	1.15	3.2(5)	7.8
1.60	6.0(4)	1.2	1.10	5.5(5)	13.8
1.55	1.2(5)	2.5	1.05	1.2(6)	31.3
1.50	1.2(5)	2.5	1.01	4.05(6)	107.5

^aRead as 1.0×10^4 .

$$\int_0^T dx \int_{1 \text{ MeV}}^{\infty} \varphi(E, x) dE \simeq \frac{1.05 T}{\cos 45^\circ} \int_{1 \text{ MeV}}^{\infty} J(E) dE =$$

$$= 8.37 \times 10^{13} \text{ neut. cm/100 Kw/200 min.}$$

(The factor ~ 1.05 was calculated by ANISN.)

Thus,

$$\bar{\sigma}(E_Y) = 7.30 \times 10^{-6} \phi_Y(E_Y)/A_Y(E_Y) \text{ millibarns.}$$

Only the easily recognizable ~ 7.72 -MeV gamma ray, which is the most intense in the capture spectrum of thermal neutrons, has been designated as a capture gamma ray in Table XIII. Multiplying the intensity of this gamma ray by 2.2, the intensities of all the others by 1.33, adding in the intensities arising from $1/V$ capture and folding, results in the upper curves indicated in Figs. 16 and 17 by the open circles. The gamma-ray intensities arising from $1/V$ capture were calculated using

$$\sigma_{n\gamma}(E) = \frac{0.235 \sqrt{0.0253}}{\sqrt{E}} = 0.0374/\sqrt{E} \text{ barns:}$$

$$\int_0^T dx \int_{0.5 \text{ eV}}^{\infty} \sigma_{n\gamma}(E) \varphi(E, x) dE = \frac{T}{\cos 45^\circ} \times 1.07 \int_{0.5 \text{ eV}}^{\infty} \sigma_{n\gamma}(E) J(E) dE =$$

$$\frac{T}{\cos 45^\circ} \times 1.07 \int_{0.5 \text{ eV}}^{\infty} \frac{0.0374}{\sqrt{E}} \times \frac{4.0 \times 10^{12}}{E} dE$$

$$= 2.54 \times 0.25 \times 1.414 \times 1.07 \times 0.0374 \times 4.0 \times 10^{12} \times 2 \times 1.414$$

$$= 4.06 \times 10^{11} \frac{\text{neutron barn cm}}{100 \text{ Kw } 200 \text{ min}} .$$

Thus,

$$\begin{aligned}
d_{\gamma}(E_{\gamma}) \text{ photons/100 Kw/200 min} &= \frac{4.06 \times 10^{11}}{4\pi} \times \frac{(\pi/4) \times 25}{144 \times 400} \times \frac{0.6023}{27} \\
&\times 2.7 I_{\gamma}(E_{\gamma}) A_{\gamma}(E_{\gamma}) \\
&= 6.63 \times 10^5 I_{\gamma}(E_{\gamma}) A_{\gamma}(E_{\gamma}) .
\end{aligned}$$

The values of $I_{\gamma}(E_{\gamma})$, in photons/capture, were again taken from ref. 6 and values of $A_{\gamma}(E_{\gamma})$ from Fig. 18.

A comparison of the calculated response with the measured response for the cadmium-filtered beam is adequate, being within $\pm 10\%$ everywhere below 6.3 MeV, and $\pm 20\%$ everywhere above 6.3 MeV except around 7.0 MeV. As with iron, the conclusion may be drawn that gamma rays arising from any resonance capture below ~ 10 keV are small compared with those arising from $1/V$ capture. Besides the ~ 7.72 MeV gamma ray, there is definite evidence in Fig. 17 that a resonance capture gamma ray or rays occurs in the vicinity of 7.5 MeV and/or 7.0 MeV. This would better the agreement between the calculated and measured responses for the cadmium-filtered beam in these regions.

A comparison of the average production cross sections summed over 0.5 MeV gamma-ray intervals with the data from ref. 5 averaged over the neutron spectrum above 1 MeV in Fig. 4 is presented in Table XIV. The agreement, as in the case of iron, is generally quite good. The only significant difference is in the region 3-3.5 MeV, where a strong ~ 3.02 -MeV gamma ray lies. The data on individual lines in ref. 1 are too incomplete to settle the question, but an earlier evaluation by Trubetzkoy² leads to an average value of 25 mb for neutrons up to 5 MeV compared to our value of 31 mb. In any event, the data of ref. 5 are deemed adequate for most shielding applications.¹⁵

From an inspection of Table XIII, relatively strong gamma rays arising from fast-neutron interactions in aluminum occur in the vicinity of 1.01, 1.72, 2.23, 2.72, 3.02, 3.97, 4.40, and 4.61 MeV. The remaining lines in Table XIII are a combination of weaker lines and a continuum.

Fig. 18. Attenuation Function for Secondary Gamma Rays from Aluminum.

Table XIV. Comparison of Average Production Cross Sections in Millibarns from Present Results with Data of Celnik and Spielberg for Aluminum

	Gamma-Ray Energy Range (MeV)													
	1-1.5	1.5-2	2-2.5	2.5-3	3-3.5	3.5-4	4-4.5	4.5-5	5-5.5	5.5-6	6-6.5	6.5-7	7-7.5	Total
Present Results	181	36.3	102.5	26.3	42.4	8.8	5.8	6.9	4.9	2.0	< 1.7	< 1.5	< 0.3	≤ 420
Celnik and Spielberg ⁵	208	60.8	106	25.5	11.0	4.5	4.7	7.3	2.7	1.1	1.1	0.6	0	433

A calculation similar to the one described for iron of the production cross section for the ~ 7.72 -MeV gamma ray arising from capture of neutrons averaged over the energy range ~ 10 keV to 200 keV yields the value 2.5 mb. If this value is compared with a value of ~ 7 mb obtained for this spectrum from data in ref. 1, the tentative conclusion may be reached that the intensity of this gamma ray, $I_\gamma(E_\gamma)$, is roughly 0.30 photons per capture -- the value for thermal-neutron capture.

Copper. The foreground runs for 0.069 in. of copper were 312-C and 313-A for the boron-filtered beam and 312-B for the cadmium-filtered beam. They were then normalized to 100 Kw and 200 min (see appendix). In the expression $E_c = G_c (C-2)$ used to convert pulse height to energy, $0.0212 \leq G_c \leq 0.0221$ MeV/channel for the cadmium-filtered run and $0.0214 \leq G_c \leq 0.0223$ MeV/channel for the boron-filtered run. The net responses were formed and appear as the solid lines in Figs. 19 and 20.

The contribution to the net response for each beam filter from capture of the incident thermal neutrons was calculated by folding the intensities determined from evaluating

$$\mathcal{J}_\gamma(E_\gamma) = 6.16 \times 10^5 I_\gamma(E_\gamma) A_\gamma(E_\gamma) \text{ photons/100 Kw/200 min}$$

for the boron-filtered beam and

$$\mathcal{J}_\gamma(E_\gamma) = 1.25 \times 10^6 I_\gamma(E_\gamma) A_\gamma(E_\gamma) \text{ photons/100 Kw/200 min}$$

for the cadmium-filtered beam. In these expressions, $I_\gamma(E_\gamma)$ in photons/capture was taken from ref. 6 and $A_\gamma(E_\gamma)$ is plotted in Fig. 21. The result of subtracting these responses from the net responses appears as a smooth curve through the open diamonds appearing in Figs. 19 and 20.

The unfolded intensities derived from the incident thermal-neutron-capture-corrected response using the boron-filtered beam produced a response shown by the open circles in the lower curve of Figs. 19 and 20. These spectral intensities are presented in Table XV along with values of their production cross sections averaged over the incident spectrum above 1 MeV presented in Fig. 4. These cross sections were calculated from the spectral intensities in the following manner:

Fig. 19. Measured and Calculated Net Responses from 0.069 inches of Copper for the Energy Region 1.0-4.0 MeV.

ORNL-DWG 69-14642

Fig. 20. Measured and Calculated Net Responses from 0.069 inches of Copper for the Energy Region 3.8-8.2 MeV.

Fig. 21. Attenuation Function for Secondary Ga-ma Rays from Copper.

Table XV. Unfolded Intensities and Average Production Cross Sections for Copper

E_γ (MeV)	$\mathcal{L}_\gamma$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb	E_γ (MeV)	$\mathcal{L}_\gamma$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb
8.12	3.4(4) ^a	capture	4.80	1.0(5)	3.3
8.02	1.0(4)	capture	4.70	5.0(4)	1.6
7.92	1.3(5)	capture	4.60	1.0(5)	3.4
7.72	4.0(4)	capture	4.50	1.15(5)	3.9
7.62	5.5(4)	capture	4.40	1.4(5)	4.7
7.52	1.0(4)	capture	4.30	1.2(5)	4.1
7.32	3.0(4)	capture	4.20	1.25(5)	4.3
7.22	6.5(4)	capture	4.10	1.6(5)	5.5
7.08	3.4(4)	capture	4.00	1.5(5)	5.3
6.88	1.1(4)	capture	3.90	1.5(5)	5.3
6.78	5.0(4)	capture	3.80	1.55(5)	5.5
6.68	6.0(4)	capture	3.70	1.8(5)	6.4
6.48	2.5(4)	0.8	3.60	2.0(5)	7.3
6.38	5.5(4)	1.7	3.50	3.0(5)	11.0
6.28	1.2(4)	0.4	3.40	2.25(5)	8.3
6.18	3.5(4)	1.1	3.30	2.0(5)	7.5
6.08	5.7(4)	1.8	3.20	2.0(5)	7.5
6.00	1.5(4)	0.4	3.10	3.0(5)	11.5
5.90	5.5(4)	1.7	3.00	2.6(5)	10.1
5.80	1.0(4)	0.3	2.95	2.3(5)	9
5.70	4.8(4)	1.5	2.87	4.0(5)	16
5.60	3.0(4)	0.9	2.80	2.5(5)	10
5.50	4.5(4)	1.5	2.75	6.0(4)	2.5
5.40	5.0(4)	1.6	2.70	2.9(5)	11.5
5.30	5.0(4)	1.6	2.60	4.0(5)	16.5
5.20	7.5(4)	2.4	2.51	5.2(5)	21.5
5.10	3.0(4)	1.0	2.42	3.6(5)	15.5
5.00	8.0(4)	2.6	2.34	5.0(5)	21.5
4.90	4.0(4)	1.3	2.28	2.0(5)	8.5

Table XV (cont.)

E_γ (MeV)	γ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb	E_γ (MeV)	γ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb
2.22	2.5(5)	11	1.55	7.5(5)	39
2.15	5.5(5)	25	1.50	8.5(5)	45
2.08	5.8(5)	26	1.45	9.5(5)	52
2.03	4.3(5)	20	1.40	1.4(6)	77
1.96	5.0(5)	23	1.33	2.4(6)	137
1.88	9.3(5)	45	1.28	7.0(5)	40
1.83	1.0(5)	5	1.23	4.0(5)	24
1.80	2.8(5)	13	1.17	1.2(6)	74
1.75	4.0(5)	20	1.11	2.7(6)	170
1.70	3.3(5)	16	1.07	7.0(5)	45
1.65	3.7(5)	19	~ 0.98	$\sim 7.0(6)$	~ 470
1.60	6.0(5)	31			

^aRead as 3.5×10^4 .

$$J_Y(E_Y) \text{ photons/100 Kw/200 min} = \frac{\bar{\sigma}(E_Y) \times 10^{-3}}{4\pi} \times \frac{0.6023}{63.5} \times 8.8 \times \frac{(\pi/4) \times 25}{144 \times 400}$$

$$\times \int_0^T dx \int_{1 \text{ MeV}}^{\infty} \varphi(E, x) dE \times A_Y(E_Y) ,$$

where $\bar{\sigma}(E_Y)$ is the average production cross section above 1 MeV in millibarns, $A_Y(E_Y)$ is taken from Fig. 21, and

$$\int_0^T dx \int_{1 \text{ MeV}}^{\infty} \varphi(E, x) dE \simeq \frac{1.05 T}{\cos 45^\circ} \int_{1 \text{ MeV}}^{\infty} J(E) dE = 2.31 \times 10^{13} \text{ neut. cm/100 Kw/200 min.}$$

$$\text{Thus, } \bar{\sigma}(E_Y) = 1.90 \times 10^{-5} J_Y(E_Y)/A_Y(E_Y) \text{ millibarns.}$$

All of the gamma rays lying above 6.5 MeV were assigned capture status in Table XV for two reasons. First, the thermal-neutron capture spectrum has many reasonably intense lines in this region.⁶ Second, the presence of several strong lines in the boron filter response is evident in this region and these in all likelihood arise as a result of capture, not inelastic scattering, because gamma rays arise from de-excitation of a discrete level from capture in elements of low Z whereas they arise from a continuum of levels for these high gamma-ray energies in the case of inelastic scattering.

To calculate the cadmium filter response, then, the intensities in Table XV lying above 6.5 MeV were multiplied by 2.2, those below 6.5 MeV by 1.33, the intensities from 1/V capture calculated, and the entire spectrum folded.

$$\text{The 1/V capture intensities were calculated using } \sigma_{n\gamma}(E) = \frac{3.8 \sqrt{0.0253}}{\sqrt{E}}$$

= 0.604/ $\sqrt{E}$ barns, so that

$$\int_0^T dx \int_{0.5 \text{ eV}}^{\infty} \sigma_{n\gamma}(E) \varphi(E, x) dE \simeq \frac{T \times 1.05}{\cos 45^\circ} \int_{0.5 \text{ eV}}^{\infty} \sigma_{n\gamma}(E) J(E) dE =$$

$$\frac{T \times 1.05}{45 \cos^\circ} \int_{0.5 \text{ eV}}^{\infty} \frac{0.604}{\sqrt{E}} \times \frac{4.0 \times 10^{12}}{E} dE$$

$$= 2.54 \times 0.069 \times 1.05 \times 1.414 \times 0.604 \times 4.0 \times 10^{12} \times 2 \times 1.414$$

$$= 1.78 \times 10^{12} \frac{\text{neutron barn cm}}{100 \text{ Kw } 200 \text{ min}} .$$

Thus,

$$\begin{aligned} \mathcal{J}_{\gamma}(E_{\gamma}) &= \frac{1.78 \times 10^{12}}{4\pi} \times \frac{(\pi/4) \times 25}{144 \times 400} \times \frac{0.6023}{63.5} \times 8.8 I_{\gamma}(E_{\gamma}) A_{\gamma}(E_{\gamma}) \\ &= 4.03 \times 10^6 I_{\gamma}(E_{\gamma}) A_{\gamma}(E_{\gamma}) \text{ photons/100 Kw/200 min} , \end{aligned}$$

where $I_{\gamma}(E_{\gamma})$, in photons/capture, is taken from ref. 6 and $A_{\gamma}(E_{\gamma})$ from Fig. 21.

A comparison of the calculated and measured responses for the cadmium-filtered beam in Figs. 19 and 20 indicates that some gamma rays arising from resonance capture in the neutron energy region below 10 keV exist that have not been accounted for. Data from ref. 1 indicate that several strong resonances do indeed occur in this region for both the ^{63}Cu and ^{65}Cu isotopes. Since the complete gamma-ray spectra arising from resonance capture in copper are still under investigation by several researchers,¹⁶⁻¹⁷ no attempt has been made by us to incorporate these incomplete data into our calculations.

The results obtained by summing the average production cross sections appearing in Table XV over 0.5-MeV gamma-ray intervals appear in Table XVI. Unfortunately, no other data exist with which to compare these results.

Comparisons can be made between the cross sections of the two most intense lines with production cross sections averaged over the incident spectrum of Fig. 4 for a considerably shorter range in neutron energies (from 1.0 to 2.6 MeV) available from ref. 1. Corrected for isotopic abundance in natural copper, they are presented in Table XVII along with the values from Table XV, which are, of course, averages over the entire incident neutron spectrum above 1 MeV in Fig. 4.

An important cause of the disagreement in Table XVII lies in the different neutron ranges over which the averages are made. The data of ref. 1 indicates that if the range were extended to higher values, the average values of both production cross sections would increase. Thus it is felt there is no inconsistency between the two sets of data.

Table XVI. Average Production Cross Sections in Millibarns for Copper

	Gamma-Ray Energy Range (MeV)											Total
	0.98-1.5	1.5-2	2-2.5	2.5-3	3-3.5	3.5-4	4-4.5	4.5-5	5-5.5	5.5-6	6-6.5	
Present Results	~ 1090	256	127	87	45	35	24	13.5	9.1	< 5.9	< 6.2	≤ 1700

Table XVII. Comparison of Average Production Cross Sections in Millibarns from Present Results with BNL Data for Two Intense Gamma Rays from Copper

Gamma-Ray Energy	0.961 MeV	1.115 MeV
Isotope	^{63}Cu	^{65}Cu
Present Results	~ 470	170
BNL-325 ¹	250	100

From an inspection of Table XV, relatively strong gamma rays arising from fast-neutron interactions in copper occur in the vicinity of 0.98, 1.11, 1.17, 1.33, 1.40, 1.88, 2.10, 2.34, 2.51, 2.87, 3.10, and 3.50 MeV. Many of these lines, and even some of the weaker lines also appearing in Table XV, may be readily identified from the decay schemes of ^{63}Cu and ^{65}Cu (ref. 18). The remaining background lines in Table XV again represent a continuum contribution.

A calculation similar to the one described for iron of the production cross sections of those gamma rays lying above 6.5 MeV in Table XV arising from capture of neutrons averaged over the energy range 10 keV to 200 keV yields the values appearing in Table XVIII. Since these gamma rays probably account for most of the binding energy available, which is in the vicinity of 8 MeV, the average total capture cross section should

approximate $\sum_i E_{\gamma_i} \bar{\sigma}_{n\gamma}(E_{\gamma_i})/8 = 92 \text{ mb}$, which is in good agreement with

the value 96 mb found by assuming none of these gamma rays are in cascade. Thus the average capture cross section for copper in the region ~ 10 -200 keV for the incident spectrum of Fig. 8 is $\approx 95 \text{ mb}$, and this is in fair agreement with a value of $\sim 50 \text{ mb}$ evaluated from data in ref. 1. The very approximate value of $\bar{I}_{\gamma}(E_{\gamma})$ in photons per 100 captures obtained by evaluating $[\bar{\sigma}_{n\gamma}(E_{\gamma})/95] \times 100$ are compared with corresponding values from thermal-neutron captures in the last two columns of Table XVIII. As in the case of iron and aluminum, the most intense gamma ray observed in resonance capture is emitted with about the same probability as in thermal-neutron capture, although important differences are found for most of the other gamma rays.

Table XVIII. Capture Cross Sections for Copper Averaged Over the
Neutron Spectrum from 10-200 keV

E_γ (MeV)	$\bar{\sigma}_{n\gamma}(E_\gamma)$ (Millibarns)	$\bar{\sigma}_{n\gamma}(E_\gamma) \times E_\gamma$ (MeV·Millibarns)	$\bar{I}_\gamma(E_\gamma)$ (photons/100 captures)	$I_\gamma(E_\gamma)$ (photons/100 thermal-neutron captures)
8.12	6	50	6	-
8.02	2	15	2	-
7.92	24	190	25	25.4
7.72	7	55	7	2.2
7.62	10	80	10	12.9
7.52	2	15	2	-
7.32	5	40	5	9.8
7.22	12	90	12	~2.7
7.08	6	45	6	3.5
6.88	2	15	2	-
6.78	9	65	9	-
6.68	<u>11</u>	<u>75</u>	11	4.5
TOTAL	96	735		

Titanium. The foreground runs using a 0.265-in. slab of titanium were 362-C for the cadmium-filtered beam and 363-A for the boron-filtered beam. They were then normalized to 100 Kw and 200 min (see appendix). The value of the gain varied between 0.0219 MeV/channel at low channel numbers to 0.0229 MeV/channel at high channel numbers, for both runs. The net responses in units of counts/MeV/100 Kw/200 min were formed and appear as the solid lines in Figs. 22 and 23.

The contribution to the net response for each beam filter arising from capture of the incident thermal neutrons was calculated and subtracted from the net response. For the boron-filtered beam, the intensities were calculated from

$$\mathcal{I}_\gamma(E_\gamma) = 2.2 \times 10^6 I_\gamma(E_\gamma) A_\gamma(E_\gamma) \text{ photons/100 Kw/200 min,}$$

where $I_\gamma(E_\gamma)$ in photons/capture was taken from ref. 6 and $A_\gamma(E_\gamma)$ appears in Fig. 24, and folded. For the cadmium-filtered beam, the intensities were calculated from

$$\mathcal{I}_\gamma(E_\gamma) = 4.5 \times 10^6 I_\gamma(E_\gamma) A_\gamma(E_\gamma) \text{ photons/100 Kw/200 min ,}$$

and then folded. The result of this subtraction for each beam filter appears as a smooth curve through the open diamonds in Figs. 22 and 23, and it is evident that the effect of these captures is small except in the energy region centered around 1.4 MeV and 6 MeV.

The net response corrected for capture of the incident thermal neutrons from the boron-filtered beam was unfolded using the fitted response depicted by the open circles in the lower curve of Figs. 22 and 23. These unfolded intensities are presented in Table XIX along with the corresponding gamma-ray production cross sections averaged over the incident spectrum above 1 MeV presented in Fig. 4. These cross sections were calculated from the unfolded intensities in the following manner:

Fig. 22. Measured and Calculated Net Responses from 0.265 inches of Titanium for the Energy Range 1.0-3.9 MeV.

Fig. 23. Measured and Calculated Responses from 0.265 inches of Titanium for the Energy Range 3.9-9.1 MeV.

Table XIX. Unfolded Intensities and Average Production
Cross Sections for Titanium

E_{γ} (MeV)	ϕ_{γ} (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_{\gamma})$ mb	E_{γ} (MeV)	ϕ_{γ} (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_{\gamma})$ mb
9.10	1.5(4) ^a	capture	5.30	7.0(4)	0.85
9.00	1.5(4)	capture	5.20	6.0(4)	0.70
8.90	1.0(4)	capture	5.10	1.0(5)	1.25
8.80	9.0(3)	capture	5.00	7.5(4)	0.90
8.70	5.5(3)	capture	4.90	9.2(4)	1.2
8.30	5.5(3)	capture	4.80	6.5(4)	0.8
8.20	1.5(4)	capture	4.70	8.0(4)	1.0
8.10	7.0(3)	capture	4.60	8.0(4)	1.0
8.00	1.3(4)	capture	4.50	8.5(4)	1.1
7.90	3.5(3)	capture	4.40	1.0(5)	1.3
7.70	8.0(3)	capture	4.32	2.5(5)	3.2
7.60	1.2(4)	capture	4.20	1.3(5)	1.6
7.50	2.0(4)	capture	4.10	9.0(4)	1.2
7.40	1.5(4)	capture	4.00	1.7(5)	2.2
7.30	2.0(4)	capture	3.90	1.1(5)	1.5
7.20	2.5(4)	capture	3.80	1.8(5)	2.4
7.10	1.2(4)	capture	3.71	7.0(5)	9.5
6.77	3.7(5)	capture	3.60	3.0(5)	4.1
6.60	3.5(4)	capture	3.50	1.9(5)	2.7
6.50	2.0(5)	capture	3.40	2.4(5)	3.4
6.40	1.4(5)	capture	3.35	2.3(5)	3.2
6.35	4.5(4)	0.50	3.30	1.5(5)	2.1
6.16	2.5(4)	0.30	3.20	3.4(5)	4.9
6.10	1.1(5)	1.3	3.10	2.75(5)	4.0
6.00	2.0(4)	0.25	3.00	2.85(5)	6.6
5.80	8.3(4)	1.0	2.90	4.4(5)	6.6
5.70	8.0(4)	0.95	2.84	5.0(5)	7.5
5.60	2.0(4)	0.25	2.75	5.6(5)	8.5
5.50	5.0(4)	0.60	2.69	2.65(5)	4.1
5.40	6.0(4)	0.70	2.63	9.0(5)	14.0

Table XIX (cont.)

E_γ (MeV)	$\mathcal{Q}_\gamma$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb	E_γ (MeV)	$\mathcal{Q}_\gamma$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb
2.55	4.5(5)	7.1	1.62	7.1(5)	14.1
2.48	2.6(5)	4.1	1.57	1.45(6)	29.3
2.42	7.7(5)	12.5	1.52	1.15(6)	23.8
2.35	1.9(6)	31.0	1.47	1.1(6)	23
2.28	5.9(5)	9.8	1.42	2.1(6)	45
2.22	1.05(6)	17.6	1.37	1.1(6)	24
2.16	4.0(5)	6.8	1.31	2.7(6)	60
2.10	4.0(5)	6.9	1.26	7.5(5)	17
2.05	5.0(5)	8.7	1.21	6.0(5)	14
1.99	6.5(5)	11.5	1.16	1.3(6)	31
1.95	1.5(5)	2.7	1.13	3.0(5)	7
1.90	3.1(5)	5.7	1.11	2.5(5)	6
1.83	4.3(5)	8.0	1.07	2.8(6)	70
1.75	7.0(5)	13.3	1.03	1.0(6)	26
1.69	4.0(5)	7.8	0.99	1.37(7)	365

^aRead as 1.5×10^4 .

$$J_Y(E_Y) \text{ photons/100 Kw/200 min} = \frac{\bar{\sigma}(E_Y) \times 10^{-3}}{4\pi} \times \frac{0.6023}{47.9} \times 4.5 \times \frac{(\pi/4) \times 25}{144 \times 400}$$

$$\times \int_0^T dx \int_{1 \text{ MeV}}^{\infty} \varphi(E, x) dE \times A_Y(E_Y) ,$$

where $\bar{\sigma}(E_Y)$ is the average production cross section above 1 MeV in millibarns, $A_Y(E_Y)$ is taken from Fig. 24, and

$$\int_0^T dx \int_{1 \text{ MeV}}^{\infty} \varphi(E, x) dE \simeq \frac{1.1 T}{\cos 45^\circ} \int_{1 \text{ MeV}}^{\infty} J(E) dE = 9.34 \times 10^{13}$$

neutrons cm/100 Kw/200 min .

Thus, $\bar{\sigma}(E_Y) = 6.97 \times 10^{-6} J_Y(E_Y)/A_Y(E_Y)$ millibarns.

Somewhat arbitrarily, all gamma rays lying above 6.4 MeV were designated as arising from resonance capture in Table XIX. Multiplying the intensities of these gamma rays by 2.2, the intensities of all the others by 1.33, adding in the intensities arising from 1/V capture and folding, results in the upper curves indicated in Figs. 22 and 23 by the open circles. The gamma-ray intensities arising from 1/V capture were calculated using $\sigma_{n\gamma}(E) = \frac{6.1 \sqrt{0.0253}}{\sqrt{E}} = 0.970/\sqrt{E}$ barns:

$$\int_0^T dx \int_{0.5 \text{ eV}}^{\infty} \sigma_{n\gamma}(E) \varphi(E, x) dE \simeq \frac{T \times 1.2}{\cos 45^\circ} \int_{0.5 \text{ eV}}^{\infty} \sigma_{n\gamma}(E) J(E) dE =$$

$$\frac{1.2 T}{\cos 45^\circ} \int_{0.5 \text{ eV}}^{\infty} \frac{0.970}{\sqrt{E}} \times \frac{4.0 \times 10^{12}}{E} dE$$

$$= 2.54 \times 0.265 \times 1.414 \times 1.2 \times 0.970 \times 4.0 \times 10^{12} \times 2 \times 1.414$$

$$= 1.25 \times 10^{13} \frac{\text{neutron barn cm}}{100 \text{ Kw } 200 \text{ min}} .$$

Thus, $J_Y(E_Y) \text{ photons/100 Kw/200 min} = \frac{1.25 \times 10^{13}}{4\pi} \times \frac{(\pi/4) \times 25}{144 \times 400}$

$$\times \frac{0.6023}{47.9} \times 4.5 I_Y(E_Y) A_Y(E_Y)$$

$$= 1.93 \times 10^7 I_Y(E_Y) A_Y(E_Y) .$$

where the values of $I_{\gamma}(E_{\gamma})$ in photons/capture were again taken from ref. 6 and values of $A_{\gamma}(E_{\gamma})$ from Fig. 24.

A comparison of the calculated response with the measured response for the cadmium-filtered beam shows that the agreement is to within $\pm 15\%$ everywhere. Thus any gamma rays from resonance capture in the region below ~ 10 keV are less than 15% of those from $1/V$ capture in titanium.

A comparison of the average production cross sections summed over 0.5 MeV gamma-ray intervals with the data from ref. 4 averaged over the neutron spectrum above 1 MeV in Fig. 4 is presented in Table XX. Here the agreement is not good, although the sum of the production cross sections over all energies checks almost exactly. The region 1.5-2.5 MeV is greatly underestimated by the data of ref. 4, and the region above 4 MeV somewhat overestimated. It is conjectured that the UNC data omitted any contributions from the isotopes ^{46}Ti , ^{47}Ti , ^{49}Ti , and ^{50}Ti , each of which occurs in natural titanium with a relative abundance of about 6%, and which can contribute appreciably to the region 1.5-2.5 MeV. The relatively small discrepancy in the region 0.99-1.5 MeV can easily be explained by anisotropy of the gamma rays, but there is no such logical explanation for the discrepancies above 4 MeV, which would control the accuracy of a transport calculation through a thick shield. Unfortunately, there is no data in ref. 1 with which to compare average intensities of strong individual lines, which from Table XIX occur in the vicinity of 0.99, 1.07, 1.16, 1.31, 1.42, 1.57, 1.75, 1.99, 2.22, 2.35, 2.63, 3.71, and 4.32 MeV. The remaining intensities in Table XIX are a combination of weaker lines superimposed upon a continuum.

Assuming all of the gamma rays lying above 6.4 MeV in the unfolded response from the boron-filtered beam occur as a result of resonance capture in the energy range 10 keV to 200 keV, average capture cross sections can be deduced in a manner completely analogous to the one described for iron. These values appear in Table XXI.

The value of $\sum_i E_{\gamma i} \bar{\sigma}_{n\gamma}(E_{\gamma i})/8 = 60$ mb reasonably checks the value of $\sum_i \bar{\sigma}_{n\gamma}(E_{\gamma i}) = 69$ mb, so that the average capture cross section for

Fig. 24. Attenuation Function for Secondary Gamma Rays from Titanium.

Table XX. Comparison of Average Production Cross Sections in Millibarns from Present Results with Data of Troubetzkoy for Titanium

	Gamma-Ray Energy Range (MeV)											Total
	0.99-1.5	1.5-2	2-2.5	2.5-3	3-3.5	3.5-4	4-4.5	4.5-5	5-5.5	5.5-6	6-6.5	
Present Results	688	116	97	48	21.8	20.2	9.5	5.1	< 4.4	< 2.8	< 2.4	≤ 1015
Troubetzkoy <u>et al.</u> ⁴	846	12	15	24	63	32		29		7.2		1028

Table XXI. Capture Cross Sections for Titanium Averaged Over the
Neutron Spectrum from 10-200 keV

E_γ (MeV)	$\bar{\sigma}_{n\gamma}(E_\gamma)$ (Millibarns)	$\bar{\sigma}_{n\gamma}(E_\gamma) \times E_\gamma$ (MeV·Millibarns)	$\bar{I}_\gamma(E_\gamma)$ (photons/100 captures)	$I_\gamma(E_\gamma)$ (photons/100 thermal-neutron captures)
9.10	1.0	9	1	0.20
9.00	1.0	9	1	
8.90	0.7	6	1	
8.80	0.6	5	1	
8.70	0.4	4	0.5	
8.30	0.4	3	0.5	0.20
8.20	1.0	9	1	
8.10	0.5	4	0.5	
8.00	0.9	7	1	0.15
7.90	0.3	2	0.5	0.08
7.70	0.6	5	1	
7.60	0.8	7	1	0.10
7.50	1.4	10	2	
7.40	1.1	8	1	0.10
7.30	1.4	11	2	0.15
7.20	1.8	13	3	0.65
7.10	0.9	6	1	
6.77	27	183	39	46.2
6.60	2.6	17	4	
6.50	15	98	21	8.7
6.40	10	67	14	32.0
Total	69	483		

titanium in the region 10-200 keV for the incident spectrum of Fig. 8 is ≈ 65 mb, compared with a value in the neighborhood of 10 mb from data in ref. 1. This is an alarming discrepancy, but comparison of the most intense resonance capture lines with values from thermal-neutron capture in the last column of Table XXI again shows that the relative intensities are about equal (39 vs 46 for the 6.77-MeV line and 35 vs 41 for the sum of the 6.40-MeV and 6.50-MeV lines). These three lines account for about 80% of the average capture cross section value of ~ 65 mb, and are fairly well established as capture lines. Consequently, there seems to be real evidence here that the $\sigma_{n\gamma}(E)$ values in ref. 1 are low in the region around 100 keV.

Zinc. The foreground runs for 0.0657 in. of zinc were 325-D and 326-A for the boron-filtered beam and 326-C for the cadmium-filtered beam. They were then normalized to 100 Kw and 200 min (see appendix). The gain for the boron-filtered runs varied between 0.0215 MeV/channel at the low channel numbers to 0.0221 MeV/channel at the high channel numbers; for the cadmium-filtered run, $0.0217 \leq G_c \leq 0.0224$ MeV/channel. The net responses were formed and expressed in units of counts/MeV/100 Kw/200 min. They appear as the solid lines in Figs. 25 and 26.

The contribution to the net response for each beam filter arising from capture of the incident thermal neutrons was calculated and subtracted from the net response. For the boron-filtered beam, the intensities were calculated from

$$Q_Y(E_Y) = 1.33 \times 10^5 I_Y(E_Y) A_Y(E_Y) \text{ photons/100 Kw/200 min ,}$$

where $I_Y(E_Y)$ in photons/capture was taken from ref. 6 and $A_Y(E_Y)$ appears in Fig. 27, and folded. For the cadmium-filtered beam, the intensities were calculated from

$$Q_Y(E_Y) = 2.7 \times 10^5 I_Y(E_Y) A_Y(E_Y) \text{ photons/100 Kw/200 min ,}$$

and then folded. The result of this subtraction for each beam filter appears as a smooth curve through the open diamonds in Figs. 25 and 26, and it is evident that the effect of these captures is small everywhere.

Fig. 25. Measured and Calculated Net Responses from 0.0657 inches of Zinc for the Energy Range 1.0-4.0 MeV.

Fig. 26. Measured and Calculated Net Responses from 0.0657 inches of Zinc for the Energy Range 4.0-9.4 MeV.

Fig. 27. Attenuation Function for Secondary Gamma Rays from Zinc.

The unfolded intensities derived from the net response corrected for incident thermal-neutron capture using the boron-filtered beam produced a response shown by the open circles in the lower curve of Figs. 25 and 26. These spectral intensities are presented in Table XXII along with values of their production cross sections averaged over the incident spectrum above 1 MeV presented in Fig. 4. These cross sections were calculated from the spectral intensities in the following manner:

$$\mathcal{J}_\gamma(E_\gamma) \text{ photons/100 Kw/200 min} = \frac{\bar{\sigma}(E_\gamma) \times 10^{-3}}{4\pi} \times \frac{0.6023}{65.4} \times 7.1$$

$$\times \frac{(\pi/4) \times 25}{144 \times 400} \times \int_0^T dx \int_{1 \text{ MeV}}^{\infty} \varphi(E, x) dE \times A_\gamma(E_\gamma) ,$$

where $\bar{\sigma}(E_\gamma)$ is the average production cross section above 1 MeV in millibarns, $A_\gamma(E_\gamma)$ is taken from Fig. 27, and

$$\int_0^T dx \int_{1 \text{ MeV}}^{\infty} \varphi(E, x) dE \approx \frac{1.05 T}{\cos 45^\circ} \int_1^{\infty} J(E) dE = 2.21 \times 10^{13} \text{ neut. cm/100 Kw/200 min.}$$

Thus, $\bar{\sigma}(E_\gamma) = 2.56 \times 10^{-5} \mathcal{J}_\gamma(E_\gamma)/A_\gamma(E_\gamma)$ millibarns.

All of the gamma rays lying above 6 MeV were assigned capture status in Table XXII for the same reasons as were stated for copper. In addition, since for zinc the response due to gamma rays from 1/V capture in the cadmium-filtered beam is a relatively small part of the total response (see below), the choice of where to assign a cutoff energy to the resonance captures is dictated to some extent by the ensuing agreement between the calculated and measured response for the cadmium-filtered beam. The cutoff energy cannot be too low, say 4 MeV, however, because the production cross section from resonance captures would then be unreasonably high.

To calculate the cadmium-filtered response, then, the intensities in Table XXII lying above 6 MeV were multiplied by 2.2, those below 6 MeV by 1.33, the intensities from 1/V capture calculated, and the entire spectrum folded.

Table XXII. Unfolded Intensities and Average Production
Cross Sections for Zinc

E_γ (MeV)	γ_γ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb	E_γ (MeV)	γ_γ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb
9.20	7.0(3) ^a	capture	5.40	2.0(4)	0.9
9.10	4.0(3)	capture	5.30	2.4(4)	1.0
9.00	2.8(3)	capture	5.20	2.9(4)	1.2
8.90	9.0(3)	capture	5.10	4.0(4)	1.7
8.60	2.0(3)	capture	5.00	3.6(4)	1.6
8.50	3.0(3)	capture	4.90	5.7(4)	2.5
8.40	2.0(3)	capture	4.80	3.8(4)	1.7
8.30	5.0(3)	capture	4.70	6.7(4)	3.0
8.20	1.1(4)	capture	4.60	3.7(4)	1.6
8.10	5.0(3)	capture	4.50	7.7(4)	3.5
8.00	1.7(4)	capture	4.40	6.1(4)	2.8
7.86	3.9(4)	capture	4.30	7.1(4)	3.2
7.70	1.0(4)	capture	4.20	6.5(4)	3.0
7.60	2.0(3)	capture	4.10	8.0(4)	3.7
7.50	8.0(3)	capture	4.00	7.3(4)	3.4
7.35	7.5(4)	capture	3.90	7.0(4)	3.3
7.20	3.5(4)	capture	3.85	1.5(4)	0.7
7.00	4.0(4)	capture	3.80	1.0(5)	4.7
6.60	3.0(4)	capture	3.75	8.0(4)	3.8
6.40	2.8(4)	capture	3.70	5.0(4)	2.4
6.30	1.5(4)	capture	3.60	1.3(5)	6.3
6.20	1.25(4)	capture	3.50	1.1(5)	5.4
6.10	2.5(3)	capture	3.43	2.0(5)	9.8
6.00	3.7(4)	capture	3.35	7.0(4)	3.5
5.90	1.5(4)	0.6	3.30	1.2(5)	6.0
5.80	2.65(4)	1.1	3.20	1.2(5)	6.0
5.70	2.2(4)	0.9	3.10	1.4(5)	7.1
5.60	4.0(4)	1.7	3.00	1.6(5)	8.2
5.50	3.0(4)	1.3	2.90	2.15(5)	11

Table XXII (cont.)

E_γ (MeV)	$\mathcal{Q}_\gamma$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb	E_γ (MeV)	$\mathcal{Q}_\gamma$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb
2.80	2.3(5)	12	1.75	3.0(5)	20
2.70	2.0(5)	11	1.70	2.5(5)	17
2.65	1.3(4)	0.7	1.65	2.9(5)	20
2.60	2.2(5)	12	1.60	1.7(5)	12
2.50	2.2(5)	12	1.55	1.8(5)	13
2.42	2.3(5)	13	1.50	2.5(5)	18
2.34	2.55(5)	15	1.45	3.6(5)	26
2.27	2.9(5)	17	1.39	4.6(5)	35
2.20	1.3(5)	7.7	1.33	7.5(5)	57
2.15	2.6(5)	15	1.26	4.8(5)	38
2.10	2.1(5)	13	1.20	5.5(5)	45
2.05	3.5(5)	21	1.15	2.0(5)	17
2.00	1.8(5)	11	1.11	1.6(6)	136
1.95	3.4(5)	21	1.10	4.0(5)	34
1.90	3.0(5)	19	1.07	7.0(5)	60
1.85	3.9(5)	25	1.03	4.3(6)	380
1.81	4.8(5)	31	1.00	9.0(5)	81

^aRead as 7.0×10^3 .

The $1/V$ capture intensities were calculated using $\sigma_{n\gamma}(E) = \frac{1.1 \sqrt{0.0253}}{\sqrt{E}} = 0.175/\sqrt{E}$ barns, so that

$$\int_0^T dx \int_{0.5 \text{ eV}}^{\infty} \sigma_{n\gamma}(E) \varphi(E, x) dE \approx \frac{T \times 1.05}{\cos 45^\circ} \int_{0.5 \text{ eV}}^{\infty} \frac{0.175}{\sqrt{E}} \times \frac{4.0 \times 10^{12}}{E} dE$$

$$= 2.54 \times 0.0657 \times 1.05 \times 1.414 \times 0.175 \times 4.0 \times 10^{12} \times 2 \times 1.414$$

$$= 4.90 \times 10^{11} \frac{\text{neutron barn cm}}{100 \text{ Kw } 200 \text{ min}} .$$

Thus

$$\gamma_{\gamma}(E_{\gamma}) = \frac{4.90 \times 10^{11}}{4\pi} \times \frac{(\pi/4) \times 25}{144 \times 400} \times \frac{0.6023}{65.4} \times 7.1 I_{\gamma}(E_{\gamma}) A_{\gamma}(E_{\gamma})$$

$$= 8.69 \times 10^5 I_{\gamma}(E_{\gamma}) A_{\gamma}(E_{\gamma}) \text{ photons/100 Kw/200 min} ,$$

where $I_{\gamma}(E_{\gamma})$, in photons/capture, is taken from ref. 6 and $A_{\gamma}(E_{\gamma})$ from Fig. 27.

A comparison of the calculated and measured responses for the cadmium-filtered beam in Figs. 25 and 26 indicates that the agreement is to within $\pm 25\%$ everywhere. However, it is also evident that there are gamma rays present which are not accounted for in the calculation, that they most likely arise from resonance capture in the neutron energy region below 10 keV, and that the capture cross section averaged over the resonances is roughly equal to the $1/V$ cross section from 0.5 eV to infinity, since roughly doubling the $1/V$ intensities would result in a good fit with the measured response for the cadmium-filtered beam. The data in ref. 1 shows several capture resonances in this energy region for zinc, perhaps the strongest ones occurring at 282 eV in ^{64}Zn , at 323 eV in ^{66}Zn , and at 514 eV in ^{68}Zn .

The average production cross sections summed over 0.5- MeV gamma-ray intervals are presented in Table XXIII. Unfortunately, no data are available with which to compare these values.

Table XXIII. Average Production Cross Sections in Millibarns for Zinc

	Gamma-Ray Energy Range (MeV)										
	1-1.5	1.5-2	2-2.5	2.5-3	3-3.5	3.5-4	4-4.5	4.5-5	5-5.5	5.5-6	Total
Present Results	909	195	113	59.4	40.6	26.6	16.1	< 13.3	< 6.4	< 5.6	1385

One comparison can be made from data in ref. 1 between the average cross section obtained in this experiment for the ~ 1.03 -MeV gamma ray arising from inelastic scattering and that averaged over a considerably shorter range in neutron energies (1.0-2.6 MeV). This comparison is shown in Table XXIV.

Table XXIV. Comparison of Average Production Cross Sections in Millibarns from Present Results with BNL Data for the 1.03-MeV Gamma Ray in Zinc

Present Results:	380
BNL-325 ¹	500

Since the two values of the cross sections in Table XXIV are both evaluated as $4\pi d\bar{\sigma}/d\Omega$ (90 deg), and the value from ref. 1 is already larger than our value and would probably increase as the neutron energy range increases, a disagreement between the two values of about a factor of 2 would probably occur if the cross sections were averaged over the same neutron energy range.

From an inspection of Table XXII, relatively strong gamma rays arising from fast-neutron interactions in zinc produce gamma rays in the vicinity of 1.03, 1.11, 1.33, 1.81, and 3.43 MeV. The remaining intensities in Table XXII represent weaker lines superimposed upon a continuum.

A rough calculation similar to the one described for iron of the production cross sections of those gamma rays lying above 6 MeV in Table XXII arising from resonance capture of neutrons averaged over the energy range 10 keV to 200 keV yields the values appearing in Table XXV.

The value of $\sum_i E_{\gamma i} \bar{\sigma}_{n\gamma}(E_{\gamma i})/8 = 87$ mb reasonably checks the value of

$\sum_i \bar{\sigma}_{n\gamma}(E_{\gamma i}) = 93$ mb, so that the gamma rays listed in Table XXV are not

in cascade and the average capture cross section for zinc in the region 10-200 keV for the incident spectrum of Fig. 8 is ≈ 90 mb, compared with a value of ~ 30 mb obtained from data in ref. 1. The values of $I_{\gamma}(E_{\gamma})$

Table XXV. Capture Cross Sections for Zinc Averaged Over the
Neutron Spectrum from 10-200 keV

E_γ (MeV)	$\bar{\sigma}_{n\gamma}(E_\gamma)$ (Millibarns)	$\bar{\sigma}_{n\gamma}(E_\gamma) \times E_\gamma$ (MeV·Millibarns)	$\bar{I}_\gamma(E_\gamma)$ (photons/100 captures)	$I_\gamma(E_\gamma)$ (photons/100 Thermal-Neutron captures)
9.20	2	20	2	
9.10	1	10	1	1.2
9.00	1	10	1	
8.90	2	20	2	
8.60	<0.5	5	<0.5	
8.50	1	10	1	0.1
8.40	<0.5	5	<0.5	
8.30	1	10	1	1.0
8.20	2	15	2	
8.10	1	10	1	
8.00	4	30	4	
7.86	9	70	10	13.9
7.70	2	15	2	0.8
7.60	<0.5	5	<0.5	
7.50	2	15	2	
7.35	18	130	20	
7.20	8	60	9	~5.6
7.00	9	65	10	~9.7
6.60	7	45	8	~3.5
6.40	7	45	8	~2.6
6.30	3	20	3	~2.7
6.20	3	20	3	
6.10	1	5	1	~2.2
6.00	<u>9</u>	<u>55</u>	10	~1.9
Total	93	695		

for the stronger gamma rays from thermal-neutron capture are about the same as the values of $\bar{I}_\gamma(E_\gamma)$ for resonance capture (comparing, for example, the values for the 7.86- and ~ 7.00 -MeV gamma rays), but there are important differences in the two spectra.

Calcium. The foreground runs for 1 in. of calcium oxide enclosed in an aluminum container were 323-A for the boron-filtered beam and 324-B for the cadmium-covered beam. They were then normalized to 100 Kw and 200 min (see appendix). The gain varied between 0.0213 MeV/channel at low channel numbers to 0.0220 MeV/channel at high channel numbers for both runs. The net responses were formed and expressed in units of counts/MeV/100 Kw/200 min. They appear as the dashed lines in Figs. 28, 29, and 30.

ANISN calculations were made of the above configuration to obtain accurate values of the spectral fluxes from 15 MeV to thermal energies integrated both over the aluminum container and over the calcium oxide. The spatially integrated aluminum fluxes were then compared with those from a similar calculation for a 0.25-in.-thick aluminum slab. The contributions to the net responses from the aluminum container filled with calcium oxide were then obtained as the ratio of the integrated fluxes above 1 MeV in the two aluminum configurations times the responses of the aluminum slab corrected for incident thermal-neutron capture (see Figs. 16 and 17). This ratio was so closely just the ratio of the total thickness of the aluminum container (twice the values in the last column of Table III) to that of the slab for the ANISN calculations run that it was approximated in this fashion. Thus the contributions from the aluminum container to the net responses in Figs. 28, 29, and 30 were calculated to be $2 \times 0.073/0.25 = 0.583$ of the responses corrected for incident thermal-neutron capture in Figs. 16 and 17.

The contribution to the net responses of capture from the incident thermal neutrons was calculated for each beam filter in a manner completely analogous to that previously described for iron, i.e., the number of captures occurring in both the aluminum and calcium oxide were calculated using the techniques described in ref. 6, the number of photons

Fig. 28. Measured and Calculated Net Responses from 1 inch of Calcium Oxide in an Aluminum Container for the Energy Region 1.1-4.0 MeV.

Fig. 29. Measured and Calculated Net Responses from 1 inch of Calcium Oxide in an Aluminum Container for the Energy Range 4.0-7.0 MeV.

Fig. 30. Measured and Calculated Net Responses from 1 inch of Calcium Oxide in an Aluminum Container for the Energy Range 7.0-8.2 MeV.

reaching the crystal calculated by using the spectral intensities, $I_Y(E_Y)$, from ref. 6, attenuating the gamma rays by using values of $A_Y(E_Y)$ from Fig. 31, and folding. The contribution to the net response of gamma rays arising from capture of the slowed-down-to-thermal neutrons was calculated by ANISN to be negligible in the case of the boron-filtered beam even though the calcium oxide contained 2.7 weight percent water. This contribution had to be considered, however, in the calculation of the response for the cadmium-filtered beam, and will be discussed later.

The result of subtracting both the contributions from the aluminum container and capture of the incident thermal neutrons appears as the two solid curves through the open diamonds in Figs. 28, 29, and 30. It is to be observed that the response of the aluminum (which is the major part involved in the subtraction since the contribution from incident thermal-neutron capture in calcium is small) contributes roughly one-half of the response obtained from the filled container. In regions where the aluminum response is particularly intense, as around 7.72 MeV, 2.22 MeV, 3.02 MeV, and 1.72 MeV, small errors in calculating the aluminum response result in large errors in the calcium response, since in these regions the aluminum contributes up to 80% of the total. Thus, discrepancies in the responses and corresponding uncertainties in the unfolded intensities will occur with the result that the values of production cross sections for calcium should be attended by large errors in these regions.

The unfolded intensities derived from the corrected net response for the boron-filtered beam produced a response denoted by the open circles in the lower curve of Figs. 28, 29, and 30. These spectral intensities are presented in Table XXVI along with values of their production cross sections averaged over the incident spectrum above 1 MeV presented in Fig. 4. These cross sections were calculated from the corresponding intensities in the following manner:

$$I_Y(E_Y) \text{ photons/100 Kw/200 min} = \frac{\bar{\sigma}(E_Y) \times 10^{-3}}{4\pi} \times \frac{0.6023}{56.1} \times 1.24 \times 0.94$$

$$\times \frac{(\pi/4) \times 25}{144 \times 400} \times \int_0^T dx \int_{1 \text{ MeV}}^{\infty} \phi(E, x) dE \times A_Y(E_Y) ,$$

Fig. 31. Attenuation Function for Secondary Gamma Rays from Calcium Oxide.

Table XXVI. Unfolded Intensities and Average Production
Cross Sections for Calcium and Oxygen

E_{γ} (MeV)	γ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_{\gamma})$ mb	E_{γ} (MeV)	γ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_{\gamma})$ mb
8.20	1.7(4) ^a	0.22	4.15	4.0(4)	0.63
8.10	1.9(4)	0.25	4.05	1.7(4)	0.27
8.00	1.0(4)	0.13	3.95	4.4(5)	5.7
7.50	1.0(4)	0.13	3.89	7.15(5)	11.5
7.40	1.3(4)	0.17	3.75	1.22(6)	20.0
7.30	1.2(4)	0.16	3.63	1.5(5)	2.5
7.20	2.3(4)	0.31	3.55	9.0(4)	1.5
7.14	6.8(4)	34*	3.45	1.3(5)	2.2
7.02	2.5(4)	0.34	3.35	9.0(4)	1.5
6.90	9.5(4)	47*	3.24	1.8(5)	3.1
6.76	1.9(4)	0.26	3.16	2.0(4)	0.35
6.44	6.0(4)	0.83	3.10	6.0(4)	1.0
6.30	4.3(4)	0.60	3.00	3.0(4)	0.55
6.13	4.0(5)	110*	2.90	9.0(4)	1.6
5.94	8.5(4)	1.2	2.80	2.2(5)	4.0
5.83	2.0(4)	0.28	2.70	1.6(5)	3.0
5.66	9.0(4)	1.3	2.65	5.2(4)	1.0
5.54	2.0(4)	0.29	2.60	1.75(5)	3.3
5.34	1.5(4)	0.22	2.50	2.4(5)	4.6
5.26	7.0(4)	1.0	2.44	1.0(5)	2.0
5.20	1.3(4)	0.19	2.23	3.2(5)	6.6
5.00	2.0(4)	0.30	2.13	1.0(5)	2.1
4.90	2.4(4)	0.36	2.06	4.2(5)	9.0
4.80	5.0(4)	0.75	1.98	2.6(5)	5.7
4.68	4.0(4)	0.60	1.91	1.9(5)	4.3
4.52	2.5(4)	0.38	1.83	1.9(5)	4.4
4.45	3.0(4)	0.46	1.77	3.1(5)	7.3
4.35	3.3(4)	0.51	1.71	2.0(4)	0.5
4.25	5.0(4)	0.78	1.63	5.0(5)	12.2

Table XXVI (cont.)

E_γ (MeV)	$\mathcal{Q}_\gamma$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb	E_γ (MeV)	$\mathcal{Q}_\gamma$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb
1.56	2.4(5)	6.0	1.27	2.4(5)	6.9
1.50	2.0(5)	5.1	1.22	2.2(5)	6.5
1.44	3.3(5)	8.7	1.17	6.65(5)	20.1
1.38	5.4(5)	14.5	1.11	3.85(5)	12.0
1.32	3.0(5)	8.4			

^aRead as 1.7×10^4 .

* Inelastic lines from oxygen.

where $\bar{\sigma}(E_Y)$ is the average production cross section above 1 MeV in millibarns, $A_Y(E_Y)$ is taken from Fig. 31, and

$$\int_0^T dx \int_{1 \text{ MeV}}^{\infty} \varphi(E, x) dE \approx \frac{1.13 T}{\cos 45^\circ} \int_{1 \text{ MeV}}^{\infty} J(E) dE = 3.61 \times 10^{14}$$

neutron cm/100 Kw/200 min ,

where the factor 1.13 was calculated by ANISN. Thus,

$$\bar{\sigma}(E_Y) = 8.15 \times 10^{-6} \varphi_Y(E_Y)/A_Y(E_Y) \text{ millibarns for calcium.}$$

Using a factor of 1.0 for neutrons above 6.3 MeV that was calculated by ANISN,

$$\int_0^T dx \int_{6.5 \text{ MeV}}^{\infty} \varphi(E, x) dE = \frac{T}{\cos 45^\circ} \int_{6.5 \text{ MeV}}^{\infty} J(E) dE = 1.65 \times 10^{13}$$

neutron cm/100 Kw/200 min ,

so that

$$\begin{aligned} \varphi_Y(6.13) &= \frac{\bar{\sigma}(6.13) \times 10^{-3}}{4\pi} \times 0.6023 \times 1.24 \times \left[\frac{0.94}{56.1} + \frac{0.027}{18} \right] \\ &\times \frac{(\pi/4) \times 25}{144 \times 400} \times 1.65 \times 10^{13} \times A_Y(6.13) \end{aligned}$$

and

$$\bar{\sigma}(6.13) = 1.63 \times 10^{-4} \varphi_Y(6.13)/A_Y(6.13) \text{ for oxygen,}$$

where $\bar{\sigma}(6.13)$ is the average production cross section for neutrons above 6.5 MeV. Similarly,

$$\bar{\sigma}(E_Y) = 3.00 \times 10^{-4} \varphi_Y(E_Y)/A_Y(E_Y) \text{ for oxygen,}$$

where $\bar{\sigma}(E_Y)$ is the average production cross section for neutrons above 7.4 MeV and E_Y has the values 6.90 and 7.14 MeV.

Since none of the gamma rays appearing in Table XXVI were designated as arising from resonance capture, the cadmium-filtered response was calculated by multiplying the intensities in Table XXVI by 1.33, adding in the intensities from $1/V$ capture and also from capture of those neutrons slowed down by the water present to thermal energies, and folding.

The intensities from $1/V$ capture were calculated using $\sigma_{n\gamma}(E) = \frac{0.44 \sqrt{0.0253}}{\sqrt{E}} = 0.070/\sqrt{E}$ barns,

so that

$$\int_0^T dx \int_{0.5 \text{ eV}}^{\infty} \sigma_{n\gamma}(E) \varphi(E, x) dE = \frac{1.22 T}{\cos 45^\circ} \int_{0.5 \text{ eV}}^{\infty} \frac{0.070}{\sqrt{E}} x \frac{4.0 x 10^{12}}{E} dE =$$

$$3.47 \times 10^{12} \text{ neutron barn cm/100 Kw 200 min ,}$$

where the factor 1.22 was calculated by ANISN. Thus

$$\phi_{\gamma 1/V \text{ Ca}}(E_\gamma) = \frac{3.47 \times 10^{12}}{4\pi} \times \frac{(\pi/4) \times 25}{144 \times 400} \times \frac{0.6023}{56.1} \times 1.24 \times 0.94 I_{\gamma \text{ Ca}}(E_\gamma) A_\gamma(E_\gamma)$$

$$= 1.18 \times 10^6 I_{\gamma \text{ Ca}}(E_\gamma) A_\gamma(E_\gamma) \text{ photons/100 Kw/200 min ,}$$

where $I_{\gamma \text{ Ca}}(E_\gamma)$, in photons/capture, is taken from ref. 6 and $A_\gamma(E_\gamma)$ from Fig. 31.

The intensities arising from capture in the aluminum as well as in the calcium oxide at energies below 0.4 eV due to slowing down were calculated with the help of ANISN. For the aluminum

$$\phi_{\gamma \text{ Al}}(E_\gamma) \approx 8.0 \times 10^4 I_{\gamma \text{ Al}}(E_\gamma) A_\gamma(E_\gamma) \text{ photons/100 Kw/200 min ,}$$

and for the calcium,

$$\phi_{\gamma \text{ Ca}}(E_\gamma) \approx 3.0 \times 10^5 I_{\gamma \text{ Ca}}(E_\gamma) A_\gamma(E_\gamma) \text{ photons/100 Kw/200 min .}$$

The calculated net response for the cadmium-filtered beam is indicated by the upper series of open circles appearing in Figs. 28, 29, and 30. A comparison of this response with the measured response shows that below 7 MeV the agreement is within $\pm 20\%$ everywhere except in the regions of intense aluminum response centered around 1.72 MeV and 2.22 MeV. There is strong evidence that gamma rays in the region 4.0-5.5 MeV arising from resonance capture occur, and it seems likely that these captures occur at neutron energies below 10 keV.

The average production cross sections for calcium, summed over 0.5-MeV gamma-ray intervals and corrected for impurities, appear in Table XXVII, where they are compared with values from ref. 3 averaged over the neutron spectrum above 1 MeV in Fig. 4.

The agreement between the two sets of cross sections is considered adequate especially when any type of transport problem is considered, since the important cross sections are those above 2 MeV and here the agreement is quite good. From an inspection of Table XXVI, relatively intense gamma rays are observed in the vicinity of 1.17, 1.38, 1.63, 2.06, 3.75, and 3.89 MeV. The remaining intensities in Table XXVI are from weaker transitions superimposed upon a continuum. There is no data in ref. 1 with which to compare the average production cross sections for the more intense lines.

The values obtained for the average production cross sections above threshold for the 6.13 MeV and ~ 6.92 MeV + ~ 7.12 MeV gamma rays from oxygen agree reasonably well with values found earlier from borated water. The values obtained from calcium oxide are less reliable than those from the earlier measurement using borated water, however, because of the presence of gamma rays from the calcium.

Potassium. The foreground runs for 1 in. of potassium carbonate in an aluminum container were 372-A for the boron-filtered beam and 371-C for the cadmium-filtered beam. They were then normalized to 100 Kw and 200 min (see appendix). The gain for the cadmium-filtered run varied between 0.0217 MeV/channel at the low channel numbers to 0.0230 MeV/channel at the high channel numbers; for the boron-filtered run, $0.0217 \leq G_c \leq 0.0232$ MeV/channel. The net responses were formed and appear in Figs. 32, 33, and 34 as the dashed lines.

Table XXVII. Comparison of Average Production Cross Sections in Millibarns from Present Results with Data of Troubetzkoy for Calcium

	Gamma-Ray Energy Range (MeV)												Total
	1.1-1.5	1.5-2	2-2.5	2.5-3	3-3.5	3.5-4	4-4.5	4.5-5	5-5.5	5.5-6	6-6.5	6.5-7.5	
Present Results	64	36.5	18	16.5	8.0	40.5	2.5	1.9	1.6	< 3.0	< 1.4	< 1.1	≤ 195
Troubetzkoy <u>et al.</u> ³	20	24	20	15	11.5	27		3.4		1.3		0.3	123

Fig. 32. Measured and Calculated Net Responses from 1 inch of Potassium Carbonate in an Aluminum Container for the Energy Range 1.1-3.9 MeV.

Fig. 33. Measured and Calculated Net Responses from 1 inch of Potassium Carbonate in an Aluminum Container for the Energy Range 3.9-6.9 MeV.

Fig. 34. Measured and Calculated Net Responses from 1 inch of Potassium Carbonate in an Aluminum Container for the Energy Range 6.9-8.0 MeV.

The contribution of the aluminum container to the net response for each beam filter was scaled to the net response from 0.25 in. of aluminum with the help of calculations using ANISN in a completely analogous manner to that described for the calcium oxide. Again it was found that the contribution of the container in the present geometry would be adequately approximated as $2 \times 0.0625/0.25 = 0.50$ times the net response corrected for incident thermal-neutron capture for each beam filter plotted in Figs. 16 and 17.

The contribution from capture of the incident thermal neutrons was calculated for each beam filter in the same manner as described earlier for calcium oxide. The contribution to the net response of gamma rays arising from capture of the slowed-down-to-thermal neutrons was calculated by ANISN to be negligible in the case of the boron-filtered beam even though the potassium carbonate contained 5.4 weight percent water. As with the calcium oxide, however, this contribution had to be considered in the case of the cadmium-filtered beam, and will be discussed later.

The result of subtracting both the contributions from the aluminum container and capture of the incident thermal neutrons appears as the two solid curves through the open diamonds in Figs. 32, 33, and 34. Again the contribution from the aluminum is roughly one-half of the total response from the filled container, making it difficult to obtain good data for potassium in the regions centered around 1.72, 2.22, and 3.02 MeV.

The unfolded intensities derived from the corrected net response for the boron-filtered beam produce a response denoted by the open circles in the lower curve of Figs. 32, 33, and 34. These spectral intensities are presented in Table XXVIII along with values of their production cross sections averaged over the incident spectrum above 1 MeV for potassium and above threshold for oxygen and carbon. These cross sections were calculated from the corresponding intensities thusly. For potassium,

$$I_Y(E_Y) \text{ photons/100 Kw/200 min} = \frac{\bar{\sigma}(E_Y) \times 10^{-3}}{4\pi} \times \frac{0.6023}{138} \times 2 \times 0.819 \times 0.946$$

$$\times \frac{(\pi/4) \times 25}{144 \times 400} \times \int_0^T dx \int_{1 \text{ MeV}}^{\infty} \phi(E, x) dE \times A_Y(E_Y) ,$$

Table XXVIII. Unfolded Intensities and Average Production Cross Sections for Potassium, Oxygen, and Carbon

E_γ (MeV)	γ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb	E_γ (MeV)	γ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb
7.86	2.5(4) ^a	capture	4.16	8.0(4)	2.3
7.60	6.7(3)	0.15	4.05	1.0(5)	2.9
7.50	1.5(4)	0.35	3.95	1.5(5)	4.3
7.40	4.0(3)	0.10	3.91	5.0(4)	1.5
7.25	3.5(4)	0.85	3.87	2.4(5)	7.0
7.11	4.0(4)	22*	3.80	1.0(4)	0.3
7.00	2.5(4)	0.35	3.70	1.2(5)	3.6
6.92	4.5(4)	26*	3.60	2.2(5)	6.6
6.70	1.8(4)	0.45	3.48	5.0(4)	1.5
6.60	2.8(4)	0.70	3.40	8.0(4)	2.4
6.50	1.2(4)	0.30	3.30	3.0(4)	1.0
6.40	3.0(4)	0.75	3.20	7.0(4)	2.2
6.30	4.3(4)	1.1	3.10	1.0(5)	3.2
6.13	2.8(5)	90*	3.02	5.0(5)	16
6.00	5.0(4)	1.3	2.90	2.5(5)	8.2
5.90	3.0(4)	0.75	2.82	1.1(6)	36
5.80	1.0(4)	0.25	2.70	2.3(5)	7.7
5.75	4.0(4)	1.0	2.60	2.5(5)	8.6
5.56	8.0(4)	2.1	2.52	1.0(6)	35
5.50	2.0(4)	0.50	2.40	1.5(5)	5.3
5.33	2.5(4)	0.65	2.24	1.5(5)	5.5
5.25	9.0(4)	2.4	2.17	4.0(5)	15
5.15	2.0(4)	0.55	2.10	2.0(5)	7.6
4.97	6.0(4)	1.6	2.05	1.0(5)	3.8
4.92	1.3(4)	0.35	2.00	7.0(4)	2.7
4.85	2.0(4)	0.55	1.95	2.0(5)	7.9
4.74	1.4(5)	3.8	1.90	1.0(5)	4.0
4.55	8.5(4)	2.4	1.85	1.5(5)	6.1
4.42	4.4(5)	180†	1.78	1.0(5)	4.1
4.30	3.2(4)	0.90	1.72	4.3(5)	18

Table XXVIII (cont.)

E_γ (MeV)	$\mathcal{Q}_\gamma$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb	E_γ (MeV)	$\mathcal{Q}_\gamma$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb
1.65	2.0(5)	8.6	1.35	3.5(5)	17
1.60	2.2(5)	9.6	1.29	7.0(5)	35
1.55	2.5(5)	11	1.24	3.5(5)	18
1.50	2.2(5)	10	1.18	3.8(5)	20
1.45	1.7(5)	8	1.12	4.4(5)	24
1.40	2.5(5)	12			

^aRead as 2.5×10^4 .

* Inelastic lines from oxygen.

[†]Inelastic line from carbon.

where $\bar{\sigma}(E_Y)$ is the average production cross section above 1 MeV in millibarns, $A_Y(E_Y)$ is taken from Fig. 35, and

$$\int_0^T dx \int_{1 \text{ MeV}}^{\infty} \varphi(E, x) dE \approx \frac{1.10 T}{\cos 45^\circ} \int_{1 \text{ MeV}}^{\infty} J(E) dE = 3.51 \times 10^{14} \text{ neut. cm/100 Kw/200 min,}$$

where the factor 1.10 was calculated by ANISN. Hence,

$$\bar{\sigma}(E_Y) = 1.53 \times 10^{-5} \mathcal{A}_Y(E_Y)/A_Y(E_Y) \text{ millibarns}$$

for gamma rays from fast-neutron interactions in potassium.

For gamma rays from oxygen, the corresponding expressions for average production cross sections above threshold are

$$\begin{aligned} \mathcal{A}_Y(E_Y) &= \frac{\bar{\sigma}(E_Y) \times 10^{-3}}{4\pi} \times 0.6023 \times 0.819 \times \left[\frac{3 \times 0.946}{138} + \frac{0.054}{18} \right] \\ &\times \frac{(\pi/4) \times 25}{144 \times 400} \times \frac{0.99 T}{\cos 45^\circ} \int_{\text{threshold}}^{\infty} J(E) dE \times A_Y(E_Y) , \end{aligned}$$

whence

$$\bar{\sigma}(6.13) = 1.92 \times 10^{-5} \mathcal{A}_Y(6.13)/A_Y(6.13) \text{ millibarns}$$

for neutrons above 6.5 MeV, and

$$\bar{\sigma}(E_Y) = 3.53 \times 10^{-4} \mathcal{A}_Y(E_Y)/A_Y(E_Y) \text{ millibarns}$$

for neutrons above 7.4 MeV, and where $E_Y = 6.92 \text{ MeV}$ and 7.11 MeV .

For the strong inelastic gamma ray from carbon

$$\begin{aligned} \mathcal{A}_Y(4.42) &= \frac{\bar{\sigma}(4.42) \times 10^{-3}}{4\pi} \times \frac{0.6023}{138} \times 0.819 \times 0.946 \times \frac{(\pi/4) \times 25}{144 \times 400} \times \frac{1.02 T}{\cos 45^\circ} \\ &\times \int_{4.8 \text{ MeV}}^{\infty} J(E) dE \times A_Y(4.42) \text{ photons/100 Kw/200 min} , \end{aligned}$$

whence

$$\bar{\sigma}(4.42) = 2.25 \times 10^{-4} \mathcal{A}_Y(4.42)/A_Y(4.42) \text{ millibarns}$$

Fig. 35. Attenuation Function for Secondary Gamma Rays from Potassium Carbonate.

for neutrons above 4.8 MeV.

The cadmium-filtered response was calculated by multiplying the 7.86-MeV gamma-ray intensity by 2.2, all of the remaining intensities appearing in Table XXVIII by 1.33, adding in the intensities from 1/V capture and from capture of those neutrons slowed down by the water present to thermal energies, and folding.

The intensities from 1/V capture were calculated using $\sigma_{n\gamma}(E) = \frac{2.08 \sqrt{0.0253}}{\sqrt{E}} = 0.331/\sqrt{E}$ barns, so that

$$\int_0^T dx \int_{0.5 \text{ eV}}^{\infty} \sigma_{n\gamma}(E) \varphi(E, x) dE = \frac{1.2 T}{\cos 45^\circ} \int_{0.5 \text{ eV}}^{\infty} \frac{0.331}{\sqrt{E}} \times \frac{4.0 \times 10^{12}}{E} dE$$

$$= 1.61 \times 10^{13} \text{ neutron barn cm/100 Kw 200 min ,}$$

where the factor 1.2 was calculated by ANISN. Thus

$$\mathcal{A}_{\gamma_{1/V} K}^{(E_Y)} = \frac{1.61 \times 10^{13}}{4\pi} \times \frac{(\pi/4) \times 25}{144 \times 400} \times \frac{0.6023}{138} \times 2 \times 0.819 \times 0.945$$

$$\times I_{\gamma_K}(E_Y) A_Y(E_Y)$$

$$= 2.95 \times 10^6 I_{\gamma_K}(E_Y) A_Y(E_Y) \text{ photons/100 Kw/200 min ,}$$

where $I_{\gamma_K}(E_Y)$, in photons/capture, is taken from ref. 6 and $A_Y(E_Y)$ from Fig. 35.

The intensities arising from capture in the aluminum as well as the potassium carbonate at energies below 0.4 eV due to slowing down by the water present were also calculated with the help of ANISN. For the aluminum,

$$\mathcal{A}_{\gamma_{Al}}(E_Y) \approx 1.5 \times 10^5 I_{\gamma_{Al}}(E_Y) A_Y(E_Y) \text{ photons/100 Kw/200 min,}$$

and for the potassium

$$\mathcal{A}_{\gamma_K}(E_Y) \approx 6.0 \times 10^5 I_{\gamma_K}(E_Y) A_Y(E_Y) \text{ photons/100 Kw/200 min .}$$

The calculated net response of the cadmium-filtered beam is indicated by the upper series of open circles appearing in Figs. 32, 33, and 34. A comparison of this response with the measured response shows that the agreement is to within $\pm 25\%$ everywhere except in the region of strong aluminum response centered around 1.72 and 2.30 MeV. The entire calculated response averages about 15% above the measured response for all energies below 3.1 MeV, which probably indicates an error in the relative normalizations of the boron-filtered and cadmium-filtered runs. No evidence of any resonance captures below 10 keV in potassium appears in Figs. 32, 33, and 34, perhaps because of the possible relative error mentioned above, and only the 7.86-MeV gamma ray was assigned capture status arising from neutrons above 10 keV.

The average production cross sections for potassium summed over 0.5-MeV gamma-ray intervals appear in Table XXIX, where they are compared with values from ref. 4 averaged over the neutron spectrum above 1 MeV in Fig. 4.

The two sets of values appearing in Table XXIX make a fascinating comparison. For gamma rays above 2.5 MeV the agreement is extremely good, but for gamma rays below 2.5 MeV there is no agreement at all. We suspect that some of the disagreement is caused by the neglect in the calculations of gamma rays from inelastic scattering in ^{41}K , which occurs with a relative abundance of about 7% in natural potassium. Thus the relatively strong gamma-ray transition at 1.29 MeV which is seen experimentally and fits the decay scheme from ^{41}K (see ref. 18) was not included in the data of ref. 4. Another possible source of the disagreement is the relatively large cross section for charged particle reactions in potassium, i.e., (n,p) , (n,α) , etc., which were probably not considered as gamma-ray producing reactions in the calculations, but which well might contribute to the gamma rays in the 1.1-2.5 MeV range observed experimentally.

From Table XXVIII, the stronger gamma rays observed from fast-neutron interactions in potassium occur in the vicinity of 1.12, 1.29, 1.72, 2.17, 2.52, 2.82, 3.02, 3.60, 3.87, 3.95, 4.74, and 5.25 MeV. The 1.72-MeV line may be spurious, but the 2.17-MeV and 3.02-MeV lines are real even though they occur in a region of strong aluminum intensity.

Table XXIX. Comparison of Average Production Cross Sections in Millibarns from Present Results with Data of Troubetzkoy for Potassium

	Gamma-Ray Energy Range (MeV)													Total
	1.1-1.5	1.5-2	2-2.5	2.5-3	3-3.5	3.5-4	4-4.5	4.5-5	5-5.5	5.5-6	6-6.5	6.5-7	7-7.5	
Present Results	133	80	39	96	26	23.3	6.1	8.7	3.6	4.6	< 3.1	< 1.4	< 1.5	≤ 425
Troubetzkoy et al. ⁴	2	3.5	3.5	70	22	30		11.5		6.8		2.0		150

A comparison of the average production cross sections for three relatively intense lines in Table XXVIII with values obtained by averaging the data in ref. 1 over the incident spectrum from 1-4 MeV appears in Table XXX.

Table XXX. Comparison of Average Production Cross Sections in Millibarns from Present Results with BNL Data for Several Intense Gamma Rays in Potassium

Gamma-Ray Energy	2.52 MeV	2.81 MeV	3.02 MeV
Present Results	35	36	16
BNL-325 ¹	19	16	7

The disagreement in Table XXX is due primarily to the shorter range of neutron energies over which the averaging was performed. Since the production cross sections for all three gamma rays are still increasing at 4 MeV, it is reasonable to conclude that averaging over a greater range of neutron energies would increase the BNL-325 values sufficiently to bring about good agreement with the present results. As a matter of fact, were the production cross sections to remain constant at their value at 4 MeV for all energies above 4 MeV, the BNL-325 values averaged over the complete range above 1 MeV would be 36 mb, 35 mb, and 21 mb respectively. Thus, it is concluded that the two sets of data of $4\pi (d\sigma/d\Omega)$ (90 deg) are essentially in agreement.

The values obtained for the average production cross sections above threshold for the 6.13-MeV and 6.92 MeV + 7.12-MeV gamma rays from oxygen in potassium carbonate are in very good agreement with the values previously found from analysis of the borated water. The present analysis yields the values 90 mb and 48 mb respectively, compared with the more accurately determined earlier values of 90 mb and 55 mb. For carbon, the value of 180 mb for $\bar{\sigma}(4.42)$ determined from the analysis of the potassium carbonate compares reasonably well with the more accurately determined value of 130 mb obtained from analysis of the polyethylene slab.

The average production cross section over the range 10 keV-200 keV of the 7.86-MeV gamma ray that arises from resonance capture in potassium is ~ 3.5 mb, but it is likely that this gamma ray is spurious.

Silicon. The foreground runs for 1 in. of silicon dioxide in an aluminum container were 311-A for the boron-filtered beam and 310-A for the cadmium-filtered beam. They were then normalized to 100 Kw and 200 min (see appendix). For the boron-filtered run, $0.0211 \leq G_c \leq 0.0220$ MeV/channel, and for the cadmium-filtered run, $0.0206 \leq G_c \leq 0.0216$. The net responses for each run were formed and are plotted in units of counts/MeV/100 Kw/200 min in Figs. 36, 37, and 38 as the dashed lines.

The contribution of the aluminum container to the net response for each beam filter was scaled to the corrected net response from 0.25 in. of aluminum with the help of calculations using ANISN. It was found that the scaling factor $(2 \times 0.073/0.25) = 0.583$ was again adequate, and the contribution of the container to the net response for each beam filter was calculated in a manner completely analogous to that described for the case of calcium oxide.

The contribution from capture of incident thermal neutrons was calculated in the following manner. First, the number of captures occurring in the aluminum container and 1 in. of sand were calculated using the code developed and described in ref. 6. The number of photons of energy E_γ reaching the crystal was then calculated using the spectral data of ref. 6 for both aluminum and silicon and the attenuation function appearing in Fig. 39. These intensities were found to be, for the case of the boron-filtered beam,

$$\phi_{\gamma_{Si}}(E_\gamma) = 8.65 \times 10^4 I_\gamma(E_\gamma) A_\gamma(E_\gamma) \text{ photons/100 Kw/200 min}$$

for silicon, and

$$\phi_{\gamma_{Al}}(E_\gamma) \approx 5.0 \times 10^4 I_{\gamma_{Al}}(E_\gamma) A_\gamma(E_\gamma) \text{ photons/100 Kw/200 min}$$

for aluminum, and very closely twice these values for the cadmium-filtered beam. These intensities were then folded to find the contribution to the net response for each beam filter.

Fig. 36. Measured and Calculated Net Responses from 1 inch of Silicon Dioxide in an Aluminum Container for the Energy Range 1.1-4.0 MeV.

Fig. 37. Measured and Calculated Net Responses from 1 inch of Silicon Dioxide in an Aluminum Container for the Energy Range 4.0-7.0 MeV.

Fig. 38. Measured and Calculated Net Responses from 1 inch of Silicon Dioxide in an Aluminum Container for the Energy Range 7.0-8.2 MeV.

Fig. 39. Attenuation Function for Secondary Gamma Rays from Silicon Dioxide.

The result of subtracting both the contributions from the aluminum container and capture of the incident thermal neutrons appears as the two solid curves through the open diamonds in Figs. 36, 37, and 38. Except in the regions centered around 2.22 and 3.02 MeV, where the aluminum response is relatively intense, the response from the silicon dioxide should be intense enough to yield accurate unfolded spectral intensities.

The unfolded intensities derived from the corrected net response for the boron-filtered beam produced a response denoted by the open circles in the lower curve of Figs. 36, 37, and 38. These spectral intensities are presented in Table XXXI along with values of their corresponding production cross sections averaged over the incident spectrum above 1 MeV for silicon and above threshold for oxygen. These cross sections were calculated from the intensities using the following expressions. For silicon,

$$\begin{aligned} \mathcal{I}_Y(E_Y) \text{ photons/100 Kw/200 min} &= \frac{\bar{\sigma}(E_Y) \times 10^{-3}}{4\pi} \times \frac{0.6023}{60} \times 1.65 \\ &\times \frac{(\pi/4) \times 25}{144 \times 400} \times \int_0^T dx \int_{1 \text{ MeV}}^{\infty} \varphi(E, x) dE \times A_Y(E_Y) \quad , \end{aligned}$$

where $\bar{\sigma}(E_Y)$ is the average cross section above 1 MeV in millibarns, $A_Y(E_Y)$ is taken from Fig. 39, and

$$\int_0^T dx \int_{1 \text{ MeV}}^{\infty} \varphi(E, x) dE \simeq \frac{1.15 T}{\cos 45^\circ} \int_{1 \text{ MeV}}^{\infty} J(E) dE$$

$$= 3.68 \times 10^{14} \text{ neutron cm/100 Kw/200 min} \quad ,$$

where the factor 1.15 was calculated by ANISN. Hence,

$$\bar{\sigma}(E_Y) = 6.05 \times 10^{-6} \mathcal{I}_Y(E_Y) / A_Y(E_Y) \text{ millibarns}$$

for gamma rays from fast-neutron interactions in silicon. For oxygen, the expressions for average production cross sections above threshold are

Table XXXI. Unfolded Intensities and Average Production Cross Sections for Silicon and Oxygen

E_Y (MeV)	$\mathcal{Q}_Y$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_Y)$ mb	E_Y (MeV)	$\mathcal{Q}_Y$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_Y)$ mb
8.50	4.0(4) ^a	capture	3.90	7.0(4)	0.85
8.30	1.0(4)	capture	3.80	5.0(4)	0.6
8.20	1.5(4)	capture	3.71	1.75(5)	2.15
7.60	1.0(4)	0.1	3.60	8.0(4)	1.0
7.50	3.5(4)	0.35	3.50	6.0(4)	0.75
7.40	6.5(4)	0.65	3.40	2.5(4)	0.3
7.20	1.4(5)	1.4	3.30	5.5(4)	0.7
7.10	1.0(5)	22*	3.22	2.2(5)	2.9
7.00	2.5(4)	0.25	3.14	8.0(4)	1.05
6.93	2.0(5)	44*	3.07	3.0(4)	0.4
6.60	2.0(4)	0.2	2.90	4.0(4)	0.55
6.50	5.0(4)	0.5	2.84	4.3(5)	6.0
6.30	1.0(5)	1.05	2.76	7.0(4)	1.0
6.20	4.0(4)	0.4	2.68	4.0(4)	0.6
6.13	8.5(5)	105*	2.60	9.0(4)	1.3
5.96	2.0(4)	0.2	2.52	1.1(5)	1.6
5.60	5.0(4)	0.5	2.44	2.8(5)	4.2
5.50	8.0(4)	0.85	2.30	1.3(5)	2.0
5.20	1.5(4)	0.15	2.25	6.2(5)	9.6
5.10	7.0(4)	0.8	2.15	8.0(4)	1.3
5.00	2.0(4)	0.2	2.08	3.5(5)	5.7
4.90	6.3(4)	0.7	2.02	4.45(5)	7.4
4.80	1.5(4)	0.2	1.95	2.7(5)	4.6
4.70	1.0(4)	0.1	1.86	1.3(6)	22
4.65	1.0(4)	0.1	1.78	1.3(7)	231
4.51	1.8(5)	2.1	1.70	6.5(5)	12
4.30	4.0(4)	0.45	1.64	1.4(5)	2.6
4.20	2.0(4)	0.25	1.60	5.4(5)	10
4.10	3.0(4)	0.35	1.50	2.4(5)	4.8
4.00	1.5(4)	0.2	1.45	1.2(5)	2.4

Table XXXI (cont.)

E_γ (MeV)	$\mathcal{Q}_\gamma$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb	E_γ (MeV)	$\mathcal{Q}_\gamma$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb
1.40	3.9(5)	8.0	1.23	2.4(5)	5.3
1.34	4.5(5)	9.5	1.17	5.2(5)	12
1.28	7.5(5)	16	1.12	3.4(5)	8.1

^aRead as 4.0×10^4 .

* Inelastic lines from oxygen.

$$\begin{aligned} \phi_Y(E_Y) \text{ photons/100 Kw/200 min} &= \frac{\bar{\sigma}(E_Y) \times 10^{-3}}{4\pi} \times \frac{0.6023}{60} \times 2 \times 1.65 \\ &\times \frac{(\pi/4) \times 25}{144 \times 400} \times \frac{0.95 T}{\cos 45^\circ} \int_{\text{threshold}}^{\infty} J(E) dE \times A_Y(E_Y) \end{aligned}$$

whence

$$\bar{\sigma}(6.13) = 7.09 \times 10^{-5} \phi_Y(6.13)/A_Y(6.13) \text{ millibarns}$$

for neutrons above 6.5 MeV, and

$$\bar{\sigma}(E_Y) = 1.30 \times 10^{-4} \phi_Y(E_Y)/A_Y(E_Y) \text{ millibarns}$$

for neutrons above 7.4 MeV, and where $E_Y = 6.93 \text{ MeV}$ and 7.10 MeV .

The response for the cadmium-filtered beam was calculated by multiplying the intensities of the 8.2-, 8.3-, and 8.5-MeV gamma rays by 2.2, all of the remaining intensities appearing in Table XXXI by 1.33, adding in the intensities from 1/V capture and from capture of those neutrons slowed down to thermal energies, and folding.

The intensities from 1/V capture were calculated using $\sigma_{n\gamma}(E) = \frac{0.16 \sqrt{0.0253}}{\sqrt{E}} = 0.0255/\sqrt{E}$ barns, so that

$$\begin{aligned} \int_0^T dx \int_{0.5 \text{ eV}}^{\infty} \sigma_{n\gamma}(E) \phi(E, x) dE &= \frac{1.25 T}{\cos 45^\circ} \int_{0.5 \text{ eV}}^{\infty} \frac{0.0255}{\sqrt{E}} \times \frac{4.0 \times 10^{12}}{E} dE \\ &= 1.30 \times 10^{12} \text{ neutron barn cm/100 Kw 200 min} , \end{aligned}$$

where the factor 1.25 was calculated by ANISN. Thus,

$$\begin{aligned} \phi_{Y1/V}(E_Y) &= \frac{1.30 \times 10^{12}}{4\pi} \times \frac{(\pi/4) \times 25}{144 \times 400} \times \frac{0.6023}{60} \times 1.65 \times I_{Y\text{Si}}(E_Y) A_Y(E_Y) \\ &= 5.84 \times 10^5 I_{Y\text{Si}}(E_Y) A_Y(E_Y) \text{ photons/100 Kw/ 200 min} , \end{aligned}$$

where $I_{Y\text{Si}}(E_Y)$, in photons/capture, is taken from ref. 6 and $A_Y(E_Y)$ from

Fig. 39.

The intensities arising from capture in the aluminum container as well as the silicon dioxide at energies below 0.4 eV due to slowing down were also calculated with the help of ANISN. For the aluminum

$$\mathcal{J}_{\gamma\text{Al}}(E_{\gamma}) \approx 5.0 \times 10^4 I_{\gamma\text{Al}}(E_{\gamma}) A_{\gamma}(E_{\gamma}) \text{ photons/100 Kw/200 min} ,$$

and for the silicon,

$$\mathcal{J}_{\gamma\text{Si}}(E_{\gamma}) \approx 1.5 \times 10^5 I_{\gamma\text{Si}}(E_{\gamma}) A_{\gamma}(E_{\gamma}) \text{ photons/100 Kw/200 min.}$$

The calculated net response for the cadmium-filtered beam is indicated by the upper series of open circles appearing in Figs. 36, 37, and 38. A comparison of this response with the corrected measured response shows that the agreement is to within +15% everywhere below 7.2 MeV except in the region of high aluminum response centered around 2.25 and 3.0 MeV. No evidence of any resonance captures below 10 keV in silicon appears from Figs. 36, 37, and 38.

The average production cross sections for silicon summed over 0.5-MeV gamma-ray intervals appear in Table XXXII, where they are compared with values from ref. 2 averaged over the neutron spectrum above 1 MeV in Fig. 4.

The two sets of values appearing in Table XXXII are in reasonable agreement for gamma rays in the region 1.5-4.5 MeV and again above 5.5 MeV. They are in disagreement in the regions 1.1-1.5 MeV and 4.5-5.5 MeV, perhaps because the calculations only considered production from ^{28}Si , which occurs with a relative abundance of 92% in natural silicon. However, this does not explain the serious disagreement in the region 4.5-5.5 MeV, where the calculated values are a factor of six higher than the experimental values, and it is this discrepancy which would cause serious disagreement in a penetration calculation through a thick shield.

From Table XXXI, relatively intense gamma rays arising from fast-neutron interactions in silicon occur in the vicinity of 1.17, 1.28, 1.60, 1.78, 2.02, 2.25, 2.44, 2.84, 3.22, 3.71, 4.51, 4.90, and 5.10 MeV. The ~ 2.25 -MeV line is real even though it occurs in a region of strong aluminum response. The remaining intensities appearing in Table XXXI are weaker lines superimposed upon a continuum.

Table XXXII. Comparison of Average Production Cross Sections in Millibarns for Present Results with Data of Troubetzkoy for Silicon

	Gamma-Ray Energy Range (MeV)													Total
	1.1-1.5	1.5-2	2-2.5	2.5-3	3-3.5	3.5-4	4-4.5	4.5-5	5-5.5	5.5-6	6-6.5	6.5-7	7-7.5	
Present Results	61	287	30	11.1	5.4	5.4	1.2	3.2	1.2	1.5	< 1.5	< 0.7	< 2.6	≤ 412
Troubetzkoy <u>et al.</u> ²	3	356	14	13.4		12.7		16.3	8.3	4.6		1.8		431

A comparison of the average production cross sections for several relatively intense lines appearing in Table XXXI with values obtained by averaging the data in ref. 1 over the incident spectrum from 1-3 MeV and supplemented by averaging the data in a recently published Texas Nuclear report¹⁹ over the range 5-11 MeV is shown in Table XXXIII.

All of the cross sections appearing in Table XXXIII are values of $4\pi (d\bar{\sigma}/d\Omega)$ (90 deg), and the agreement between the present results and the data from refs. 1 and 19 is most satisfactory. Data from ref. 19 also indicate that the cross section averaged over the incident spectrum above 1 MeV in Fig. 4 for production of gamma rays in the interval 4.5-5.5 MeV is approximately 6 millibarns, a value which agrees much better with the present data (4.4 mb) than with the data of ref. 2 (24.6 mb). The present value of $4\pi (d\bar{\sigma}/d\Omega)$ (90 deg) for the 1.78-MeV gamma ray is about 20% lower than the value obtained from refs. 1 and 19, and this is consistent with the $\sim 20\%$ low value of 0.82 photons/capture obtained for this gamma ray from analysis of thermal-neutron capture in aluminum discussed in ref. 6. It is felt that this consistent 20% error is the result of an error in the interpolated response function for 1.78-MeV source photons. Since 1.78 MeV is about midway between the two calibrated response functions for 1.38-MeV and 2.22-MeV source photons, each of which is known to $\pm 10\%$, the interpolated response will be least accurate here, and can lead to additional errors of $\sim 10\%$ high in the response, so that it is reasonable that an unfolded intensity for photons in the vicinity of 1.78 MeV be $\sim 20\%$ low.

The anisotropy of the 1.78-MeV gamma ray is such that the cross section obtained as $4\pi (d\bar{\sigma}/d\Omega)$ (90 deg) should be increased by $\sim 30\%$ to obtain the true value.¹⁹

The values obtained for the average production cross sections above threshold for the 6.13-MeV and 6.93 MeV + 7.10-MeV gamma rays from oxygen in silicon dioxide are in reasonable agreement with the values previously found from analysis of the borated water. The present analysis yields the values 105 mb and 66 mb respectively, compared with the more accurately determined earlier values of 90 mb and 55 mb.

Table XXXIII. Comparison of Average Production Cross Sections in Millibarns from Present Results with Other Data for Several of the More Intense Gamma Rays in Silicon

Gamma-Ray Energy	1.28 MeV	1.78 MeV	2.02 MeV	2.20 MeV	2.41 MeV	2.84 MeV	3.20 MeV	4.58 MeV	4.91 MeV	5.10 MeV
Isotope	^{29}Si	^{28}Si	^{29}Si	^{30}Si	^{29}Si	^{28}Si	^{28}Si	^{28}Si	^{29}Si	^{28}Si
Present Results	16	231	7.4	9.6	4.2	6.0	2.9	2.1	0.7	0.8
Refs. 1 and 19	11	285	7	12	3	~ 6	~4	~2	~ 0.9	~ 2.5

Sulfur. The foreground runs for 1 in. of sulfur enclosed in an aluminum container were 505-A and 504-B for the boron-filtered beam and 504-A for the cadmium-filtered beam. They were then normalized to 100 Kw and 200 min (see appendix). The gain for these runs was the same, varying between 0.0222 MeV/channel at low channel numbers and 0.0231 MeV/channel at the high channel numbers. The net responses were formed and plotted in units of counts/MeV/100 Kw/200 min. These appear in Figs. 40 and 41 as the dashed lines.

The contribution of the aluminum container to the net response for each beam filter was scaled to the net response corrected for capture of incident thermal neutrons from 0.25 in. of aluminum. The scaling factor used was $(2 \times 0.0625/0.25) = 0.50$.

The contribution from capture of incident thermal neutrons was calculated in identically the same manner as described for the case of silicon dioxide. These intensities were found to be, for the case of the boron-filtered beam

$$\mathcal{I}_{\gamma S}(E_{\gamma}) = 1.52 \times 10^5 I_{\gamma S}(E_{\gamma}) A_{\gamma}(E_{\gamma}) \text{ photons/100 Kw/200 min}$$

for sulfur, and

$$\mathcal{I}_{\gamma Al}(E_{\gamma}) = 4.85 \times 10^4 I_{\gamma Al}(E_{\gamma}) A_{\gamma}(E_{\gamma}) \text{ photons/100 Kw/200 min}$$

for aluminum, where values of $I_{\gamma S}(E_{\gamma})$ and $I_{\gamma Al}(E_{\gamma})$, in photons/capture for sulfur and aluminum respectively, were taken from ref. 6 and $A_{\gamma}(E_{\gamma})$ is plotted in Fig. 42. For the cadmium-filtered beam, the corresponding intensities were very closely twice the above values.

The result of folding the above intensities, adding in the response of the aluminum container, and then subtracting the sum for each beam filter from the net response is denoted in Figs. 40 and 41 by the two solid curves through the open diamonds. Except in the vicinity of 3.0 MeV, the contribution from the aluminum container is about 50% or less of the total response from the filled container for pulse-height energies below 7 MeV.

ORNL-DWG 69-14660

Fig. 40. Measured and Calculated Net Responses from 1 inch of Sulfur Enclosed in an Aluminum Container for the Energy Range 1.1-3.4 MeV.

Fig. 41. Measured and Calculated Net Responses from 1 inch of Sulfur Enclosed in an Aluminum Container for the Energy Range 3.4-8.3 MeV.

Fig. 42. Attenuation Function for Secondary Gamma Rays from Sulfur.

The unfolded intensities derived from the corrected net response for the boron-filtered beam produced a response denoted by the lower series of open circles in Figs. 40 and 41. These spectral intensities are presented in Table XXXIV along with values of their corresponding cross sections averaged over the incident spectrum above 1 MeV presented in Fig. 4. These cross sections were calculated from the intensities in the following manner:

$$\begin{aligned} \phi_Y(E_Y) \text{ photons/100 Kw/200 min} &= \frac{\bar{\sigma}(E_Y) \times 10^{-3}}{4\pi} \times \frac{0.6023}{32} \times 0.5833 \times \\ &\times \frac{(\pi/4) \times 25}{144 \times 400} \times \int_0^T dx \int_{1 \text{ MeV}}^{\infty} \varphi(E, x) dE \times A_Y(E_Y) , \end{aligned}$$

where $\bar{\sigma}(E_Y)$ is the average cross section above 1 MeV in millibarns, $A_Y(E_Y)$ is taken from Fig. 42, and

$$\int_0^T dx \int_{1 \text{ MeV}}^{\infty} \varphi(E, x) dE \simeq \frac{1.1 T}{\cos 45^\circ} \int_{1 \text{ MeV}}^{\infty} J(E) dE = 3.52 \times 10^{14} \text{ neutron cm/100 Kw/200 min}$$

Hence, $\bar{\sigma}(E_Y) = 9.55 \times 10^{-6} \phi_Y(E_Y)/A_Y(E_Y)$ millibarns.

The response for the cadmium-filtered beam was calculated by multiplying the intensities appearing in Table XXXIV by 1.33, adding in the intensities from 1/V capture, and folding.

The intensities arising from 1/V capture were calculated using

$$\sigma_{n\gamma}(E) = \frac{0.51 \sqrt{0.0253}}{\sqrt{E}} = 0.081/\sqrt{E} \text{ barns, so that}$$

$$\begin{aligned} \int_0^T dx \int_{0.5 \text{ eV}}^{\infty} \sigma_{n\gamma}(E) \varphi(E, x) dE &\simeq \frac{1.2 T}{\cos 45^\circ} \int_{0.5 \text{ eV}}^{\infty} \frac{0.081}{\sqrt{E}} \times \frac{4.0 \times 10^{12}}{E} dE = \\ &= 3.95 \times 10^{12} \text{ neutron barn cm/100 Kw 200 min} . \end{aligned}$$

Table XXXIV. Unfolded Intensities and Average Production
Cross Sections for Sulfur

E_γ (MeV)	γ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb	E_γ (MeV)	γ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb
8.70	2.0(4) ^a	0.3	4.55	8.0(3)	0.1
8.30	5.0(3)	0.1	4.50	2.0(4)	0.3
8.10	5.0(3)	0.1	4.40	1.0(4)	0.2
8.00	5.0(3)	0.1	4.30	3.9(5)	6.8
7.60	7.0(3)	0.1	4.20	9.0(4)	1.6
7.50	1.9(4)	0.3	4.10	2.0(4)	0.3
7.40	1.5(4)	0.2	4.04	1.1(5)	1.9
7.30	1.0(4)	0.15	3.96	6.0(4)	1.1
7.20	1.15(4)	0.2	3.90	1.5(4)	0.3
6.93	5.05(4)	0.75	3.85	1.0(4)	0.2
6.56	3.25(4)	0.45	3.82	3.5(4)	0.6
6.34	1.3(4)	0.2	3.70	3.5(4)	0.6
6.25	1.5(4)	0.2	3.60	2.5(4)	0.5
6.05	2.0(4)	0.3	3.40	7.0(4)	1.3
5.90	5.0(4)	0.8	3.30	8.5(4)	1.6
5.80	4.0(4)	0.6	3.21	1.35(5)	2.6
5.70	4.3(4)	0.7	3.15	1.0(4)	0.2
5.60	1.5(4)	0.2	3.08	2.5(4)	0.5
5.50	4.6(4)	0.7	3.01	2.5(5)	5.0
5.40	7.5(3)	0.1	2.90	1.0(5)	2.0
5.32	2.9(4)	0.5	2.82	1.55(5)	3.1
5.20	1.3(4)	0.2	2.76	1.9(5)	3.9
5.10	2.5(4)	0.4	2.70	1.45(5)	3.0
4.97	7.0(4)	1.1	2.65	8.5(4)	1.8
4.90	2.2(4)	0.4	2.55	1.2(5)	2.6
4.80	2.0(4)	0.3	2.48	4.0(4)	0.8
4.75	8.5(3)	0.1	2.44	4.3(5)	9.4
4.70	6.0(4)	1.0	2.30	5.0(5)	11
4.60	3.0(4)	0.5	2.24	5.8(6)	131

Table XXXIV (cont.)

E_γ (MeV)	γ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb	E_γ (MeV)	γ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb
2.16	1.2(6)	28	1.58	2.5(5)	6.8
2.10	2.0(5)	4.7	1.52	1.5(5)	4.2
2.03	6.0(5)	14	1.46	1.5(5)	4.3
1.96	1.0(5)	2.4	1.40	1.5(5)	4.4
1.88	1.0(5)	2.5	1.35	3.5(5)	10
1.80	1.5(5)	3.8	1.29	9.0(5)	28
1.76	1.0(5)	2.6	1.22	4.5(5)	14
1.72	2.0(5)	5.2	1.17	1.6(5)	5.2
1.65	3.5(5)	9.3	1.13	2.5(5)	8.3

^aRead as 2.0×10^4 .

Thus

$$\begin{aligned} \phi_{Y_{1/V}}(E_Y) &= \frac{3.95 \times 10^{12}}{4\pi} \times \frac{(\pi/4) \times 25}{144 \times 400} \times \frac{0.6023}{32} \times 0.5833 \times I_{Y_S}(E_Y) A_Y(E_Y) \\ &= 1.18 \times 10^6 I_{Y_S}(E_Y) A_Y(E_Y) \text{ photons/100 Kw/200 min} , \end{aligned}$$

where $I_{Y_S}(E_Y)$, in photons/capture, is taken from ref. 6 and $A_Y(E_Y)$ from Fig. 42.

Since no ANISN calculations were performed for sulfur, the small contribution to the response from capture of neutrons slowed down to thermal energies was not explicitly calculated, but the factor 1.2 assumed above for the $1/V$ capture calculation was made a little high in order to include the effect of capture at thermal energies.

The calculated net response of the cadmium-filtered beam is indicated by the upper series of open circles appearing in Figs. 40 and 41. A comparison of this response with the corrected measured response shows that the agreement is to within $\pm 20\%$ everywhere below 7 MeV except in the vicinity of 3.0 MeV. There is some evidence of resonance capture below 10 keV in sulfur from this comparison, especially in the region above 7 MeV.

The average production cross sections for sulfur summed over 0.5-MeV gamma-ray intervals appear in Table XXXV, where they are compared with values from ref. 3 averaged over the neutron spectrum above 1 MeV in Fig. 4.

Except in the interval containing the intense 2.24-MeV gamma ray, the present results are about a factor of two higher than the calculations, and this could cause similar errors in thick shield penetration calculations.

From Table XXXIV, relatively intense gamma rays arising from fast-neutron interactions in sulfur occur in the vicinity of 1.29, 1.63, 2.03, 2.24, 2.44, 2.76, 3.01, 3.21, 4.04, 4.30, 4.70, 4.97, 5.50, 5.70, 5.90, 6.56, and 6.93 MeV. The ~ 3.01 -MeV line is probably real even though it occurs in a region of strong aluminum response. The remaining intensities appearing in Table XXXIV are weaker lines superimposed upon a continuum.

Table XXXV. Comparison of Average Production Cross Sections in Millibarns from Present Results with Data of Troubetzkoy for Sulfur

	Gamma-Ray Energy Range (MeV)													Total
	1.1-1.5	1.5-2	2-2.5	2.5-3	3-3.5	3.5-4	4-4.5	4.5-5	5-5.5	5.5-6	6-6.5	6.5-7	7-7.5	
Present Results	73	37	199	16.4	11.2	3.3	10.8	3.8	1.2	< 3.0	< 0.7	< 1.2	< 0.6	≤ 361
Troubetzkoy et al. ³	32	17	161	8	6.5	14.5		2.1		0.7		0.2		242

A comparison of the production cross section for the 2.24-MeV gamma ray averaged over the incident spectrum above 1 MeV obtained in this study with a value obtained by averaging the data in ref. 1 over the incident spectrum from 1 MeV to 5 MeV is shown in Table XXXVI. Considering the limited range of the latter, the agreement is considered adequate.

Table XXXVI. Comparison of Average Production Cross Section in Millibarns from Present Results with BNL Data for the 2.24-MeV Gamma Ray in Sulfur

Present Results	131
BNL-325 ¹	100

VI. Gamma-Ray Spectra at 90 deg from Fast-Neutron Interactions Using Response Functions Altered to Account for Effects of Gamma-Ray Scattering in Thick Slabs

In the preceding section all of the slabs that were analyzed, including those supported by an aluminum container, were sufficiently thin that the effects of slab scattering of the secondary gamma rays could be safely neglected. The elements to be analyzed in this section, however, were, either from design or availability, in such a form that these effects should be considered. Table XXXVII lists the four elements (including stainless steel) that were analyzed with modified response functions, along with other information pertaining to the slabs used.

The calculations necessary to obtain the altered response functions for the nickel, stainless steel, and barium oxide slabs have been described in great detail in ref. 6. To summarize, the first step was the calculation of the scattered gamma-ray spectrum escaping at 45 deg to the slab normal, by Monte Carlo techniques, that accompanied a spatially averaged source gamma ray produced in the slab and escaping in the same direction. These spectra were obtained for each of the source energies used to obtain the original response functions. The next step was to fold these spectra along with the source gamma ray with the original response functions. These altered response functions were then used in place of the

Table XXXVII. Elements Analyzed Using Altered Response Functions

Element	Form	Compound	Density (grams/cm ³)	Thickness (in.)	Main Impurities (% by wt)	Aluminum Container Side Thickness (in.)
Nickel	Metal		8.85	1.0	0.13% Mn, 0.05% Fe	-
Stainless Steel	Metal		7.88*	1.0	Type 321	-
Barium	Powder	BaO	2.73*	1.0	0.3% Sr	0.066
Sodium	Powder	Na ₂ CO ₃	0.737*	4.0	9.0% H ₂ O, 0.15% Cl	0.073

*Measured value.

original response functions for the particular slab that the calculations were performed. The altered response functions for the first three slabs in Table XXXVII that were used in this study were identically the same as those described for the same slabs in ref. 6. There is a small error involved in this, since the relative attenuation of 6 in. of lithium hydride to the scattered photons compared to that of the source photons was not considered. Thus all scattered photons accompanying a source photon were considered to be attenuated by the lithium hydride as the source photons are. This effect contributes at most a 20% error in the intensities of the scattered photons relative to the source photons reaching the crystal and at most a 10% error in the altered response function, which would occur at low pulse-height energies.

The remaining series of altered response functions to be described are those for the 4-in. sample of sodium carbonate. The response functions used in this case were identical to those used for the barium oxide (and for the 4-in. sulfur slab used and described in ref. 6), since the thickness of both slabs is very closely the same in photon scattering mean free paths.

The analysis of each of the four slabs appearing in Table XXXVI proceeded in an identical fashion to that described for all but the first two slabs presented in Table III. The only difference was that the response functions used in the unfolding of the boron-filtered response and the folding of the gamma-ray intensities to obtain a cadmium-filtered response were altered.

Nickel. The foreground runs for a 1-in.-thick nickel slab were 559-B and 560-A for the boron-filtered beam and 559-A for the cadmium-filtered beam. They were then normalized to 100 Kw and 200 min (see appendix). The gains for both runs varied between 0.0224 MeV/channel at the lower channel numbers to 0.0228 MeV/channel at the higher channel numbers. The net responses for each beam filter were formed and expressed in units of counts/MeV/100 Kw/200 min, and appear as the solid lines in Figs. 43, 44, and 45.

The contribution to the net response for each beam filter from capture of incident thermal neutrons was calculated in a manner com-

Fig. 43. Measured and Calculated Net Responses from 1 inch of Nickel for the Energy Range 1.0-4.6 MeV.

Fig. 44. Measured and Calculated Net Responses from 1 inch of Nickel for the Energy Range 4.6-8.2 MeV.

Fig. 45. Measured and Calculated Net Responses from 1 inch of Nickel for the Energy Range 8.2-9.2 MeV.

pletely analogous to that described for iron. For the boron-filtered beam, the intensities

$$\mathcal{J}_Y(E_Y) = 5.44 \times 10^6 I_Y(E_Y) A_Y(E_Y) \text{ photons/100 Kw/200 min} ,$$

where $I_Y(E_Y)$, in photons/capture, was taken from ref. 6 and $A_Y(E_Y)$ from Fig. 46, were folded and the resulting response subtracted from the net response. For the cadmium-filtered beam,

$$\mathcal{J}_Y(E_Y) = 1.1 \times 10^7 I_Y(E_Y) A_Y(E_Y) \text{ photons/100 Kw/200 min.}$$

The result of these subtractions from the net responses appear as smooth curves through the open diamonds presented in Figs. 43, 44, and 45.

The net response corrected for capture of incident thermal neutrons obtained for the boron-filtered beam was unfolded using the fitted response depicted by the open circles in the lower curve of Figs. 43, 44, and 45. The resulting intensities are shown in Table XXXVIII, along with the corresponding average production cross sections. These cross sections were calculated from the unfolded spectral intensities in the following manner:

$$\mathcal{J}_Y(E_Y) \text{ photons/100 Kw/200 min} = \frac{\bar{\sigma}(E_Y) \times 10^{-3}}{4\pi} \times \frac{0.6023}{58.7} \times 8.85 \times \frac{(\pi/4) \times 25}{144 \times 400} \\ \times \int_0^T dx \int_{1 \text{ MeV}}^{\infty} \varphi(E, x) dE \times A_Y(E_Y) ,$$

where $\bar{\sigma}(E_Y)$ is the average production cross section to neutrons above 1 MeV in millibarns, $A_Y(E_Y)$ is taken from Fig. 46, and

$$\int_0^T dx \int_{1 \text{ MeV}}^{\infty} \varphi(E, x) dE \simeq \frac{1.1 T}{\cos 45^\circ} \int_{1 \text{ MeV}}^{\infty} J(E) dE \\ = 3.52 \times 10^{14} \text{ neutron cm/100 Kw/200 min} ,$$

where the factor 1.1 was calculated by ANISN. Thus

$$\bar{\sigma}(E_Y) = 1.15 \times 10^{-6} \mathcal{J}_Y(E_Y) / A_Y(E_Y) \text{ millibarns.}$$

Fig. 46. Attenuation Function for Secondary Gamma Rays from Nickel.

Table XXXVIII. Unfolded Intensities and Average Production Cross Sections for Nickel

E_{γ} (MeV)	$\mathcal{L}_{\gamma}$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_{\gamma})$ mb	E_{γ} (MeV)	$\mathcal{L}_{\gamma}$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_{\gamma})$ mb
9.10	8.0(5) ^a	capture	5.70	2.0(5)	0.75
9.00	6.0(5)	capture	5.60	2.0(5)	0.75
8.90	5.0(5)	capture	5.50	1.5(5)	0.60
8.70	3.0(5)	capture	5.40	2.5(5)	1.0
8.60	2.0(5)	capture	5.30	1.7(5)	0.7
8.40	1.0(5)	capture	5.20	2.5(5)	1.0
8.30	2.0(5)	capture	5.10	2.0(5)	0.8
8.10	2.5(5)	capture	5.00	3.1(5)	1.2
8.00	2.5(5)	capture	4.90	2.6(5)	1.1
7.90	2.0(5)	capture	4.80	3.0(5)	1.2
7.80	2.0(5)	capture	4.70	2.0(5)	0.8
7.70	2.5(5)	capture	4.60	4.0(5)	1.7
7.60	2.0(5)	capture	4.45	4.5(5)	1.9
7.50	1.0(5)	capture	4.37	2.2(5)	1.0
7.40	2.0(5)	capture	4.25	4.5(5)	1.9
7.30	1.0(5)	capture	4.15	6.8(5)	3.0
7.20	1.0(5)	capture	4.05	8.3(5)	3.7
7.10	1.0(5)	capture	3.95	6.0(5)	2.7
7.00	2.0(5)	capture	3.85	7.0(5)	3.2
6.85	4.0(5)	capture	3.75	2.6(5)	1.2
6.70	2.5(5)	0.9	3.70	6.0(5)	2.8
6.60	7.0(4)	0.25	3.63	1.0(6)	4.7
6.50	1.5(5)	0.55	3.55	5.6(5)	2.7
6.40	2.5(5)	0.95	3.45	7.0(5)	3.4
6.30	1.0(5)	0.35	3.35	9.0(5)	4.4
6.20	2.0(5)	0.75	3.26	1.85(6)	9.2
6.10	2.0(5)	0.75	3.15	1.35(6)	6.9
6.00	2.2(5)	0.85	3.05	2.6(6)	13.5
5.90	1.5(5)	0.60	2.95	1.35(6)	7.2
5.80	1.8(5)	0.70	2.85	1.35(6)	7.3

Table XXXVIII (cont.)

E_γ (MeV)	$\mathcal{L}_\gamma$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb	E_γ (MeV)	$\mathcal{L}_\gamma$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb
2.75	1.35(6)	7.5	1.80	2.5(6)	20
2.70	9.0(5)	5.1	1.74	1.5(6)	12
2.65	9.5(5)	5.4	1.68	1.25(6)	11
2.62	3.5(5)	2.0	1.62	2.2(6)	19
2.55	1.35(6)	8.0	1.57	7.0(5)	6
2.45	1.6(6)	10	1.52	5.0(6)	47
2.40	2.0(5)	1	1.45	3.4(7)	335
2.35	1.65(6)	10	1.39	2.5(6)	25
2.30	2.5(5)	2	1.33	2.22(7)	240
2.25	2.0(6)	13	1.28	1.5(6)	17
2.19	2.0(5)	1	1.20	3.0(6)	36
2.15	2.5(6)	17	1.16	4.0(6)	50
2.07	1.95(6)	14	1.12	2.0(6)	26
1.98	1.7(6)	12	1.04	1.0(6)	14
1.92	1.15(6)	8	1.00	3.5(6)	52
1.86	1.85(6)	14			

^aRead as 8.0×10^5 .

As can be seen from Table XXXVIII all gamma rays lying below 6.8 MeV were assumed to arise from fast-neutron interactions in nickel, whereas all gamma rays above 6.8 MeV were assumed to arise from resonance capture in the region 10 keV-200 keV. This rather arbitrary division was dictated to some extent by the relatively strong spectral intensities lying above 6.8 MeV that occur from thermal-neutron capture.⁶ No clue is provided by the cadmium-filtered response, since it is dominated by 1/V captures above pulse-height energies of ~ 4 MeV.

The cadmium-filtered response was calculated by multiplying the intensities of all gamma rays above 6.8 MeV in Table XXXVIII by 2.2, those below 6.8 MeV by 1.33, adding in the intensities from 1/V capture and capture at thermal energies from slowing down, and the entire spectrum folded.

The 1/V capture intensities were calculated using $\sigma_{n\gamma}(E) =$

$$\frac{4.6 \sqrt{0.0253}}{\sqrt{E}} = 0.73/\sqrt{E} \text{ barns, so that}$$

$$\int_0^T dx \int_{0.5 \text{ eV}}^{\infty} \sigma_{n\gamma}(E) \varphi(E, x) dE = \frac{1.2 T}{\cos 45^\circ} \int_{0.5 \text{ eV}}^{\infty} \frac{0.731}{\sqrt{E}} \times \frac{4.0 \times 10^{12}}{E} dE$$

$$= 3.56 \times 10^{13} \text{ neutron barn cm/100 Kw 200 min}$$

where the factor 1.2 was calculated by ANISN. Thus,

$$\mathcal{J}_{\gamma_{1/V}}(E_\gamma) = \frac{3.56 \times 10^{13}}{4\pi} \times \frac{(\pi/4) \times 25}{144 \times 400} \times \frac{0.6023}{58.7} \times 8.85 \times I_\gamma(E_\gamma) A_\gamma(E_\gamma)$$

$$= 8.76 \times 10^7 I_\gamma(E_\gamma) A_\gamma(E_\gamma) \text{ photons/100 Kw/200 min ,}$$

where $I_\gamma(E_\gamma)$, in photons/capture, is taken from ref. 6 and $A_\gamma(E_\gamma)$ from Fig. 46.

The intensities arising from capture below 0.4 eV due to slowing down were also calculated with the help of ANISN:

$$\mathcal{J}_\gamma(E_\gamma) \approx 1.0 \times 10^7 I_\gamma(E_\gamma) A_\gamma(E_\gamma) \text{ photons/100 Kw/200 min .}$$

The calculated net response for the cadmium-filtered beam is indicated by the upper series of open circles appearing in Figs. 43, 44, and 45. A comparison of this response with the corrected measured response shows that the agreement is to within $\pm 15\%$ everywhere, and that there is evidence of resonance capture below 10 keV which contributes of the order of 20% as many gamma rays as those from $1/V$ capture.

The average production cross sections for nickel summed over 0.5-MeV gamma-ray intervals appear in Table XXXIX, where they are compared with values from ref. 5 averaged over the neutron spectrum above 1 MeV in Fig. 4.

The two sets of values appearing in Table XXXIX are in excellent agreement, differing by no more than 30% in each energy group. Thus the data appearing in ref. 5 may be used to predict secondary gamma-ray spectra as a function of neutron energy with sufficient accuracy for shielding applications.¹⁵

From an inspection of Table XXXVIII, relatively intense gamma rays arising from fast-neutron interactions in nickel are observed in the vicinity of 1.00, 1.16, 1.33, 1.45, 1.62, 1.80, ~ 2.1 , 2.25, 2.35, ~ 2.5 , 3.05, 3.26, 3.63, 4.05, and ~ 4.95 MeV. The remaining gamma rays in Table XXXVIII are composed of weaker lines superimposed upon a continuum.

A comparison of the production cross sections averaged over the incident spectrum above 1 MeV for several of the more intense lines obtained in this study with values obtained by averaging the data in ref. 1 over the incident spectrum from 1 MeV to E_u and correcting to natural nickel appears in Table XL.

Considering the limited range of the BNL-325 data, the agreement between the two sets of values for the 1.45- and 3.04-MeV gamma rays is considered adequate, since the production cross sections are still rising at 4 MeV. The ~ 1.33 -MeV gamma ray is really composed of two gamma rays, one with a threshold of ~ 2.80 MeV from ^{58}Ni , and the other with a threshold of ~ 1.35 MeV from ^{60}Ni . The latter is still by far the more intense when natural nickel is subjected to the neutron spectrum of Fig. 4, even though it occurs in the less abundant isotope. Certainly the extremely limited range of the BNL-325 data is at least partially responsible for the discrepancy in the values appearing in Table XL for the 1.33-MeV

Table XXXIX. Comparison of Average Production Cross Sections in Millibarns from Present Results with Data of Celnik and Spielberg for Nickel

	Gamma-Ray Energy Range (MeV)											Total
	1-1.5	1.5-2	2-2.5	2.5-3	3-3.5	3.5-4	4-4.5	4.5-5	5-5.5	5.5-6	6-7	
Present Results	795	149	68	42.5	37.4	17.3	11.5	4.8	< 4.7	< 3.4	< 5.3	≤ 1138
Celnik and Spielberg ⁵	840		45	40	27	21		13			4	990

Table XL. Comparison of Average Production Cross Sections in Millibarns from Present Results with BNL Data for Several Intense Gamma Rays in Nickel

Gamma-Ray Energy	1.33 MeV	1.45 MeV	3.04 MeV
Isotope	⁵⁸ Ni, ⁶⁰ Ni	⁵⁸ Ni	⁵⁸ Ni
Present Results	240	335	13.5
BNL-325 ¹	10, 110	250	10
E _u	4.0 MeV, 2.5 MeV	4.0 MeV	4.0 MeV

gamma ray, for if the cross section remained constant at its 2.5-MeV value for all energies greater than 2.5 MeV, the average value for the 1.33-MeV gamma ray would increase to around 180 mb. Comparing this value with the value of 240 mb obtained in this study suggests the possibility that the production cross section for this gamma ray above 2.5 MeV approaches values somewhat higher than an extrapolation of the BNL-325 data would indicate, although the two values are now within reasonable agreement.

A rough calculation similar to the one described for iron of the production cross sections of those gamma rays lying above 6.8 MeV in Table XXXVII arising from resonance capture of neutrons averaged over the neutron energy range 10 keV-200 keV yields the values appearing in Table XLI.

The value of $\sum_i E_{\gamma_i} \bar{\sigma}_{n\gamma}(E_{\gamma_i})/8 = 80$ mb checks the value of

$\sum_i \bar{\sigma}_{n\gamma}(E_{\gamma_i}) = 78$ mb, so that the gamma rays listed in Table XLI are not

in cascade and the average capture cross section for nickel in the region 10-200 keV for the incident spectrum of Fig. 8 is ≈ 80 mb, compared with a value of ~ 15 mb obtained from data in ref. 1. The disagreement is alarming, but it must be kept in mind that the values obtained in this study are very approximate since positive identification of these capture gamma rays is impossible in the present series of experiments. It is interesting, however, that the strongest gamma ray in the resonance

Table XLI. Capture Cross Sections for Nickel Averaged Over the
Neutron Spectrum from 10-200 keV

E_γ (MeV)	$\bar{\sigma}_{n\gamma}(E_\gamma)$ (Millibarns)	$\bar{\sigma}_{n\gamma}(E_\gamma) \times E_\gamma$ (MeV·Millibarns)	$\bar{I}_\gamma(E_\gamma)$ (photons/100 captures)	$I_\gamma(E_\gamma)$ (photons/100 thermal-neutron captures)
9.10	12	110	15	31.9 } 2.7 }
9.00	9	80	11	
8.90	7	60	9	
8.70	4	35	5	13.4
8.60	3	25	4	
8.40	1	10	2	6.1
8.30	3	25	4	
8.10	4	30	5	1.5
8.00	4	30	5	
7.90	3	25	4	2.3
7.80	3	25	4	
7.70	4	30	5	1.8 } 3.1 } 3.2 }
7.60	3	25	4	
7.50	1	10	2	
7.40	3	20	4	2.9
7.30	2	15	2	
7.20	1	10	2	
7.10	2	15	2	8.9
7.00	3	20	4	
6.85	6	40	7	
Total	78	640		

capture spectrum continues to occur with about the same probability per capture as in thermal-neutron capture. In fact, for nickel, there are many similarities between the values of $\bar{I}_Y(E_Y)$ and $I_Y(E_Y)$ when they are grouped according to the scheme indicated in Table XLI.

Stainless Steel. The foreground runs for 1 in. of type 321 stainless steel were 327-C for the boron-filtered beam and 326-D and 327-A for the cadmium-filtered beam. They were then normalized to 100 Kw and 200 min (see appendix). The gain for the boron-filtered run varied between 0.0217 MeV/channel at low channel numbers and 0.0224 MeV/channel at the high channel numbers. For the cadmium-filtered run, $0.0220 \leq G_c \leq 0.0227$ MeV/channel. The net responses were formed and appear in Figs. 47, 48, and 49 as the solid lines, plotted in units of counts/MeV/100 Kw/200 min.

The contribution to the net response for each beam filter from capture of incident thermal neutrons was calculated in a manner identical to that described earlier for iron. For the boron-filtered beam, the intensities

$$\mathcal{I}_Y(E_Y) = 4.90 \times 10^6 I_Y(E_Y) A_Y(E_Y) \text{ photons/100 Kw/200 min} ,$$

where $I_Y(E_Y)$, in photons/capture, was taken from ref. 6 and $A_Y(E_Y)$ from Fig. 50, were folded and the resulting response subtracted from the net response. For the cadmium-filtered beam,

$$\mathcal{I}_Y(E_Y) = 1.0 \times 10^7 I_Y(E_Y) A_Y(E_Y) \text{ photons/100 Kw/200 min} .$$

The results of these subtractions from the net responses appear as smooth curves through the open diamonds presented in Figs. 47, 48, and 49.

The net response corrected for capture of incident thermal neutrons obtained for the boron-filtered beam was unfolded using the fitted response shown by the open circles in the lower curve of Figs. 47, 48, and 49. The resulting intensities are presented in Table XLII, along with the corresponding average production cross sections which were calculated from the unfolded intensities in the following manner:

Fig. 47. Measured and Calculated Net Responses from 1 inch of Stainless Steel for the Energy Range 1.0-3.8 MeV.

Fig. 48. Measured and Calculated Net Responses from 1 inch of Stainless Steel for the Energy Range 3.8-6.8 MeV.

Fig. 49. Measured and Calculated Net Responses from 1 inch of Stainless Steel for the Energy Range 6.8-9.8 MeV.

Fig. 50. Attenuation Function for Secondary Gamma Rays from Stainless Steel.

Table XLIII. Unfolded Intensities and Average Production
Cross Sections for Stainless Steel

E_{γ} (MeV)	γ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_{\gamma})$ mb	E_{γ} (MeV)	γ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_{\gamma})$ mb
9.80	1.0(5) ^a	capture	6.70	1.0(5)	0.32
9.70	6.0(4)	capture	6.60	1.0(5)	0.32
9.60	7.0(4)	capture	6.50	2.3(5)	0.75
9.50	5.0(4)	capture	6.40	1.8(5)	0.59
9.40	6.0(4)	capture	6.30	1.3(5)	0.43
9.30	3.0(4)	capture	6.20	2.2(5)	0.73
9.20	3.0(4)	capture	6.10	2.5(5)	0.83
9.00	2.0(5)	capture	5.97	4.0(5)	capture
8.90	9.0(4)	capture	5.90	1.5(5)	0.50
8.80	1.35(5)	capture	5.80	1.3(5)	0.44
8.70	2.5(4)	capture	5.70	1.1(5)	0.37
8.60	1.2(5)	capture	5.60	2.5(5)	0.85
8.50	7.0(4)	capture	5.50	2.6(5)	0.89
8.40	1.1(5)	capture	5.40	1.6(5)	0.55
8.30	5.5(4)	capture	5.30	3.0(5)	1.05
8.10	1.1(5)	capture	5.20	4.0(5)	1.4
8.04	2.0(4)	capture	5.10	3.5(5)	1.2
8.00	1.6(5)	capture	5.00	2.5(5)	0.9
7.90	1.0(5)	capture	4.90	3.5(5)	1.3
7.80	1.3(5)	capture	4.80	2.0(5)	0.7
7.67	9.0(5)	capture	4.73	4.0(5)	1.4
7.50	8.0(4)	0.25	4.60	3.0(5)	1.1
7.40	2.0(5)	0.63	4.50	3.4(5)	1.3
7.30	1.5(5)	0.47	4.40	5.0(5)	1.9
7.20	1.3(5)	0.41	4.30	3.5(5)	1.3
7.10	1.5(5)	0.48	4.20	3.5(5)	1.3
7.00	1.5(5)	0.48	4.10	6.0(5)	2.3
6.90	1.8(5)	0.58	4.00	5.0(5)	2.0
6.80	1.5(5)	0.48	3.90	7.5(5)	3.0

Table XLIII (cont.)

E_γ (MeV)	$\bar{\nu}_\gamma$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb	E_γ (MeV)	$\bar{\nu}_\gamma$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb
3.80	9.0(5)	3.6	2.12	5.6(6)	32
3.70	8.5(5)	3.5	2.06	3.0(5)	1.8
3.63	1.8(6)	7.4	2.03	2.0(5)	1.2
3.57	1.75(6)	7.3	2.00	1.15(6)	6.9
3.46	2.4(6)	10	1.94	5.0(5)	3.1
3.40	1.0(5)	0.4	1.88	2.4(6)	15
3.36	5.0(5)	2.1	1.82	6.3(6)	41
3.30	1.5(6)	6.5	1.76	1.8(6)	12
3.20	2.0(6)	8.8	1.70	7.0(5)	4.9
3.10	1.3(6)	5.8	1.64	9.5(5)	6.8
3.00	2.0(6)	9.1	1.58	1.2(6)	8.8
2.90	8.0(5)	3.7	1.52	3.1(6)	24
2.80	2.1(6)	10	1.45	1.65(7)	131
2.70	1.7(6)	8.2	1.40	1.1(6)	9.0
2.65	5.0(5)	2.4	1.35	5.5(6)	47
2.60	3.9(6)	19	1.30	2.7(6)	24
2.55	3.0(5)	1.5	1.25	6.75(6)	61
2.50	1.6(6)	8.2	1.20	2.05(6)	19
2.40	1.5(6)	7.9	1.15	2.0(6)	20
2.35	4.5(5)	2.4	1.10	1.0(6)	10
2.30	3.5(6)	19	1.05	2.5(6)	27
2.23	1.1(6)	6.0	1.00	3.0(6)	35
2.17	1.2(6)	6.8			

^aRead as 1.0×10^5 .

$$I_Y(E_Y) \text{ photons/100 Kw/200 min} = \frac{\bar{\sigma}(E_Y) \times 10^{-3}}{4\pi} \times \frac{0.6023}{55} \times 7.88 \times \frac{(\pi/4) \times 25}{144 \times 400}$$

$$\times \int_0^T dx \int_{1 \text{ MeV}}^{\infty} \varphi(E, x) dE \times A_Y(E_Y)$$

where $\bar{\sigma}(E_Y)$ is the average production cross section to neutrons above 1 MeV in millibarns, $A_Y(E_Y)$ is taken from Fig. 50, and

$$\int_0^T dx \int_{1 \text{ MeV}}^{\infty} \varphi(E, x) dE \approx \frac{1.1 T}{\cos 45^\circ} \int_{1 \text{ MeV}}^{\infty} J(E) dE = 3.52 \times 10^{14}$$

neutron cm/100 Kw/200 min ,

where the factor 1.1 was calculated by ANISN. Thus,

$$\bar{\sigma}(E_Y) = 1.21 \times 10^{-6} I_Y(E_Y)/A_Y(E_Y) \text{ millibarns} .$$

As can be seen from Table XLII, all gamma rays lying below 7.5 MeV were assumed to arise from fast-neutron interactions in stainless steel, with the exception of the 5.97-MeV gamma ray, whereas all gamma rays above 7.5 MeV were assumed to arise from resonance capture in the region 10 keV-200 keV. This somewhat arbitrary division was dictated by the previous analyses of iron and nickel. Again, the cadmium-filtered response is dominated by 1/V captures above pulse-height energies of ~ 4 MeV, so that no clue is afforded by this response as to whether the intensities in the boron-filtered response should be multiplied by 1.33 or 2.2.

The cadmium-filtered response was calculated by multiplying the intensities of all gamma rays above 7.5 MeV in Table XLII as well as that of the 5.97-MeV gamma ray by 2.2, those below 7.5 MeV excluding the 5.97-MeV gamma ray by 1.33, adding in the intensities from 1/V capture and capture at thermal energies from slowing down, and folding.

The 1/V capture intensities were calculated using⁶

$$\sigma_{n\gamma}(E) = \frac{3.05 \sqrt{0.0253}}{\sqrt{E}} = 0.485/\sqrt{E} \text{ barns,}$$

so that

$$\int_0^T dx \int_{0.5 \text{ eV}}^{\infty} \sigma_{n\gamma}(E) \varphi(E, x) dE \approx \frac{1.3 T}{\cos 45^\circ} \int_{0.5 \text{ eV}}^{\infty} \frac{0.485}{\sqrt{E}} \times \frac{4.0 \times 10^{12}}{E} dE$$

$$= 2.56 \times 10^{13} \text{ neutron barn cm/100 Kw 200 min} ,$$

where the factor 1.3 was calculated by ANISN. Thus,

$$\mathcal{A}_{\gamma_{1/V}}(E_\gamma) = \frac{2.56 \times 10^{13}}{4\pi} \times \frac{(\pi/4) \times 25}{144 \times 400} \times \frac{0.6023}{55} \times 7.88 \times I_\gamma(E_\gamma) A_\gamma(E_\gamma)$$

$$= 6.00 \times 10^7 I_\gamma(E_\gamma) A_\gamma(E_\gamma) \text{ photons/100 Kw/200 min} ,$$

where $I_\gamma(E_\gamma)$, in photons/capture, is taken from ref. 6 and $A_\gamma(E_\gamma)$ from Fig. 50.

The intensities arising from capture below 0.4 eV due to slowing down were also calculated with the help of ANISN:

$$\mathcal{A}_\gamma(E_\gamma) \approx 6.0 \times 10^6 I_\gamma(E_\gamma) A_\gamma(E_\gamma) \text{ photons/100 Kw/200 min} .$$

The calculated net response for the cadmium-filtered beam is indicated by the upper series of open circles appearing in Figs. 47, 48, and 49. A comparison of this response with the corrected measured net response shows that the agreement is to within $\pm 20\%$ everywhere, and that there is some evidence that resonance captures below 10 keV contribute of the order of 30% as many gamma rays as those from 1/V capture.

The average production cross sections for stainless steel summed over 0.5-MeV gamma-ray intervals are presented in Table XLIII, where they are compared with values from ref. 5 averaged over the neutron spectrum above 1 MeV in Fig. 4.

The two sets of values appearing in Table XLIII are in excellent agreement, differing by no more than 30% in each energy group. The calculated results were compounded from the data in ref. 5 by adding the contributions from iron multiplied by 0.7, nickel multiplied by 0.1, and chromium

Table XLIII. Comparison of Average Production Cross Sections in Millibarns from Present Results with Data of Celnik and Spielberg for Stainless Steel

	Gamma-Ray Energy Range (MeV)												Total
	1-1.5	1.5-2	2-2.5	2.5-3	3-3.5	3.5-4	4-4.5	4.5-5	5.5-5	5.5-6	6-6.5	6.5-7.5	
Present Results	383	116	84	53.4	42.8	24.8	8.8	5.8	5.1	3.0	< 2.6	< 4.9	≤ 734
Celnik and Spielberg ⁵	390		139		59.8		19.2		12.6			3.3	624

multiplied by 0.2. Since the data appearing in ref. 5 for iron and nickel have already been found to be in satisfactory agreement with the measurements previously described for these materials, the conclusion may be drawn that the data in ref. 5 for chromium are satisfactory as well.¹⁵

From an inspection of Tables IX, XXXIX, and XLII, relatively intense gamma rays arising from fast-neutron interactions in stainless steel are observed in the vicinity of 1.00 MeV (Fe+Ni), 1.05 MeV (Fe), 1.25 MeV (Fe), 1.35 MeV (Ni+Cr), 1.45 MeV (Ni+Cr), 1.52 MeV (Cr), 1.82 MeV (Fe), 2.12 MeV (Fe), 2.30 MeV (Fe), 2.60 MeV (Fe), 2.80 MeV (Fe), 3.00 MeV (Fe), 3.20 MeV (Fe), 3.46 MeV (Fe+Cr), 3.63 MeV (Fe), 3.80 MeV (Cr), 4.40 MeV (Fe), 4.73 MeV (Fe), and 5.20 MeV (Cr). The remaining intensities in Table XLII arise from weaker lines superimposed upon a continuum.

A comparison of the production cross sections obtained from the stronger iron lines in stainless steel with the values obtained earlier for iron multiplied by 0.7 is shown in Table XLIV.

The agreement in Table XLIV is considered adequate, since there is some contribution to each of the intensities in the stainless steel from the nickel and chromium. Only in the case of the 3.63-MeV gamma ray is there any real disagreement, and even here it is not serious.

A rough calculation similar to the one described for iron of the production cross sections of those gamma rays labeled "capture" in Table XLII and which possibly arise from resonance capture in the neutron energy range 10 keV-200 keV yields the values appearing in Table XLV.

The value of $\sum_i E_{\gamma_i} \bar{\sigma}_{n\gamma}(E_{\gamma_i})/8 = 39 \text{ mb}$ checks the value of

$\sum_i \bar{\sigma}_{n\gamma}(E_{\gamma_i})$, so that the gamma rays listed in Table XLV are not in cascade

and the average capture cross section for stainless steel in the region 10-200 keV for the incident spectrum of Fig. 8 is $\approx 40 \text{ mb}$, compared with a value of $\sim 20 \text{ mb}$ obtained from data in ref. 1. Again the almost exact agreement of the intense $\sim 7.65\text{-MeV}$ gamma-ray transition probability per capture for thermal-neutron and resonance capture is apparent, substantiating the analysis from the iron slab presented earlier. A comparison of the relative intensities in the two spectra is shown by the grouping indicated in Table XLV.

Table XLIV. Comparison of Average Production Cross Sections in Millibarns of Several Intense Gamma Rays from Iron Obtained from Analysis of Iron and Stainless Steel Slabs

E_{γ} (MeV)	1.05	1.25	1.82	2.12	2.30	2.60	2.80	3.00	3.20	3.63	4.40	4.73
Stainless Steel	27	61	41	32	19	19	10	9.1	8.8	7.4	1.9	1.4
0.7 x Iron	22	64	39	33	13	21	9	6.4	6.3	10.2	2.2	1.1

Table XLV. Capture Cross Sections for Stainless Steel Averaged Over the
Neutron Spectrum from 10-200 keV

E_γ (MeV)	$\bar{\sigma}_{n\gamma}(E_\gamma)$ (Millibarns)	$\bar{\sigma}_{n\gamma}(E_\gamma) \times E_\gamma$ (MeV·Millibarns)	$\bar{I}_\gamma(E_\gamma)$ (photons/100 capture)	$I_\gamma(E_\gamma)$ (photons/100 thermal-neutron captures)
9.80	1.2	10	3	1.8
9.70	0.7	5	2	
9.60	0.9	10	2	
9.50	0.6	5	1	
9.40	0.7	10	2	1.9
9.30	0.4	5	1	
9.20	0.4	5	1	
9.00	2.5	20	6	
8.90	1.1	10	3	4.6 5.3 0.4
8.80	1.7	15	4	
8.70	0.3	5	1	
8.60	1.5	10	4	4.2
8.50	0.9	10	2	
8.40	1.4	10	4	
8.30	0.7	5	2	
8.10	1.4	10	3	1.0 0.2
8.04	0.3	5	1	
8.00	2.0	15	5	
7.90	1.3	10	3	2.6 0.3 0.3
7.80	1.6	15	4	
7.67	11.5	90	29	
5.97	5.5	30	14	29.4 ~11
Total	39	310		

Barium. The foreground runs for 1 in. of barium oxide enclosed in an aluminum container were 515-A for the boron-filtered beam and 514-B for the cadmium-filtered beam. They were then normalized to 100 Kw and 200 min (see appendix). The gain for the cadmium-filtered run varied between 0.0221 MeV/channel at the low channel numbers and 0.0233 MeV/channel at the high channel numbers; for the boron-filtered run, $0.0220 \leq G_c \leq 0.0232$ MeV/channel. The net responses were formed and appear plotted in units of counts/MeV/100 Kw/200 min as the dashed lines in Figs. 51, 52, and 53.

The contribution of the aluminum container to the net response for each beam filter was scaled to the corrected net response from 0.25 in. of aluminum with the help of calculations using ANISN. It was found that the scaling factor $(2 \times 0.066/0.25) = 0.528$ was again adequate, and the contribution of the container to the net response for each beam filter was calculated in a manner similar to that described for the calcium oxide. (Since the contribution from the aluminum is relatively small compared to that of the barium oxide everywhere, no significant error is incurred in using the unaltered response functions for the aluminum contribution.)

The contribution from capture of the incident thermal neutrons was calculated for each beam filter in the same manner as described earlier for calcium oxide, i.e., captures in both the aluminum and barium oxide were calculated, the spectral intensities determined from the data in ref. 6 and the attenuation function in Fig. 54, and then folded with the altered response functions for barium oxide. This contribution is very small compared to the measured net response for both beam filters and could have been safely neglected. The effect of slowing down to thermal energies and subsequent capture was calculated by ANISN to be negligible for the case of the boron-filtered beam, and although small was included for the case of the cadmium-filtered beam.

The result of subtracting both the contribution from the aluminum container and capture of the incident thermal neutrons appears as the solid curve through the open diamonds for each beam filter in Figs. 51, 52, and 53. As previously mentioned, the contribution from the aluminum was relatively small compared to that from the barium oxide everywhere below ~ 7.5 MeV,

Fig. 51. Measured and Calculated Net Responses from 1 inch of Barium Oxide Enclosed in an Aluminum Container for the Energy Range 1.1-3.4 MeV.

Fig. 52. Measured and Calculated Net Responses from 1 inch of Barium Oxide Enclosed in an Aluminum Container for the Energy Range 3.4-7.0 MeV.

Fig. 53. Measured and Calculated Net Responses from 1 inch of Barium Oxide Enclosed in an Aluminum Container for the Energy Range 7.0-8.9 MeV.

Fig. 54. Attenuation Function for Secondary Gamma Rays from Barium Oxide.

averaging around 20% of the total response except in the regions centered around 2.22 MeV and 3.02 MeV where the contribution increases to at most ~ 40%.

The unfolded intensities derived from the corrected net response for the boron-filtered beam produce a response denoted by the lower series of open circles in Figs. 51, 52, and 53. These spectral intensities are presented in Table XLVI along with values of the corresponding production cross sections averaged over the incident spectrum of Fig. 4 above 1 MeV for barium and above threshold for oxygen. These cross sections were calculated in the following manner. For barium,

$$\mathcal{J}_Y(E_Y) \text{ photons/100 Kw/200 min} = \frac{\bar{\sigma}(E_Y) \times 10^{-3}}{4\pi} \times \frac{0.6023}{153.3} \times 2.73 \times \frac{(\pi/4) \times 25}{144 \times 400} \\ \times \int_0^T dx \int_{1 \text{ MeV}}^{\infty} \varphi(E, x) dE \times A_Y(E_Y) ,$$

where $\bar{\sigma}(E_Y)$ is the average production cross section to neutrons above 1 MeV in millibarns, $A_Y(E_Y)$ is taken from Fig. 54, and

$$\int_0^T dx \int_{1 \text{ MeV}}^{\infty} \varphi(E, x) dE \simeq \frac{1.15 T}{\cos 45^\circ} \int_1^{\infty} J(E) dE \\ = 3.68 \times 10^{14} \text{ neutron cm/100 Kw/200 min} ,$$

where the factor 1.15 was calculated by ANISN. Hence,

$$\bar{\sigma}(E_Y) = 9.45 \times 10^{-6} \mathcal{J}_Y(E_Y) / A_Y(E_Y) \text{ millibarns.}$$

For the gamma rays from oxygen,

$$\mathcal{J}_Y(E_Y) = \frac{\bar{\sigma}(E_Y) \times 10^{-3}}{4\pi} \times \frac{0.6023}{153.3} \times 2.73 \times \frac{(\pi/4) \times 25}{144 \times 400} \times \frac{1.01 T}{\cos 45^\circ} \\ \times \int_{\text{threshold}}^{\infty} J(E) dE \times A_Y(E_Y) ,$$

whence

Table XLVI. Unfolded Intensities and Average Production Cross Sections for Barium and Oxygen

E_Y (MeV)	$\mathcal{Q}_Y$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_Y)$ mb	E_Y (MeV)	$\mathcal{Q}_Y$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_Y)$ mb
8.90	1.2(4) ^a	capture	5.25	4.0(4)	0.75
8.80	1.5(4)	capture	5.16	1.1(5)	2.1
8.70	1.0(4)	capture	5.08	1.3(5)	2.5
8.60	3.5(4)	capture	4.95	1.4(5)	2.7
7.80	3.5(4)	capture	4.84	2.3(5)	4.4
7.70	2.5(3)	capture	4.70	2.2(5)	4.3
7.60	2.5(4)	capture	4.60	1.2(5)	2.3
7.50	1.5(4)	capture	4.50	3.7(5)	7.3
7.40	2.0(4)	capture	4.40	1.8(5)	3.5
7.30	3.5(4)	capture	4.30	3.0(5)	6.0
7.20	1.8(4)	capture	4.20	1.8(5)	2.9
7.11	9.5(4)	65*	4.10	3.3(5)	6.7
7.00	1.5(4)	capture	4.00	1.6(5)	3.2
6.94	1.6(5)	115*	3.93	3.8(5)	7.8
6.80	1.0(4)	0.18	3.80	4.2(5)	8.8
6.70	8.5(4)	1.5	3.70	4.8(5)	10.0
6.60	4.0(4)	0.70	3.60	4.65(5)	9.8
6.42	1.0(5)	1.8	3.50	5.0(5)	10.6
6.30	1.0(5)	1.8	3.40	4.0(5)	8.6
6.13	4.5(5)	180*	3.35	5.5(5)	12
6.00	9.0(4)	1.6	3.28	2.5(5)	5.5
5.90	7.0(4)	1.3	3.20	7.0(5)	15
5.80	2.0(4)	0.37	3.10	6.5(5)	14.5
5.70	1.2(5)	2.2	3.00	1.0(6)	23
5.60	7.5(4)	1.4	2.90	8.0(5)	18
5.54	5.0(4)	0.93	2.80	1.0(6)	23
5.50	1.15(5)	2.1	2.70	1.0(6)	24
5.40	1.1(5)	2.0	2.65	8.5(5)	20
5.30	1.0(5)	1.9	2.60	1.0(6)	24

Table XLVI (cont.)

E_γ (MeV)	$\mathcal{L}_\gamma$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb	E_γ (MeV)	$\mathcal{L}_\gamma$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_\gamma)$ mb
2.50	1.4(6)	34	1.75	1.45(6)	43
2.40	1.3(6)	33	1.68	1.9(6)	58
2.30	1.7(6)	44	1.62	2.3(6)	72
2.20	4.1(6)	108	1.55	1.85(6)	59
2.15	8.5(5)	23	1.50	3.0(6)	98
2.10	1.3(6)	35	1.43	2.05(7)	692
2.05	8.0(5)	22	1.32	3.5(6)	125
2.00	1.0(6)	28	1.27	2.5(6)	90
1.95	1.05(6)	30	1.23	3.0(6)	110
1.90	1.1(6)	31	1.18	3.2(6)	120
1.85	1.5(6)	43	1.14	2.1(6)	83
1.80	9.0(5)	27	1.10	3.0(6)	120

^aRead as 1.2×10^4 .

* Inelastic lines from oxygen.

$$\bar{\sigma}(6.13) = 2.08 \times 10^{-4} \mathcal{J}_{\gamma}(6.13)/A_{\gamma}(6.13) \text{ millibarns}$$

for neutrons above 6.5 MeV, and

$$\bar{\sigma}(E_{\gamma}) = 3.83 \times 10^{-4} \mathcal{J}_{\gamma}(E_{\gamma})/A_{\gamma}(E_{\gamma}) \text{ millibarns}$$

for neutrons above 7.4 MeV, and where $E_{\gamma} = 6.94 \text{ MeV}$ and 7.11 MeV .

The cadmium-filtered response was calculated by multiplying the intensities of all gamma rays below 7.0 MeV appearing in Table XLVI by 1.33, the intensities of all gamma rays above 7.0 MeV except the 7.11-MeV gamma ray by 2.2, adding in the intensities from 1/V capture and from capture of those neutrons slowed down below 0.4 eV, and folding.

The intensities from 1/V capture were calculated using $\sigma_{n\gamma}(E) =$

$$\frac{1.2 \sqrt{0.0253}}{\sqrt{E}} = 0.191/\sqrt{E} \text{ barns, so that}$$

$$\begin{aligned} \int_0^T dx \int_{0.5 \text{ eV}}^{\infty} \sigma_{n\gamma}(E) \varphi(E, x) dE &= \frac{1.22 T}{\cos 45^\circ} \int_{0.5 \text{ eV}}^{\infty} \frac{0.191}{\sqrt{E}} \times \frac{4.0 \times 10^{12}}{E} dE \\ &= 9.47 \times 10^{12} \text{ neutron barn cm/100 Kw/200 min ,} \end{aligned}$$

where the factor 1.22 was calculated by ANISN. Thus,

$$\begin{aligned} \mathcal{J}_{\gamma_{1/V} \text{ Ba}}(E_{\gamma}) &= \frac{9.47 \times 10^{12}}{4\pi} \times \frac{(\pi/4) \times 25}{144 \times 400} \times \frac{0.6023}{153.3} \times 2.73 \times I_{\gamma_{\text{Ba}}}(E_{\gamma}) A_{\gamma}(E_{\gamma}) \\ &= 2.76 \times 10^6 I_{\gamma_{\text{Ba}}}(E_{\gamma}) A_{\gamma}(E_{\gamma}) \text{ photons/100 Kw/200 min ,} \end{aligned}$$

where $I_{\gamma_{\text{Ba}}}(E_{\gamma})$, in photons/capture, is taken from ref. 6 and $A_{\gamma}(E_{\gamma})$ from Fig. 54.

The intensities arising from capture in the barium oxide at energies below 0.4 eV due to slowing down were also calculated with the help of ANISN. They were extremely small:

$$\mathcal{L}_{\gamma\text{Ba}}(E_{\gamma}) \approx 6.0 \times 10^4 I_{\gamma\text{Ba}}(E_{\gamma}) A_{\gamma}(E_{\gamma}) \text{ photons/100 Kw/200 min} .$$

The calculated net response for the cadmium-filtered beam is indicated by the upper series of open circles appearing in Figs. 51, 52, and 53. It in general falls considerably below the measured response everywhere above ~ 2.5 MeV. Since the two essentially agree in the region below 2 MeV where the capture contribution is small, the conclusion may be drawn that there is an appreciable contribution from resonance capture in the pulse-height region above ~ 2.5 MeV. Since, further, there is considerably disagreement in the region above 7 MeV even after the intensities from the boron-filtered beam have been multiplied by 2.2, the added conclusion may be drawn that most of the resonance captures that are contributing to the cadmium-filtered response occur at energies below ~ 10 keV. Unfortunately, no data for natural barium pertinent to these conclusions is available in ref. 1, but considerable resonance capture below 10 keV occurs in both natural elements on either side of barium in the periodic table -- lanthanum and cesium. Our tentative conclusion is that resonance capture below 10 keV produces of the order of 5 to 10 times the number of gammas from a $1/E$ neutron spectrum as that arising from $1/V$ capture above 0.5 eV.

The average production cross sections summed over 0.5-MeV gamma-ray intervals are presented in Table XLVII. Unfortunately, no data are available with which to compare these values.

The cross sections in Table XLVII are somewhat larger than those of previously analyzed elements, but this is attributable to multiplicity in the gamma-ray cascades in barium rather than to a higher nonelastic cross section. About 90% of the total production cross section above ~ 1.1 MeV from neutron interactions above 1 MeV occurs for gamma rays of less than 3 MeV. Since in this region the resonance capture contribution is relatively small, the total production cross-section value of 2440 mb above ~ 1.1 MeV is reasonably well established.

Reasonably intense gamma rays arising from fast-neutron interactions in barium are observed in the vicinity of 1.10, 1.20, 1.32, 1.43, 1.62, 2.10, 2.20, 2.30, 3.35, 4.10, 4.30, and 4.50 MeV. The remaining intensities observed in Table XLVI represent somewhat weaker lines superimposed upon a continuum.

Table XLVII. Average Production Cross Sections in Millibarns for Barium

	Gamma-Ray Energy Range (MeV)												Total
	~1.1-1.5	1.5-2	2-2.5	2.5-3	3-3.5	3.5-4	4-4.5	4.5-5	5-5.5	5.5-6	6-6.5	6.5-7	
Present Results	1345	460	290	144	< 79	< 47	< 22.3	< 21.0	< 9.2	< 8.3	< 5.2	< 2.4	≤ 2440

Table XLVIII. Capture Cross Sections for Barium Averaged Over the Neutron Spectrum From 10 - 200 keV

E_{γ} MeV	$\bar{\sigma}_{n\gamma}(E_{\gamma})$ millibarns	E_{γ} MeV	$\bar{\sigma}_{n\gamma}(E_{\gamma})$ millibarns
8.90	1.0	7.60	2.1
8.80	1.3	7.50	1.3
8.70	0.8	7.40	1.7
8.60	2.9	7.30	3.0
7.80	3.0	7.20	1.5
7.70	0.2	7.00	1.3

The values of the average production cross sections above threshold obtained for the three inelastic gamma rays from oxygen in barium oxide are in serious disagreement with those found earlier from analysis of the borated water, being of the order of a factor of two higher. Since so many secondary gamma rays are observed from barium, it is probable that there is an appreciable contribution from the barium in the immediate vicinity of the oxygen gamma rays. Thus the intensities at 6.13, 6.94 and 7.11 MeV should perhaps not all be ascribed to oxygen.

It should be mentioned that both the cadmium-filtered and boron-filtered runs were repeated in order to check a possible error in the power levels of the two runs, but after corrections to the density of the barium oxide and the attenuation through an additional 6 in. of lithium hydride were made, the foreground counting rates expressed in units of counts/channel/100 Kw/200 min agreed to within several percent of the original values.

A rough calculation was made of the production cross sections of those gamma rays labelled "capture" in Table XLVI and which possibly arise from resonance capture in the neutron energy range 10 keV-200 keV, and these appear in Table XLVIII. No calculation of $\bar{\Gamma}_Y(E_Y)$ was made however, because too much of the capture spectrum probably lies below 7 MeV.

The production cross section from resonance capture in barium for gamma rays above 7 MeV is thus ≈ 20 mb. If the spectrum is at all similar to that arising from thermal-neutron capture, considerably more than this lies below 7 MeV. A very rough value of ≈ 40 mb for natural barium is estimated from the data of Macklin, Gibbons, and Inada in ref. 1.

Sodium. The foreground runs for 4 in. of sodium carbonate containing 9% by weight water enclosed in an aluminum container were 581-B and 582-A for the cadmium-filtered beam and 581-A for the boron-filtered beam. They were then normalized to 100 Kw and 200 min (see appendix). The gain for both runs varied between 0.0223 MeV/channel at the lower channel numbers and 0.0226 MeV/channel at the higher channel numbers. The net responses were formed and appear plotted in units of counts/MeV/100 Kw/200 min as dashed lines in Figs. 55, 56, and 57.

The contribution of the aluminum container to the net response for each beam filter from neutrons above 1 MeV was scaled to the corrected net response from 0.25 in. of aluminum with the help of calculations using ANISN. It was found that the scaling factor 0.50 was adequate, and the contribution of the container from fast neutrons to the net response for the boron-filtered beam was calculated by folding one-half the intensities in Table XIII with the response functions used for the barium oxide, and 1.33 times this response was taken to be the fast-neutron aluminum contribution for the cadmium-filtered beam. The contributions from the container to the net response arising from capture of incident thermal neutrons, capture of neutrons slowed down to thermal energies by the water present and 1/V capture were calculated for both beam filters by means of the code described in ref. 6 and ANISN respectively. For the boron-filtered beam, the total contribution was obtained by folding the intensities

$$\mathcal{J}_{\text{Al}}(E_Y) = 1.14 \times 10^5 I_{\text{Al}}(E_Y) A_Y(E_Y) \text{ photons/200 min/100 Kw}$$

with the barium oxide response functions, where $I_{\text{Al}}(E_Y)$ in photons/capture was taken from ref. 6 and $A_Y(E_Y)$ from Fig. 58. For the cadmium filtered beam, the sum of the contributions yielded

$$\mathcal{J}_{\text{Al}}(E_Y) = 1.03 \times 10^6 I_{\text{Al}}(E_Y) A_Y(E_Y) \text{ photons/200 min/100 Kw}$$

and these intensities were folded to obtain the response.

Fig. 55. Measured and Calculated Net Responses from 4 inches of Sodium Carbonate Enclosed in an Aluminum Container for the Energy Range 1.1-4.0 MeV.

Fig. 56. Measured and Calculated Net Responses from 4 inches of Sodium Carbonate Enclosed in an Aluminum Container for the Energy Range 4.0-7.0 MeV.

Fig. 57. Measured and Calculated Net Responses from 4 inches of Sodium Carbonate Enclosed in an Aluminum Container for the Energy Range 7.0-8.6 MeV.

Fig. 58. Attenuation Function for Secondary Gamma Rays from Sodium Carbonate.

The contributions of the "wet" sodium carbonate and the chlorine impurity to the net response for each beam filter arising from incident thermal-neutron capture were calculated using the code described in ref. 6, as well as the contributions arising from $1/V$ capture and capture of neutrons slowed down to thermal energies by the water present for the boron-filtered beam using ANISN. These total contributions were obtained by folding the following intensities determined for the boron-filtered beam:

$$\mathcal{J}_{\gamma_{Na}}(E_{\gamma}) = 1.52 \times 10^6 I_{\gamma_{Na}}(E_{\gamma}) A_{\gamma}(E_{\gamma}) \text{ photons/200 min/100 Kw} ,$$

$$\mathcal{J}_{\gamma_H}(2.23) = 1.9 \times 10^5 \text{ photons/200 min/100 Kw} ,$$

and

$$\mathcal{J}_{\gamma_{Cl}}(E_{\gamma}) = 2.3 \times 10^5 I_{\gamma_{Cl}}(E_{\gamma}) A_{\gamma}(E_{\gamma}) \text{ photons/200 min/100 Kw} ,$$

where the values of $I_{\gamma_{Na}}(E_{\gamma})$ and $I_{\gamma_{Cl}}(E_{\gamma})$, in photons/capture, were taken from ref. 6 and $A_{\gamma}(E_{\gamma})$ from Fig. 58. For the cadmium-filtered beam, the contribution from incident thermal-neutron capture was obtained by folding the following intensities:

$$\mathcal{J}_{\gamma_{Na}}(E_{\gamma}) = 1.14 \times 10^6 I_{\gamma_{Na}}(E_{\gamma}) A_{\gamma}(E_{\gamma}) \text{ photons/200 min/100 Kw} ,$$

$$\mathcal{J}_{\gamma_H}(2.23) = 1.42 \times 10^5 \text{ photons/200 min/100 Kw} ,$$

and

$$\mathcal{J}_{\gamma_{Cl}}(E_{\gamma}) = 1.16 \times 10^5 I_{\gamma_{Cl}}(E_{\gamma}) A_{\gamma}(E_{\gamma}) \text{ photons/200 min/100 Kw} .$$

The result of subtracting the response from capture of incident thermal neutrons, $1/V$ capture, and capture of neutrons slowed to thermal energies by the water present, as well as the total response from the aluminum container, is shown in Figs. 55, 56, and 57 for the boron-filtered beam as the lower solid curve. This curve thus represents the response of the sodium carbonate to gamma rays arising from fast-neutron interactions and resonance capture.

The result of subtracting the response from capture of incident thermal neutrons as well as the total response from the aluminum container is shown in Figs. 55, 56, and 57 for the cadmium-filtered beam as the upper solid curve. This curve thus represents the response of the sodium carbonate to gamma rays arising from fast-neutron interactions and resonance capture, as well as from capture of neutrons slowed down to thermal energies by the water present and $1/V$ capture. The effect of the aluminum has been completely removed in both solid curves.

The unfolded intensities derived from the corrected net response for the boron-filtered beam produce a response denoted by the lower series of open circles in Figs. 55, 56, and 57. These spectral intensities are presented in Table XLIX along with values of the corresponding production cross sections averaged over the incident spectrum of Fig. 4 above 1 MeV for sodium and above threshold for carbon and oxygen. These cross sections were calculated in the following manner. For sodium,

$$\begin{aligned} \mathcal{A}_\gamma(E_\gamma) \text{ photons/100 Kw/200 min} &= \frac{\bar{\sigma}(E_\gamma) \times 10^{-3}}{4\pi} \times \frac{0.6023}{106} \times 0.737 \times 0.91 \\ &\times 2 \times \frac{(\pi/4) \times 25}{144 \times 400} \times \int_0^T dx \int_{1 \text{ MeV}}^{\infty} \varphi(E, x) dE \times A_\gamma(E_\gamma) , \end{aligned}$$

where $\bar{\sigma}(E_\gamma)$ is the average production cross section for neutrons above 1 MeV in millibarns, $A_\gamma(E_\gamma)$ is taken from Fig. 58, and

$$\begin{aligned} \int_0^T dx \int_{1 \text{ MeV}}^{\infty} \varphi(E, x) dE &\simeq \frac{1.05 T}{\cos 45^\circ} \int_{1 \text{ MeV}}^{\infty} J(E) dE \\ &= 1.34 \times 10^{15} \text{ neutron cm/100 Kw/200 min} , \end{aligned}$$

where the factor 1.05 was calculated by ANISN. Hence,

$$\bar{\sigma}(E_\gamma) = 3.60 \times 10^{-6} \mathcal{A}_\gamma(E_\gamma) / A_\gamma(E_\gamma) \text{ millibarns.}$$

For gamma rays from oxygen,

$$\begin{aligned} \mathcal{A}_\gamma(E_\gamma) &= \frac{\bar{\sigma}(E_\gamma) \times 10^{-3}}{4\pi} \times 0.6023 \times 0.737 \times \left(\frac{3 \times 0.91}{106} + \frac{0.09}{18} \right) \times \frac{\pi}{4} \times \frac{25}{144 \times 400} \\ &\times \frac{0.88T}{\cos 45^\circ} \int_{\text{threshold}}^{\infty} J(E) dE \times A_\gamma(E_\gamma) , \end{aligned}$$

whence $\bar{\sigma}(6.13) = 4.63 \times 10^{-5} \mathcal{J}_Y(6.13)/A_Y(6.13)$ millibarns for neutrons above 6.5 MeV,

and $\bar{\sigma}(E_Y) = 8.52 \times 10^{-5} \mathcal{J}_Y(E_Y)/A_Y(E_Y)$ millibarns for neutrons above 7.4 MeV, and where $E_Y = 6.92$ MeV and 7.10 MeV.

For the gamma ray from carbon,

$$\begin{aligned} \sigma_Y(E_Y) &= \frac{\bar{\sigma}(E_Y) \times 10^{-3}}{4\pi} \times \frac{0.6023}{106} \times 0.737 \times 0.91 \times \frac{\frac{\pi}{4} \times 25}{144 \times 400} \times \frac{0.90T}{\cos 45^\circ} \\ &\times \int_{\text{threshold}}^{\infty} J(E) dE \times A_Y(E_Y), \end{aligned}$$

whence $\bar{\sigma}(4.44) = 5.75 \times 10^{-5} \mathcal{J}_Y(4.44)/A_Y(4.44)$ millibarns for neutrons above 4.8 MeV.

The cadmium-filtered response was calculated by multiplying the intensities in Table XLIX by 1.33, adding in the intensities from 1/V capture and from capture of those neutrons slowed down to thermal energies by the water present for sodium, hydrogen, and chlorine, and folding.

The intensities from 1/V capture in sodium were calculated using

$$\sigma_{nY}(E) = \frac{0.534\sqrt{0.0253}}{\sqrt{E}} = 0.085/\sqrt{E} \text{ barns,}$$

$$\begin{aligned} \text{so that } \int_0^T dx \int_{0.5 \text{ eV}}^{\infty} \sigma_{nY}(E) \varphi(E, x) dE &= \frac{1.41T}{\cos 45^\circ} \int_{0.5 \text{ eV}}^{\infty} \frac{0.085}{\sqrt{E}} \times \frac{4.0 \times 10^{12}}{E} dE \\ &= 1.95 \times 10^{13} \frac{\text{neut. barn cm.}}{100 \text{ Kw } 200 \text{ min}}, \end{aligned}$$

where the factor 1.41 was calculated by ANISN.

$$\begin{aligned} \text{Thus, } \mathcal{J}_{Y_{1/V \text{ Na}}}(E_Y) &= \frac{1.95 \times 10^{13}}{4\pi} \times \frac{\frac{\pi}{4} \times 25}{144 \times 400} \times \frac{0.6023}{106} \times 0.737 \\ &\times 0.91 \times 2 \times I_{Y_{\text{Na}}}(E_Y) A_Y(E_Y) \\ &= 4.03 \times 10^6 I_{Y_{\text{Na}}}(E_Y) A_Y(E_Y) \text{ photons/100 Kw/200 min.} \end{aligned}$$

Table XLIX. Unfolded Intensities and Average Production Cross Sections for Sodium, Oxygen, and Carbon

E_{γ} (MeV)	$\mathcal{Q}_{\gamma}$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_{\gamma})$ mb	E_{γ} (MeV)	$\mathcal{Q}_{\gamma}$ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_{\gamma})$ mb
8.60	4.0(4) ^a	0.24	4.90	1.5(5)	1.05
8.50	2.5(4)	0.15	4.80	1.0(5)	0.7
8.40	1.5(4)	0.09	4.70	5.0(4)	0.35
8.30	2.0(4)	0.12	4.60	1.2(5)	0.9
8.20	5.0(3)	0.03	4.52	2.2(5)	1.6
7.90	1.0(4)	0.06	4.44	1.5(6)	178 [†]
7.80	1.0(4)	0.06	4.30	3.5(5)	2.6
7.70	4.0(4)	0.25	4.20	1.2(5)	0.9
7.60	3.0(4)	0.19	4.10	8.0(4)	0.6
7.50	3.0(4)	0.19	4.00	4.0(5)	3.1
7.40	5.0(4)	0.31	3.90	1.05(6)	8.2
7.30	2.0(4)	0.13	3.80	1.5(5)	1.2
7.20	1.2(5)	0.76	3.73	2.5(5)	2.0
7.10	2.8(5)	42*	3.67	1.5(5)	1.2
7.00	7.0(4)	0.45	3.60	1.7(5)	1.4
6.92	3.25(5)	49*	3.52	1.0(5)	0.8
6.52	1.2(5)	0.78	3.44	4.0(5)	3.3
6.40	8.0(4)	0.52	3.36	1.3(5)	1.1
6.30	1.1(5)	0.72	3.28	6.7(5)	5.7
6.20	2.5(5)	1.65	3.22	4.0(5)	3.4
6.13	1.0(6)	85*	3.14	3.0(5)	2.6
6.00	2.0(5)	1.3	3.06	6.3(5)	5.5
5.80	2.0(5)	1.4	2.99	1.0(6)	8.8
5.70	1.0(5)	0.7	2.92	5.0(4)	0.4
5.60	1.0(5)	0.7	2.85	3.9(5)	3.5
5.50	1.5(5)	1.0	2.78	2.1(5)	1.9
5.40	1.0(5)	0.7	2.71	7.5(5)	7.0
5.30	1.0(5)	0.7	2.64	1.5(6)	14.2
5.20	1.2(5)	0.85	2.57	8.5(5)	8.2
5.00	1.0(5)	0.7	2.51	6.0(5)	5.8

Table XLIX (cont.)

E_{γ} (MeV)	γ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_{\gamma})_{\text{mb}}$	E_{γ} (MeV)	γ (photons/ 200 min/100 Kw)	$\bar{\sigma}(E_{\gamma})_{\text{mb}}$
2.44	6.0(5)	5.9	1.70	9.0(5)	11.0
2.39	1.05(6)	10.5	1.63	4.65(6)	58.3
2.32	5.0(5)	5.1	1.56	5.0(5)	6.4
2.27	4.0(5)	4.1	1.50	3.0(5)	4.0
2.22	3.5(5)	3.6	1.44	3.0(5)	4.1
2.16	6.3(5)	6.7	1.38	2.4(5)	3.4
2.10	4.0(5)	4.3	1.32	4.0(5)	5.8
2.04	6.0(5)	6.6	1.27	4.0(5)	6.0
1.98	3.5(5)	3.9	1.22	2.0(5)	3.1
1.93	6.0(5)	6.8	1.18	5.0(5)	7.9
1.88	3.0(5)	3.5	1.14	5.0(5)	8.1
1.82	4.0(5)	4.7	1.10	1.25(6)	20.7
1.76	9.0(5)	10.8			

^aRead as 4.0×10^4 .

* Inelastic lines from oxygen.

[†]Inelastic line from carbon.

The intensities arising from capture in sodium of neutrons slowed down below 0.4 eV were calculated using

$$\bar{\sigma}_{n\gamma} \simeq 0.534/2 = 0.267 \text{ barns,}$$

so that

$$\bar{\sigma}_{n\gamma} \int_0^T \phi(x) dx = 4.00 \times 10^{13} \frac{\text{neut. barn cm.}}{100 \text{ Kw } 200 \text{ min.}},$$

using the results of ANISN. Thus

$$\gamma_{\text{th Na}}(E_\gamma) = 8.27 \times 10^6 I_{\gamma\text{Na}}(E_\gamma) A_\gamma(E_\gamma) \text{ photons/100 Kw/200 min.}$$

The sum of the two contributions from sodium capture is consequently

$$\gamma_{\text{Na}}(E_\gamma) = 1.23 \times 10^7 I_{\gamma\text{Na}}(E_\gamma) A_\gamma(E_\gamma) \text{ photons/100 Kw/200 min.}$$

Similarly, using for chlorine

$$\sigma_{n\gamma}(E) = \frac{33.2\sqrt{0.0253}}{\sqrt{E}} = 5.28/\sqrt{E} \text{ barns,}$$

leads to

$$\gamma_{1/V \text{ Cl}}(E_\gamma) = 6.1 \times 10^5 I_{\gamma\text{Cl}}(E_\gamma) A_\gamma(E_\gamma) \text{ photons/100 Kw/200 min.,}$$

and

$$\bar{\sigma}_{n\gamma} \simeq 33.2/2 = 16.6 \text{ barns}$$

leads to

$$\gamma_{\text{th Cl}}(E_\gamma) = 1.26 \times 10^6 I_{\gamma\text{Cl}}(E_\gamma) A_\gamma(E_\gamma) \text{ photons/100 Kw/200 min.}$$

Thus,

$$\gamma_{\text{Cl}}(E_\gamma) = 1.87 \times 10^6 I_{\gamma\text{Cl}}(E_\gamma) A_\gamma(E_\gamma) \text{ photons/100 Kw/200 min.}$$

Similarly, using for water,

$$\sigma_{n\gamma}(E) = \frac{0.663\sqrt{0.0253}}{\sqrt{E}} = 0.105/\sqrt{E} \text{ barns/molecule}$$

leads to

$$\phi_{\gamma_{1/V} \text{ H}_2\text{O}}(2.23) = 5.00 \times 10^5 \text{ photons/100 Kw/200 min.},$$

and

$$\bar{\sigma}_{n\gamma} \simeq 0.663/2 = 0.332 \text{ barns/molecule}$$

leads to

$$\phi_{\gamma_{th} \text{ H}_2\text{O}}(2.23) = 1.03 \times 10^6 \text{ photons/100 Kw/200 min.}$$

Thus,

$$\phi_{\gamma_{\text{H}_2\text{O}}}(2.23) = 1.53 \times 10^6 \text{ photons/100 Kw/200 min.}$$

The calculated net response for the cadmium-filtered beam is indicated by the upper series of open circles appearing in Figs. 55-57. In general, the agreement between the calculated and measured cadmium-filtered responses averages within $\pm 15\%$, with the calculated response being the lower above 7 MeV. Since the amount of water present in the sample is appreciable, the contribution to the cadmium-filtered response from capture of neutrons slowed down to thermal energies is of the same order of magnitude as that arising from fast-neutron (>1 MeV) interactions in the slab sample. Since, further, the output from ANISN contained only those neutrons slowed down below 0.4 eV as one group, the value of the average capture cross section for neutrons in this group is difficult to estimate since a $1/V$ dependence would alter the cross section by a factor of 4 between 0.0253 and 0.4 eV. Our estimate of a factor of 2 for the ratio of the capture cross section at 0.0253 eV to the average capture cross section for this group (i.e.,

$$\bar{\sigma}_{n\gamma} = \frac{\sigma_{n\gamma}(0.0253)}{2}) \text{ is thus conceivably in error, but the agreement with}$$

experiment of the calculated cadmium-filtered response is such that this error is probably not too significant. (This estimate could have been made more accurately if several groups below 0.4 eV had been incorporated into ANISN.) In any event, the conclusion may be drawn that there is no evidence of any resonance capture in sodium from the data appearing in Figs. 55-57, the disagreement in the cadmium-filtered responses being totally explainable in terms of errors involved in thermal-neutron capture calculations.

The average production cross sections summed over 0.5 MeV gamma-ray intervals for sodium are presented in Table L, where they are compared with values obtained from an earlier experiment performed with 1 inch of sodium carbonate and values obtained by averaging the data of reference 3 over the incident spectrum above 1 MeV of Fig. 4.

The two sets of experimental values are in reasonable agreement, even though the relatively large contribution from the aluminum in the 1-inch case caused the values for sodium to be somewhat inaccurate. The mutual disagreement with the values of reference 3 is striking - the calculated values averaging factors of 4 less than experimental values. Thus, the data in reference 3 for sodium are not recommended for shielding calculations.

From an inspection of Table XLIX, relatively intense gamma rays arising from fast-neutron interactions in sodium are observed in the vicinity of 1.10, 1.63, 1.76, 1.93, 2.04, 2.39, 2.64, 2.99, 3.28, and 3.90 MeV. The remaining intensities in Table XLIX arise from weaker lines superimposed upon a continuum.

There are sufficient data in reference 1 to calculate the average production cross section of only the 1.63-MeV gamma ray for neutrons up to 3.5 MeV for the incident spectrum above 1 MeV of Fig. 4. Table LI shows the comparison of this value with the experimental values.

The agreement between the present experimental values and the BNL-325 value is considered adequate, since the latter value would probably become larger if the neutron energy range were extended above 3.5 MeV. In any event, the value of 40 mb is already in excess of the value calculated for the entire 1.5- to 2.0-MeV interval from the data of reference 3.

Table L. Comparison of Average Production Cross Sections in Millibarns From Present Results With Data of Troubetzkoy for Sodium

	Gamma-Ray Energy Range, MeV												Total
	1.1-1.5	1.5-2	2-2.5	2.5-3	3-3.5	3.5-4	4-4.5	4.5-5	5-5.5	5.5-6	6-6.5	6.5-7.5	
Present Results (4")	59	109	47	50	21.6	14.8	7.3	4.6	2.9	3.7	4.2	2.4	326
Present Results (1")	91	118	68	46	34.4	16.0	8.2	3.4	3.7	5.1	3.3	-	397
Troubetzkoy ³	12.0	32.1	20.5	9.0	8.5		3.4		1.4		0.8		88

Table LI. Comparisons of Average Production Cross Sections
in Millibarns From Present Results With BNL Data for
the 1.63-MeV Gamma Ray in Sodium

Present Results (4")	58
Present Results (1")	70
BNL-325 ¹	40

The average production cross sections above threshold for oxygen obtained from the sodium carbonate analysis are in fair agreement with the more accurate values obtained from the water analysis described earlier. For the 6.13-MeV gamma ray, the value of 85 mb is in excellent agreement with the earlier value, but for the 6.92 + 7.11-MeV gamma rays the present value of 90 mb is considerably in excess of the earlier value of 55 mb. The latter disagreement may be due to the presence of gamma rays from sodium in this vicinity.

The average production cross section above threshold for the 4.43-MeV gamma ray from carbon from the analysis of the sodium carbonate, 180 mb, agrees exactly with the value obtained from the analysis of the potassium carbonate, and is somewhat in excess of the value of 130 mb obtained from a more accurate analysis of the polyethylene described earlier. A possible explanation for this disagreement is that the small contribution to the 4.43-MeV gamma-ray production that arises from the (n, α) reaction in oxygen is sufficient to elevate the production cross section in the carbonates to the measured values. Also, the contribution from gamma rays in this vicinity arising from interactions in the sodium or potassium renders the values obtained for the carbonate analyses somewhat inaccurate.

VII. Conclusions and Summary

The average production cross sections obtained for fast-neutron (i.e., >1 MeV) interactions from the preceding analyses in general should be accurate enough to provide adequate checks of data differential in both the gamma-ray energy and the neutron energy. The only exception to this generality lies in the cross sections obtained for gamma-ray energies above ~ 6 MeV, where gamma rays arising from resonance capture can compete in magnitude and consequently render an inaccurate determination of the fast-neutron production cross sections. The average production cross sections presented for resonance capture in the region 10-200 keV are not intended to be taken too seriously, since their determination in the present experiment is fraught with uncertainties and guesswork. Not the least of these uncertainties is the range of neutron energies over which the resonance captures are observed for the production of a particular gamma ray. Another uncertainty is recognizing a gamma ray as arising from resonance capture; the criterion of matching its attenuation through the boron filter to one of

two values not particularly different, in the presence of other gamma rays, is admittedly inferior and in fact was not often used. Even the conclusion that the number of photons/capture for the more intense gamma rays arising from thermal-neutron capture is about the same as that arising from resonance capture in the neutron-energy region 10-200 keV should be questioned, since it depends to a large extent on knowing the range of neutron energies over which the resonance captures are observed for production of the given gamma ray and on a very accurate determination of the magnitude of the incident thermal-neutron current. Thus, we might estimate that the average production cross-section spectral data, $\int_{E_Y} 4\pi d\bar{\sigma}(E_Y)/d\Omega$ at 90° , arising from fast-neutron interactions is accurate to within $\pm 30\%$ for each gamma-ray interval below 6 MeV, and the average production cross sections arising from resonance capture in the region 10-200 keV is accurate to about a factor of 4.

Comparison of the present data with the work of United Nuclear indicates that the production cross sections arising from fast-neutron interactions previously published by United Nuclear are generally adequate for iron,⁵ aluminum,⁵ calcium,³ nickel,⁵ and chromium,⁵ and are inadequate for titanium,⁴ potassium,⁴ silicon,² sulfur,³ and sodium.³ No calculations exist for copper, zinc, or barium.

VIII. Acknowledgments

The authors wish to acknowledge the work of all of the personnel of the Tower Shielding Facility without whom the measurements described in this report could never have been made. In particular, the following people should be mentioned: G. M. Argo, T. V. Blosser, L. W. Gilley, K. M. Henry, L. B. Holland, J. L. Hull, J. J. Manning, J. N. Money, F. E. Richardson, and D. R. Ward. The authors also wish to acknowledge C. E. Clifford of this laboratory whose continued interest and support in both the measurements and analyses were greatly appreciated, and whose counsel throughout the course of the experiments was invaluable.

IX. Appendix

In the following pages the analyzer data alluded to in Sections V and VI are shown in the form of counts per channel as a function of channel number. For all these runs, the expression $E_c = G_c \times (C - 2)$ was used to convert channel number to energy in MeV, where G_c is mentioned in the text for each run. The data that were used commence in channel 40 (always so indicated) and end in channel 511. The data are presented in increasing order of channel number and proceed across the page from left to right. Each column thus represents data from channel numbers eight channels apart. The normalization of each sheet of data to analyzer live time and reactor power is mentioned in the text, and any additional factors that need to be applied to the data or have already been incorporated into the normalization are also mentioned in the text.

Run 320-B

	006390	006219	005911	005408	005109	004876	004519	004155
	003755	003398	003190	002890	002669	002431	002158	001910
	001792	001656	001599	001567	001594	001691	002527	005269
40	051515	516834	761103	016789	663505	598716	541614	487330
→	440692	404175	369678	337027	307044	081945	262375	243277
	226156	211854	199263	185966	174650	166366	159536	153270
	147519	142441	134655	125333	118114	113551	111030	108237
	104190	099801	094574	087541	082470	078777	075332	070899
	067564	064539	061789	058342	057135	055564	053456	050993
80	-047853	045847	043625	041924	039864	037930	035951	034011
	032132	030720	028835	027589	026763	025501	024335	023038
	022003	021749	021203	020828	020920	020984	021271	021667
	022250	023200	023227	022918	021168	020175	018879	017229
	015685	014337	013379	012540	011939	011925	011430	011698
120	-011174	011121	010916	010641	010438	010602	010198	009956
	010023	009705	009722	009464	009508	009372	009194	009101
	009016	009105	008921	008866	008675	008751	008661	008356
	008405	008446	008313	008114	008274	007998	007964	007961
	007775	007710	007694	007461	007590	007522	007531	007542
160	-007694	007636	007557	007250	007335	007245	007169	007103
	007162	007259	007000	006986	006977	007030	006926	006741
	006720	006783	006629	006765	006736	006837	006703	006912
	006882	006758	006800	006617	006500	006499	006483	006419
	006293	006301	006268	006208	006198	006177	006185	006044
200	-006000	006039	006217	006234	006398	006468	006639	006669
	006445	006490	006627	006266	006053	005964	006029	005856
	005925	005719	005784	005654	005671	005572	005679	005454
	005516	005552	005390	005465	005546	005439	005448	005569
	005413	005495	005480	005613	005617	005505	005476	005243
240	-005069	005086	005156	005008	004962	005028	004747	004778
	004794	004717	004610	004745	004674	004688	004653	004627
	000000	004695	004794	004724	004629	004632	004627	004418
	004377	004347	004409	004289	004493	004263	004366	004273
	004285	004177	004236	004104	004205	004159	004151	004057
280	-004168	004044	004019	003937	003990	004036	003866	003995
	003910	004023	004035	004231	004207	004090	003970	003912
	003729	003581	003426	003143	003114	002812	002740	002484
	002363	002473	002460	002521	002597	002771	002908	003078
	003161	003351	003318	003320	003190	002853	002626	002288
320	-001988	001693	001352	001181	001082	001021	000925	000855
	000841	000825	000693	000714	000674	000671	000680	000666
	000604	000573	000536	000501	000437	000420	000428	000386
	000463	000456	000425	000463	000484	000483	000483	000436
	000380	000367	000340	000298	000244	000278	000235	000200
360	-000235	000184	000205	000209	000178	000151	000139	000173
	000169	000150	000142	000156	000121	000135	000159	000165
	000162	000188	000166	000169	000187	000179	000174	000188
	000166	000177	000158	000137	000140	000138	000139	000125
	000106	000121	000125	000110	000132	000131	000134	000118
400	-000124	000122	000109	000111	000115	000119	000103	000097
	000089	000101	000097	000091	000096	000073	000077	000087
	000088	000064	000085	000097	000095	000078	000089	000073
	000072	000070	000071	000088	000083	000070	000077	000080
	000089	000089	000083	000070	000084	000092	000087	000088

440 -000088 000072 000081 000067 000091 000064 000073 000061
 000090 000063 000078 000079 000072 000092 000107 000097
 000119 000119 000126 000174 000172 000222 000270 000323
 000397 000507 000530 000667 000757 000367 001031 001266
 001526 002021 002733 003667 005601 037190 165136 312908
 480 -131609 029009 009471 001209 002540 001806 001444 001096
 000861 000673 000564 000454 000373 000263 000184 000152

Run 319-D

015962 015418 014884 013698 013302 012356 011182 010778
 210178 009232 008454 007698 007150 006488 005788 005008
 004518 004310 004122 004008 003964 004548 006438 015440
 168126 1271562 1688122 1613452 1481948 1350018 1226016 1109686
 40 → 1009312 926236 851636 782002 714928 662492 613046 567802
 528492 495980 467360 438056 411534 392008 375584 360102
 346420 334934 317698 296692 281350 269376 262788 255724
 245360 234862 221048 208090 196914 189106 181092 171252
 161440 154074 147872 141042 137358 132896 128538 123258
 80 - 117166 111990 106824 101544 097246 092972 088180 083646
 078524 074944 071232 068058 065496 063140 059514 056796
 054680 053414 051982 051362 051748 051558 052388 053858
 056620 060892 062198 060514 055800 051956 048066 044444
 039288 036512 033194 031936 030636 029300 029508 029016
 120 - 028822 028490 028762 027026 027110 026572 025714 025420
 025030 024770 024780 024624 024148 024290 024266 023770
 022940 023306 023208 022772 022936 022200 023016 022062
 022120 021772 021990 021350 020986 020934 020530 020850
 020002 020490 020020 019742 020176 020010 019966 019754
 160 - 019540 019644 019696 019210 019212 019070 018788 018738
 018192 018926 018782 018380 018148 018108 018462 017952
 017618 017516 017744 017682 017468 017574 017508 018002
 017354 017920 017932 017400 017214 017242 017186 016710
 016342 016586 016588 016456 016322 016396 015910 015996
 200 - 016140 015900 016056 016176 016506 016764 016760 017276
 016972 016790 016792 016522 016058 015604 015878 015314
 015046 015786 015620 015040 014874 015222 015336 015172
 014600 014694 014526 014600 014498 014634 014568 014486
 014648 014490 014656 014578 014742 014634 014288 014522
 240 - 014052 013848 013896 013284 013312 013338 013348 013266
 012480 012790 012482 012394 012316 012522 012168 012492
 000000 012586 012516 012354 011954 012174 011856 011366
 011350 011330 011458 011394 011270 011212 011348 011210
 011348 011082 011228 010854 011066 010672 010846 010552
 280 - 010330 010828 010338 010220 010012 010334 010358 010076
 010356 010380 010782 010770 010836 010770 010430 010010
 009624 009562 008760 008228 007972 007470 007108 006864
 006490 006324 006398 006598 006924 007368 007480 008036
 008324 008648 008542 008470 008176 007518 006898 005824
 320 - 005152 004470 003752 003336 002856 002738 002460 002394
 002258 002070 002152 001946 002048 001794 001928 001774
 001722 001646 001440 001332 001224 001226 001220 001234
 001186 001272 001372 001300 001384 001326 001352 001200
 001176 001006 000932 000822 000740 000692 000602 000570
 360 - 000626 000510 000534 000536 000452 000444 000434 000512
 000420 000416 000426 000376 000388 000374 000382 000388
 000400 000444 000474 000442 000456 000462 000456 000512
 000464 000426 000462 000400 000420 000394 000372 000314
 000400 000346 000264 000300 000306 000302 000286 000338

~~400-000340 000366 000340 000334 000306 000378 000326 000304~~
~~000310 000276 000296 000248 000228 000258 000250 000210~~
~~000218 000256 000224 000240 000230 000218 000164 000200~~
~~000228 000184 000206 000208 000176 000188 000208 000218~~
~~000190 000232 000278 000228 000200 000218 000196 000228~~
~~440-000212 000240 000266 000210 000206 000202 000174 000252~~
~~000238 000204 000226 000238 000272 000278 000244 000280~~
~~000282 000286 000406 000388 000464 000560 000702 000816~~
~~000932 001058 001246 001484 001676 001834 002274 002656~~
~~003116 004142 005040 007274 012016 054760 258016 557554~~
~~480-357646 089100 027974 011160 006266 004312 003184 002584~~
~~002198 001744 001406 001110 000936 000628 000466 000368~~
~~000306 000872 000210 000178 000184 000140 000120 000114~~

Run 551-C + 552A

000000 412000 002059 002256 002983 003712 003999 004336
 004394 004690 004714 004779 004956 005511 005872 006733
 110406 009569 011847 014300 018116 022368 028007 034269
 041427 050131 059715 068978 078253 088874 099884 111606
 122644 137144 152038 170507 190486 215741 252840 333343
 40 → 463160 600759 705764 775377 827152 856260 843908 778835
 685346 602755 531976 480111 440310 412365 387727 365042
 343287 325407 309717 298880 286000 276499 267097 256798
 245869 240292 231178 223388 215999 210455 204397 198734
 194639 193031 192831 195612 203101 213056 221802 224602
 80- 218561 208624 198377 187367 179457 174724 170746 167339
 162800 159782 154441 148420 143506 140168 140131 145411
 158462 187562 229796 278869 319017 335536 321910 278409
 227246 180654 142982 115465 098473 087687 082473 079583
 076939 075668 073707 070684 068314 065676 063764 062240
 120- 060644 059904 059411 057814 056515 054332 051789 049346
 047366 046538 047425 049278 052693 057108 061915 064088
 064708 062790 058334 052208 046806 042178 038238 035778
 033637 032621 031368 030315 029574 028637 028141 027541
 026911 027124 026805 026327 025847 025320 024872 024312
 160- 024012 024050 024050 024179 024365 024473 024823 024742
 023946 023771 023805 023137 023297 022999 023194 022813
 022763 022679 022621 021780 021443 021000 020790 020223
 019701 019982 019400 019299 019025 018837 018848 018719
 018466 018491 018354 018284 018453 018743 018693 018594
 200- 018700 018582 018526 018442 018536 018253 017940 017744
 016967 017250 016493 016218 016128 016186 015850 015519
 015741 015474 015661 015492 015567 015887 015752 016085
 015878 015768 016169 016456 016350 016474 016414 016538
 016055 016404 015970 015958 015683 015911 015888 015972
 240- 015865 016033 016742 016887 017576 017901 018768 018928
 019431 020502 020622 020464 020555 020119 019829 018966
 000000 017602 016774 016229 015792 015506 015391 014894
 015028 015421 015779 015936 016740 017459 018463 019023
 019744 019668 019785 018861 018061 017030 015954 014351
 280- 013492 012657 011960 011334 011107 010895 010547 010509
 010070 009989 009605 009374 009075 009197 008951 008864
 008314 008223 008148 007842 007871 007818 007670 007635
 007701 007668 007796 007671 007506 007558 007044 006892
 006621 006570 006311 006077 006025 005813 005505 005288

320-005088 004771 004295 004129 003721 003478 003208 003035
 002775 002690 002514 002349 002384 002360 002340 002202
 002241 002326 002342 002255 002214 002230 002170 002057
 001954 001900 001814 001568 001515 001215 001128 001123
 000968 000905 000805 000746 000690 000683 000624 000594
 360-000582 000584 000514 000526 000514 000513 000483 000495
 000457 000477 000501 000461 000432 000408 000429 000395
 000444 000360 000388 000380 000353 000398 000336 000346
 000316 000343 000331 000290 000299 000313 000284 000291
 000294 000288 000272 000277 000227 000254 000241 000208
 400-000232 000214 000210 000208 000190 000195 000165 000166
 000170 000164 000177 000149 000147 000146 000156 000129
 000129 000146 000112 000134 000103 000106 000098 000111
 000088 000108 000120 000090 000120 000096 000108 000112
 440-000108 000093 000091 000085 000094 000088 000094 000086
 000077 000083 000066 000089 000088 000091 000094 000078
 000108 000083 000097 000103 000105 000088 000097 000097
 000087 000093 000088 000083 000084 000090 000080 000082
 000075 000072 000081 000066 000060 000066 000079 000069
 480-000069 000068 000078 000078 000069 000074 000061 000053
 000060 000042 000049 000046 000041 000056 000034 000040
 000039 000036 000032 000030 000033 000039 000036 000032

Run 551-B

200000 001937 549748 001906 002561 002873 003137 003123
 003081 003121 003031 002973 003044 003096 003202 003662
 261930 004671 005761 007359 010357 013466 017515 022032
 027639 033099 038673 044120 049854 056312 063216 071125
 080425 090614 102455 116734 133296 161262 282741 580675
 40 → 821932 942274 989929 1025213 1078567 1124083 1068149 922441
 764955 668238 608260 572660 556067 543435 525477 499084
 482021 472198 458949 456660 448348 444744 433581 425234
 414552 410426 404467 396036 392755 383590 381076 377460
 377893 384680 393816 418719 463593 515205 553539 553341
 80- 525748 491654 459641 435664 419281 412681 404845 398749
 381646 365253 338059 315482 294384 285758 293514 332212
 427733 612502 869478 130391 1283837 1264591 1061859 782647
 530843 352854 239141 177518 144230 128831 120359 117687
 114918 112796 108593 103622 099049 094359 093048 090462
 120- 088363 087297 085591 083151 078650 073969 068732 065542
 062317 063634 065682 073049 082173 092653 099681 102948
 097821 090539 077631 067115 058780 052199 048325 046394
 044979 043394 042316 040766 040204 040314 039391 038864
 039016 038989 039149 039609 037728 037214 036861 036592
 160- 035456 036644 036987 036996 037233 038105 037545 037168
 036848 035425 035221 035217 035269 035525 035048 035450
 034890 034982 033554 032562 031765 031174 030741 030421
 030271 030094 030752 030402 030219 029886 029778 029204
 200- 028800 028417 028750 028338 028694 028987 028673 028700
 028290 028047 028207 027989 026990 026975 027181 027248
 026609 025783 026124 025101 025251 025026 024320 023567
 023370 023259 023220 023018 023197 023199 023586 023646
 023790 024266 024518 024391 024314 025005 023865 024096
 023347 023426 022837 022722 022034 022812 022664 023022

240-023114 023917 023561 024335 025365 026636 027575 028303
 029046 030030 030485 030292 028821 027706 026875 025446
 000000 022679 022760 021711 021158 020731 020390 020251
 020877 021395 021957 022910 024404 026998 028022 029137
 029614 028848 028251 026526 023904 022298 020178 018407
 280-017402 016516 016121 015680 015020 014806 014989
 014500 014161 013971 013734 013570 013753 013314 013191
 012567 012710 012488 012396 012465 012538 012234 012577
 012513 012598 012783 012767 012319 012255 012101 011905
 012217 011899 011797 011955 012032 011647 011485 011387
 320-010642 010120 009234 008668 008000 007374 006741 006115
 005794 005486 005326 005276 005120 005335 005436 005605
 005938 006225 006745 006897 006677 006862 006720 006127
 005472 004927 004386 003653 003177 002598 002195 001849
 001469 001558 001356 001275 001182 001142 001009 001017
 360-000951 000928 000861 000815 000838 000899 000842 000778
 000855 000834 000840 000807 000851 000784 000788 000659
 000718 000676 000709 000676 000599 000591 000628 000622
 000582 000618 000541 000595 000501 000570 000532 000557
 000551 000553 000532 000493 000412 000397 000412 000412
 400-000439 000347 000368 000345 000343 000326 000347 000343
 000285 000274 000318 000326 000287 000274 000268 000202
 000235 000241 000235 000233 000225 000239 000204 000179
 000204 000229 000241 000225 000233 000227 000206 000243
 000225 000154 000183 000210 000270 000227 000216 000214
 440-000210 000243 000239 000218 000208 000218 000216 000175
 000193 000295 000231 000266 000225 000239 000193 000227
 000239 000262 000168 000212 000212 000179 000166 000225
 000156 000154 000175 000191 000189 000177 000216 000185
 000181 000191 000225 000204 000162 000166 000160 000152
 480-000141 000139 000108 000091 000114 000100 000060 000075
 000081 000060 000056 000075 000068 000050 000054 000056

Run 365-A

284014 030596 034754 030520 027284 023314 019648 015562
 011578 008494 006430 004826 003904 003416 003104 006424
 40 → 042826 230600 894176 1244750 1228408 1132022 1015590 909040
 814804 742076 674360 614178 559642 518536 477536 442338
 412224 389594 367412 347994 332184 320784 318516 318536
 317938 305882 291334 275544 269232 266000 265922 262022
 254052 246600 240666 233222 229296 225392 223058 218116
 216018 216262 213332 214150 216118 226834 245396 276600
 80 - 302424 309938 295480 273434 256900 249590 246918 243562
 242556 237410 225808 208384 186346 164802 146334 131664
 123404 135044 183714 296014 488080 727432 916056 955716
 834006 621088 400696 233092 131204 077890 052970 042662
 037330 034274 033110 032410 031128 030624 029916 029542
 120 - 028794 027430 026598 026762 025654 024974 024250 024002
 023328 022962 022846 023440 022504 021986 022552 022296
 022332 022834 023042 023226 023032 023668 023880 023894
 023436 023720 023808 023818 023668 024146 025152 026166
 027778 029316 030706 031096 031058 030594 030180 029696
 160 - 029464 029400 030024 030362 031074 032054 032568 032492
 032104 032066 031918 031466 032562 036026 040508 047158
 054536 060970 065872 066074 062888 057174 051336 045800
 042420 039364 037678 034894 032810 030452 028358 025606
 025096 025556 028878 035016 044224 057736 072922 083114

200-089946 090202 082720 068708 052914 039068 027752 019704
 014170 011586 010194 009042 008510 008122 007968 007986
 007536 007440 007664 008116 008082 008638 008314 008032
 007914 007536 007130 006918 006290 006350 006054 005910
 005648 005678 005660 005600 005662 005722 005570 005186
 240-005056 005404 005234 005018 005296 005154 005118 004972
 005118 004864 004930 005108 004762 004698 004634 004656
 000000 004372 004480 004432 004408 004472 004330 004336
 004258 004230 004392 004352 004326 004280 004348 004090
 004184 004128 004042 003976 004042 003902 004140 004190
 280-004124 004052 004164 003950 004104 004032 003706 003674
 003418 003306 003104 002952 002848 002618 002922 002640
 002762 002782 002932 002992 003088 003568 003174 003328
 003282 003158 002992 002814 002576 002256 001928 001744
 001494 001352 001376 001254 001268 001100 001196 001052
 320-001054 001048 000972 000940 000826 000808 000760 000798
 000748 000772 000668 000678 000726 000650 000650 000672
 000696 000760 000644 000724 000648 000626 000620 000548
 000464 000470 000358 000344 000360 000356 000272 000328
 000278 000274 000280 000290 000254 000282 000250 000276
 360-000262 000232 000296 000262 000286 000288 000300 000232
 000252 000262 000278 000250 000260 000218 000232 000248
 000184 000196 000176 000194 000232 000152 000200 000192
 000178 000184 000186 000182 000186 000208 000206 000174
 000228 000202 000200 000204 000196 000154 000134 000184
 400-000204 000194 000152 000170 000146 000138 000206 000138
 000150 000136 000158 000146 000178 000176 000152 000160
 000154 000162 000200 000182 000168 000192 000154 000180
 000180 000158 000206 000248 000182 000228 000308 000330
 000424 000576 000724 000848 001094 001316 001492 001660
 001744 001894 002106 002098 002414 002484 002934 003202
 003820 004374 005032 005768 047616 411490 536876 362344
 190740 067288 021522 008532 005322 004072 003486 003230
 002708 002436 002170 001936 001772 001696 001538 001360
 001140 000928 000880 000698 000576 000370 000314 000228
 000202 000134 000132 000108 000106 000066 000088 000106
 000062 000084 000076 000062 000078 000062 000064 000062
 000052 000064 000056 000048 000072 000074 000042 000032

Run 364-A x 1/10

001658 000210 101096 000716 000606 000546 000450 000571
 002690 017640 100896 154758 153861 146492 135169 124658
 40 → 115876 109523 102830 097244 092013 088010 084127 080240
 077224 074774 073060 070375 068411 067945 069518 072093
 074413 073451 070505 067544 066287 066277 067344 066953
 065751 065195 064533 063697 063732 063997 064198 063767
 063802 064067 064734 065375 067284 070941 080460 093757
 80 - 106066 110169 104834 096056 089132 086337 085861 085806
 085786 084378 080220 073130 064343 055896 047845 042068
 038566 041848 059047 102078 179603 281040 367057 391541
 347633 257879 162313 087945 044052 021873 012224 008186
 006518 005691 005425 005090 004959 004779 004624 004483
 120-004383 004288 004118 003972 003837 003672 003547 003436
 003386 003321 003261 003271 003216 003131 003216 003101
 003151 003111 003171 003156 003336 003256 003351 003466
 003386 003391 003431 003366 003371 003316 003421 003602
 003732 003917 004073 004238 004208 004143 003997 003982

160- 003927 003962 004028 004063 004158 004393 004463 004569
 004614 004509 004473 004358 004313 004654 005060 005846
 006718 007705 008396 008517 008226 007565 006853 006112
 005511 005265 004839 004584 004373 004113 003852 003486
 003276 003266 003622 004248 005375 006963 008857 010526
 200- 011733 012159 011307 009719 007610 005786 004033 002950
 002209 001743 001538 001412 001367 001287 001292 001277
 001277 001297 001402 001513 001643 001738 001863 001793
 001678 001508 001452 001287 001152 001032 000926 000921
 000896 000911 000916 000866 000916 000896 000906 000851
 240- 000841 000791 000761 000771 000771 000761 000781 000811
 000736 000736 000731 000721 000686 000646 000651 000641
 000000 000626 000626 000636 000631 000631 000646 000596
 000606 000606 000606 000586 000546 000576 000566 000561
 000551 000586 000541 000546 000536 000516 000576 000566
 280- 000566 000576 000566 000576 000581 000576 000541 000516
 000470 000415 000430 000410 000385 000375 000370 000350
 000385 000365 000365 000420 000415 000430 000460 000470
 000470 000440 000455 000395 000365 000330 000315 000265
 000260 000235 000200 000215 000205 000225 000200 000215
 320 - 000215 000210 000185 000180 000180 000160 000145 000140
 000125 000105 000105 000115 000125 000100 000130 000125
 000135 000145 000140 000140 000130 000130 000120 000105
 000095 000080 000070 000060 000055 000055 000045 000045
 000045 000050 000045 000055 000040 000045 000045 000045
 000040 000045 000060 000050 000050 000045 000050 000055
 000050 000045 000035 000050 000045 000040 000050 000040
 000035 000035 000045 000035 000045 000035 000035 000040
 000035 000035 000035 000035 000040 000040 000045 000030
 000035 000030 000030 000035 000035 000030 000035 000030
 000040 000030 000040 000035 000035 000025 000035 000030
 000030 000030 000025 000035 000030 000030 000030 000030
 000030 000025 000025 000035 000030 000035 000040 000035
 000030 000035 000040 000045 000035 000045 000040 000050
 000050 000065 000065 000095 000110 000120 000150 000170
 000185 000225 000225 000235 000255 000240 000325 000340
 000360 000415 000450 000506 000626 000726 011092 108216
 122945 062715 032575 006377 001653 000771 000556 000460
 000390 000335 000315 000290 000245 000255 000235 000190
 000190 000175 000145 000125 000105 000085 000060 000050
 000035 000025 000025 000025 000015 000020 000015 000010
 000015 000015 000015 000015 000020 000010 000010 000015
 000010 000010 000015 000015 000010 000015 000010 000015

Run 322-E

```

011886 010612 009381 008486 007349 006620 006188 004949
004284 004022 003764 003585 003509 003354 004381 006437
040786 515604 1195731 1188073 1101018 1050601 1100313 1371428
40 → 1858853 1241274 1013538 1326215 795159 571421 486937 446955
423226 410652 396917 371860 341947 318890 308225 306053
313817 333144 365053 397605 398733 355738 298634 260354
242752 235014 234638 231737 224389 206303 185009 167335
157293 152154 148211 145079 141754 138542 134302 129833
80 -126278 126539 130101 141177 160637 182658 197275 190937
168042 139355 114258 098215 089417 085182 084872 086492
090563 098377 110161 121745 129623 125259 113727 099824
089002 084553 087190 088813 088241 084276 076838 066896
060286 055833 053397 053013 054380 057135 061473 067068
120 -070597 075314 077358 076133 072630 066932 060181 054929
052051 051435 051674 050405 048841 045132 043026 039315
039789 040208 042162 043627 044755 044571 043682 042110
040994 039883 039954 039074 039059 039334 039159 038596
160 -037417 036365 034113 033011 031866 031254 032145 032675
033660 035275 036507 036044 035562 035621 035659 035852
035453 035168 033032 029828 026725 024203 022258 019843
019432 019222 019247 018628 019060 018358 017631 017054
016406 016539 016497 015656 015566 015310 014905 014247
013266 013538 013161 013186 012759 012863 013094 013067
200 -012832 013025 013197 013609 013625 013306 013125 013507
012782 012191 012285 011790 011645 011543 011190 011350
011312 011109 011079 011287 011312 010918 010897 011006
010880 010788 010976 010545 10622 010524 010192 010541
010199 010377 010346 010084 010050 009731 010228 010119
240 -009857 009777 009951 009811 009828 009752 009754 009291
009467 009251 009769 009467 009463 009402 009230 009303
000000 008970 008989 008654 008572 008094 008140 007899
008182 007819 008381 008066 008172 008683 008305 008750
008800 008610 008920 008905 008312 008209 008172 007899
280 -007683 007266 007299 007039 007156 007031 006863 006647
006884 006941 007000 007140 007123 007285 007058 006993
006614 006756 006740 006511 006117 006067 006150 006123
005975 005855 005959 005874 006270 006123 006603 006476
006364 006553 006503 006591 006370 006476 006140 006083
320 -006012 005979 006285 006291 006416 006488 006230 005951
005373 005173 004685 004635 004398 003999 003849 003571
003324 003157 003109 003058 002836 003385 003559 003909
004270 004710 004817 005081 004999 004922 004207 003589
002930 002563 002123 001781 001567 001349 001144 001104
360 -001006 001006 000888 001056 001002 000930 000958 000865
000869 000850 000740 000779 000819 000836 000773 000790
000802 000825 000863 000907 000802 000821 000834 000763
000735 000752 000717 000727 000735 000647 000668 000678
000725 000731 000765 000752 000699 000691 000752 000723
400 -000649 000777 000651 000601 000620 000618 000563 000488
000559 000410 000446 000482 000462 000446 000425 000517
000494 000532 000553 000561 000460 000509 000462 000389
000402 000370 000316 000264 000335 000293 000312 000239
000239 000301 000268 000266 000251 000249 000213 000211

```

440- 000198 000219 000188 000219 000251 000278 000217 000171
 000213 000200 000221 000169 000202 000171 000175 000213
 000239 000253 000239 000219 000274 000243 000316 000314
 000404 000446 000603 000804 000903 001195 001303 001517
 001844 002050 002297 002635 002878 003440 003853 005058
 006643 045265 265511 562169 530468 181107 045518 015308
 006616 004379 003406 002966 002530 002376 001779 001611
 001146 000775 000595 000349 000303 000270 000247 000243

Run 366-A + 365-B

095587 090792 084824 080678 077292 074002 072279 070873
 070181 068860 064684 059785 053690 048467 045734 040145
 029639 020548 014776 011953 009852 008775 008005 013424
 067782 459618 1973951 756951 766657 1642043 2647401 3167956
 4476849 5705922 5360398 3866093 2498064 1698020 1346089 1199130
 1125864 1093563 1043939 966326 886884 838644 815956 811680
 846535 933557 1024607 1043584 951462 820384 713546 652713
 628189 627554 626356 594424 543463 487272 451805 419373
 409892 401386 401196 392679 385514 373371 366654 358406
 80- 366658 399890 444018 492505 515211 499278 437643 365682
 306001 268248 246326 234934 236796 242842 261010 287400
 314680 340305 342842 316611 282436 256062 245846 245605
 244907 242460 229072 209634 186839 167864 158330 150745
 152964 157705 167323 178223 189790 199912 207991 206313
 120- 199353 187769 173483 159018 151955 149261 148634 144232
 139502 132287 125004 118662 117117 117756 121252 123481
 124504 124359 122228 118964 116476 114447 112906 113288
 111971 113588 113706 111418 107085 105507 100059 096050
 094305 095160 097148 100941 102933 105974 106193 104856
 160- 103425 103184 101629 102429 099169 094952 089131 082954
 076084 070832 066417 065131 062392 063643 063322 062765
 060924 060228 058928 056842 054537 054366 053723 052276
 051431 051247 050218 049283 047910 048463 049877 050020
 050553 050328 049763 048885 048561 048332 050292 049053
 220- 048783 048665 047487 046275 043429 044176 042364 042490
 041482 040904 040966 042129 041892 042660 042680 043019
 043299 042613 042762 043113 042811 042043 042576 042737
 041884 042229 041884 041245 040137 041133 040376 040486
 039051 039661 040023 039749 040819 040688 042019 043048
 240- 043221 045664 046556 047520 047254 047779 046787 045413
 044307 043421 041672 039778 038445 038322 037100 036165
 000000 035838 037449 039141 040278 042680 043872 045930
 047291 048112 048426 048287 045311 043258 040982 038075
 036757 035612 032705 031968 031805 031501 032350 031895
 280- 032087 033181 033838 034136 034854 034522 034697 034075
 034383 035510 035981 035497 035944 036461 036514 036875
 037261 037322 038633 039253 038071 038892 038649 039580
 038704 038933 039839 040486 041064 042974 047577 050004
 053919 056813 058428 058528 057154 054882 051496 047324
 320- 043262 040355 037208 034140 030505 028500 026910 025639
 024906 025841 028008 030837 035420 039990 044254 047250
 048536 047991 045140 039486 033042 027143 021364 015921
 011775 009377 007401 005921 005319 005078 004562 004396
 004188 004151 003925 003968 004017 003831 003839 003892

360 - 003941 003802 004070 004176 004070 004196 004500 004123
 004172 004147 004327 003766 003623 003892 003355 003672
 003692 003684 003545 003949 004143 004384 004762 004464
 004676 004672 004713 004623 004327 003972 003770 003578
 003433 002925 002794 002639 002557 002261 002302 002433
 400 - 002606 002729 003041 002996 003004 003404 003302 003323
 002984 002594 002412 001929 001782 001412 001314 001049
 000953 000912 000802 000790 000843 000765 000757 000745
 000749 000769 000765 000794 000696 000728 000875 000888
 000920 001286 001659 001806 002057 002061 002453 002578
 440 - 002855 003184 003143 003480 003843 004468 005058 005192
 006258 006815 008119 009750 047414 517011 045427 815497
 572396 229266 088508 025410 011340 007311 006348 005425
 004680 004066 004070 003274 003135 002786 002578 002355
 002049 001839 001712 001433 001208 000973 000659 000581
 000516 000408 000379 000355 000326 000302 000326 000245
 000330 000302 000224 000220 000208 000151 000183 000151
 000147 000179 000130 000155 000155 000151 000106 000118

Run 548-A

100000 000176 000155 000180 000211 000320 000377 000412
 000448 000472 000506 000532 000606 000768 000839 001099
 001338 001715 002233 002364 003613 004469 005264 005951
 007150 008397 009754 011104 012806 014985 017294 020245
 023824 023667 034919 042309 051835 089907 264003 392449
 400 - 404197 381729 350826 321650 294249 272663 251934 233280
 215083 201462 189661 178448 169802 160700 150139 139297
 130320 124270 119091 114414 109848 104284 096799 090388
 084789 080543 075964 072021 068490 064734 061717 058781
 056083 053931 051159 049094 046438 044864 043069 040913
 80 - 038704 036486 034448 032745 030967 029654 028048 026854
 025221 024500 023313 021825 021059 020228 019176 018526
 018205 018031 018163 018231 018500 017878 017214 015506
 013894 013066 012313 011560 011169 010788 010523 010114
 009728 009718 009236 009222 008962 008628 008496 008278
 120 - 008040 008252 007924 007517 007492 007151 007178 006945
 006661 006650 006597 006351 006248 006273 006166 006108
 005702 005891 005771 005543 005429 005375 005368 005311
 005219 005261 005241 005195 005116 004981 004980 004776
 004940 004926 004839 004825 004741 004711 004588 004488
 160 - 004429 004270 004324 004113 004168 004170 004154 004107
 004090 004088 004062 003981 004039 004025 004056 003922
 004033 003872 003742 003808 003635 003626 003594 003596
 003448 003492 003492 003615 003573 003590 003645 003633
 003532 003509 003545 003538 003582 003490 003564 003525
 200 - 003451 003519 003483 003310 003203 003102 003177 002960
 003031 003049 002947 002938 002965 003029 002961 002994
 003068 002950 002979 002885 002945 002928 002953 002943
 002863 002884 002873 002886 002834 002822 002834 002891
 002941 002821 002824 002765 002903 002907 002906 002904
 240 - 002941 002952 002922 003040 003111 003179 003119 003095
 002980 002987 002955 002712 002824 002726 002676 002651
 002591 002615 002621 002650 002636 002700 002791 002772
 002871 002930 002978 003036 003031 002976 002888 002744
 002596 002552 002440 002309 002343 002176 002239 002248

~~250~~-002225 002312 002263 002231 002257 002390 002380 002435
 002367 002321 002393 002302 002237 002264 002230 002193
 002113 002192 002180 002208 002254 002372 002219 002309
 002395 002438 002461 002619 002552 002569 002538 002477
 002436 002434 002460 002552 002536 002600 002593 002584
 320-002578 002493 002399 002232 002013 001926 001713 001658
 001547 001445 001305 001415 001310 001383 001335 001455
 001625 001650 001929 002055 001994 002058 001915 001855
 001577 001358 001150 000938 000725 000649 000468 000339
 000320 000253 000238 000218 000176 000204 000214 000207
 360-000160 000174 000184 000153 000167 000168 000196 000185
 000175 000215 000192 000187 000168 000137 000188 000173
 000169 000174 000185 000171 000193 000178 000168 000169
 000165 000178 000170 000185 000203 000199 000189 000219
 000189 000211 000192 000204 000143 000160 000137 000148
 400-000125 000117 000131 000131 000121 000110 000126 000116
 000090 000115 000110 000141 000140 000124 000128 000108
 000111 000084 000066 000073 000064 000052 000036 000030
 000032 000035 000027 000029 000036 000028 000028 000035
 000023 000026 000027 000026 000023 000033 000027 000031
 440-000032 000024 000028 000026 000034 000028 000024 000019
 000029 000027 000032 000030 000026 000029 000036 000027
 000028 000030 000022 000026 000027 000038 000022 000033
 000023 000016 000021 000022 000023 000029 000022 000026
 000029 000021 000012 000020 000022 000024 000022 000026
 480-000020 000022 000018 000023 000019 000018 000021 000015
 000017 000014 000011 000009 000015 000009 000011 000014
 000008 000010 000012 000007 000008 000019 000011 000016
 000007 000014 000017 000004 000005 000014 000015 000014

Run 548-B

100000 000104 000112 000162 000224 000298 000296 000353
 000396 000414 000376 000403 000460 000528 000653 000777
 000948 001198 001555 001938 002384 002932 003619 004239
 004907 005910 006819 008083 009297 010999 013093 015643
 018817 022728 028475 035208 043851 099543 289420 380044
 400-370211 343190 314092 286150 262349 243152 224363 207397
 191482 180546 170095 159629 151972 144467 134050 125202
 116164 110894 106399 102268 097999 092687 086278 079887
 074407 071321 066608 063485 059859 056522 054087 050615
 048172 046261 043576 041568 039543 038075 036031 034191
 440-032523 030975 029332 027469 026158 024889 023427 022260
 021179 020134 019055 018103 017050 016305 015803 015309
 014633 014667 014842 015219 015687 015076 014034 012805
 011389 010411 009705 008917 008643 008382 007950 007840
 007367 007405 007084 006923 006825 006446 006206 006281
 480-005988 005895 005696 005680 005295 005480 005238 005021
 004792 004905 004901 004648 004600 004477 004405 004397
 004276 004368 004364 004088 004170 003873 004011 003761
 003878 003571 003602 003749 003679 003599 003590 003444
 003486 003447 003491 003423 003403 003360 003307 003238

160-003138	003151	003147	003115	003156	003086	003071	003053
002964	002932	002861	002992	002854	002969	002834	002838
002907	002852	002858	002960	002793	002700	002762	002773
002767	002717	002646	002622	002601	002548	002638	002602
002522	002574	002514	002547	002437	002569	002631	002569
200-002725	002638	002568	002523	002533	002496	002383	002362
002252	002314	002313	002249	002245	002328	002256	002233
002239	002068	002153	002181	002079	002050	002168	002098
002054	002019	001970	001947	002064	002132	002046	001988
001961	001956	001978	001988	001979	001936	001949	001934
240-001770	001946	001908	001843	001862	001863	001851	001851
001932	001781	001786	001728	001738	001775	001692	001763
001661	001716	001614	001664	001630	001677	001648	001603
001603	001641	001570	001615	001527	001543	001591	001607
001522	001478	001487	001551	001495	001462	001414	001465
280-001471	001505	001524	001540	001534	001551	001541	001545
001451	001403	001376	001353	001229	001202	001128	001077
000984	000988	000973	000987	000977	001027	001032	001153
001082	001233	001251	001255	001240	001178	001066	000970
000856	000795	000667	000572	000488	000441	000431	000377
320-000345	000329	000303	000309	000320	000291	000310	000271
000276	000246	000269	000225	000215	000228	000180	000250
000179	000225	000199	000196	000200	000248	000189	000166
000176	000152	000172	000142	000129	000104	000098	000104
000092	000078	000088	000087	000084	000087	000067	000074
360-000061	000064	000055	000073	000065	000069	000067	000066
000067	000064	000071	000065	000077	000075	000084	000070
000061	000080	000085	000057	000047	000088	000045	000063
000075	000048	000056	000049	000045	000045	000060	000062
000058	000058	000050	000060	000051	000050	000048	000045
400-000039	000039	000033	000038	000042	000037	000040	000034
000040	000037	000021	000028	000034	000027	000027	000017
000036	000027	000038	000019	000023	000022	000026	000018
000011	000019	000022	000028	000031	000024	000026	000029
000032	000025	000021	000029	000034	000017	000015	000023
440-000030	000029	000026	000025	000021	000027	000022	000024
000032	000031	000021	000035	000032	000030	000024	000025
000029	000023	000013	000029	000034	000024	000026	000032
000028	000029	000031	000023	000027	000030	000029	000023
000033	000023	000028	000030	000025	000023	000023	000024
480-000017	000026	000018	000023	000015	000013	000010	000022
000007	000011	000010	000014	000011	000015	000020	000014

Run 307-B

	020900	019842	018255	016258	01	005575	011447	00829
	008145	007182	006619	006281	005	005574	005405	005196
	005021	004985	004790	004633	004539	004486	004477	004355
40	004310	004260	004311	004383	004567	006209	014720	080069
	396273	727550	989225	044073	866584	625152	451803	356003
	311585	293299	281556	270152	258734	248622	238050	227752
	220355	217212	212998	206407	200075	194037	188299	182108
	176500	172704	168683	163083	157079	153015	149686	148793
80	151107	162075	185161	215783	242195	250056	235818	206156
	179044	160691	148578	141619	137116	135118	132750	128604
	122816	116601	109984	103397	099267	096883	098776	106526
	127025	167356	226265	288858	327894	327553	288334	229938
	174346	134037	106861	090004	081305	077298	076767	079195
120	083934	089035	091234	088353	083617	078008	074236	071606
	071621	072856	075915	078338	078432	076717	072970	067284
	061674	056371	053544	052551	056093	065374	080718	102159
	122185	136633	139605	128807	108497	085640	065571	050567
	041584	037476	035629	035026	033953	032369	031129	029615
160	028395	027943	028008	028252	028871	028934	028563	027854
	026973	025764	024692	023682	022998	022758	022197	021840
	021950	022214	022748	023414	023521	023646	023963	024386
	023907	024863	025515	026699	027414	027979	028425	027417
	026485	025482	024391	023318	022414	021876	021141	020493
200	020344	020278	020269	020283	020105	020051	020005	020085
	020407	020544	020705	020774	020626	020547	020474	020468
	020595	021244	021284	020896	019900	018986	017865	016436
	015270	015145	014690	014770	014557	014684	014479	014175
	014032	013799	013724	013665	013334	013361	013358	013688
240	013320	013489	013540	013317	013176	012666	012500	012002
	011329	011030	010896	010548	010702	010848	011007	011009
	011263	011337	011322	011359	011115	010745	010928	010471
	009935	009710	009436	009357	009064	008968	008698	008690
	008286	007938	008083	007915	007835	007774	007733	007622
280	007752	007515	007465	007520	007474	007389	007284	007275
	007083	006947	006963	006777	006826	006715	006669	006576
	006607	006592	006672	006627	006753	006600	006452	006307
	006258	006040	005899	005568	005239	005114	004980	004828
	004674	004521	004618	004353	004590	004682	004797	004786
320	004927	005044	004997	004927	004562	004459	004084	003723
	003520	003142	003037	002869	002763	002666	002618	002640
	002475	002537	002419	002369	002274	002231	002060	002041
	001808	001906	001840	001765	001750	001717	001683	001625
	001672	001679	001612	001675	001735	001752	001729	001677
360	001638	001630	001549	001365	001276	001046	001032	000986
	000902	000796	000736	000681	000646	000610	000632	000597
	000628	000621	000560	000582	000544	000475	000481	000482
	000495	000475	000477	000469	000476	000494	000465	000455
	000426	000373	000423	000404	000432	000385	000360	000421
400	000393	000381	000393	000388	000350	000328	000338	000379
	000318	000353	000334	000337	000309	000320	000309	000329
	000330	000293	000335	000313	000304	000286	000253	000224
	000242	000243	000228	000230	000229	000241	000216	000216
	000222	000222	000237	000233	000213	000231	000259	000249

440- 000248 000248 000271 000296 000288 000359 000455 000498
 000679 000808 000983 001116 001284 001451 001643 001788
 001930 002216 002450 002788 003153 003452 004221 005017
 005955 010049 057939 189554 200543 093265 041179 017640
 008203 005093 004003 003451 002987 002563 002263 002030
 001807 001595 001428 001190 000988 000764 000580 000385
 000348 000243 000245 000232 000155 000169 000157 000130
 000126 000102 000125 000107 000093 000085 000083 000074

Run 550-A

001814 001875 002001 001792 001724 001726 001622 001750
 265287 002330 002842 003589 004766 005870 008144 010299
 012383 015318 018211 020831 023459 027482 031505 036402
 042208 049174 058169 068546 080028 099331 277130 789979
 40 → 895221 795365 764165 813008 1000930 1338465 1558814 1402394
 995960 670985 505845 431945 402151 390035 376137 358666
 342322 327271 318209 311505 305006 302156 291471 279582
 268347 264992 257147 250732 242678 235781 226303 218935
 214454 213430 216574 230230 258371 307195 352218 371352
 80- 353019 314890 272455 238630 218166 205097 196665 192346
 186197 181525 173171 161156 153224 146062 142346 144631
 159855 196643 258608 349774 436444 486181 465422 389327
 293298 218731 163057 132639 117267 108440 108731 111648
 116968 124611 127651 123535 117649 109074 103975 100628
 120- 100845 103999 107139 108162 108373 105047 098052 089421
 082043 077169 077737 083067 096289 117110 145482 171658
 189020 190827 173348 144822 112109 084966 065904 054854
 049443 047777 046534 045968 043724 041905 040437 039449
 039690 039863 040879 040706 040357 040118 038697 037271
 160- 035462 034587 033585 033091 032162 032220 032085 032537
 032274 033256 034047 033607 034312 034838 035585 036229
 037189 038415 038905 039281 038620 038699 036791 034758
 033483 032680 031100 030803 029907 029618 029546 030441
 029582 029439 029060 028869 028725 029084 028923 029407
 200- 029124 029251 028986 028582 028534 028982 028697 029114
 028546 027703 027010 025430 023557 022854 022541 022148
 021397 021140 020989 020118 019869 019726 019265 019122
 018994 019570 018717 018976 018982 019070 019106 018203
 017910 017982 017307 017137 016637 015874 015432 015266
 240- 015328 015460 015633 015930 015579 016149 015749 016293
 015545 015697 015241 014623 014328 014083 014011 013025
 000000 012672 012383 012371 012491 011859 011272 011240
 011375 011383 011298 010869 010897 010925 011017 011290
 010829 010692 010483 010457 010244 010186 010200 010080
 280- 009961 009955 010040 010092 009905 009933 009871 009797
 009582 009500 009281 009241 008775 008622 008169 007934
 007685 007687 007448 007213 007193 007327 007378 007556
 007478 007879 007986 007801 008002 007747 007372 007357
 007006 006550 006450 005924 006040 005761 005555 005673
 320- 005816 005906 005802 006059 005876 005376 005272 005113
 004645 004541 004218 004037 003900 003892 003686 003348
 003607 003605 003533 003981 004069 004091 004352 004503
 004567 004458 004252 004147 003832 003457 003067 002397
 002142 001802 001567 001292 001146 001062 000977 000971

360-000895 000871 000855 000817 000696 000742 000793 000734
 000636 000644 000672 000646 000632 000584 000562 000596
 000562 000630 000485 000560 000501 000461 000461 000457
 000447 000451 000463 000395 000453 000435 000431 000453
 000423 000383 000385 000383 000401 000377 000351 000351
 400-000345 000343 000301 000281 000289 000281 000297 000252
 000295 000269 000240 000269 000214 000186 000263 000218
 000190 000222 000198 000210 000184 000208 000158 000158
 000142 000132 000170 000156 000164 000132 000166 000170
 000140 000148 000146 000140 000164 000110 000134 000130
 440-000148 000134 000114 000120 000112 000106 000156 000156
 000138 000134 000112 000110 000114 000120 000138 000122
 000118 000130 000136 000118 000086 000108 000136 000134
 000092 000088 000078 000094 000104 000044 000084 000092
 000054 000082 000104 000088 000082 000118 000094 000096
 000060 000086 000080 000062 000060 000080 000062 000080
 000044 000082 000064 000044 000042 000056 000054 000058

Run 312-C + 313-A

010046 003350 008518 007370 006478 005700 005406 004226
 003550 003204 002980 002930 002726 002524 002390 002488
 002426 002444 002258 002154 002086 002152 002026 001942
 720-002008 002038 002000 002012 002378 003710 018060 404778
 863136 720274 612226 591040 647634 733364 771784 702750
 583736 477896 411828 376726 367226 388942 441920 507990
 534238 508888 457408 420388 396560 376646 357850 337088
 318194 303854 286790 267490 243424 223176 207940 1656110
 80-181218 174614 173472 165652 159168 155280 155878 166396
 174836 175172 165834 149704 136110 129798 128156 129250
 130664 131038 129954 126494 121638 112896 105238 097392
 093384 093768 092280 090114 092362 093614 095352 093248
 090120 084212 078912 076602 075820 078812 079558 079372
 120-077768 074176 070472 065938 062960 061518 059608 058936
 057914 057510 058388 057860 059326 060514 060982 059770
 059374 058542 055984 054462 052788 051362 049692 048270
 047512 046420 044384 043600 041238 040540 039728 039324
 038438 038458 037736 037886 038392 038886 038686 038768
 160-039128 038902 039176 038476 037200 035310 035266 033360
 033332 032356 030808 030660 030018 029940 029092 028854
 028460 028384 028034 027924 027716 027490 026600 026498
 026584 026182 026048 025524 024558 025070 025010 024654
 023732 023276 022580 022254 021644 021016 021334 021188
 200-021202 020610 021200 020742 020944 020324 020016 019940
 019400 019188 018794 018318 017848 017532 017208 017040
 016644 016178 016236 016324 015688 015898 015878 015578
 015224 014970 014682 014650 014474 014358 014020 013856
 014272 013984 014044 014256 013544 013540 013238 013134
 240-013456 012386 012730 012762 012460 012332 012250 012302
 012040 011720 011550 011382 011376 011540 011450 011172
 000000 010984 010994 011082 010790 010848 010710 010386
 010496 010410 010578 010028 010236 010588 010480 010264
 010212 009864 010094 010448 001008 010248 010456 010076
 280-010374 010228 009954 009906 009518 009418 009690 009772
 009396 009946 009410 009712 009564 009384 009178 008872
 008886 008898 008712 008836 008386 008760 008738 008606
 008622 008556 008976 008556 008596 008618 008864 008906
 008430 008482 008798 008516 008238 007398 007566 007180

~~310~~ 007116 006814 006784 006890 006798 006780 006698 006492
 006328 006384 006284 006178 006454 006210 006150 005858
 005808 005518 005134 004938 004690 004574 004666 004454
 004482 004516 004178 003966 004108 003922 003612 003356
 003130 003318 003134 003328 003376 003448 003640 003454
~~360~~ 003278 002840 002516 002124 001828 001548 001152 001088
 000870 000772 000670 000692 000686 000680 000610 000554
 000602 000546 000550 000550 000518 000540 000640 000454
 000520 000474 000500 000446 000396 000426 000368 000388
 000372 000368 000334 000298 000298 000298 000316 000282
~~400~~ 000292 000260 000294 000272 000298 000256 000268 000232
 000262 000228 000292 000230 000250 000248 000218 000226
 000232 000200 000246 000208 000224 000232 000232 000226
 000268 000196 000246 000176 000256 000210 000184 000194
 000178 000202 000196 000176 000184 000198 000128 000154
~~440~~ 000178 000202 000178 000200 000184 000222 000208 000272
 000216 000204 000192 000210 000316 000270 000342 000370
 000450 000586 000830 000890 001078 001344 001554 001614
 001970 002104 002350 002602 003106 003584 004226 005114
 006724 042884 230398 553010 568034 167862 402968 018560
 007868 005074 003736 003150 002838 002528 002154 001900
 001638 001334 001034 000714 000540 000388 000276 000252
 000210 000220 000192 000192 000180 000150 000158 000158

Run 312-B

000000 006691 007456 009114 013083 014806 015464 016154
 016883 017531 017706 018239 017852 017775 017256 016550
 266010 014641 013293 011635 010575 009593 008377 006593
 005533 005104 004708 004558 004281 004114 003879 003852
 003729 003737 003581 003585 003500 002214 003308 003260
~~40~~ 003360 003008 003087 003256 003685 005531 024175 538843
 021315 021731 824016 758691 001458 909031 998502 955987
 810175 664058 559160 502104 478677 489029 541441 626454
 684477 685727 626443 568066 534479 510785 484083 454054
~~80~~ 33766 416206 395135 372416 343062 317291 294302 272183
 255112 243987 239452 232952 224412 216033 215439 221304
 233339 241429 237887 221218 201785 186504 180391 182100
 183164 186377 185706 181256 176354 168858 159133 148089
 141200 140741 137779 136654 135664 135400 137218 137741
 134985 129100 122137 1 7231 1 2856 1 4750 119106 121458
~~120~~ 120435 1 7481 110964 106381 100660 097212 095325 092925
 091885 090231 091066 091275 091791 091941 093625 093112
 092877 092387 090985 089477 086164 082925 081908 079989
 078664 078468 075875 074741 072858 070958 069914 068439
 066766 067241 066981 066185 065593 065877 066125 066335
~~160~~ 066564 066691 066435 066002 066156 064104 063166 062666
 060108 058643 058175 056937 055725 055831 055337 054185
 053641 054506 053287 052877 052491 052383 051452 051887
 051985 051237 050535 050275 050558 048714 048381 047943
 046933 047377 046377 045268 044910 044700 043329 043060
~~200~~ 042725 043200 043083 043104 042762 042777 041958 041568
 040700 0411 4 039843 039975 038404 038718 038135 037552
 036685 036381 036160 036183 036200 036333 035816 034666
 033881 034118 033629 033368 032506 031956 032252 032258
 032127 031914 032677 032177 031875 031785 031152 031241

~~140~~- 030833 030368 030533 030454 ~~030054~~ 030091 030158 029210
~~029285~~ 029125 028675 028364 ~~028308~~ 028433 027968 028079
~~006000~~ 027708 027835 027366 ~~026702~~ 026477 026381 025854
~~025689~~ 025681 026158 026168 ~~026108~~ 026083 026902 026316
~~026243~~ 026618 027250 027506 ~~027581~~ 027627 028591 029035
~~280~~- 029131 028883 029181 028779 ~~028106~~ 027641 027300 027231
~~026481~~ 027191 026881 026808 ~~026952~~ 026891 027233 026743
~~026456~~ 026733 027175 027214 ~~027377~~ 028364 028989 029614
~~029837~~ 030539 031304 031093 ~~030225~~ 030885 030231 029677
~~029193~~ 028304 028343 027477 ~~026764~~ 026541 025404 025654
~~320~~- 026089 026602 026218 026689 ~~027335~~ 027764 028008 028014
~~027793~~ 028241 027710 028352 ~~028766~~ 028883 028950 028835
~~027816~~ 027891 026350 025277 ~~023531~~ 021991 020716 019981
~~019364~~ 018802 019008 018908 ~~018241~~ 018518 017693 016735
~~015435~~ 014725 014368 013989 ~~013883~~ 014114 015075 015581
~~360~~- 015800 015541 014810 013268 ~~011731~~ 009958 007912 006104
~~004558~~ 003331 002637 001964 001725 001439 001264 001177
~~001031~~ 001093 000991 000827 ~~000933~~ 000862 000841 000958
~~000804~~ 000841 000793 000756 ~~000775~~ 000729 000577 000633
~~000539~~ 000566 000458 000458 ~~000479~~ 000487 000412 000483
~~400~~- 000435 000408 000391 000437 ~~000439~~ 000429 000387 000381
~~000404~~ 000400 000333 000379 ~~000308~~ 000389 000333 000337
~~000302~~ 000333 000310 000314 ~~000318~~ 000304 000312 000287
~~000312~~ 000281 000214 000283 ~~000293~~ 000314 000308 000241
~~440~~- 000266 000216 000285 000227 ~~000241~~ 000260 000204 000225
~~000243~~ 000268 000283 000239 ~~000287~~ 000266 000289 000250
~~000289~~ 000260 000304 000302 ~~000329~~ 000352 000433 000441
~~000660~~ 000754 001010 001181 001408 001714 001714 001935
~~002306~~ 002510 002727 003077 003560 004183 004641 005693
~~007335~~ 038735 290533 736008 470066 1 0656 038664 014154
~~006900~~ 004864 004029 003458 002927 002660 002491 001975
~~001710~~ 001464 001175 000835 000529 000437 000333 000306
~~000266~~ 000258 000237 000252 000250 000200 000191 000158

Run 362-C x 1/10

006478 006437 006094 005608 004996 004494 004331 003788
 002739 001865 001371 001102 000869 000751 000714 000792
 002767 017503 121740 218544 231513 220524 202865 186223
~~40~~ → 174500 173606 180206 188150 202612 261817 374908 453355
 413911 303465 200743 138248 113866 104204 096534 088366
 080847 075447 072140 075014 084427 098481 112609 127618
 141889 148849 140485 120079 097906 079594 069336 067185
~~80~~- 068903 069475 066299 058882 050620 042742 037615 036064
 036088 038076 038876 037460 035411 033035 031072 029247
 027847 027259 026977 027222 028035 028749 028488 028010
 026745 025798 025128 024618 024912 025900 027308 027753
 027733 027137 027810 029165 031035 032619 031704 029206
 025149 021548 018732 017062 016103 016291 016687 017642
~~120~~- 018324 018528 018373 017781 017132 016193 015662 015303
 014989 015009 015299 015283 014891 014323 013674 013107
 012670 012678 012764 012482 012254 012025 011719 011466
 011413 011580 011539 011503 011176 011127 010964 010850
 010809 010821 010935 010772 010507 010364 009984 009719
~~160~~- 009376 009229 009196 009119 009184 009486 010009 010470
 010911 011237 011074 010686 010160 009621 009045 008727
 008262 008233 008147 007886 007825 007670 007519 007331
 007217 007008 007082 007286 007188 007188 007278 007441
 007788 007882 008233 008343 008592 008674 008690 008433

200-008270 008213 007984 007645 007523 007433 007417 007433
 007384 007368 007315 007282 007437 007384 007470 007682
 007882 008249 008637 008743 008923 008915 008841 008527
 008115 007817 007584 007339 007229 007143 007082 007004
 007180 007131 007429 007772 007919 007996 008384 008596
 240-008723 008670 008976 008915 009017 009025 009343 009421
 009470 009886 010115 010405 010837 011078 011352 011903
 000000 012854 013866 014613 015511 016156 016544 016605
 016148 015785 015152 014393 014152 013838 014009 014368
 014809 015887 016968 017830 018389 018658 018720 018797
 280-018650 018585 018601 018540 017793 017230 015789 014119
 012401 010919 009882 009188 009111 009543 010564 011617
 012915 013948 014707 014274 013144 011405 009482 007302
 005445 004004 002812 002090 001555 001310 001110 000996
 000869 000894 000841 000792 000751 000730 000661 000665
 320-000616 000559 000567 000530 000498 000477 000457 000420
 000400 000408 000367 000408 000404 000392 000379 000359
 000347 000367 000367 000355 000347 000343 000330 000306
 000302 000298 000281 000281 000265 000269 000261 000281
 000249 000245 000228 000232 000220 000216 000228 000228
 360-000220 000220 000228 000216 000232 000224 000204 000220
 000208 000204 000179 000171 000175 000179 000167 000163
 000167 000163 000159 000171 000155 000163 000171 000159
 000159 000159 000159 000151 000147 000151 000134 000122
 000110 000118 000130 000110 000106 000114 000118 000118
 400-000114 000130 000126 000110 000106 000110 000110 000085
 000106 000081 000077 000081 000077 000061 000073 000061
 000065 000057 000061 000057 000065 000065 000069 000065
 000065 000061 000065 000053 000057 000065 000057 000065
 000065 000069 000069 000073 000069 000081 000098 000098
 440-000110 000155 000155 000171 000183 000208 000228 000224
 000232 000257 000298 000306 000355 000371 000404 000506
 000514 000563 000906 007221 052024 112131 089419 051996
 022373 005482 001555 000751 000567 000453 000379 000355
 000298 000294 000253 000220 000228 000224 000196 000175
 000130 000122 000114 000098 000073 000061 000049 000036
 000028 000028 000024 000020 000016 000028 000024 000016
 000020 000016 000016 000024 000020 000016 000016 000016

Run 363-A x 1/10

006895 006996 006654 006134 005325 004927 004606 004062
 003062 002112 001534 001168 000945 000799 000708 000905
 004048 020173 101423 176557 189326 180832 165932 150213
 410 → 139305 134919 136777 139867 147562 184243 008564 063695
 292386 → 043382 (270819) 148235 100941 079236 070059 064442 058820
 053984 050653 049032 050402 056336 063521 066886 064641
 061057 059661 059715 059338 054924 049020 044283 043030
 043028 042693 040376 036364 031980 027902 025263 023598
 80 → 023301 023470 023502 022783 021777 020875 019805 019175
 018558 018122 017895 018179 018895 019309 019134 018604
 221228 → 017697 017213 016562 016223 016339 016839 017656 018255
 018267 018181 018480 019389 020970 021823 021309 019299
 016572 013960 011745 010363 009763 009612 009974 010462

120- 010915 011333 011158 010863 010209 009688 009237 008941
 008624 008779 008889 008763 008361 007895 007227 006628
 006297 006048 006002 005923 005907 005883 005927 005976
 005901 005887 005984 005813 005789 005702 005646 005550
 005602 005634 005554 005333 005116 004893 004654 004361
 160- 004210 004064 004114 004144 004265 004556 004859 005142
 005343 005490 005486 005233 004783 004425 003961 003616
 003407 003289 003212 003126 002941 002935 002877 002805
 002769 002700 002660 002626 002574 002588 002618 002684
 002698 002851 002865 002919 002861 002885 002749 002682
 200- 002594 002508 002451 002437 002369 002367 002353 002341
 002281 002224 002224 002224 002188 002148 002152 002076
 002152 002140 002224 002210 002208 002210 002190 002184
 002074 002074 002004 002050 001975 001927 001977 001943
 001903 001939 001965 001873 001963 001953 002004 001965
 240- 001915 001939 001927 001939 001937 001939 001961 001963
 001994 002002 002106 002114 002142 002128 002112 002210
 000000 002295 002397 002443 002566 002670 002660 002624
 002626 002572 002498 002397 002319 002353 002293 002373
 002417 002530 002572 002682 002781 002817 002763 002763
 280- 002805 002821 002789 002724 002650 002576 002391 002190
 001949 001803 001666 001548 001534 001580 001672 001791
 001889 002052 002114 002066 001915 001791 001528 001293
 001086 000895 000720 000658 000558 000510 000417 000397
 000363 000357 000329 000315 000315 000281 000289 000259
 320- 000253 000265 000230 000210 000208 000202 000202 000192
 000194 000196 000180 000186 000158 000172 000172 000162
 000172 000158 000156 000150 000154 000150 000150 000136
 000120 000120 000120 000128 000112 000110 000108 000106
 000104 000104 000106 000098 000090 000104 000102 000088
 360- 000090 000100 000090 000100 000090 000100 000090 000086
 000086 000082 000084 000082 000084 000074 000074 000072
 000066 000074 000066 000064 000072 000064 000072 000060
 000064 000064 000062 000060 000064 000054 000056 000056
 000050 000056 000054 000048 000048 000054 000048 000048
 400- 000042 000042 000050 000042 000048 000042 000040 000038
 000036 000040 000038 000038 000032 000028 000030 000030
 000028 000032 000032 000028 000028 000032 000032 000030
 000032 000036 000032 000034 000032 000030 000034 000032
 000030 000032 000032 000036 000042 000042 000066 000084
 440- 000106 000126 000134 000152 000172 000178 000200 000204
 000208 000224 000259 000297 000303 000339 000399 000447
 000522 000730 006972 037657 053593 034436 020179 007054
 002154 000939 000540 000441 000375 000337 000291 000273
 000230 000204 000194 000176 000162 000152 000134 000108
 000096 000082 000072 000054 000038 000032 000030 000018
 000014 000016 000014 000014 000012 000012 000012 000012
 000008 000010 000012 000010 000012 000010 000012 000008

Run 325-D + 326-A

```

010170 009009 008451 008289 008343 010080 006767 084330
683469 1384327 1458928 1390177 1288315 1207117 1156897 1133902
+8 → 105444 026361 889317 768978 699012 685260 734427 858015
1021195 112941 1077418 955521 819972 668745 519012 405423
338364 305712 288450 275220 268101 266733 270666 274140
272205 266094 249399 232011 218349 205443 192852 176715
80 - 166140 160236 155880 150534 147177 144090 142065 140103
137682 135639 133596 132210 134955 140787 144306 142308
134289 125640 118089 110439 102690 096165 092574 090441
089892 089010 086049 081630 076806 073305 071145 068670
067734 068310 068904 067941 066438 064179 061515 059400
056655 055008 052236 050094 048321 046674 045594 044460
120 - 042804 042318 041913 041166 040320 040194 039573 039042
038313 038880 039627 039465 039429 038943 038547 037287
036495 035748 034902 033903 032733 032274 031437 031185
030528 030366 029448 028692 027918 027423 027522 027423
026901 027189 027243 026928 026829 027531 027765 027900
160 - 027783 027693 027243 025857 024651 023985 022743 022050
021312 020853 020763 020754 020799 021150 021465 021717
021204 020844 020403 019467 019062 018162 017856 017199
017001 016902 016893 016848 016488 016272 016236 016101
015912 016065 015768 015606 015408 015021 014931 014571
200 - 014400 014382 014265 014130 014175 014202 014265 014094
014067 014112 013914 013545 013041 012987 012843 012438
012042 012240 012186 012051 011709 011619 011493 011151
011250 011106 011133 010872 010962 010800 010629 010665
010521 010377 010296 010431 010170 010017 010215 009783
240 - 009513 009477 009567 009243 009216 008910 008883 008694
008541 008595 008523 008532 008631 008325 008487 008487

000000 008388 008442 008118 007866 007920 007866 007542
007551 007470 007398 007281 007497 007047 007200 007092
007056 007128 006939 007002 006975 006975 006858 006714
280 - 006597 006642 006507 006408 006543 006480 006192 006291
006165 006291 006435 006534 006552 006435 006444 006507
006255 006174 006048 005958 005787 005544 005373 005157
005049 004761 004887 004770 004833 004851 004950 004986
005355 005652 005661 005850 005787 005742 005463 005112
320 - 004725 004392 004104 003681 003447 003141 002943 002799
002772 002655 002682 002700 002637 002790 002754 002907
002970 002916 002853 002808 002619 002484 002286 002043
001881 001746 001611 001557 001539 001494 001467 001467
001467 001395 001431 001404 001404 001431 001467 001422
360 - 001341 001395 001242 001152 001116 001044 000909 000846
000801 000729 000675 000648 000639 000594 000639 000603
000612 000558 000594 000594 000576 000558 000531 000531
000513 000504 000477 000459 000468 000459 000459 000414
000396 000369 000369 000369 000351 000369 000333 000324
400 - 000324 000315 000342 000315 000324 000315 000279 000270
000297 000270 000252 000252 000261 000234 000243 000225
000243 000234 000225 000225 000171 000180 000171 000153
000180 000162 000162 000171 000180 000180 000135 000153
000171 000135 000162 000162 000171 000162 000144 000180

```

440 000171 000162 000153 000153 000126 000135 000162 000144
 000117 000144 000153 000135 000171 000144 000144 000153
 000153 000144 000180 000162 000198 000216 000216 000270
 000315 000396 000495 000639 000810 000927 001098 001251
 001440 001692 001899 002115 002484 002889 003465 004266
 005859 021258 101844 226629 178002 071226 026739 010674
 005346 003690 002916 002448 002079 001746 001458 001197
 000909 000666 000450 000342 000243 000198 000180 000144

Run 326-C x 1/1.0

001551 001440 001281 001180 001053 000944 000809 000670
 000566 000520 000469 000463 000482 000560 000923 000592
 087899 162101 158471 147509 138712 133441 134775 136875
 40 → 131054 114844 096582 084965 081948 087943 106335 133976
 154557 154245 140069 121324 099127 075203 055799 044753
 039833 038066 036515 035369 035306 036385 037807 038699
 037558 035097 032446 029991 028135 026035 024039 022567
 021661 021217 020724 020214 020026 019734 019610 019253
 80 - 018874 018809 018728 019315 020242 020900 020448 019411
 018098 017176 016200 014941 013652 013220 012990 013036
 012831 012252 011603 011097 010585 010322 010203 010257
 010385 010535 010435 009954 009680 009291 009057 008660
 008202 008024 007721 007422 007146 007150 006962 006686
 120 - 006582 006552 006533 006416 006364 006313 006315 006274
 006261 006315 006431 006408 006341 006176 006076 005902
 005755 005628 005531 005390 005375 005270 005149 005097
 005023 004936 004846 004783 004708 004697 004685 004566
 004631 004606 004656 004679 004791 004791 004846 004772
 160 - 004704 004538 004367 004181 004022 003901 003803 003760
 003727 003771 003760 003771 003746 003811 003859 003744
 003631 003520 003453 003351 003326 003202 003165 003161
 003113 003115 003100 003098 003073 003052 003092 003077
 003069 003010 002985 002941 002916 002801 002797 002688
 200 - 002721 002773 002771 002757 002763 002719 002721 002665
 002640 002632 002615 002516 002462 002435 002483 002341
 002320 002341 002377 002341 002314 002287 002265 002265
 002289 002304 002287 002281 002281 002267 002199 002176
 002115 002130 002086 002053 002008 002038 001977 001973
 240 - 001958 001927 001923 001879 001862 001879 001835 001812
 001775 001850 001883 001879 001848 001860 001837 001747
 000000 001672 001639 001605 001600 001554 001531 001538
 001532 001469 001469 001442 001471 001465 001468 001448
 001427 001422 001410 001399 001403 001348 001353 001312
 280 - 001302 001270 001301 001256 001288 001267 001281 001321
 001317 001311 001339 001346 001318 001324 001315 001302
 001257 001220 001222 001150 001119 001116 001081 001046
 001033 001044 001050 001046 001107 001129 001156 001170
 001159 001157 001140 001094 001045 001019 000899 000857
 320 - 000800 000746 000756 000721 000685 000698 000696 000678
 000708 000727 000756 000814 000794 000781 000795 000734
 000713 000670 000607 000559 000476 000460 000430 000416
 000403 000401 000373 000379 000411 000403 000436 000453
 000473 000492 000524 000530 000495 000482 000426 000411

360-000356 000330 000267 000228 000208 000174 000153 000141
 000138 000127 000124 000128 000127 000137 000137 000124
 000134 000127 000109 000106 000106 000103 000098 000096
 000096 000091 000087 000084 000091 000079 000076 000070
 000067 000072 000063 000058 000067 000057 000057 000058
 400-000053 000046 000058 000049 000048 000055 000056 000046
 000061 000061 000047 000058 000052 000051 000045 000042
 000036 000034 000033 000031 000025 000028 000023 000023
 000024 000022 000018 000022 000027 000023 000026 000021
 000021 000026 000025 000020 000027 000023 000026 000026
 440-000028 000026 000022 000018 000023 000020 000020 000019
 000017 000019 000018 000020 000020 000024 000021 000029
 000027 000023 000027 000029 000026 000028 000036 000033
 000037 000041 000053 000067 000081 000108 000116 000134
 000161 000188 000218 000227 000282 000307 000380 000458
 000587 001472 014587 053616 074509 023399 005260 001671
 000666 000447 000357 000300 000263 000208 000184 000145
 000116 000085 000059 000040 000031 000023 000027 000019
 000023 000021 000024 000015 000015 000015 000014 000013

Run 323-A

009444 008512 007902 007700 007604 008396 011586 037822
 353534 1624924 2073412 1989042 1840254 1707728 1572490 1423410
 40 → 1300006 1222696 1126342 980064 840816 742778 692152 694808
 750142 820206 803434 681730 536274 448544 414848 406520
 405062 392184 361944 330822 305886 293390 285536 283944
 282302 285010 281816 273962 259780 244410 228436 217610
 210602 205572 201528 198518 198220 201028 202690 199786
 80 - 197202 202660 213418 215780 210476 196130 181030 167368
 155190 148472 146642 143852 142492 142898 140200 137052
 131892 131418 126370 123580 120718 121728 130058 150210
 181794 215282 233926 226786 198106 163488 133228 110236
 096204 088660 084258 081698 081154 083294 086000 086972
 120 - 086968 085566 082386 079048 076932 075524 076862 077524
 078502 081266 081336 079770 076886 073534 070586 066864
 065312 064890 068570 075368 085434 097186 104256 107394
 102912 091114 080518 070152 062702 060566 061132 062480
 064882 066598 066074 063674 060670 058620 059290 060384
 160 - 061214 062302 060468 057298 053554 049402 046756 043968
 042658 042984 043714 045728 049964 056822 063254 069270
 071866 071322 068140 062656 059706 058816 060780 061542
 060824 058836 053152 046032 038376 032790 028928 026642
 024246 023328 022386 022448 021720 021662 022336 021332
 200 - 021124 020622 020948 020832 020686 021008 021282 021468
 021090 020956 020958 020554 020386 020230 020732 020224
 020224 019860 018748 018130 017946 017672 017640 017342
 017458 017090 016820 016768 016328 016366 015860 016190
 016328 016332 016370 016360 016748 017148 017138 016980
 240 - 017260 017160 017142 017082 016934 016628 016508 016028
 015362 015512 015518 015060 015078 015400 015662 015696
 000000 017624 018087 019058 018750 018729 018235 017041
 015834 014836 014236 013981 013754 013898 013994 013260
 013194 013348 013431 013144 013037 013517 013756 014411

~~280~~-015153 015528 015959 016013 015234 014682 013625 012516
 011387 010941 010138 010416 009920 010236 010501 010728
 010259 010032 009403 009419 008792 008611 008069 007467
 007083 006773 006533 006510 006039 006035 006035 006325
 006354 006929 006773 006939 007033 007231 006931 006637
~~320~~-006137 005733 005258 004937 004474 004393 004272 004195
 003870 003872 003513 003184 002917 002646 002644 002390
 002217 002246 002015 001996 001967 001898 001938 001644
 001863 001721 001825 001625 001804 001929 001898 001767
 001852 001865 001577 001415 001488 001281 001177 001181
~~360~~-001146 001060 000975 001067 000910 000962 000944 000992
 000825 000837 000858 000804 000821 000808 000758 000771
 000681 000727 000673 000756 000773 000635 000702 000700
 000696 000714 000589 000735 000683 000610 000658 000506
 000558 000635 000571 000581 000548 000504 000529 000460
~~400~~-000500 000437 000510 000475 000454 000433 000477 000366
 000483 000450 000412 000433 000416 000348 000381 000352
 000402 000325 000327 000366 000339 000323 000341 000310
 000308 000379 000300 000333 000331 000325 000304 000308
 000302 000291 000387 000348 000298 000262 000300 000358
~~440~~-000304 000337 000310 000289 000250 000262 000304 000306
 000279 000306 000264 000289 000266 000306 000312 000304
 000252 000285 000358 000300 000310 000335 000391 000466
 000523 000687 000808 001100 001477 001738 002194 002548
 003015 003524 003797 004216 004749 005558 006185 007196
 008705 011762 073399 312886 577623 435085 152574 055416
 020550 010772 007640 006333 005418 004630 003976 003219
 002536 002013 001252 000894 000581 000452 000337 000291

Run 324-B x 1/10

004592 004696 004642 004523 004375 004272 004043 003921
 003666 003409 003100 002722 002391 002110 001915 001584
 001315 001194 001116 001048 001046 001206 001718 008043
 072401 218721 244117 231989 215553 201879 188322 171757
~~40~~ → 159065 150913 140741 122136 104176 092119 085870 086683
 095580 105863 105541 089790 070460 057852 052871 051857
 051899 050797 046783 042667 039777 038011 037089 036637
 036410 036936 037326 036318 034649 032718 030977 029535
~~80~~-028578 027933 027544 027060 026779 027373 027519 027301
 027152 027807 029337 030027 029648 028395 026580 024355
 022596 021476 021024 021255 022118 022976 023134 022468
 021000 019986 018753 017701 016884 016743 017793 020233
 024345 029208 032346 032186 029051 023821 019444 015890
 013601 012505 011768 011380 011238 011454 011732 012019
~~120~~-011987 011940 011515 011199 010844 010766 010699 010822
 010876 011123 011308 011207 010946 010537 010061 009599
 009268 009141 009463 010280 011511 012998 014251 014954
 014436 013322 011730 010284 009165 008686 008537 008758
 009047 009310 009270 008955 008635 008410 008394 008535
~~160~~-008587 008740 008676 008290 007730 007413 006927 006561
 006409 006362 006509 006744 007202 007816 008698 009441
 009961 009989 009706 009119 008535 008356 008631 008659
 008788 008555 007987 007025 005973 005240 004523 004049
 003831 003700 003606 003559 003525 003551 003475 003473

200- 003405 003379 003375 003401 003479 003499 003582 003648
 003614 003565 003513 003377 003323 003206 003190 003128
 003112 003041 003021 002945 002917 002969 002915 002802
 002776 002774 002738 002698 002672 002620 002604 002566
 002608 002630 002636 002628 002656 002694 002694 002696
 240- 002742 002720 002772 002762 002732 002696 002732 002670
 002644 002616 002608 002668 002594 002614 002668 002656
~~010000 002752 002622 002893 002900 002871 002808 002669~~
 000000 002765 002826 002893 002900 002887 002808 002686
 002615 002445 002361 002368 002385 002449 002524 002587
 002649 002697 002670 002665 002574 002488 002414 002380
 280- 002350 002457 002454 002428 002374 002294 002133 002021
 001872 001803 001724 001820 001850 001947 002074 002175
 002196 002192 002028 001901 001730 001533 001390 001316
 001166 001093 001040 001016 000970 000918 000926 000931
 000935 000982 000993 001010 001036 001048 001032 000950
 320- 000877 000891 000830 000789 000736 000718 000722 000697
 000736 000733 000686 000669 000613 000600 000548 000505
 000467 000440 000413 000400 000385 000344 000335 000343
 000309 000335 000335 000343 000359 000375 000385 000405
 000411 000440 000420 000400 000356 000332 000294 000248
 360- 000224 000193 000180 000158 000166 000141 000136 000130
 000130 000133 000120 000124 000116 000118 000115 000103
 000099 000103 000101 000096 000100 000098 000090 000105
 000095 000095 000092 000098 000095 000097 000087 000089
 000086 000090 000076 000085 000077 000070 000077 000076
 400- 000068 000073 000072 000070 000064 000062 000064 000068
 000057 000065 000056 000055 000062 000051 000042 000050
 000044 000047 000047 000043 000046 000047 000047 000044
 000042 000042 000047 000049 000046 000041 000036 000040
 000034 000048 000044 000041 000048 000040 000041 000051
 440- 000041 000039 000041 000037 000036 000037 000033 000041
 000042 000036 000036 000039 000043 000034 000038 000037
 000037 000041 000042 000044 000042 000046 000061 000056
 000066 000088 000108 000136 000173 000218 000277 000306
 000360 000385 000459 000483 000543 000615 000710 000831
 000977 002261 015023 048963 062285 025746 009018 003405
 001414 000920 000731 000624 000535 000462 000408 000307
 000251 000183 000122 000079 000049 000046 000039 000036

Run 372-A x 1/10

004065 004065 003915 003446 003103 002701 002416 002048
 001471 001040 000745 000564 000470 000395 000389 001180
 006693 036202 114434 144611 143016 134774 125644 118305
 40 → 114723 108496 096077 081354 070346 065471 067538 076833
 085144 082673 068878 053158 043051 038735 036982 036214
 034691 033571 032793 033563 033867 033226 031371 028815
 026274 024601 023512 022478 021872 021464 021096 020748
 020555 020170 019614 019302 019266 020357 022074 023462
 80 - 023402 021918 019687 017651 016309 015528 015029 014702
 014504 014394 014315 013957 013584 013264 013058 012773
 012829 013487 015204 017947 021258 024220 024755 022532
 019377 016061 013674 011911 010698 010021 009786 010069
 010837 012263 013412 013832 013412 012005 010627 009343

120- 008550 008140 008169 008317 008739 009457 010202 010860
 011045 010646 009746 008637 008096 008267 009359 010469
 011476 011894 011226 009728 008171 006579 005542 004787
 004383 004221 004096 003954 003798 003802 003752 003750
 003871 003927 003981 004017 003969 003942 003809 003802
 160- 003765 001109 003990 004023 003879 001292 003650 003519
 003355 003297 003290 003334 003467 003688 004015 004327
 004470 004610 004614 004352 004100 003763 003490 003359
 003176 003016 002828 002820 002670 002695 002545 002543
 002504 002545 002541 002666 002849 003022 003226 003247
 200- 003309 003213 003051 002870 002722 002537 002406 002285
 002198 002131 002152 002091 002077 002079 002029 001975
 001925 001850 001833 001771 001713 001792 001740 001713
 001736 001748 001675 001721 001729 001729 001721 001742
 001704 001725 001700 001694 001648 001617 001575 001561
 240- 001505 001536 001517 001490 001490 001550 001590 001625
 001663 001677 001646 001577 001550 001494 001446 001407
 000000 001284 001238 001209 001157 001128 001107 001067
 001103 001134 001174 001228 001267 001330 001348 001367
 001346 001190 001163 001038 000970 000907 000897 000851
 280- 000876 000853 000882 000859 000807 000811 000776 000774
 000722 000691 000662 000639 000620 000620 000553 000572
 000576 000578 000568 000599 000622 000620 000618 000603
 000601 000580 000551 000493 000462 000424 000395 000387
 000395 000343 000333 000318 000316 000291 000281 000235
 320- 000233 000218 000200 000189 000168 000179 000156 000158
 000168 000156 000160 000152 000154 000156 000164 000158
 000164 000156 000158 000154 000150 000129 000116 000112
 000091 000077 000081 000068 000066 000075 000062 000060
 000056 000052 000052 000052 000043 000054 000054 000050
 360- 000050 000050 000043 000041 000050 000050 000050 000037
 000041 000035 000039 000041 000035 000033 000041 000029
 000039 000035 000031 000033 000031 000031 000031 000035
 000029 000033 000031 000035 000025 000029 000029 000029
 000031 000027 000027 000027 000025 000018 000018 000018
 400- 000018 000016 000025 000018 000016 000018 000014 000016
 000016 000016 000018 000018 000012 000018 000018 000025
 000018 000016 000025 000027 000027 000025 000033 000037
 000050 000062 000077 000102 000112 000118 000141 000156
 000185 000193 000191 000210 000225 000262 000285 000324
 000370 000414 000481 000580 003777 015643 043813 054390
 032630 013761 003621 001259 000630 000481 000395 000349
 000322 000281 000235 000208 000202 000166 000175 000156
 000135 000106 000100 000089 000060 000056 000039 000027
 000025 000016 000016 000012 000010 000008 000008 000008
 000006 000006 000006 000008 000006 000004 000004 000006
 000006 000006 000006 000004 000006 000004 000004 000004

Run 371-C x 1/10

	005838	005804	005546	005082	004478	003918	003554	002944
	002212	001530	001144	000862	000694	000596	000526	001252
	007030	035066	117540	158414	161918	157040	148748	141522
40 →	136382	128386	113778	097202	085002	079366	083342	095418
	104462	100458	084630	066934	054886	048780	046806	045636
	043774	042760	042470	043092	043364	042252	039916	036794
	033906	031976	030508	029564	028872	028406	028116	027798
	027538	027650	027334	026816	027070	028140	030000	031310
80-	031070	029810	027480	024996	023240	021854	021008	020556
	020050	020064	020188	019836	019556	019326	019236	018990
	018820	019538	021486	024868	029060	032172	032526	029796
	025870	022010	018802	016556	015158	014264	013940	014306
	015180	016912	018232	018430	017870	016452	014858	013308
120-	012076	011792	011726	011920	012520	013422	014402	014902
	014970	014432	013386	012328	011662	012078	013382	014770
	015836	015970	015154	013450	011452	009746	008380	007456
	007124	006732	006646	006400	006308	006190	006272	006286
	006308	006466	006482	006524	006396	006468	006376	006350
160-	006380	006506	006664	006538	006456	006330	006154	005886
	005612	005450	005464	005512	005626	006000	006394	006686
	006960	007028	006982	006708	006230	005942	005586	005340
	005048	005008	004970	004720	004612	004550	004394	004298
	004224	004308	004370	004520	004708	004982	005050	005154
200-	005066	004912	004692	004502	004254	004080	003906	003828
	003714	003714	003678	003608	003550	003504	003490	003380
	003378	003340	003286	003276	003274	003236	003250	003256
	003198	003244	003232	003278	003272	003198	003274	003210
	003182	003154	003128	003040	002934	002976	002910	002836
240-	002814	002732	002684	002602	002564	002576	002564	002620
	002624	002668	002690	002674	002660	002600	002576	002444
	000000	002130	002004	001880	001756	001640	001612	001540
	001528	001586	001640	001726	001786	001820	001812	001764
	001692	001658	001584	001438	001366	001342	001286	001282
280-	001258	001260	001252	001230	001236	001234	001156	001146
	001120	001076	001060	000988	000960	000962	000950	000952
	000932	000962	001002	001000	000988	000994	001028	001028
	001006	000962	000918	000882	000834	000842	000808	000786
	000792	000792	000820	000784	000788	000766	000732	000646
320-	000624	000586	000540	000514	000448	000442	000424	000398
	000380	000374	000372	000390	000412	000420	000474	000470
	000552	000542	000524	000526	000464	000438	000372	000316
	000254	000204	000162	000148	000112	000120	000090	000096
	000088	000082	000084	000066	000064	000066	000066	000064
360-	000060	000060	000060	000062	000066	000060	000064	000062
	000062	000058	000058	000052	000050	000052	000050	000042
	000052	000038	000040	000042	000050	000050	000042	000048
	000034	000042	000040	000038	000038	000036	000038	000036
	000030	000036	000036	000036	000032	000028	000028	000032
400-	000030	000026	000026	000028	000024	000032	000030	000018
	000026	000026	000028	000028	000016	000028	000026	000024
	000026	000006	200030	000028	000030	000032	000030	000036
	000036	000042	000062	000080	000088	000104	000136	000146
	000170	000176	000216	000202	000216	000246	000260	000296
	000308	000358	000416	000460	000570	002274	014960	033340
	038500	031578	021782	010636	004932	001764	000780	000514
	000394	000342	000308	000254	000252	000232	000202	000192
	000170	000152	000132	000106	000104	000080	000064	000048

Run 311-A

	028862	030292	030821	030659	030792	031106	030514	029275
	275260	026497	023843	020823	017909	016410	014067	011451
	009166	008211	007325	007045	006343	006157	005914	005930
	005849	005517	005397	005370	005224	005075	004793	004950
40 →	004853	004627	004681	004809	005448	007404	025717	202843
	658741	891610	953165	839387	687293	580983	523173	496283
	483072	478698	466921	460186	467789	494449	519028	518881
	490411	465035	444592	430754	417565	408187	403210	393426
	385397	371430	349829	324733	301170	284046	270822	263173
80 —	266231	294193	373172	541993	816613	1131044	1300783	1196535
	903690	594475	368998	241921	183523	158207	146206	142758
	139094	137145	133965	124966	117190	111630	115584	127801
	157273	197922	233818	242909	223473	186919	147428	116171
	097183	086631	079845	077766	077940	078326	079011	080063
120 —	077651	076661	072257	067696	064772	063909	062507	062925
	063722	066381	067742	067952	067238	067292	067966	068437
	068251	065988	063876	063394	067296	076084	085676	091778
	091969	084588	073008	061642	051999	046741	044689	044352
	045388	044983	043705	040195	037440	034321	032618	031062
160 —	030510	030248	030744	030311	030610	030734	030152	029528
	029210	029198	028374	028517	028291	029202	029470	030130
	029561	028727	028069	027793	026728	026907	026950	027450
	027660	027646	028793	028791	028096	027427	026363	026053
	024362	024073	022953	022577	022566	022234	021873	021861
200 —	021605	021707	021522	022074	022151	022220	022913	024188
	024708	025518	026178	025781	026068	025632	025308	024887
	024002	022566	021516	019949	019299	018471	018114	018288
	018131	018482	018602	018633	019104	019052	019443	019529
	019945	019924	020310	020281	020530	021381	021750	021107
240 —	020734	020663	019391	018924	018357	017317	017184	017033
	017002	017118	016906	017178	017344	017475	018108	018984
	000000	021419	023619	024785	026066	026417	026447	025011
	023891	022224	020479	019353	019056	018208	016989	016607
	016555	015866	015497	015347	015545	016096	016553	018131
280 —	020173	022442	024513	026045	025572	026213	024825	023010
	020244	017894	015613	014262	013228	013025	012815	012830
	012674	012450	012070	011968	011669	011428	010846	010986
	010507	010360	009963	009349	009289	009035	008784	008633
	008604	008931	009282	009864	010285	010434	010378	010302
320 —	009660	009316	008684	007736	007304	006912	006827	006675
	006102	006218	005916	005465	004880	004564	003952	003873
	003472	003388	003053	003070	003215	002931	002773	002651
	002499	002387	002235	002194	002166	002007	002061	001982
	001929	001939	001833	001818	001829	001715	001640	001550
360 —	001401	001453	001418	001353	001335	001349	001245	001198
	001156	001063	001013	000957	000974	000955	000903	000899
	000903	000855	000891	000853	000843	000949	000909	000943
	000922	000845	000924	000874	000901	000741	000818	000710
	000731	000751	000725	000700	000751	000635	000698	000669
400 —	000608	000625	000662	000719	000652	000608	000656	000631
	000727	000600	000637	000558	000559	000627	000571	000512
	000571	000486	000581	000465	000505	000463	000504	000482
	000527	000502	000527	000521	000498	000571	000498	000386
	000484	000488	000453	000436	000382	000405	000361	000378

440- 000397 000476 000415 000363 000390 000480 000480 000490
 000457 000376 000496 000527 000525 000558 000679 000801
 001029 001270 001870 002078 002404 002821 003273 003740
 004228 004880 005150 005720 006599 007485 008736 009683
 011933 035622 409397 606026 330866 128940 050957 020902
 011715 008600 007149 006136 005245 004897 004116 003701
 002806 002427 001793 001326 000807 000633 000465 000384

Run 310-A

042044 043958 044324 045304 045142 045018 044120 043010
 297794 039224 036854 031966 029006 024694 022216 018294
 014766 012526 011506 010488 009838 009632 008706 008492
 008388 008326 008014 007706 007622 007256 007316 007178
 40- 007226 007156 006952 007058 007518 010026 029978 206364
 694460 969972 1151172 1185136 1052334 872106 736036 657858
 620018 604026 591200 582332 577680 585942 611218 644546
 649076 629778 593186 566208 546138 532294 523088 509450
 504374 496366 482004 455392 427208 397494 372396 354074
 80- 343240 343656 361480 420804 563728 830716 1194972 1507982
 1598856 1414966 1045030 687556 432978 300086 235372 205124
 190754 185884 183078 180398 173712 165664 158574 152862
 156138 175318 212486 258638 297886 312786 291674 246718
 199220 160956 133470 116790 108458 103360 103538 103970
 120- 104348 106132 104616 101828 092268 092942 089172 086844
 084258 084856 086786 087836 090190 090302 090386 089626
 089914 091356 091834 089380 087470 085618 088602 096060
 107086 117770 123942 120096 110914 095840 082444 071124
 065616 063600 062518 062910 061428 059678 055690 052210
 160- 048064 046110 044560 044522 041622 045048 045402 046818
 048506 049340 050744 051212 050156 048858 046882 045448
 044000 043630 042682 041462 040220 040440 039510 038548
 038582 038436 038668 039134 039952 040254 040054 039978
 038772 037796 036390 035922 034640 034364 033684 032964
 200- 032008 032786 032774 032466 032344 032218 032466 033086
 033032 034972 036548 038058 038470 038918 039276 038226
 037716 037372 036752 035330 033774 032934 031522 030044
 028386 027600 027872 026950 027136 027218 028294 029354
 030272 031724 032708 033680 033420 032952 032222 031318
 240- 030638 029494 029890 029530 028192 028220 026810 026254
 024258 024174 023366 023172 023330 023000 023296 023102
 000000 023280 023814 024402 025716 027552 029218 030932
 032222 033928 033284 033324 031850 030604 028672 026918
 025474 024566 024050 023460 022752 022250 022078 021332
 280- 020896 021308 021626 022556 023870 026610 028590 031108
 032726 033568 033034 031684 029418 026404 023992 021722
 020208 019226 018392 018286 018412 018264 017648 017236
 016732 015804 015600 015438 014634 015056 014652 014218
 013802 013264 013282 012528 012780 012436 012622 012968
 320- 013148 013650 014196 013824 014076 013550 012854 012596
 011590 010612 010464 010344 010116 010142 009838 009860
 009666 009182 008586 008278 007670 006896 006470 006048
 005550 005332 005230 004748 004688 004514 004174 004130
 003722 003856 003756 003688 003756 003782 004134 004064

360-004150 004208 003850 003700 003544 003184 002848 002648
 002442 002196 002238 002138 001974 002004 001750 001874
 001806 001680 001540 001496 001414 001288 001334 001388
 001166 001184 001308 001346 001448 001384 001344 001386
 001334 001394 001362 001236 001318 001236 001196 001242
 400-001688 001144 001042 000986 000880 000950 000884 000978
 000884 001008 000984 000950 000930 000836 000906 000982
 000848 000934 000890 000880 000794 000848 000798 000670
 000746 000702 000720 000736 000724 000708 000674 000628
 000556 000578 000696 000578 000568 000650 000616 000598
 440-000596 000608 000624 000564 000556 000680 000668 000722
 000700 000726 000784 000916 001170 001416 001724 002206
 002526 002786 003512 003778 004096 004716 005340 005768
 006532 007362 008044 009244 010492 012170 031930 263392
 650522 420728 159778 065736 026588 013570 009756 008340
 006824 006520 005534 004954 004458 004064 003266 002770
 002100 001438 001076 000860 000722 000590 000528 000480

Run 505-A + 504-B

073094 069853 066365 064568 061242 058722 055861 054401
 052081 051954 052444 051269 049093 046744 042235 042493
 048401 229567 804920 024534 878296 417011 467129 708467
 40 → ~~10558~~ 1207273 1874480 1620799 1427880 1289772 1202690 1167321
 134600 1038068 880907 747797 680359 682025 760686 863414
 890984 783252 612488 486601 420841 398700 392061 387967
 380770 384926 395966 403786 402188 381659 350530 322024
 293369 278983 266244 257639 251654 249730 248110 247972
 244576 240342 236964 242493 254518 280353 306366 323428
 80 - 315729 293898 263412 238016 221306 212595 207588 202315
 200358 196755 192719 186126 179291 174054 169491 164755
 171382 199092 262217 359822 478069 567125 577746 499781
 373631 257006 169718 119525 097199 089733 088146 089044
 088259 089129 083577 078419 071407 066278 063418 063027
 120 - 064495 065955 068891 072373 073787 073683 071659 065197
 058908 054609 052571 053928 060230 070861 083467 092353
 095739 091555 079360 064214 050915 042454 038022 037102
 036700 038355 038118 036731 035881 034536 033303 032093
 031414 031185 030246 029996 028982 028865 027766 026485
 160 - 026329 025729 025723 025340 025752 025706 026641 027364
 028447 029478 029846 029873 029361 028732 028707 028066
 028822 029588 029336 029228 028026 027470 026054 025315
 024136 023982 024082 024305 025250 026048 027970 029105
 030073 030438 029807 028280 026583 024584 022884 022139
 200 - 020394 020177 019692 019546 019417 019182 019590 019184
 019042 018605 018051 016756 016481 016547 015666 015350
 015375 015331 016091 015698 015658 015904 015596 015564
 014442 015073 014144 013811 014132 013917 013511 013696
 013099 012526 012849 012745 012095 012195 012457 012751
 240 - 012776 012726 012828 013544 012876 012930 011989 012041
 011649 010743 010777 010275 010073 009713 009979 009765
~~009000 009000 009000~~
 000000 009827 009742 009469 009217 008730 008809 008480
 008257 008255 008009 008180 007466 007557 007601 007241
 007289 007116 007130 006951 006895 007118 007251 007153

280-007130 007157 006976 007003 006753 006616 006537 006260
 005910 005918 005589 005181 005029 004935 004808 004579
 004765 004846 005116 005158 005431 005635 005487 005618
 005637 005414 004929 004456 003896 003575 003240 002930
 002888 002655 002684 002457 002495 002503 002309 002284
 320-002234 002166 001993 001976 002103 001891 001953 001866
 001847 001612 001697 001728 001714 001730 001649 001672
 001522 001545 001637 001522 001410 001301 001208 001076
 001112 000958 001008 000931 000870 000885 000791 000824
 000758 000733 000743 000608 000733 000662 000645 000708
 360-000635 000662 000604 000685 000554 000718 000620 000639
 000631 000631 000560 000512 000535 000531 000535 000495
 000493 000454 000495 000454 000477 000454 000429 000441
 000485 000408 000485 000404 000335 000356 000379 000362
 000293 000331 000304 000287 000302 000281 000237 000204
 400-000270 000212 000243 000291 000233 000200 000256 000210
 000231 000258 000252 000216 000235 000243 000175 000181
 000206 000214 000166 000166 000191 000158 000177 000183
 000118 000150 000125 000181 000183 000177 000156 000162
 000154 000177 000137 000162 000137 000183 000127 000160
 440-000158 000131 000133 000131 000108 000143 000152 000139
 000125 000108 000089 000162 000089 000093 000068 000079
 000089 000108 000087 000102 000100 000110 000102 000085
 000075 000114 000087 000102 000083 000081 000077 000077
 000068 000093 000064 000058 000058 000075 000066 000079
 480-000054 000054 000066 000081 000062 000039 000066 000060

Run 504-A

200000 038411 044771 057612 084435 096526 101105 098627
 093311 090970 088322 083320 080782 077335 074790 072350
 070043 070735 072196 070275 066909 062551 057264 054088
 057085 126843 060927 588447 852440 260196 868202 091349
 454275 006451 1154013 1883675 1673648 1526695 1449106 1444557
 40 → 1438550 1326594 1115124 927879 839202 844435 952361 1109752
 1161945 1027577 800812 619768 531093 500149 490642 484184
 481001 486487 502716 520065 516821 489450 452068 413867
 382230 359967 348856 336563 330475 326810 327164 324880
 320149 315085 314216 317709 333587 365975 404531 430632
 80-428853 403499 364946 330963 305097 289259 280479 274013
 268359 266004 257707 250789 243743 236020 226582 222320
 228368 262092 337152 464722 623186 749936 779045 683195
 519465 358959 242778 177725 147091 135085 131718 130707
 127212 124095 118262 109001 100638 093446 090045 088301
 120-089001 092312 095549 100002 102646 103347 099952 092859
 085131 078806 076974 078525 086844 099850 115776 128694
 131999 124874 110689 090352 073294 061457 055404 054461
 055779 058211 059511 057869 055641 052569 050642 048089
 046390 045715 044533 044056 042303 041512 040424 038905
 160-038117 037008 037233 037083 036614 037204 038644 039940
 040641 043221 042823 043875 042679 041970 041666 041093
 041820 042312 041993 042343 041078 039711 038461 037212
 036177 035954 035433 035456 037000 038819 040780 042831
 044477 045711 045157 043783 042114 039715 037473 035454
 200-033593 032672 031765 031659 031075 031627 031498 030952
 030037 029787 029702 028585 027505 027130 027561 028685
 028377 030663 032186 033018 032703 033343 031827 029970
 028591 027507 026663 025834 024525 024173 023071 022569
 021393 021512 020487 020539 020191 020737 022235 024300

240- 025661 027912 029602 029581 028445 027011 024406 021666
 019234 017788 016667 015671 014826 014289 014039 014353
 000000 014028 013912 013387 013189 012928 012411 012067
 011842 011519 011182 011255 010946 011013 011111 010728
 010369 010592 010407 010103 010196 010505 010130 010346
 280- 010536 010588 010463 010290 009973 009812 009475 009696
 008958 008764 008452 008112 007785 007848 007652 007571
 008073 008035 008102 008121 008477 008854 008612 008658
 008521 008154 007825 007257 006995 006291 005924 005793
 005545 005478 005470 005745 005703 005672 005557 005505
 320- 005218 005045 004969 004534 004057 004067 003942 003815
 003486 003500 003586 003382 003550 003542 003865 003907
 004300 004265 004192 004025 003915 003444 003271 002884
 002521 001987 001954 001900 001612 001527 001466 001475
 001358 001466 001350 001310 001262 001252 001225 001054
 360- 001133 001075 001079 001110 001010 001060 000979 001112
 001054 000977 001075 001025 000931 001016 001018 001018
 000962 000981 000939 000841 000760 000802 000702 000712
 000658 000631 000714 000606 000577 000612 000506 000529
 000481 000475 000389 000427 000412 000379 000387 000375
 400- 000393 000389 000339 000291 000377 000314 000325 000302
 000289 000337 000304 000285 000314 000252 000268 000239
 000314 000302 000283 000293 000227 000233 000237 000266
 000214 000204 000250 000243 000204 000206 000183 000231
 000185 000181 000206 000187 000204 000206 000208 000216
 440- 000193 000227 000216 000243 000177 000214 000208 000189
 000200 000143 000135 000185 000162 000156 000154 000135
 000143 000133 000133 000116 000156 000154 000164 000164
 000152 000156 000160 000133 000139 000127 000112 000077
 000108 000089 000083 000093 000089 000083 000075 000077
 480- 000066 000068 000125 000108 000085 000079 000087 000083

Run (559-B + 560-A) x 1/10

001178 001142 001223 001160 001250 001403 001754 002158
 102732 003624 004541 005602 006717 007940 009108 010475
 012084 013307 014899 016760 018810 021174 023961 027855
 032242 066030 312554 524715 552839 546869 518907 430362
 40 → 446466 429096 418891 416526 413775 410376 398859 375275
 344364 322723 309749 308796 319423 330896 320700 287469
 255111 245203 263786 336219 418657 453119 405854 343106
 342809 431230 514870 499693 384797 250130 161254 120093
 103999 096303 039623 084012 073582 076172 074689 075291
 80 - 076316 078249 078798 075066 070796 065985 062371 059062
 056437 055169 053488 052413 052058 052364 052202 052013
 051141 050997 050242 048408 046043 044416 042631 041071
 040631 039974 038770 038050 037187 036864 036594 035533
 034805 034085 033636 032863 032395 032188 032278 032278
 120 - 032431 033253 033015 032773 031460 030399 029167 027765
 026695 025395 025445 025418 025751 025350 027054 027477
 028556 029077 028277 027027 024897 023233 021858 021129
 020815 021300 021579 021974 021462 020320 018792 017263
 015780 014773 014044 013721 013640 013505 013721 014197
 160 - 014386 014494 014881 014701 014017 013397 012498 011779
 010943 010583 010511 010412 010196 009900 009936 010142
 009837 009801 009855 009675 009810 009855 009792 009603
 009387 009495 008821 008677 008146 007953 007526 007328
 007094 007391 007157 007157 006995 006906 006807 006798

200-006690 006726 006618 006798 006393 006285 006339 006222
 005953 006105 005890 005953 006006 005890 005818 005791
 005710 005354 005917 005737 005512 005584 005440 005458
 005557 005440 005395 005368 005152 005161 005269 005170
 005251 005116 005260 005089 005044 005197 005026 005035
 240-005008 004901 004919 004865 004757 004811 004703 004730
 004694 004622 004658 004703 004694 004649 004703 004730
 000000 004649 004613 004793 004631 004730 004559 004550
 004433 004460 004397 004568 004514 004451 004406 004451
 004307 004352 004271 004244 004163 004343 004316 004334
 280-004343 004424 004136 004442 004307 004451 004307 004262
 004037 004055 004037 003939 003894 003849 003768 003741
 003759 003741 003921 003894 004019 004055 004172 004163
 004289 004388 004325 004136 004109 004023 004100 003822
 003732 003714 003642 003543 003498 003426 003507 003507
 320-003426 003471 003462 003588 003597 003588 003633 003552
 003543 003471 003552 003615 003552 003525 003570 003705
 003597 003624 003597 003561 003516 003597 003489 003489
 003597 003534 003588 003588 003651 003732 003732 003642
 003732 003642 003786 003741 003633 003642 003615 003606
 360-003444 003462 003219 003210 003354 003192 003165 003210
 003138 003264 003264 003354 003399 003624 003849 003822
 003903 004046 003939 003921 003822 003606 003318 003066
 002824 002599 002446 002329 002194 002131 002032 002050
 002041 002059 002248 002329 002347 002419 002455 002419
 400-002365 002176 002059 001835 001709 001412 001241 001043
 000864 000837 000756 000639 000603 000558 000549 000477
 000495 000459 000450 000450 000441 000405 000423 000414
 000405 000342 000360 000351 000342 000315 000297 000306
 000297 000252 000252 000270 000234 000252 000225 000198
 440-000234 000225 000225 000198 000180 000234 000198 000180
 000162 000171 000153 000144 000180 000162 000135 000126
 000135 000117 000117 000126 000108 000108 000099 000108
 000108 000099 000081 000072 000072 000099 000081 000072
 000072 000063 000072 000054 000072 000054 000063 000063
 480-000054 000063 000054 000054 000054 000054 000063 000054
 000036 000054 000054 000063 000036 000045 000036 000036
 000027 000026 000026 000026 000025 000026 000026 000026

Run 559-A x 1/10

200000 000702 000679 000631 000750 000725 000818 000800
 000725 000858 000862 000729 000706 000925 001102 00138
 252587 002010 002565 003152 003784 004513 005195 00590
 006511 007478 008520 009458 010567 012015 013722 01587
 019819 084609 517635 727107 697714 674314 634647 58537
 40 → 546546 527579 520225 522222 519479 521604 509276 47977
 439683 414185 400533 403444 427830 447785 433779 38315
 332394 316259 357107 461207 587808 632484 550246 44679
 451413 592004 719367 692137 509255 311297 195049 14806
 133612 126033 118975 112984 106792 104954 103610 1043
 80-107336 111506 109755 106842 099315 094470 089579 0861
 083528 081900 080218 078966 078418 078404 078422 0781
 078864 077313 077149 073877 070645 068755 066396 064
 063116 062224 061532 060745 058836 059300 058059 057
 056437 055773 054964 054739 053530 053455 052723 052

120- 053689 054253 054407 053097 051944 050533 049193 048008
 046066 045955 045059 044555 044721 045057 045793 046766
 048137 048227 047543 045038 043257 040471 038787 038101
 037912 038837 039726 039837 038645 037534 035211 033329
 030787 030028 029259 028124 028067 027699 028378 028742
 160- 029336 030318 030118 030001 029113 028392 026979 026258
 025250 024539 024610 024047 023908 023814 023260 023658
 023135 023431 023249 023543 023456 023570 023385 023383
 023187 022883 022376 022115 021534 021230 020830 020027
 020409 020647 020336 020234 020349 020309 020476 020023
 180- 019856 019852 019773 020043 019288 019773 019665 019388
 019443 018965 019588 019765 019669 020111 020280 020303
 020372 020807 020303 020422 020376 020211 020027 019631
 019652 019394 019673 019356 019417 020313 019325 019700
 020184 020234 020355 020474 020809 020955 021324 021509
 240- 021390 020982 020476 020176 020313 020157 019671 019854
 020023 020149 019867 020257 020505 020409 020651 020534
 000000 021886 022501 022003 021976 022159 021288 021430
 021036 021026 020782 020663 020755 020653 020597 020872
 020913 021588 021547 021197 021572 021872 022011 022480
 260- 022591 022885 023562 023639 023535 023043 022849 021878
 021376 021324 020882 020384 020753 020838 020351 020665
 020836 020247 020997 021655 022272 022799 023439 024312
 025150 025049 025165 024908 024206 023843 022791 021638
 020980 020657 020109 020276 019602 019340 019492 020174
 320- 019927 020576 020799 021049 021186 021399 021572 021555
 022140 021878 021451 021705 022708 022484 022762 022949
 022747 022939 023474 022922 022486 022843 022336 023089
 022635 023731 023724 024841 024922 025723 026550 026958
 026677 027738 027321 027558 028107 027579 027129 026475
 360- 025858 025433 024383 023722 023043 022587 022509 022253
 022749 023437 023883 025931 026929 028292 030007 031666
 032510 033219 032802 032210 030324 028386 026219 023864
 021211 018917 017014 015564 014184 013507 013484 013382
 013920 014570 015863 017137 017675 019021 019013 019175
 400- 018500 017397 015537 013703 011300 008952 007253 005722
 004338 003225 002435 001881 001487 001266 001237 001091
 000975 001004 000868 000891 000829 000727 000727 000729
 000668 000608 000639 000585 000533 000581 000477 000462
 000456 000431 000404 000393 000404 000429 000416 000383
 440- 000350 000368 000291 000375 000316 000354 000281 000308
 000264 000277 000258 000283 000250 000256 000283 000229
 000254 000206 000227 000200 000187 000214 000200 000214
 000168 000187 000185 000202 000135 000135 000133 000175
 000156 000141 000133 000125 000127 000125 000114 000114
 480- 000116 000103 000118 000125 000091 000035 000083 000087
 000102 000083 000083 000077 000108 000100 000062 000089
 79 91 102 77 108 100 62 89
 64 77 56 64 56 56 82 50

Run 327-C x 1/10

	00805+	007471	006913	006085	005666	004674	003865
	003205	002934	002786	002540	002613	004613	034727
	537681	923484	882173	801456	728610	200274	693365 826053
40	122452	290756	102988	010863	436199	343239	310883 293226
	281478	272990	261813	242407	224547	215728	213239 216331
	222957	231218	241160	250218	245901	225707	209791 206865
	205089	203904	213982	246633	286178	296769	260238 199311
	149011	122101	108275	098825	092257	087235	084177 081680
80	-080823	083412	089201	097888	109057	122120	126343 119187
	102690	085743	072487	063929	059173	058824	058527 059591
	062054	066945	073570	079087	081206	078609	070921 061696
	055735	053914	054760	056175	056796	054578	050745 046183
	042492	039281	038177	037028	038133	039579	042267 044007
120	-046434	048472	049189	048359	046295	043109	038834 036616
	035435	035180	034904	034052	032682	030730	028609 027375
	027281	027688	028716	029494	029975	029960	029211 028414
	028146	027856	027773	027294	027706	027308	027411 026473
	025701	024519	023634	022216	020983	020636	020480 020567
160	-021155	021748	022055	022056	021407	021640	021758 021695
	022006	021319	019992	017969	016287	014733	013411 012590
	012463	012078	012012	011579	011222	011057	010404 010080
	009731	009540	009412	009379	009294	008902	008672 008481
	008184	007958	007867	007805	007789	007638	007510 007447
200	-007451	007565	007538	007422	007367	007386	007316 007166
	006824	006775	006515	006597	006385	006155	006328 006195
	006171	006399	006348	006282	006258	006062	005945 005790
	005781	005833	005705	005631	005719	005593	005552 005466
	005396	005417	005414	005482	005346	005338	005424 005335
240	-005299	005302	005191	005135	005118	004927	004857 004815
	004875	004972	005089	004901	005050	004974	004977 004887
	004803	004615	004551	004370	004317	004149	004091 004036
	003909	003994	004045	004130	004146	004169	004463 004310
	004467	004400	004479	004293	004225	004101	004069 003962
280	-003680	003642	003605	003496	003653	003535	003522 003391
	003490	003600	003558	003570	003406	003605	003611 003502
	003656	003488	003569	003603	003567	003550	003598 003522
	003435	003554	003558	003560	003505	003592	003569 003601
	003527	003385	003556	003438	003334	003388	003458 003527
320	-003722	003872	004102	003992	004120	004057	003963 003775
	003697	003474	003286	003268	003105	002917	002960 002741
	002759	002641	002470	002557	002601	002631	002928 003054
	003166	003416	003347	003423	003276	003089	002926 002608
	002248	002148	001988	001835	001814	001818	001739 001743
360	-001731	001721	001738	001687	001613	001545	001561 001446
	001330	001312	001310	001248	001250	001281	001356 001338
	001348	001419	001510	001422	001535	001416	001443 001501
	001492	001355	001373	001296	001251	001125	001142 001027
	000961	000999	000972	000962	000993	001026	001001 000990
400	-000980	001030	001023	001023	000994	000935	000891 000800
	000728	000675	000699	000650	000575	000553	000544 000517
	000491	000491	000448	000469	000454	000455	000427 000334
	000350	000301	000311	000255	000277	000257	000254 000259
	000292	000292	000254	000268	000263	000250	000273 000242

440-000257 000226 000196 000198 000168 000145 000149 000153
 000149 000118 000130 000122 000144 000122 000118 000113
 000116 000119 000100 000130 000128 000142 000128 000146
 000179 000205 000239 000308 000388 000463 000546 000533
 000621 000794 000836 000901 001008 001128 001242 001428
 001530 003202 033619 174206 294023 125018 026550 008173
 002868 001826 001460 001203 001101 000951 000890 000675
 000578 000488 000331 000226 000193 000157 000150 000157
 000130 000111 000088 000093 000084 000083 000083 000083

Run (326-D + 327-A) x 1/100

000577 000525 000498 000427 000402 000364 000325 000252
 000227 000214 000193 000189 000191 000214 000302 003504
 068323 113858 106659 096550 089890 089289 101387 131126
 40-164880 164784 118401 070641 049314 043569 041657 040095
 039061 037988 035755 033224 031836 031619 032507 034162
 035492 036366 037058 036424 034281 032213 032165 032515
 032109 032453 035909 041054 043081 038926 030487 023065
 80-018735 016959 015887 015018 014443 013940 013640 013609
 -013903 014878 016223 017449 018831 019265 018208 015783
 013352 011637 010582 010121 010033 010079 010311 010538
 011211 012179 012894 013069 012437 011370 010229 009450
 009441 009598 009731 009621 009170 008672 008030 007515
 007071 007067 006988 007136 007325 007626 008007 008211
 120-008422 008476 008207 007805 007392 006979 006769 006577
 006548 006562 006393 006079 005766 005524 005384 005418
 005489 005676 005741 005716 005649 005497 005422 005405
 005280 005284 005355 005357 005299 005334 005174 005003
 004759 004726 004509 004438 004419 004494 004557 004653
 160-004619 004578 004519 004357 004419 004361 004348 004246
 004111 003865 003612 003408 003302 003156 003106 003137
 003045 003054 003004 002893 002866 002749 002749 002660
 002603 002618 002524 002518 002478 002441 002416 002416
 002447 002399 002462 002468 002441 002462 002428 002418
 200-002368 002403 002357 002372 002299 002272 002218 002126
 002095 002072 002059 002053 002043 002051 002076 002028
 002059 002080 002082 002082 002061 002101 002103 002047
 002049 002082 002051 002022 002011 001999 002032 001976
 001968 001961 001947 001959 001934 001938 001936 001899
 240-001886 001911 001915 002018 002020 002049 002101 002136
 002186 002174 002207 002132 002072 002074 001974 001901
 000000 001839 001837 001733 001739 001724 001734 001728
 001830 001801 001944 002023 002100 002136 002162 002188
 002098 002047 002007 001887 001811 001746 001654 001620
 280-001549 001589 001566 001564 001593 001592 001648 001629
 001636 001642 001611 001641 001682 001717 001737 001757
 001784 001836 001860 001917 001943 001974 001949 002013
 002000 001979 001977 001982 001905 001904 001884 001890
 001867 001898 001992 002094 002162 002326 002541 002660
 320-002693 002705 002715 002589 002437 002302 002174 002079
 001984 001857 001759 001631 001596 001548 001513 001485
 001519 001615 001756 001895 002070 002210 002271 002335
 002240 002067 001877 001607 001382 001202 001054 001026
 000937 000975 000970 000962 000960 000972 000899 000896

360- 000863 000786 000779 000740 000686 000678 000658 000671
 000663 000701 000736 000748 000788 000837 000861 000896
 000923 000951 000945 000933 000907 000867 000827 000773
 000689 000642 000594 000557 000542 000519 000514 000531
 000553 000580 000602 000632 000646 000624 000631 000624
 400- 000582 000520 000511 000457 000411 000366 000330 000288
 000280 000270 000266 000245 000253 000237 000227 000206
 000187 000182 000139 000130 000119 000109 000095 000088
 000079 000082 000089 000091 000106 000098 000097 000109
 000100 000113 000090 000086 000073 000065 000050 000046
 440- 000039 000037 000029 000028 000020 000021 000023 000022
 000019 000018 000022 000018 000016 000019 000014 000015
 000018 000014 000021 000020 000014 000017 000018 000019
 000021 000029 000037 000042 000052 000059 000071 000082
 000098 000100 000113 000116 000135 000152 000167 000178
 000209 001210 009339 066065 071640 004505 002206 000618
 000278 000202 000167 000146 000126 000122 000092 000090
 000070 000057 000034 000027 000022 000024 000023 000016
 000019 000015 000015 000014 000012 000012 000014 000016

Run 515-A x 1/10

111845 008655 008708 008743 008482 008575 008276 009843
 030099 363434 167328 279343 049497 379960 085653 917860
 769681 663212 575187 507204 458359 423683 401703 392940
 40 → 379144 349832 309146 274202 252966 243167 243147 250664
 256833 245728 219556 194417 181803 176869 175558 172323
 164560 156716 148347 143269 137017 133121 131077 132915
 148357 191557 254460 288312 262060 195429 134153 098641
 084498 078819 075610 073049 070290 068176 067536 067812
 80- 066803 064294 060059 056632 054899 053506 053094 051385
 050250 048073 046288 043533 041346 039720 038339 037224
 036398 036393 039050 043679 050426 057694 059552 055633
 047413 039421 032785 028540 026117 024776 023960 023639
 023227 023134 022855 022373 021946 021760 022092 022245
 120- 022855 022594 021989 021027 019735 018703 017741 016845
 015966 015627 015316 014851 015120 015504 016318 017548
 018133 018693 018279 016312 015203 013656 012787 011642
 011095 010942 010786 010235 010202 009803 009833 009853
 009976 009976 009780 009554 009160 008801 008472 008151
 160- 007739 007666 007749 007430 007304 007179 007367 007149
 006943 006850 006777 006614 006398 006272 006202 006044
 006157 006064 005971 006044 006024 005788 005788 005539
 005519 005220 005130 004972 004859 004653 004610 004487
 004507 004560 004497 004424 004364 004517 004447 004334
 200- 004364 004374 004208 004188 004053 004138 004073 003980
 003754 003754 003641 003508 003508 003259 003332 003166
 003071 003312 003176 003071 003031 002973 002918 002928
 002918 002805 002805 002845 002815 002712 002845 002865
 002762 002619 002579 002526 002506 002453 002433 002383
 240- 002280 002383 002227 002237 002300 002237 002350 002383
 002330 002393 002463 002443 002486 002556 002413 002290
 000000 002127 002093 001982 001756 001776 001780 001701
 001719 001660 001649 001677 001746 001694 001757 001951
 001937 001944 002050 002002 001902 001797 001616 001483

280- 001424 001270 001216 001323 001299 001336 001318 001198
 001173 001113 001075 001101 001034 001044 000984 000937
 000908 000838 000844 000857 000830 000854 000888 000874
 000930 000867 000869 000854 000794 000775 000671 000641
 000609 000541 000506 000491 000482 000452 000414 000417
 320- 000375 000339 000315 000304 000290 000254 000255 000266
 000224 000222 000244 000255 000243 000208 000222 000202
 000229 000237 000217 000238 000196 000200 000208 000184
 000160 000180 000135 000120 000140 000138 000111 000099
 000130 000116 000107 000094 000111 000112 000112 000099
 360- 000086 000110 000105 000109 000095 000081 000087 000082
 000077 000075 000088 000090 000103 000099 000084 000092
 000081 000105 000068 000081 000069 000064 000059 000047
 000049 000064 000075 000055 000051 000050 000053 000049
 000047 000043 000042 000062 000051 000048 000031 000059
 400- 000045 000038 000041 000046 000034 000028 000030 000028
 000028 000024 000015 000023 000023 000023 000025 000024
 000024 000011 000020 000035 000029 000024 000032 000023
 000019 000017 000017 000019 000014 000014 000022 000018
 000017 000023 000022 000017 000017 000015 000015 000018
 440- 000022 000016 000019 000019 000022 000015 000012 000011
 000018 000015 000016 000009 000017 000024 000023 000021
 000014 000013 000015 000011 000022 000014 000017 000018
 000025 000015 000020 000013 000018 000018 000020 000015
 000018 000016 000009 000010 000020 000006 000011 000010
 480- 000009 000011 000011 000012 000011 000015 000007 000012
 000013 000021 000014 000018 000015 000015 000018 000010
 000009 000008 000012 000013 000012 000016 000010 000017

Run 514-B x 1/10

112387 011647 011357 011597 011597 011000 010610 011147
 016212 104873 872392 085448 912752 461296 201303 005815
 856330 746787 655400 538031 539025 511439 506641 513453
 40 → 491666 437628 377955 333457 310646 303010 308202 321166
 327028 311293 275871 245888 231383 227335 224961 219239
 209009 202130 196845 189636 181830 165498 173594 180513
 213644 284344 361903 381853 324604 235031 166138 121060
 117562 112257 107302 102517 098089 096842 096365 096485
 80- 095098 090860 086682 082754 080740 079316 077839 075855
 073134 070640 067979 064311 061350 059933 057979 057142
 056842 058476 062407 070590 080423 087282 086802 077282
 065875 055258 048129 043461 041130 039503 038746 038316
 037979 037999 037469 037182 036522 036852 036792 037329
 120- 037152 037052 035715 034328 032314 031180 029933 028776
 027759 027389 026762 026792 026822 027979 029683 030620
 031357 031487 030210 027879 025665 023931 022714 021927
 020890 020670 020600 019833 019363 019293 019133 019443
 019203 019046 018486 018596 018126 017419 017169 016752
 160- 016282 016025 015925 015765 015715 015545 015285 014808
 014668 014368 014158 014011 013741 013601 013341 013161
 012994 012814 013181 013121 013121 012814 012884 012784
 012624 012314 012057 011817 011317 011287 010770 010660
 010470 010370 010300 010170 010090 010180 009943 009893

200- 009493 009633 009373 009263 009093 009093 008996 008926
 008676 008476 008386 008056 003019 007959 007699 007459
 007389 007179 007159 007079 006942 006842 006842 006712
 006652 006562 006722 006692 006592 006432 006562 006362
 006222 006015 006122 005715 005465 005595 005355 005355
 240- 005325 005085 005225 005135 005075 005065 004948 005185
 005195 005135 005165 005205 005175 005018 004778 004648
 000000 004402 004223 004030 003779 003602 003752 003467
 003448 003466 003472 003562 003536 003618 003670 003805
 003890 003765 003612 003430 003105 003010 002948 002764
 280- 002711 002612 002743 002545 002630 002498 002437 002345
 002312 002262 002166 002225 002034 001971 001949 001890
 001796 001793 001737 001717 001725 001734 001741 001820
 001770 001739 001756 001535 001541 001495 001347 001282
 001277 001240 001183 001154 001130 001117 001062 000975
 320- 000902 000851 000778 000739 000696 000720 000658 000637
 000601 000574 000646 000558 000646 000708 000662 000727
 000730 000728 000779 000735 000695 000623 000574 000538
 000571 000480 000457 000434 000405 000433 000370 000372
 000392 000393 000348 000394 000384 000424 000405 000399
 360- 000379 000440 000411 000448 000389 000418 000363 000401
 000362 000355 000387 000406 000394 000440 000435 000462
 000390 000371 000362 000305 000252 000272 000230 000224
 000211 000196 000173 000180 000207 000192 000241 000249
 000238 000249 000277 000259 000275 000239 000231 000197
 400- 000163 000135 000111 000091 000058 000050 000047 000040
 000034 000049 000024 000032 000042 000044 000043 000037
 000038 000023 000023 000039 000032 000032 000037 000032
 000039 000023 000029 000027 000023 000023 000020 000019
 000022 000023 000028 000022 000026 000026 000013 000022
 440- 000014 000032 000018 000023 000019 000024 000028 000029
 000016 000021 000030 000031 000017 000012 000012 000020
 000023 000018 000021 000016 000016 000023 000026 000027
 000011 000028 000016 000016 000015 000022 000013 000012
 000017 000011 000026 000009 000021 000019 000016 000016
 480- 000015 000017 000016 000009 000017 000006 000009 000011
 000017 000012 000013 000015 000021 000019 000015 000015
 000009 000006 000009 000018 000013 000018 000008 000011

Run 581-A

500000 002557 002168 001901 002254 002007 001611 001616
 001397 001286 001117 000875 000443 000312 000443 000933
 533441 001210 002378 017503 060048 142643 217361 296463
 379094 467279 546264 732634 100403 128233 157603 170649
 193451 217972 259938 281259 322478 341213 3650 402304 1866332
 40- 167302 1530456 1405409 1320401 1232120 131511 1245643 1235120
 1133738 1140107 1100142 941335 359274 800225 734623 676936
 25428 593740 579434 564016 560005 551940 534235 506954
 478118 455886 440951 424762 410756 401597 404632 421276
 435604 619969 742699 731540 706932 506790 513343 461177
 80- 426151 404534 365955 336539 312743 303263 274955 302199
 306349 306326 297306 285846 275153 274747 275909 270254
 268310 270526 237777 313855 347303 370337 367226 343130
 305126 270091 260315 251591 250225 246112 234037 222483
 215113 215088 216429 219104 217871 220975 225425 232331

120- 232177 22127 211233 191544 177375 163221 147570 137071
 125035 125035 122233 125440 135025 151541 173176 190699
 141572 186330 169631 143311 125773 115090 107700 103961
 101637 107259 112239 111994 104951 107215 152229 399012
 071296 071603 071789 071764 094109 039703 057127 031736
 160- 071264 076900 076345 073940 074941 072029 072851 071983
 071174 070356 071406 071212 072052 035321 032033 101359
 107927 111790 111031 105610 093321 066370 073166 070463
 055531 059561 056223 054579 052126 051571 050164 047221
 050444 051493 053933 056760 052339 056303 073123 071321
 200- 036932 070011 077130 072039 032743 050252 050023 043704
 041560 039814 036428 035465 031225 030767 033576 033372
 033405 033826 034267 034063 031233 031051 034610 033432
 034760 031779 035629 034537 035037 035044 034277 034347
 033955 034421 034963 035205 031357 034757 034015 035611
 240- 034063 034703 034595 031161 033037 031222 034502 035273
 035023 035118 040417 041917 044534 044539 044567 043548
 037000 040162 033710 037207 035447 035065 033037 031582
 030576 030735 030117 026846 023526 021166 030437 032003
 033710 037767 039741 040383 041637 033063 037421 033049
 280- 036430 027197 025463 023360 022644 022575 022722 022432
 021617 021340 020723 019731 019442 019764 017713 017942
 016731 016356 015176 014553 014036 013552 012942 012261
 012630 013295 013547 013635 014405 014573 014452 014200
 013245 013265 012637 012191 011451 011192 011074 010367
 320- 000600 009402 003477 007594 006601 006004 005625 005194
 004561 004720 004173 004310 003999 003951 003946 003457
 003424 003285 002940 003157 002973 003157 003039 002925
 002859 002960 002829 002524 002544 002312 002239 002093
 002163 001906 001323 001727 001576 001606 001523 001430
 360- 001206 001296 001263 001205 001276 001152 001190 000996
 001044 001039 001039 000996 000961 001047 001132 000830
 001014 000986 000918 000961 000913 000833 000764 000312
 000769 000731 000726 000721 000625 000691 000600 000678
 000500 000512 000577 000512 000464 000562 000439 000416
 400- 000427 000371 000454 000444 000343 000290 000429 000338
 000338 000323 000333 000376 000300 000363 000280 000232
 000247 000280 000252 000295 000237 000222 000242 000247
 000242 000237 000235 000179 000373 000270 000252 000199
 000222 000159 000204 000134 000199 000160 000174 000139
 440- 000131 000159 000217 000217 000137 000174 000121 000169
 000136 000121 000141 000164 000169 000141 000126 000121
 000159 000136 000111 000136 000111 000169 000136 000033
 000146 000141 000146 000126 000131 000126 000121 000111
 000131 000164 000121 000106 000126 000173 000116 000141
 480- 000106 000131 000131 000106 000111 000073 000131 000111
 000116 000093 000033 000111 000063 000063 000063 000063
 000073 000033 000093 000063 000073 000063 000131 000063
 000053 000033 000033 000033 000053 000063 000033 000033

Run 581-B + 582-A

000000 004390 003870 003575 004125 003500 002505 002445
 002130 001960 001665 001675 001605 001540 001565 002135
 003625 006120 013450 053425 130320 240505 337120 420955
 520 00 623415 729070 970155 213536 320721 014120 03427140
 337027532371053043245235690523030525353452500052353205
 40 → 0121430 017720 751830 1634565 1592735 160370 057505 1552505
 1410150 1237790 1213845 1100360 1030015 990200 930065 870390
 319315 794610 769260 763005 763200 753365 717360
 675100 647075 622210 606330 590330 530045 530375 600995
 637950 351535 1016505 1077340 906005 550310 3450 677205
 80 — 640000 607390 563245 519330 480035 42510 420010 431460
 435445 401460 470410 465730 450005 440570 443435 436050
 421265 432365 451310 501900 560450 601105 603215 553610
 409355 407300 413990 401360 393750 391025 330150 364770
 355670 362000 367995 366650 357365 360200 360100 361470
 120 - 357335 349335 336140 318315 307320 291090 275750 259710
 242370 234900 223575 225500 232335 250000 242140 291800
 293990 295110 273520 252615 227440 203975 193370 131040
 180040 131945 182170 183010 170730 176130 189395 166945
 162740 166015 165220 164360 161310 150800 155265 152565
 160 - 143670 148210 149740 145835 143895 141300 134335 123020
 123545 120255 117395 120335 123630 131370 142170 154360
 165200 174340 175890 170775 163515 152100 133630 123420
 112235 104540 096730 092245 087005 074620 073035 091175
 081175 081675 084385 087375 091200 093665 105300 111220
 200 - 113960 116135 113115 105710 095740 087495 076305 074070
 067755 064995 062045 060060 057335 052140 057700 056470
 057110 056705 057300 057445 057065 057030 053500 060135
 059195 060380 060365 062300 061100 063120 062700 060380
 060330 061360 060395 060700 060375 061020 061035 061315
 240 - 061670 061260 061370 059940 059705 060700 060325 063470
 064515 063595 072350 075315 077325 077105 070230 075765
 070000 073005 069390 069360 069655 063245 063920 063045
 067060 064560 064265 062175 059230 053355 059035 059940
 280 - 063300 067010 063015 070500 067300 065005 062960 056130
 053930 049920 049015 048630 049030 051155 053155 053730
 053615 051830 049075 045165 04105 037230 035005 031315
 030155 028340 027160 026175 024620 023930 021435 021440
 021100 021035 021165 022035 023010 022740 022445 022130
 021240 021110 020000 019550 018970 019540 019175 013635
 320 - 017265 016940 016465 015320 014470 013510 012630 012215
 011365 010800 010700 009800 009420 007230 003645 007360
 007300 007410 007200 007505 007190 007200 007345 003030
 003230 007380 003070 007805 006960 006635 006275 005335
 004790 004275 003810 003495 003030 003035 003010 002890
 360 - 002455 002615 002340 002260 002140 001925 001905 001965
 001815 001930 001665 001305 001695 001605 001555 001570
 001540 001415 001550 001575 001665 001555 001550 001670
 001550 001510 001295 001095 001030 001120 001000 000930
 000915 000905 000395 000900 000730 000730 000730 000655

400-000765 000695 000630 000535 000500 000515 000500 000540
 000425 000415 000520 000405 000375 000400 000465 000400
 000390 000315 000415 000395 000340 000300 000400 000295
 000290 000230 000340 000310 000340 000330 000310 000290
 000315 000375 000265 000320 000290 000350 000350 000295
 440-000225 000220 000195 000290 000205 000250 000245 000265
 000200 000240 000340 000230 000210 000225 000295 000265
 000350 000290 000200 000255 000255 000255 000220 000275
 000195 000220 000200 000155 000130 000165 000225 000145
 000150 000175 000235 000250 000190 000235 000235 000240
 460-000170 000235 000150 000135 000155 000150 000140 000140
 000155 000100 000125 000115 000135 000075 000075 000075
 000100 000065 000035 000045 000135 000115 000075 000100
 000110 000075 000095 000075 000075 000075 000100 000085

References and Footnotes

1. J. R. Stehn, M. D. Goldberg, B. A. Magurno, and R. Wiener-Chasman, "Neutron Cross Sections," Vols. I, IIA, IIB, IIC, Brookhaven National Laboratory 325, Second Edition, Supplement No. 2 (1964-1966).
2. E. S. Troubetzkoy, "Fast Neutron Cross Sections of Iron, Silicon, Aluminum, and Oxygen," NDA-2111-3, Vol. C (1959).
3. E. S. Troubetzkoy, M. H. Kalos, H. Lustig, J. H. Ray, B. H. Trupin, "Fast Neutron Cross Sections of Manganese, Calcium, Sulfur, and Sodium," NDA 2133-4 (1961).
4. N. Tralli, M. H. Kalos, A. J. Longano, J. H. Ray, R. P. Sullivan, E. S. Troubetzkoy, and B. H. Trupin, "Neutron Cross Sections for Titanium, Potassium, Magnesium, Nitrogen, Aluminum, Silicon, Sodium, Oxygen, and Manganese," UNC-5002 (1962).
5. J. Celnik and D. Spielberg, "Gamma Spectral Data for Shielding and Heating Calculations," UNC 5140 (1966).
6. R. E. Maerker and F. J. Muckenthaler, "Gamma-Ray Spectra Arising from Thermal-Neutron Capture in Elements Found in Soils, Concretes, and Structural Materials," ORNL-4382 (1969).
7. E. A. Straker, "Experimental Evaluation of Minima in the Total Neutron Cross Sections of Several Shielding Materials," Nucl. Sci. Eng. 34, 114 (1968).
8. R. E. Maerker and F. J. Muckenthaler, "Differential Fast-Neutron Albedos for Concrete, Vol. 1. Description of Monte Carlo Calculation and TSF Experiment and Comparison of Calculated and Experimental Results," ORNL-3822 (1965).
9. B. J. Henderson, "Conversion of Neutron or Gamma Ray Flux to Absorbed Dose Rate," XDC 59-8-179 (1959).
10. The authors are indebted to T. V. Blosser for obtaining these data and R. M. Freestone for performing the analysis.
11. The argument for assuming that one-dimensional geometry is adequate has already been presented in ref. 6. Interested readers are referred to Section III of this reference.

12. The authors are indebted to L. R. Williams of the ORGDP Computing Group for making these calculations.
13. H. E. Hall and T. W. Bonner, "Gamma Radiations from Inelastic Scattering of Fast Neutrons in C^{12} , N^{14} , and O^{16} ," Nucl. Phys. 14, 295 (1959).
14. J. K. Dickens and F. G. Perey, to be published.
15. The data in reference 5 are expressed in units of gamma rays per neutron producing reaction, and must be multiplied by σ_{in} for each neutron energy. To prepare Tables X, XIV, XXXIX, and XLIII, the values of $\sigma_{in}(E)$ were taken from reference 1.
16. B. J. Allen, Nucl. Phys. A111, 1 (1968).
17. K. J. Yost and S. M. Kremer, Nucl. Sci. Eng. 36, 220 (1969).
18. C. M. Lederer, J. M. Hollander, and I. Perlman, Table of Isotopes, Sixth Edition, John Wiley and Sons, Inc. (1967).
19. W. E. Tucker, P. S. Buchanon, T. C. Martin, D. O. Nellis, and G. H. Williams, "Neutron-Induced Gamma-Ray Production Cross Sections for Silicon and Tungsten," Nuclear-Chicago Corporation, Texas Nuclear Division, DASA-2333 (1969).

INTERNAL DISTRIBUTION

1. Biology Library	74. R. F. Hibbs
2-4. Central Research Library	75. H. G. MacPherson
5-6. ORNL - Y-12 Technical Library	76-81. R. E. Maerker
Document Reference Section	82. F. C. Maienschein
7-56. Laboratory Records Department	83-92. F. J. Muckenthaler
57. Laboratory Records, ORNL R.C.	93. J. P. Nichols
58. L. S. Abbott	94. S. K. Penny
59. J. E. Bigelow	95. R. L. Shepard
60. J. A. Biggerstaff	96. M. J. Skinner
61. A. R. Buhl	97. E. Silver
62. V. R. Cain	98. O. Sisman
63. T. A. Carlson	99. G. G. Slaughter
64. M. M. Chiles	100. E. A. Straker
65-66. C. E. Clifford	101. D. A. Sundberg
67. W. K. Crowley	102. D. K. Trubey
68. F. F. Dyer	103. D. Vondy
69. J. S. Eldridge	104. H. Wright
70. J. R. Engel	105. J. R. Beyster (consultant)
71. J. H. Gibbons	106. E. R. Cohen (consultant)
72. A. Houtzeel	107. H. Feshbach (consultant)
73. W.H. Jordan	108. H. Goldstein (consultant)

EXTERNAL DISTRIBUTION

- 109-255. Given DASA Shielding Distribution (updated February 5, 1970).
256. W. H. Hannum, Division of Reactor Development and Technology, U. S. Atomic Energy Commission, Washington, D. C.
257. P. B. Hemmig, Division of Reactor Development and Technology, U. S. Atomic Energy Commission, Washington, D. C.
258. Kermit Laughon, AEC Site Representative.
259. G. A. Bartholomew, Chalk River Nuclear Laboratories, Atomic Energy of Canada Limited, Chalk River, Ontario, Canada
260. P. S. Bland, Westinghouse Electric Corp., Advanced Reactor Division, Pittsburgh, Pa. 15236
261. W. L. Bunch, Battelle Northwest, Radiation and Shield Analysis, P. O. Box 999, Richland, Wash. 99352
262. John Butler, AERE, Harwell, Shielding Group 451-3T, Didcot, Berkshire, England
263. R. L. Detterman, Atomics International, 8900 Desota St., Box 309, Canoga Park, Calif. 91304
264. R. K. Disney, Westinghouse Astronuclear Lab., P. O. Box 10864, Pittsburgh, Pa. 15236
265. Maj. R. W. Enz, Department of Defense, Defense Atomic Support Agency, Washington, D. C. 20305
266. W. E. Ford, III, Computing Technology Center, Union Carbide Corporation, Oak Ridge, Tenn.
267. J. G. Gallagher, Manager, Thermal and Nuclear Design, Westinghouse Astronuclear Lab., P. O. Box 10864, Pittsburgh, Pa. 15236
268. G. A. Graves, N-2 Division, Los Alamos Scientific Laboratory, P. O. Box 1663, Los Alamos, New Mexico 87544

269. L. V. Groshev, I. V. Kurchatov Institute of Atomic Energy,
Moscow, U.S.S.R.
270. P. G. Johnson, Space Nuclear Propulsion Office, U. S. Atomic Energy
Commission, Washington, D. C. 20545
271. M. Kapo, Westinghouse Astronuclear Lab., P. O. Box 10864, Pitts-
burgh, Pa. 15236
272. I. M. Karp, Lewis Research Center, National Aeronautics and Space
Administration, 21000 Brookpark Road, Cleveland, Ohio 44135
273. C. D. Kemshall, United Kingdom Atomic Energy Authority, Atomic
Weapons Research Establishment, Bldg. A70.2, Aldermaston, Reading,
England
274. Martin Leimdorfer, Research Institute of National Defense, Stockholm
80, Sweden
275. C. P. McCallum, Division of Space Nuclear Systems, U. S. Atomic
Energy Commission, Washington, D. C. 20546
276. H. T. Motz, Los Alamos Scientific Laboratory, P. O. Box 1663,
Los Alamos, New Mexico 87544
277. F. R. Mynatt, Computing Technology Center, Union Carbide Corpora-
tion, Oak Ridge, Tenn.
278. V. S. Oblock, Westinghouse Astronuclear Lab., P. O. Box 10864,
Pittsburgh, Pa. 15236
279. V. J. Orphan, Gulf General Atomic, P. O. Box 608, San Diego,
Calif. 92112
280. Jean Rastoin, Centre d'Etudes Nucleaires, Boite Postale No. 6,
Fontenay-aux-Roses (S&O), France
281. Director, Institut fur Kernenergetik, Postanschrift: Herdweg 51,
7 Stuttgart, West Germany, Attn: F. A. R. Schmidt
282. Lt. C. Schoenwalter, Department of Defense, Defense Atomic Support
Agency, Washington, D. C. 20305
283. Gale Simons, Battelle Northwest, P. O. Box 999, Richland, Wash.
284. TRW, One Space Park, Redondo Beach, Calif., Attn: J. Warner and
R. M. Webb
285. Leona Stewart, P Division, Los Alamos Scientific Laboratory, P.
O. Box 1663, Los Alamos, New Mexico 87544
286. J. A. Swartout, Union Carbide Corporation, New York, N. Y.
287. E. A. Warman, Manager, Shielding Analysis, Nuclear Rocket Operations,
Aerojet-General Corporation, Box 15847, Sacramento, Calif. 95813
288. H. Woodsum, Westinghouse Astronuclear Laboratory, P. O. Box 10864,
Pittsburgh, Pa. 15236
289. Laboratory and University Division, AEC, ORO
- 290-509. Given distribution as shown in TID-4500 under Radiation Effects
on Materials category (25 copies - CFSTI)