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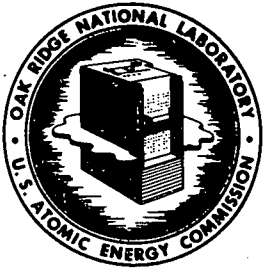
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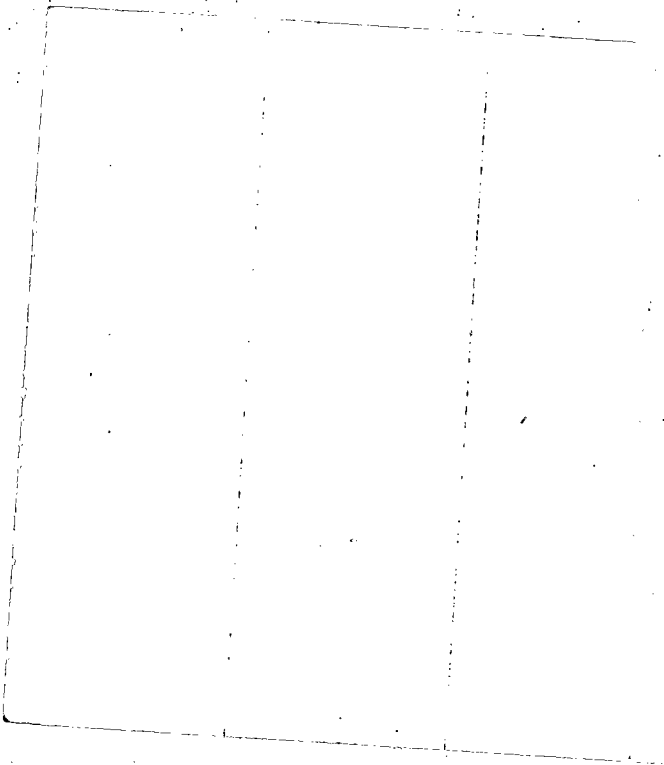
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SUBJECT: A Calculation of the Gamma-Ray Spectrum
of the Bulk Shielding Reactor

TO: Distribution

FROM: Gerard deSaussure



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Abstract

A calculation was performed to determine the high-energy portion of the gamma-ray flux at the surface of the uranium-aluminum BSF reactor. The flux was considered to consist entirely of the uncollided flux plus the Compton-scattered flux. The four sources of primary gamma rays included in the calculation were prompt fission gamma rays, capture gamma rays from aluminum, capture gamma rays from uranium, and fission-product decay gamma rays. These sources were divided into four energy groups. Although the assumptions used to calculate the Compton-scattered flux tended to overestimate the actual spectrum, the calculated results were considerably lower than a previously measured spectrum.

Introduction

The continued requirements for more refined shielding information necessitates that the gamma-ray spectrum of shield-test reactors be known. The spectrum of the Bulk Shielding Facility reactor has previously been determined experimentally with a scintillation spectrometer,¹ and calculational attempts² have been made to estimate the gamma-ray spectrum on the surface of the reactor core over the entire energy range. The calculation reported here represents an attempt to determine the high-energy end of this spectrum from the known intensity and spectra of the primary gamma-ray sources inside the core.

Method of Calculation

Only the high-energy end of the BSR spectrum, from 3.0 Mev up, was calculated, and for this portion of the spectrum the main contribution to the flux at the core surface is the uncollided flux. This can be calculated in a straightforward manner if the sources of primary gamma rays inside the core are known. In addition to the uncollided flux, the Compton-scattered flux was calculated using two simplifying assumptions: (1) The "straightahead assumption" which assumes that the energy spectrum of the scattered gamma rays is given by the correct Klein-Nishina formula, but that their direction is the same as the direction of the primary gamma rays, and (2) the "one-collision assumption" which assumes that the scattered gamma rays proceed without further absorption to the surface of the core. It is obvious that the spectrum of Compton-scattered

1. F. C. Maienschein and T. A. Love, Nucleonics 12, No. 5, 6-8 (1954).
2. G. deSaussure, ORNL-2221, p. 347 (classified); W. F. Osborne and D. W. Montgomery, Babcock and Wilcox, private communication.

gamma rays calculated under these assumptions will be an overestimate of the actual spectrum.

The Primary Sources of Gamma Radiation

There are four sources of primary gamma rays: (1) prompt fission gamma rays, (2) capture gamma rays from aluminum, (3) capture gamma rays from uranium, and (4) fission-product decay gamma rays. There are also gamma rays produced by neutron capture in water, inelastic scattering of fast neutrons, decay of Al^{28} , and annihilation radiation, but none of these sources contribute appreciably to the gamma radiation above 3 Mev.

Most fissions in the reactor occur at thermal energies. In this energy range the spectrum and intensity of the prompt fission gamma rays have been investigated by Gamble³ and can be approximated by $N(E) = 7.6e^{-1.01E}$ photons/Mev·fission.

This spectrum of uranium capture gamma rays has not been investigated; however, the total gamma-ray energy emitted per capture is known to be 6.43 Mev, which is equivalent to the binding energy of an additional neutron in U^{235} . The energy dependence of the capture gamma-ray spectrum of U^{235} was assumed to be the same as the one of the prompt fission spectrum (i.e., $\sim e^{-1.01E}$). This is probably not correct, but the intensity of this source of radiation is small compared to that of aluminum capture gamma rays and it should not affect the results greatly.

3. R. L. Gamble, Prompt Fission Gamma Rays from Uranium-235, (Dissertation presented to the faculty of the Graduate School of the University of Texas in partial fulfillment of the requirements for the Degree of Doctor of Philosophy).

The gamma-ray source associated with fission-product activity evidently depends on the past history of the fuel elements of the reactor; however, the main contribution is from the short-lived fission products. The spectrum of the short-lived fission-product decay gamma rays of U^{235} has been measured by Peelle, Zobel, and Love,⁴ who have reported a typical spectrum corresponding to about 3 hr of activation. This is the spectrum that was used in the calculation.

The energies and intensities of aluminum capture gamma rays have been investigated and tabulated by Kinsey, Bartholomew, and Walker.⁵

When combined, the spectra of these four sources of primary gamma rays gave an almost continuous spectrum of radiation. For purposes of calculation, the spectrum from 3 Mev to 11 Mev was divided into four 2-Mev wide groups, and the total energy emitted by the various sources in each of the energy groups was computed. For convenience, the total energy emitted in each group was divided by a "representative energy" inside the group to convert it into an equivalent number of photons of the representative energy. The results of this calculation are tabulated in Table 1.

The numbers presented in Table 1 were obtained in the following manner. The prompt fission gamma-ray spectrum was approximated by:

$$n^f(E) = 7.6e^{-1.01E} \text{ photons/Mev} \cdot \text{fission}$$

Since the total source is known to be:

$$E_T = \int_0^{\infty} E n(E) dE = 7.45 \text{ Mev/fission}$$

4. R. W. Peelle, W. Zobel, and T. A. Love, Appl. Nuc. Phys. Ann. Prog. Rep., Sept. 10, 1956. ORNL-2081, p. 94.

5. B. B. Kinsey, G. A. Bartholomew, and W. H. Walker, Phys. Rev. 83, 519 (1951).

Table 1. Number of Photons per Fission in the BSR Core for Various Energy Intervals

Energy Group	E_a (MeV)	E_b (MeV)	\bar{E} (MeV)	$\nu^U(\bar{E})$ (photons/fission)	$\nu^P(\bar{E})$ (photons/fission)	$\nu^{Al}(\bar{E})$ (photons/fission)	$\nu(E)$ (photons/fission)
I	3.0	5.0	4.0	0.34	0.10	0.07	0.51
II	5.0	7.0	6.0	0.046	0.002	0.021	0.069
III	7.0	9.0	8.0	0.006	0	0.036	0.042
IV	9.0	11.0	10.0	0.0008	0	0	0.0008

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the capture gamma-ray spectrum of U^{235} , assumed to have the same shape as the prompt fission spectrum, can be given by:

$$n^c(E) = \frac{6.43}{7.45} \cdot 7.6e^{-1.01E} \text{ photons/Mev.fission}$$

The prompt fission spectrum plus the capture gamma-ray spectrum of U^{235} is then:

$$n^U(E) = \left(1 + \frac{\sigma_c}{\sigma_f} \cdot \frac{6.43}{7.45} \right) 7.6e^{-1.01E} \text{ photons/Mev.fission}$$

where σ_c/σ_f , the ratio of the microscopic capture and fission cross sections of U^{235} , is 0.184. Solving,

$$n^U(E) = 8.8e^{-1.01E} \text{ photons/Mev/fission}$$

The number of photons per fission in each group is then obtained as follows:

$$\nu^U(\bar{E}) = \frac{1}{\bar{E}} \int_{E_a}^{E_b} n^U(E) \cdot E \, dE = \frac{8.8}{\bar{E}} \int_{E_a}^{E_b} Ee^{-1.01E} \, dE \text{ photons/fission}$$

where E_a and E_b are the energy limits of the group and \bar{E} is the representative energy.

The number of fission-product decay gamma rays per fission is obtained in a similar fashion:

$$\nu^P(\bar{E}) = \frac{1}{\bar{E}} \int_{E_a}^{E_b} n^P(E) \cdot E \, dE$$

where $n^P(E)$ is the number of photons of energy E per Mev per fission measured experimentally.⁴

Finally, the number $\nu^{Al}(\bar{E})$ of aluminum capture gamma rays per fission for each energy group was obtained by multiplying the number $\omega^{Al}(\bar{E})$ of aluminum capture photons per neutron capture in aluminum by the ratio of the macroscopic capture cross section of aluminum to the macroscopic fission cross section of uranium ($\Sigma_a^{Al} / \Sigma_f^{25}$) in the core material. The value of $\omega^{Al}(\bar{E})$ was obtained from the measurements of Kinsey, Bartholomew, and Walker:⁴

$$\omega^{Al}(\bar{E}) = \frac{1}{100 \bar{E}} \sum_{E_a \leq E' \leq E_b} \lambda(E') E'$$

where $\lambda(E')$ is the number of photons of energy E' per 100 captures. Values of $\omega^{Al}(\bar{E})$ for the four energy groups are given in Table 2. The macroscopic capture cross section of aluminum in the reactor core material was obtained by multiplying the number of aluminum nuclei per cubic centimeter of the reactor core (2.47×10^{22}) by the microscopic capture cross section of aluminum for thermal neutrons ($\sigma_a^{Al} = 0.204$ barns), obtaining a value of 0.00504 cm^{-1} for Σ_a^{Al} . The macroscopic fission cross section of the U^{235} in the core was obtained similarly by multiplying the number of U^{235} nuclei per cubic centimeter of the reactor core (0.94×10^{20}) by the microscopic fission cross section of U^{235} for thermal neutrons ($\sigma_f^{25} = 512$ barns), obtaining a value of 0.0481 cm^{-1} for Σ_f^{25} . It is implicitly assumed in the calculation of the number of aluminum capture photons per fission that the ratio of the number of thermal neutrons absorbed in U^{235} to that in aluminum is proportional to the ratio of their respective macroscopic absorption cross sections. This may not be strictly true because of self-shielding and flux depression effects in the heterogeneous fuel plates. However, this has been determined to be a small effect⁶ that could not affect the absorption ratios by more than 4%.

6. E. G. Silver, private communication.

Table 2. Number of Photons per Neutron Capture in Aluminum for Various Energy Intervals.

Energy Group	E_a (Mev)	E_b (Mev)	\bar{E} (Mev)	$\omega^{Al}(\bar{E})$ (Photons/Capture)
I	3.0	5.0	4.0	0.727
II	5.0	7.0	6.0	0.202
III	7.0	9.0	8.0	0.349
IV	9.0	11.0	10.0	0

The Uncollided Flux

In the calculation of the uncollided flux it was first assumed that the thermal-neutron flux was the same throughout the reactor core. Later corrections were made to take into account the actual shape of the thermal-neutron flux (see below). If the flux is the same throughout the core, the number of fissions per cubic centimeter per second in the core is obtained by dividing the total number of fissions per second by the volume of the core.

The experimental measurements¹ were normalized to 1 watt of reactor power, which corresponds to 3.1×10^{10} fissions/sec. The total volume of the reactor core for the experiment was 1.15×10^5 cc (BSR Loading 22A). Thus, there were 2.7×10^5 fissions/sec.watt.cm³. These same conditions were assumed for the calculation.

The uncollided flux at the surface of the reactor core in a direction normal to the core surface is related to the source strength per unit volume of the core by the relation:

$$\begin{aligned} \Gamma^0(E, \vec{\Omega}) dE d\vec{\Omega} &= \int_0^L \frac{n(E) dE d\vec{\Omega} e^{-\mu(E)r}}{4\pi r^2} \cdot r^2 dr \\ &= \frac{n(E) dE d\vec{\Omega}}{4\pi} L \frac{1 - e^{-\mu(E)L}}{\mu(E)L} \end{aligned}$$

where

$\Gamma^0(E, \vec{\Omega})$ = uncollided flux of energy E in the direction $\vec{\Omega}$ per square centimeter per steradian,

$n(E)$ = number of gamma rays of energy E per cubic centimeter of reactor core,

L = length of the core in the direction Ω , cm,

μ = linear absorption coefficient of the reactor core material for radiation of energy E , cm^{-1} .

Thus the uncollided flux at the surface of the reactor, in the direction determined by the centerline, per square centimeter per steradian per Mev per watt per second is related to the total number of photons per fission $\nu(E)$ by the relation:

$$\Gamma^0(E) = K \cdot \nu(E) \cdot f(E)$$

$$f(E) = \frac{1 - e^{-\mu(E)L}}{\mu(E)L}$$

where

$f(E)$ = dimensionless function of the energy and the length of the core, taking into account the attenuation of the radiation of the core,

$$K = 2.7 \times 10^5 \times \frac{4L}{4\pi \Delta E}$$

$$= 4.83 \times 10^5 \text{ fissions/watt} \cdot \text{sec} \cdot \text{cm}^2 \cdot \text{steradian} \cdot \text{Mev},$$

L = length of the reactor core on the centerline = 45 cm,

ΔE = width of each of the energy groups = 2 Mev.

The metal-to-water ratio in the BSR is about 0.7; therefore, the linear absorption coefficients $\mu(E)$ for each energy group is determined as follows:

$$\mu(E) = 0.41 \mu_{\text{Al}}(E) + 0.59 \mu_{\text{H}_2\text{O}}(E)$$

where $\mu_{\text{Al}}(E)$ and $\mu_{\text{H}_2\text{O}}(E)$ are the linear absorption coefficients of aluminum

and water, respectively (the volume occupied by uranium was neglected). Table 3 lists the values for $\mu(\bar{E})$, the product $\mu(\bar{E})L$ and the function $f(\bar{E})$ for each of the four energy groups.

Table 3. Values of $\mu(\bar{E})$, $\mu(\bar{E})L$, and $f(\bar{E})$
for Various Energy Groups

Energy Group	\bar{E} (Mev)	$\mu(\bar{E})$ (cm^{-1})	$\mu(\bar{E})L$	$f(\bar{E})$
I	4.0	0.055	2.48	0.37
II	6.0	0.046	2.09	0.42
III	8.0	0.042	1.885	0.44
IV	10.0	0.039	1.77	0.45

The results of the calculations of the uncollided flux for all of the energy groups are tabulated in Table 4.

The Compton-Scattered Flux

Since, as mentioned above, the Compton-scattered gamma-ray flux was calculated under "straightahead" and "one-collision" assumptions, only those gamma rays that originated from Compton scattering of primary gamma rays that were moving in the direction of the detector before their collision were considered. The total number of primary photons that were originally moving in the direction of the detector was obtained by setting $\mu(E) = 0$ [or $f(E) = 1$] in the equation for the uncollided flux. The difference between this number and the actual uncollided flux is the number of photons that had originally been moving in the direction toward the surface of the core but were removed by absorption or scattered in the core material. The portion that was Compton scattered was:

$$k \sqrt{(E)} \sqrt{1 - f(E)} \frac{\mu_C(E)}{\mu(E)}$$

where $\mu_C(E)$ is the linear absorption coefficient for Compton scattering alone.

If a primary photon of energy E makes a Compton scattering there is a probability

$$P(E, E') dE' = \frac{1}{\sigma_C(E)} \frac{d\sigma_C(E)}{d(E')} dE'$$

given by the Klein-Nishina law, that the scattered photon will have an energy between E' and $E' + dE'$. Since the energy spectrum for this calculation has been divided into four regions, it is possible to calculate the probability that a photon of energy \bar{E}_1 will be scattered into the k^{th} energy group by the equation,

Table 4. Gamma-Ray Flux at the Surface of the BSR,
Assuming a Flat Thermal-Neutron Flux

Energy Group	\bar{E} (Mev)	$\frac{\mu_C(\bar{E})}{\mu(\bar{E})}$	Flux (photons/cm ² ·sec·Mev·watt·steradian)		
			Uncollided	Compton-Scattered	Total
I	4.0	0.889	9.2×10^4	2.8×10^4	1.2×10^5
II	6.0	0.805	1.4×10^4	2.9×10^3	1.7×10^4
III	8.0	0.728	9.0×10^3	6.9×10^2	1.0×10^4
IV	10.0	0.658	1.8×10^2	7.0	1.9×10^2

$$\alpha_{ik} = \frac{1}{\sigma_C(\bar{E}_i)} \int_{E_{ak}}^{E_{bk}} \frac{d\sigma_C(\bar{E}_i)}{dE'} dE', \quad k \leq i$$

where

E_{ak}, E_{bk} = energy limits of the k^{th} group,

\bar{E}_i = representative energy of the i^{th} group.

The values of α_{ik} for each of the four energy groups are listed in Table 5.

The Compton spectrum can then easily be calculated as

$$P'(\bar{E}_k) = K \sum_{i \geq k} \nu(\bar{E}_i) \left[1 - f(\bar{E}_i) \right] \frac{\mu_C(\bar{E}_i)}{\mu(\bar{E}_i)} \alpha_{ik}$$

The results of the calculations of the Compton-scattered flux, along with values of the ratio $\mu_C(\bar{E})/\mu(\bar{E})$, and the uncollided flux, are given in Table 4.

Correction for the Thermal-Neutron Flux Shape

For this calculation of the gamma-ray flux at the surface of the BSR it was assumed that the thermal-neutron flux was constant throughout the reactor core. This of course, is not correct, gold foil measurements having shown that the average value of the flux along the reactor centerline ($\bar{\phi}_0$) is 11.23×10^{10} neutrons/cm²·sec while the average value of the flux throughout the core ($\bar{\phi}$) is 7.36×10^{10} neutrons/cm²·sec. Before any comparison can be made with the experimental results¹ a correction must be made for the shape of the flux. In the experiment the detector collimator was pointed toward the center of the core; therefore, the experimental results could be expected to indicate a greater intensity than the calculations. In order to correct the calculations, the calculated values of the gamma-ray flux given in

Table 5. Coefficients for Compton Scattering

\bar{E}_k (Mev)	α_{ik}			
	$E_i = 4$ Mev	$E_i = 6$ Mev	$E_i = 8$ Mev	$E_i = 10$ Mev
4	0.165	0.204	0.159	0.140
6	0	0.096	0.139	0.110
8	0	0	0.066	0.101
10	0	0	0	0.050

Table 4 were multiplied by the ratio $\bar{\phi}_0/\bar{\phi}$. The corrected calculated values for the flux on the centerline at the surface of the reactor are shown in Table 6.

Conclusion

The corrected calculated gamma-ray spectrum is plotted in Fig. 1 along with the experimental spectrum¹ and the spectrum determined in a previous calculation.² The two calculations give almost the same results; this can be attributed to the fact that the uncollided flux in both calculations was obtained in the same way, and the different methods of calculating the Compton-scattered flux had little effect on the high-energy end of the spectrum where the Compton-scattered flux is small compared to the uncollided flux.

It should be pointed out that the experimental values plotted in Fig. 1 do not represent the exact values reported previously.¹ In re-examining the experimental data for the BSR gamma-ray spectrum, it appeared probable that the error due to leakage through the collimator side walls used for this particular experiment was appreciably greater than the 20% quoted when the data was originally published. Application of a recent formula⁷ to the geometry of this experiment indicates that the effective solid angle should be increased by a factor of 1.7. Therefore, the gamma-ray flux plotted in Fig. 1 was reduced by this factor.*

It is surprising that the calculation performed above gives a much lower intensity for the spectrum than the one measured, in spite of the fact that all the assumptions made in the calculation were made in a direction that would tend to overestimate the flux. This seems to indicate a discrepancy between the experimental data¹ and the capture gamma-ray

7. Robert L. Mather, Gamma-Ray Collimator Penetration and Scattering Effect (unpublished).

* This experimental data was corrected by F. C. Maienschein and reported to the author by private communication.

Table 6. Calculated Gamma-Ray Flux on the Centerline
at the Surface of the BSR Core

Energy Group	E_a (Mev)	E_b (Mev)	\bar{E} (Mev)	Total Flux (photons/cm ² ·sec·Mev·watt·steradian)
I	3.0	5.0	4.0	1.8×10^5
II	5.0	7.0	6.0	2.6×10^4
III	7.0	9.0	3.0	1.5×10^4
IV	9.0	11.0	10.0	2.9×10^2

spectrum of aluminum reported by Kinsey, Bartholomew, and Walker.⁵ The error in the experimental data is uncertain but it would appear to be less than the marked discrepancy between the experiment and the calculation. Further experimental work is planned as soon as the new Model IV spectrometer and positioner are available.⁸

8. R. W. Peelle and T. A. Love, ORNL-2340, Part 5 (classified).

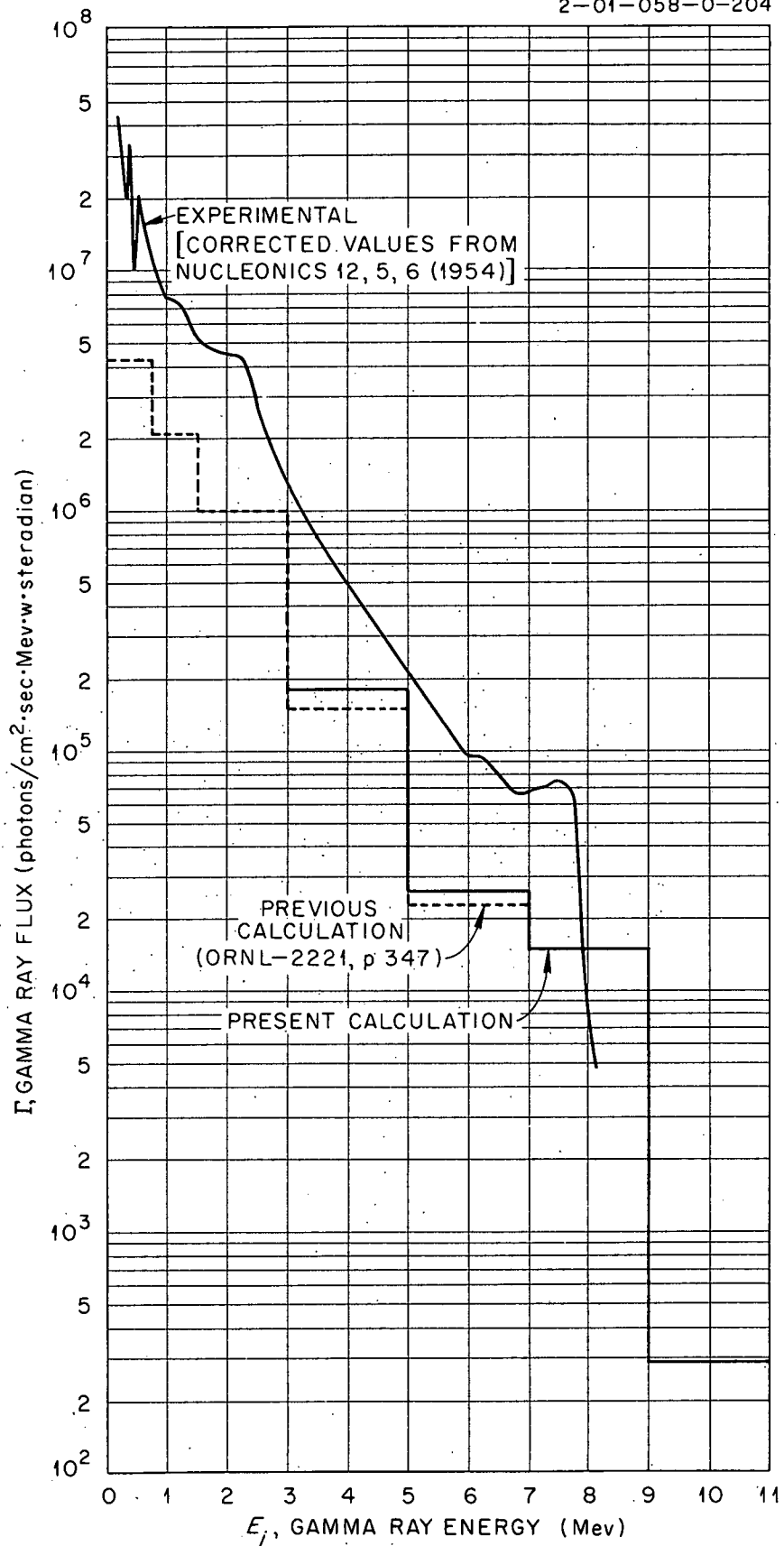


Fig. 1. Comparison of Calculated and Experimental Spectra of Gamma Rays at the Surface of the Bulk Shielding Facility Reactor.