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TECHNICAL SUMMARY OF ALL ORNL SUMMER SHIELDING SESSION (1949)

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ORNL SUMMER SHIELDING SESSION

Gale Young, Leader

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TECHNICAL SUMMARY OF ALL ORNL SUMMER SHIELDING SESSION (1949)

Ralph Zirkind

Date Issued JUN 19 1950

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This report is one of a series on shielding and represents part of the work of a summer group working principally at Oak Ridge National Laboratory. Participants are drawn from Nuclear Development Associates (Navy contract), NEPA, Bureau of Ships and Bureau of Aeronautics (Navy Department), AEC, RAND, General Electric, Argonne and Oak Ridge National Laboratory.

These reports are issued through ORNL on Contract No. W-7405, eng. 26.

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Introduction

This report summarizes the activities of the Summer (1949) Shielding Group[1] assembled at the Oak Ridge National Laboratory. Under the leadership of Gale Young, Nuclear Development Associates, this group was gathered to evaluate the theoretical and experimental aspects of reactor shielding, both neutron and gamma-ray. From this evaluation, a program was to be formulated with the purpose of closing some of the existing gaps in shielding knowledge. Special emphasis in this program was to be placed on a set of crucial experiments for the new ORNL shield testing facility, the so-called "lid tank." The latter experimental program was formulated and described in ORNL-427.

A major portion of the group's time was spent in formulating the present state of the art (theoretical and experimental) with some new work in analytical and numerical methods. It is the intent of this report to present this information in a concise and informative manner. Much detail must be omitted, but the writer hopes that sufficient is given to be of some value.

In Section 1, neutron cross-section data, gamma-ray absorption coefficients, etc. of particular interest to shielding are presented. The available analytical techniques are presented in Section 2. This is then followed in Section 3 by a summary of the best available experimental data. Finally, a development of numerical methods applicable to particle attenuation problems is discussed in Section 4.

This report has been prepared under the Bureau of Aeronautics, Research Division Project DR 1158.

The writer wishes to express his gratitude to all the participants of the Summer Session for many helpful suggestions and especially to Drs. Welton and Nelson, and Messrs. Sleeper and Kahn for critically reviewing portions of the manuscript.

1/ The list of participants is given in ORNL-437.

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Section I

Basic Data

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I. Basic Data

In view of the plethora of difficulties associated with the shielding problem one would hope that at least the basic data, i.e. neutron and gamma-ray cross sections, were well known. This is, however, not the case, with the situation worse for neutrons than for gamma-rays. The importance of reliable cross section data is clear: first, the accuracy of a calculation, analytical or numerical, is no better than the data used; and (2), the structure of the cross-section as a function of energy can aid in formulating the analytical problem. Hence, a review of the neutron and gamma-ray cross-sections was made.

A. Neutrons

Because of the enormity of the task, the review of neutron data has been guided by the role particular data plays in the shielding problem. Thus, the effort is largely confined to those nuclei which have been explored as shielding material and to cross-sections at high energies ($\gtrsim$ 2 Mev).

At the outset of the Summer Session Friedman and Feshbach[1] collected and tabulated some fast neutron data available (particularly, the work of Barschall). Subsequently, Feshbach[2] analyzed the available fast neutron data by the theory developed by Feshbach and Weisskopf (see references below). This theory provides a schematic treatment of the neutron cross sections of different nuclei making possible the analysis of experimental

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1/ F. Friedman and H. Feshbach, "Collection of Data for Shielding" ORNL-418

2/ Feshbach, Peaslee and Weisskopf, Phys. Rev., 71, 45 (1947) and Feshbach and Weisskopf, Phys. Rev. 76, 11 (1949). The latter reference contains the bulk of Feshbach's reports and therefore the discussion here is a brief outline.

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data and, interpolation and extrapolation when necessary.

The energy region considered was roughly between 200 kw and several million electron volts. Below 1 Mev, light nuclei and "magic number" nuclei, e.g. Pb, behave differently from most heavy nuclei, in that they exhibit considerable resonance regions. Thus, Feshbach discusses the behavior of the absorption, total and elastic cross-sections at or near the resonances and gives formulae for their estimation. Between the resonances the same formulae apply omitting the resonance features. The general results indicate (1) resonances for fast neutrons are scattering resonances and (2) the minima which give rise to "windows" in the energy probably will not occur for neutron energies larger than $\sim$1 Mev for light elements and about .5 Mev for magic number nuclei. It is to be noted that between resonances the total cross-section is approximately equal to the elastic cross-section.

In the non-resonance region, the behavior of the different cross-sections included are examined and formulae for calculating their values as a function of energy are given. It is to be noted that the formulae developed apply as well to the resonance region if they are interpreted as average cross-sections over an energy region containing several resonances.

A significant feature of this report for shield work is the development of a method of evaluating qualitatively the energy lost by a neutron undergoing an inelastic collision and the relative probability for a given energy loss.

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In addition, the discussion of the experimental data and the sources used is given. The data are compared with the theoretical predictions and this indicates reasonable agreement in view of the crudeness of the theory.

In addition to the above described work, the detailed structure of the cross-sections (total, scattering, inelastic) for hydrogen and oxygen was calculated by Feshbach. (3) This work was done in preparation for a Monte-Carlo calculation on a water shield because of the importance of this material in neutron shields. Also, recent unpublished neutron data (4) was obtained from the Minnesota group giving the total cross-sections in fine detail for C, Si, S, and Mg in the energy range of .5 to 5 Mev.

B. Gamma-rays

For gamma-rays the situation is immensely better since theory exists which predicts absorption coefficients to within 10% and a number of good experiments have been made for comparison with theory.

Powell and Snyder (5) attempted to collect all existing experimental data on gamma ray absorption coefficients from 10^{-2} Mev to 100 Mev and then compared these with the values expected from the best theoretical formulae. By "best" is meant the region for which the approximations made are reasonably good. The results of their findings can be summarized as follows:

(1) in general, theoretical formulae cannot be relied upon to give an accuracy of greater than 5% for the mass absorption coefficient,

[redacted]

(2) in the photoelectric region, theoretical values were comparable to the experimental ones, although for the heavy elements errors of $\sim 5\%$ could be expected.

(3) The Klein-Nishina formula (Compton Scattering) would accurately give the contribution of scattering to the total coefficient except for a minor correction due to binding ($\sim 1\%$).

(4) The theoretical pair-production cross-section (Bethe-Heitler) is found to give accurate results only for small Z whereas for lead it exceeds the experimental value by 10% at energies of ~ 10 Mev or greater. This disagreement still persists after one adds the contribution of pair creation in the field of the atomic electrons.

The values of the absorption coefficients determined by Powell and Snyder's semi-empirical approach are reproduced here in Figures 1 and 2. The former figure applies to elements with $Z \leq 13$ (Al) and the latter for elements with $Z \geq 13$.

As to the method of construction, Cu, Al, Sn and Pb were used as the reference elements, because of the available experimental information. The theoretically calculated curves were then adjusted to fit the experimental data. In this manner interpolation was made possible in the energy regions where experimental data are lacking.

It is to be noted that little or no experimental data are available in the energy region (2 - 10 Mev) of special interest to gamma-ray

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shielding problem. As is well known this is the region where for heavy elements a minimum exists in the absorption coefficient.

The last item of basic data reviewed by the Summer Group is the quantity and quality of gamma-rays to be encountered in the reactor shielding problem. The source of these gamma-rays are (a) fission and fission products and (b) n-γ reactions where the former originate within the reactor and the latter primarily in the shield (to a lesser extent in the reactor).

Blizard and Young (6) made a survey of available information on gamma-rays emitted from fission and fission products in reference (6). The data indicate that a total photon energy of ~ 5 Mev/fission is emitted as prompt gammas during "25" fission. However, due to experimental difficulties the spectral distribution is uncertain. Thus, one can only state, at present, that these gammas appear to be softer than fission product or neutron capture gammas. With respect to fission product gammas the photons emitted within the first few minutes are hard (~ 3 Mev) whereas the long period product gammas (after ten minutes) are soft. The total energy of the long period gammas is ~ 1.4 Mev. Since it is believed that ~ 5 Mev/fission is released then one may surmise that one to two photons with energies ~ 3 Mev or greater are released per fission from fission products.

The gammas emitted from (n-γ) reactions may be subdivided into two groups: a) photons emitted as a consequence of thermal neutron capture and b) those from inelastic scattering of fast neutrons. The latter did not receive attention this summer. However, Blizard in reference (7) has reviewed the literature including the latest unpublished Canadian data on

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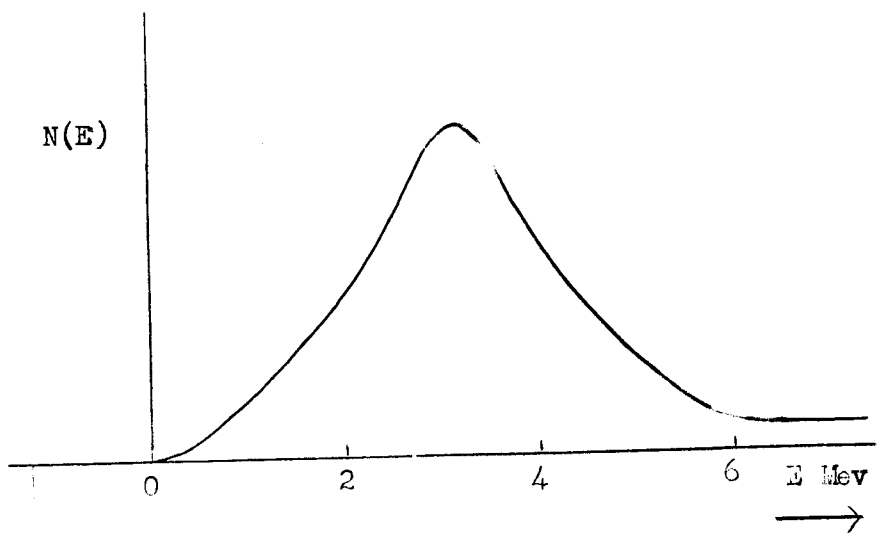
capture gamma-rays. To date detail information exists only on the spectral distribution with no precise data on the number of photons emitted.

From the experiments performed on twelve elements, one can draw the general conclusion that the largest fraction of the emitted photons have energies between 2 and 6 Mev with a maximum at $\simeq$ 3.6 Mev and a small tail extending to 11 Mev which is questionable (see schematic drawing for qualitative picture.



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7. E. P. Blizard, "Summary of Capture Gammas" ORNL-419

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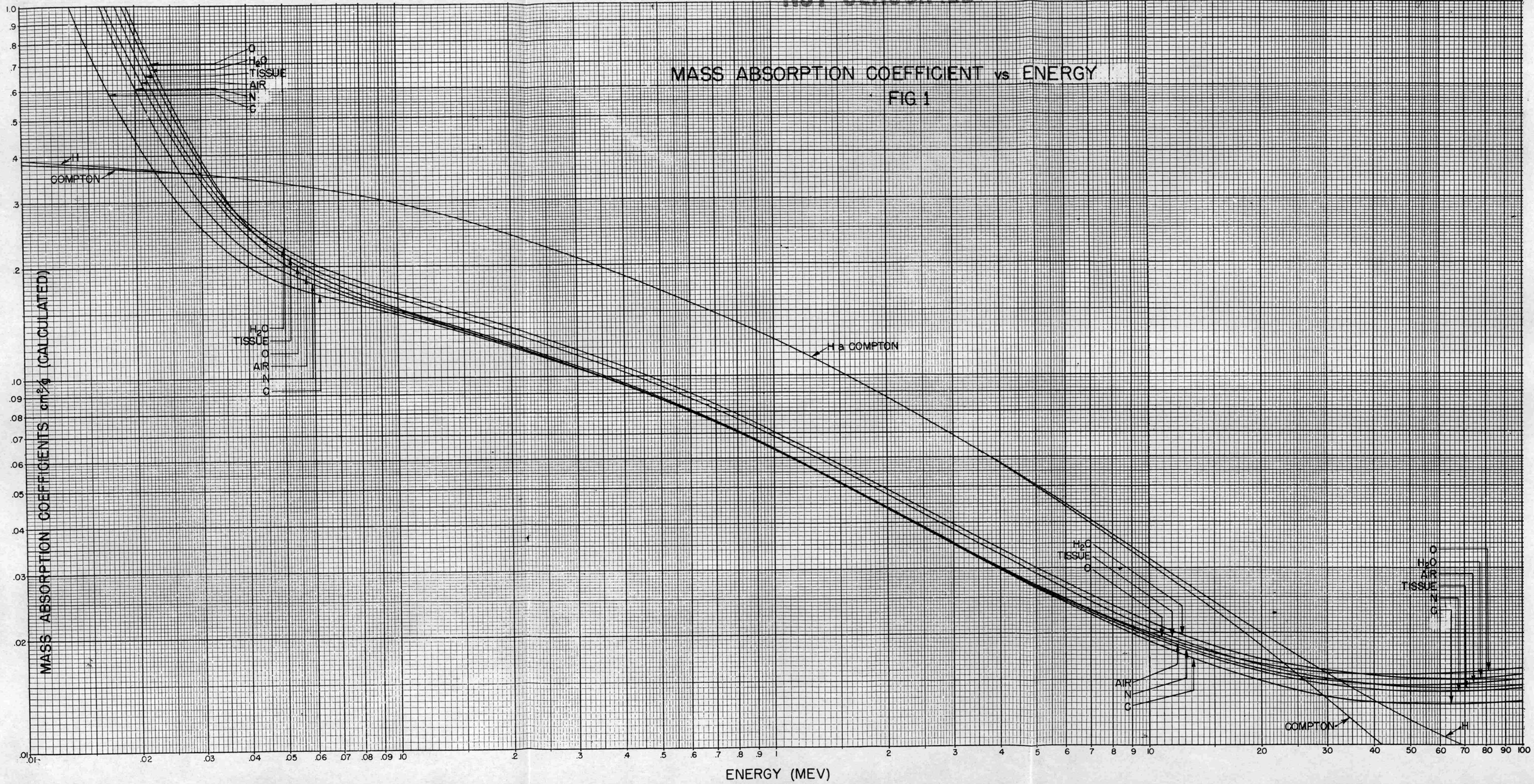

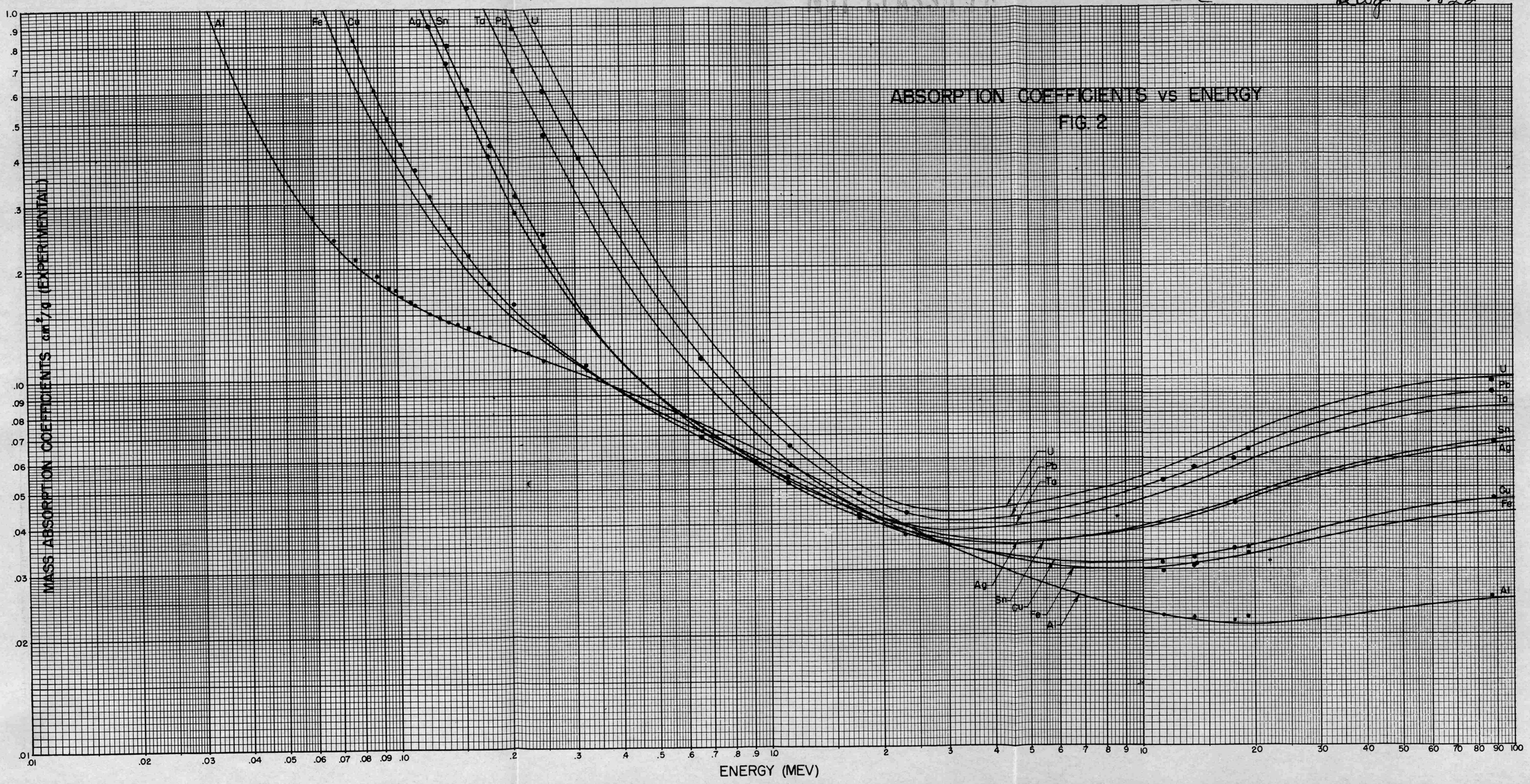
ABSORPTION COEFFICIENTS vs ENERGY
FIG. 2
MASS ABSORPTION COEFFICIENTS cm²/g (EXPERIMENTAL)
ENERGY (MEV)
Al
Fe
Cu
Ag
Sn
Ta
Pb
U
1.0
.9
.8
.7
.6
.5
.4
.3
.2
.10
.09
.08
.07
.06
.05
.04
.03
.02
.01
.01
.02
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.04
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1.0
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Section II

Analytical Methods

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II. Analytical Procedures[1]

During the Summer Shielding Session, the activities in the theoretical (analytical) phase of shielding consisted of (1) an excellent critical review by Welton of existent analytical methods and (2) some new extensions by Young, Murray and Butler. For convenience, the different procedures will be discussed as they apply to neutrons and gamma-rays separately. In each case, a summary of Welton's discussion will be given and this will be followed by the new work.

A. Neutrons

The results of Welton's study (1) will be presented here under three headings: hydrogen, hydrogenous mixtures and non-hydrogenous materials. (It is to be noted that this grouping is not based on similarity in shielding properties).

(1) Hydrogen

If one adopts the unrealistic assumption of constant cross-section for hydrogen, then the method developed by Wick of expanding the scattering function into spherical harmonics, gives a solution valid at large distances. This latter method is similar to that of Hurwitz and Tonks (KAPL-76). Wick has also found an asymptotic solution for varying cross-section.

The use of the straight ahead approximation here is another approach. This approximation is made possible because of the highly anisotropic scattering exhibited by hydrogen. If the energy dependence of

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1/ For general reference see G. C. Wick P. R. 75 (Mar. 1949), H. Hurwitz "Theory of Neutron Attenuation" (to be published in Nucleonics) and H. A. Bethe, KAPL-56 (1948).

[redacted]

the cross-section is taken as

$$\sigma = \sigma_o \, (E_o/E)^{1/n} \qquad (1)$$

(E_o = source energy)

where n is a positive number whose value depends on E_o, (for E_o = 1 Mev, n = 2, and approaches 1 with increasing E_o) then the asymptotic solution is of a Greuling type

$$x^n \, e^{-\sigma_o x} \qquad (2)$$

The angular effects can be included in the theory, and this leads to a correction to the exponent "n" . An approximate calculation indicates that these effects become less important with increasing primary energy. This has been corroborated by a comparison between a Monte Carlo calculation (by Kahn) and an analytical one where the source is 8 Mev neutrons. The comparison shows that the analytical calculation (straight ahead) leads to an excess flux by a factor of 5 for a shield thickness of 16 mean free paths. However, Kahn's work almost certainly has a large statistical error in it and, therefore, the comparison is not justified.

(2) Hydrogenous Mixtures

Let us consider here a mixture of heavy nuclei plus hydrogen. This is more complex than the above but lends itself to reasonable simplifications. These are (1), the neutron cross-section of the heavy elements vary smoothly with energy and the scattering is isotropic, (2), an inelastic collision with heavy nuclei is equivalent to a capture and (3), straight ahead treatment for hydrogen. With these simplifications Murray, in part, and Welton and Goertzel (2) have treated this

[redacted]

problem. The latter assumed the hydrogen cross-section to obey equation (1).

If we define a parameter $f = \sigma_a/\sigma$ where σ_a, the capture cross-section, is the sum of the hydrogen cross-section and inelastic scattering and σ is the total cross-section, (the two cross-sections being evaluated at the source energy) the results of the treatment indicate that at considerable distances from the source the density is of diffusion theory form. For $f \leq .5$ the diffusion type solution sets in at $\sigma r \leq 1$, however, when $f > .6$ this type of solution seems unlikely for r less than an actual shield thickness.

The asymptotic solution in the diffusion region for energies much below the source energy is

$$r^n \; e^{-Kr} \tag{3}$$

where n is the exponent in equation (2) and K is determined from the transcendental equation

$$\frac{K/\sigma}{\tanh^{-1} K/\sigma} = 1 - f \tag{4}$$

Butler (3) has attempted to give a treatment of this problem which will yield the asymptotic solution at large distances, under the condition that the cross-sections vary arbitrarily with energy (and are isotropic). His result should be of the form similar to equation (3) but more complicated due to the energy spectrum. This solution, although certainly asymptotically correct, may not be valid except for very large r if the cross-sections are changing rapidly at the source energy.

[redacted]

(3) Non-hydrogenous

For a light element shield material, like boron, which both slows and captures neutrons a one-velocity approach may be used provided the cross-sections vary slightly in the energy interval traversed by the neutron before capture. The Hurwitz-Tonks approach can yield the energy spectrum, if desired.

In the case of a pure heavy shield material, a diffusion type of theory is applicable when $\sigma_a/\sigma_s < .5$ (inelastic scattering is considered an absorption process). That is, if all processes are isotropic, then the source neutrons penetrate until elastically scattered and the scattered neutrons become a space distributed source for lower energy neutrons, etc. The space distribution is easily calculated where the diffusion length, $1/\kappa$, is determined from equation (4). If, however, $f \simeq 1$, i.e., $\sigma_a/\sigma \sim 1$, the diffusion approach is inapplicable and a more reasonable one is to consider first collision densities only.

The assumption of isotropy is probably poor for neutrons with energies of 1 Mev or greater. This objection can be removed simply, although no one has done it. At large distances from the source a diffusion type solution will apply, but the diffusion length is obtained from a somewhat more complicated equation than equation (4).

When we have a mixture of a heavy element with boron, the slowing down by boron introduces an added complication. For energies where σ_s of boron is small compared with the inelastic cross-section of the heavy material the slowing down process can be neglected and the discussion in the above paragraph applies.

If the cross-sections vary with energy the process

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is quite complicated. A first approach to the problem is to assume that cross-sections are constant in the energy region required for boron capture. The problem then resolves itself to a one-velocity problem. An improvement on this has been carried out by Hurwitz and Tonks on a suggestion of Bethe, namely expanding the neutron-boron elastic cross-section into spherical harmonics. This seems to give excellent results. It may be possible to include varying cross-sections in a procedure analogous to Wick.

In addition to the work reviewed above, some specialized extensions of neutron attenuation theory were carried out for hydrogenous mixtures and non-hydrogenous materials by Butler and Murray respectively.

Butler (3) has considered the general case of a pure scatterer, scattering isotropic with no energy loss + pure absorber + hydrogen (straight-ahead approximation) where the cross-sections vary monotonically. For this problem he has found an asymptotic solution whose form is similar to equation (1) except that it is now multiplied by a complicated function of the energy. In addition to this, asymptotic solutions have been found when some of the conditions given above are specialized: (a) the sum of the cross-sections of the three elements were held constant and (b) no scatterer is present.

Murray has found (4), using the method of moments, an asymptotic solution for the flux as a function of the energy in a non-hydrogenous material under the following conditions: (a) constant cross-section and (b) anisotropic scattering and absorption present. It is to be noted that the solution of the energy distribution is satisfactory provided the maximum energy is considerably less than the source energy.

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Further, Murray has carried out a generalization of the age equation for variable cross-section with absorption present. From the solution obtained, still in integral form, an exponential asymptotic solution is expected to result. For mixtures of heavy nuclei and hydrogen, the shadow scattering from heavy nuclei is taken into account and a method for finding the upper and lower bounds of the attenuation is found in asymptotic form.

Young (5) has presented an interesting approach to the hydrogen problem by piecing together two Greuling solutions where each solution results from a different cross-section energy dependence. By this manner of treatment better overall approximations can be made than by assuming a single cross-section variation with energy over a large energy region.

B. Gamma-rays

The major effort expended on gamma-ray attenuation theory was confined to a critical review of existing methods by Welton in reference (1). The remainder of the discussion is based on this review. For orientation purposes, we shall first consider possible solutions with the straight ahead approximation, i.e. Compton collision only degrades the photon energy, and then, solutions to include angular effects. The solutions will differ further depending on whether the source energy is above or below the minimum of the cross-section curve.

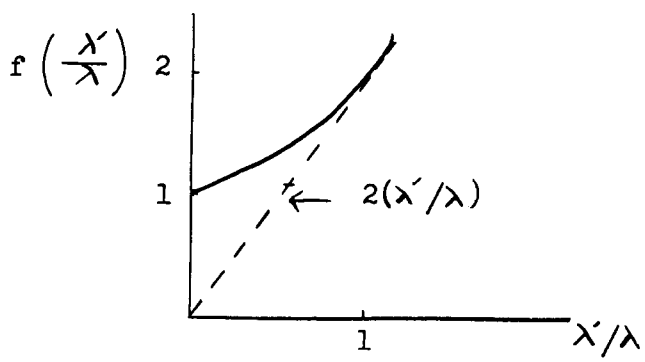
Before discussing the actual solutions a few explanatory comments regarding the cross-section curve will be made. From the Compton scattering formula the Klein-Nishina expression for the differential cross-section can be written as

[redacted]

$$\frac{\lambda'}{\lambda}\left[1 + \frac{\lambda'^2}{\lambda^2}\right] + \text{angular effects} \qquad (5)$$

where the unit of length (absorption length) is 8/3 the Thomson mean free path, and λ', λ denote the wavelengths before and after collision respectively; λ and λ' are measured in units of the Compton wavelength.

If now, one neglects angular effects ($\sin^2\theta = 0$) (as is implied in the straight ahead approximation) then equation (5) can be represented as $f(\lambda'/\lambda)$ where $f(\lambda'/\lambda)$ has a simple geometrical representation (see accompanying sketch). We see immediately that for small degradations, $\lambda'/\lambda \sim 1$ f has a value of 2 and decreases with higher energy losses/collision.



With these brief comments we now consider the different solutions of the transport equation in the following order (1) straight ahead without minimum i.e. no pair-production present (2) straight ahead with minimum (3) angular effects with no minimum.

Numerous investigations have considered the first problem using somewhat different approximations to $f(\lambda'/\lambda)$: Greuling suggested $2(\lambda'\lambda)$, Fano assumed it to be a constant with the value 2, and Bethe, Fano and Karr take $f(\lambda'/\lambda) = 1.5$.

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In addition, the variation of the cross-section with wavelength is required here for the solution of the transport equation. Greuling found that a linear variation of σ with λ will lead to a solution given by a product of a confluent hypergeometric function and an exponential. Hence, if one expands σ is a series

$$\sigma(\lambda) = \sigma_0 + \sigma_0' (\lambda - \lambda_0) + \ldots \tag{6}$$

and neglect terms beyond the second, the solution for x large can be written as

$$x^n e^{-\sigma_0 x} \tag{7}$$

where n, "is determined by the value assumed by the approximation to $1 + (\lambda'/\lambda)^2$ for λ'/λ equal to unity and by the slope of the total cross-section at the source energy." That is, n has the values $2/\sigma_0'$ or $1.5/\sigma_0'$.

With the approximations of Greuling the solution has the form

$$\psi(x, \lambda) = e^{-\sigma_0 x} F(ux) \tag{8}$$

where u is $(\lambda - \lambda_0)$ and F is a simple power series in ux. The conditions of validity with the necessary arguments are given in reference (1).

Thus, for gamma rays with source energies below the minimum the transmission can be fairly well calculated, and with some additional work reliable values of the buildup factor can be derived. However, when the source energy is near the minimum no solution is yet available <u>which takes completely into account the necessary angular effects</u>.

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Let us now examine the case where the cross-section has a minimum, i.e., when pair production is present. Fano and Hurwitz (1) have both investigated this problem and given methods of calculating the space behavior of the transmitted spectrum where the solution holds for energies less than the minimum. Although their spatial dependence is in agreement the spectra will disagree in detail.

Fano assumes (a) angular deflections negligible and (b) $f(\lambda'/\lambda)$ given by a numerical constant equal to 2 (see Figure 1). Hurwitz, on the other hand, proposed a numerical method which allows the exact cross-section to be used but also neglects angular effects. In Fano's method the first approximation is bad, but if it is replaced by Greuling's suggestion, namely $2(\lambda'/\lambda)$ one would expect to obtain useful results. Hurwitz's procedure is perhaps unduly complicated but could be used for checking purposes.

To this point we have not considered any effects due to angular deflections. The inclusion of these effects in the theory is important since even for high energy photons they cannot be neglected.

For the case below the minimum, angular deflections seem to seriously effect the space dependence of the build-up factor. The effect is to reduce the exponent n by a factor $(1 - \sigma_0/\sigma_0')$. This result has been found by Hurwitz and Fano independently and seems to be reliable for $\sigma_0/\sigma_0' \ll 1$.

A Monte Carlo calculation of transmission of 2.5 mc^2 gamma-rays through 20 relaxation lengths by Kahn does not agree with the analytical results. It appears that the Monte Carlo method is in error as the statistics used by Kahn were poor.

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Some work was accomplished for the case of source energy near the minimum and constant, and this indicated the angular effects to be quite significant. The characteristic parameter here is $\sqrt{\sigma_o/2}$. In fact the result given is as follows: if the spatial term for the straight ahead approximation is given by exp $\sqrt{8xu}$ where $u = (\lambda - \lambda_o)$ and x is the linear distance, then the corrected exponent is

$$8xu\left(1 - \frac{\sqrt{\sigma_o}}{2} + \frac{\sigma_o}{8}\right).$$

It is essential here that the coefficient "8" be really close to 8 in order for the build-up factor to be accurate.

For source energies above the minimum no information is available on angular effects.

Young has applied his piece-wise Greuling solutions to the gamma-ray problem. He approximates the cross-section with a minimum as made up of several segments and finds the Greuling solutions thereof on the assumptions (a) straight ahead scattering and (b) scattering distributes the energy of the emerging rays uniformly over the lower energies. These solutions are then matched at the intersection points.

[redacted]

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5. G. Young, "Piecewise Greuling Solutions for Hydrogen" ORNL-415 (1949)

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Section III

Experimental Results

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III. Experimental Results

In this section some indications of the essential features of the experimental results obtained from bulk shield tests will be discussed. These tests cover a large group of attenuation experiments both for neutrons and gamma-rays, performed at ORNL, using a 28" x 32" hole extending through the shield. The source used was the ORNL pile modified to give a higher fast neutron flux in the vicinity of the core hole.

This facility and its limitations are elaborated on in Reference (1). One serious limitation, streaming, or the influx of neutrons from the ORNL pile shield proper into the sample under test, introduces spurious effects for which one must be on guard in interpreting the results.

As mentioned above, the source was the ORNL pile with a slight modification. This source is neither isotropic nor collimated. Hence, to remove this uncertainty, and incidentally that of streaming, a separate fission source was employed in one of the tests on MO concrete. From the results obtained one could state with some assurance that the observed relaxation lengths using a pile source would be the same as from an isotropic fission source. With these brief general comments we now consider the summarized results of the tests.

Sleeper (1) has critically reviewed all the neutron attenuation tests made to date and assembled those experiments upon which some degree of trust can be laid. The criteria used for selecting the experiments were: (a) the attenuations had to be 10^5 or greater, with no interference due to streaming and (b) the compositions must be known with the exception of the

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hydrogen content of the concrete shields. In the latter instance the accuracy was ± 20% or better, thus introducing some degree of uncertainty in any interpretation of the results.

The variety of materials covered by these tests is quite large: concretes -- Portland, Brookhaven, W-1, MO and M1 (Fe bearing); low density materials -- boron cement (B.C.), H_2O and B_4C; and high density materials -- Fe, Pb and WC (see Table 1 for composition of materials). Further, the thicknesses of material used yielded attenuations from 10^5 to 10^8.

Let us now examine in detail the results obtained. First, in the case of uniform shields, i.e. uniform weight distribution, there exists a distinct difference between low density materials and mixtures of light and heavy materials. The low density* materials exhibit nearly exponential attenuation when the source is a collimated beam, whereas the mixtures exhibit exponential attenuation for an isotropic source (see Figures 1 and 2). Hence, in either case one can specify an aymptotic relaxation length for a known geometry. These relaxation lengths are summarized in Table 2. It is to be noted that the measured attenuation curve for the light materials (isotropic plane source) departs from an exponential (see Figure 3).

A second result obtained is that the relaxation lengths for high density concretes do not differ appreciably although the iron to hydrogen ratio covers a wide range. In Figure 4, the attenuation curves for the Brookhaven and W1 shields indicate a behavior similar to the other Fe-bearing shields, MO and M1.

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* The experiments on H_2O by Hill and Roberts were carried out with point source geometry. The results were then transformed to that expected for an isotropic or collimated source by standard mathematical techniques.

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With regard to the non-uniform shields, tests were made with a slab ($\sim$8" thick) of heavy metal -- Fe, Pb or WC, placed between a uniform shield (analyzer) and the neutron source. The results indicate that the attenuation curve in the uniform shields is not a pure exponential, i.e. transient effects are present in the initial portion of the analyzer. This is in all probability due to the inelastic scattering in the heavy slab, so that the pile neutron spectrum is moderated to some extent, and therefore a non-equilibrium condition exists at the beginning of the uniform section of the shield. After some thickness $\leq$ 5 m.f.p.'s, depending on the material, equilibrium is approached, and the attenuation curve is vertically displaced downward from that for the completely uniform shield by a constant factor (see Figure 3 and 5). In Table 3 the characteristics of the front slabs are given.

We shall now augment the above discussion with the specific results obtained by Ergen and Clifford (2) on paraffin. For pure paraffin the tests indicated that the attenuation curve was similar to water. This is to be expected since the two materials are similar. In spite of this, the results are inconclusive since the quality of the data is questionable beyond 10^4, although these experiments were carried out to an attenuation of 4×10^5.

With reference to the different arrangements of WC and B_4C, it is interesting to note the following: the overall neutron attenuation for equal thicknesses ($\sim$110 cms) is approximately the same; however, the character of the curves is radically different. This difference depends on the positioning of the boron and tuugsten layers relative to the source. If boron is placed immediately after the source, the attenuation curve

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begins with a steep slope, whereas tungsten produces only a slight attenuation. This is to be expected since these curves represent attenuation of non-thermal neutrons ($E \approx 10$ kev), at the best, and the boron capture cross section here is enormous.

Yet this is not the complete picture since it must be examined in conjunction with gamma-ray attenuation, i.e., although the neutron attenuation is the same, the gamma-ray attenuation is essentially different.

Of the gamma-ray attenuation we discuss only the tests on WC and B_4C described by Ergen and Clifford (2) where the attenuation extended to $\sim 10^6$ and the ORNL pile served as the source. The results of the tests can be summarized as follows:

(1) When sufficient B_4C is placed between the source and the tungsten layer, the neutron beam doesn't appreciably affect the gamma-ray attenuation, and therefore one is concerned essentially with primary gamma-rays.

(2) When a thick layer of tungsten follows the source directly, the secondary gamma-rays from tungsten dominate the attenuation curve and the whole shield design.

(3) The mass attenuation factors for primary gamma-rays are 27 gm/cm^2 and 48.5 gm/cm^2 for tungsten and boron respectively. For a uniform shield of B_4C and WC the mass attenuation factor is $\sim 40 \pm 5$ gm/cm^2.

In addition to the above major activities on the experimental side of shielding, Sleeper (3) reviewed two M1 shield tests. This was done because the new experiments gave detailed information on neutron and gamma-ray

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attenuations by using 2" thick slabs instead of 16" slabs, as previously. The composition of the two shield materials was somewhat different, especially the hydrogen content. The results indicate (a) the relaxation length for neutrons is the same for both cases (b) no transition region occurs near the inner face of the shield for attenuation of the modified pile neutron spectrum, and (c) the gamma-ray mass attenuation factor is about 42 gm/cm^2.

Blizard (4 and 5) has made a study of the tests conducted on the Hanford and Canadian shields from the point of view of their validity. At present, the results show that the tests made at Chalk River were satisfactory, whereas the recent experiments on the Hanford shield performed at Hanford were considered poor due to the streaming problem. Any evidence of equilibrium in either shield is inconclusive, since it has never been demonstrated.

Finally, Powell in Reference (6) has compiled a bibliography on gamma-ray attenuation experiments. From an analysis of the experiments one can hope to get some information on gamma-ray buildup.

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1. H. P. Sleeper, Jr., "A Critical Review of ORNL Shield Measurements: Neutron Attenuation" ORNL-436 (Secret)

2. W. K. Ergen and C. E. Clifford, "Neutron and Gamma Attenuation Through WC and B_4C" ORNL-435 (Secret)

3. H. P. Sleeper and C. E. Clifford, "Attenuation Measurements in Heavy Aggregate M1 Concrete" ORNL-438 (Secret)

4. E. P. Blizard, "Measurements on Hanford Type Shields" ORNL-430 (Secret)

5. E. P. Blizard, "Canadian Measurements of an Iron-Water Whield" ORNL-428 (Secret)

6. J. L. Powell, "Gamma-ray Attenuation Experiment References" ORNL-422 (Secret)

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Table 1

Shield Compositions *

Shield Material	Run No.	Density gm/cc	Elemental Components - (milligram atoms/cc)													
			Fe	H	B	O	Mg	Cl	Ca	Si	Al	Mn	S	C	W	Total
M.O.	30-31	5.6	88.3	26.3(±15%)	--	21.1	8.69	2.05	--	--	--	0.37	--	--	--	146.8
M.1.	15	4.5	55.7	63(±20%)	3.76	47.3	10.5	3.21	1.25	--	--	0.25	--	--	--	187.0
M.1.	46	4.8	68.5	35.9(±5%)	3.91	32.0	8.5	2.89	1.29	--	--	0.28	--	--	--	153.2
Brookhaven	3-4	4.3	54.7	21.1(±20%)	--	47.7	0.35	--	6.45	2.10	0.79	1.22	0.11	--	--	134.5
W1 Concrete	11	3.6	40.5	20.1(±20%)	3.28	38.8	0.62	--	12.15	3.65	1.36	0.21	0.18	--	--	120.8
WC-B_4C	42	5.0	--	--	62.0	--	--	--	--	--	8.4	--	--	34.2	18.7	123.3
B_4C-WC	42	4.2	--	--	78.6	--	--	--	--	--	7.7	--	--	38.8	14.2	134.3
H_2O	H-R	1.0	--	111.0	--	55.5	--	--	--	--	--	--	--	--	--	166.5
Boron Cement	33-34	2.0	--	79.1(±5%)	7.98	68.2	16.42	--	6.93	2.66	--	--	--	--	--	181.3
Portland Concrete	10-29	2.33	0.13	4.76(±20%)	--	71.8	0.2	--	14.5	1.56	0.44	--	.06	10.8	--	104.3
WC	37	7.9	--	5.6(±10%)	--	2.8	--	--	--	--	10	--	--	36.7	33.3	88.4
B_4C	36-37	1.9	--	7.2(±10%)	117.5	3.6	--	--	--	--	4.9	--	--	33.7	--	166.9
Hanford W	--	4.3	64.3	42.3(±10%)	--	21.7	--	--	--	--	--	.19	--	26.5	--	154.7
Paraffin	43	0.9	--	116.9	--	5.5	--	--	--	--	--	--	--	59.5	--	181.9

*Based on values given in Conf. Mem. Y-El-1 from Rockwell to Blizard, dated 9/2/49

Table 2 [redacted]

Summary of Neutron Attenuation Characteristics for Uniform Shields						
Shield Material	Run No.	Density (gm/cc)	Attenuation Range	Relaxation Length λ (cm)	Relaxation Length $\rho\lambda$ (gm/cm^2)	Reference ORNL Report
*M.O. Concrete	30	5.6	10^7	6.0	34	436
M.O. Concrete	31	5.6	10^6	6.1	34	436
*M.1. Concrete	15	4.5	10^6	5.8	26	438
M.1. Concrete	46	4.7	10^4	5.9	28	438
Brookhaven	3	4.3	10^5	6.3	27	436
Concrete	4	4.3	10^5	6.3	27	436
W.1. Concrete	11	3.6	10^6	6.6	24	209
*WC-B_4C	42	5.0	10^3	8.6	42	435
*B_4C-WC	42	4.2	10^5	7.0	29	435
**Water	H & R	1.0	10^6	8.3	8.3	181
**Boron Cement	34	2.0	10^6	7.6	15.2	436
**Portland	10	2.33	10^8	12.0	28	209
Concrete	29	2.33	10^9	10.8	25	436
Paraffin	43	0.9	10^6	7.0	6.3	NEPA STRM-22

Notes:

* "Asymptotic" relaxation length for an isotropic plane source. If a diffusion kernel is assumed, will be the same for both isotropic and collimated plane sources.

** "Asymptotic" relaxation length for a collimated plane source, with a transport kernel assumed.

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Table 3

Summary of Neutron Attenuation Characteristics

for Nonuniform Shields

Front Slab	Thickness (inches)	Density(gm/cc)	Analyzer	Attenuation Range	Relaxation Length(cm)
Fe + B_2O_3	7.5; 1.0	7.85; 1	MO	18 ± 3	7.5 ± .5
Pb	8	11.3	B.C. Cement	13 ± 3	8 ± 1
WC	10	7.9	B_4C	19 ± 2	8.6 ± 0.5

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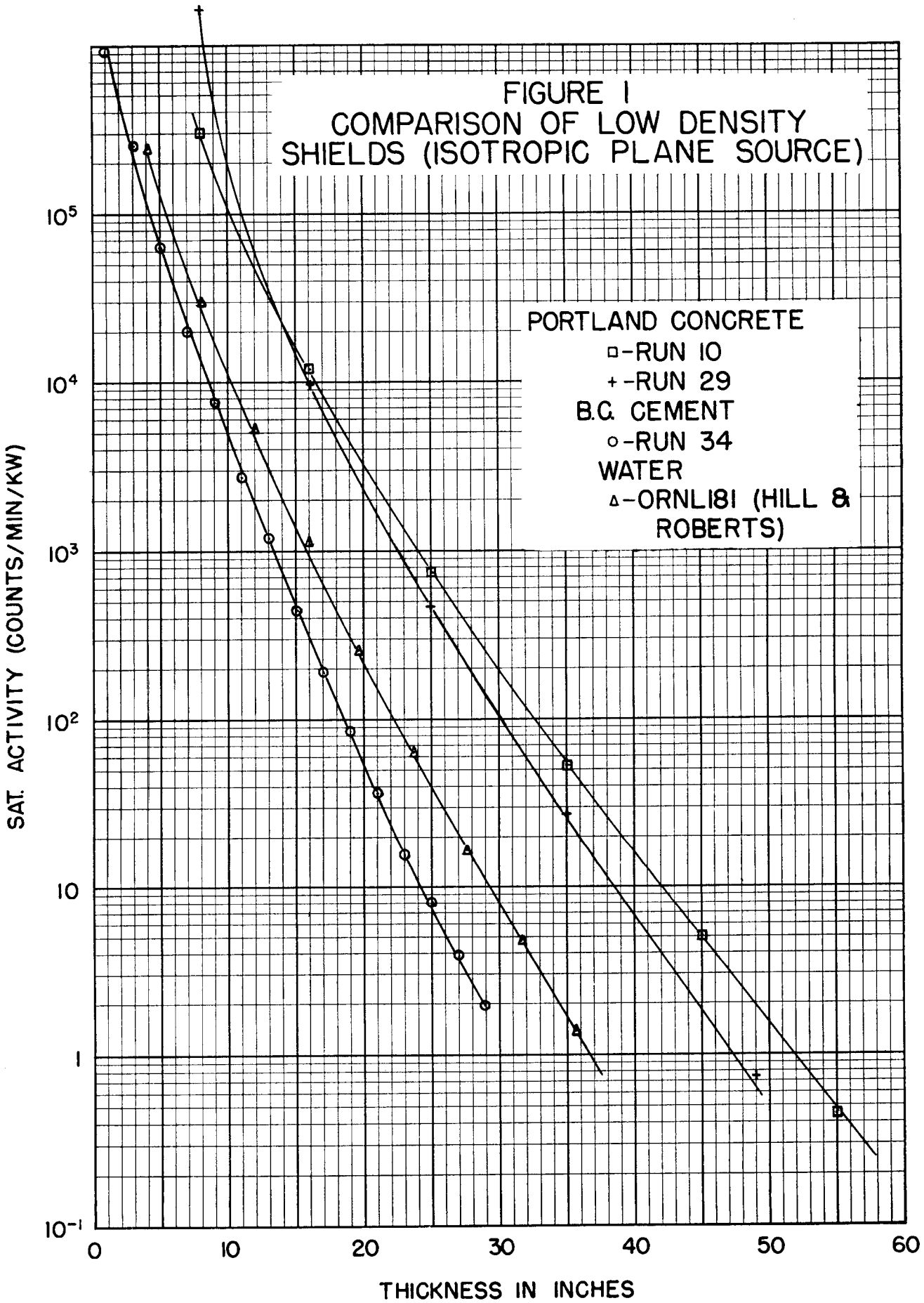

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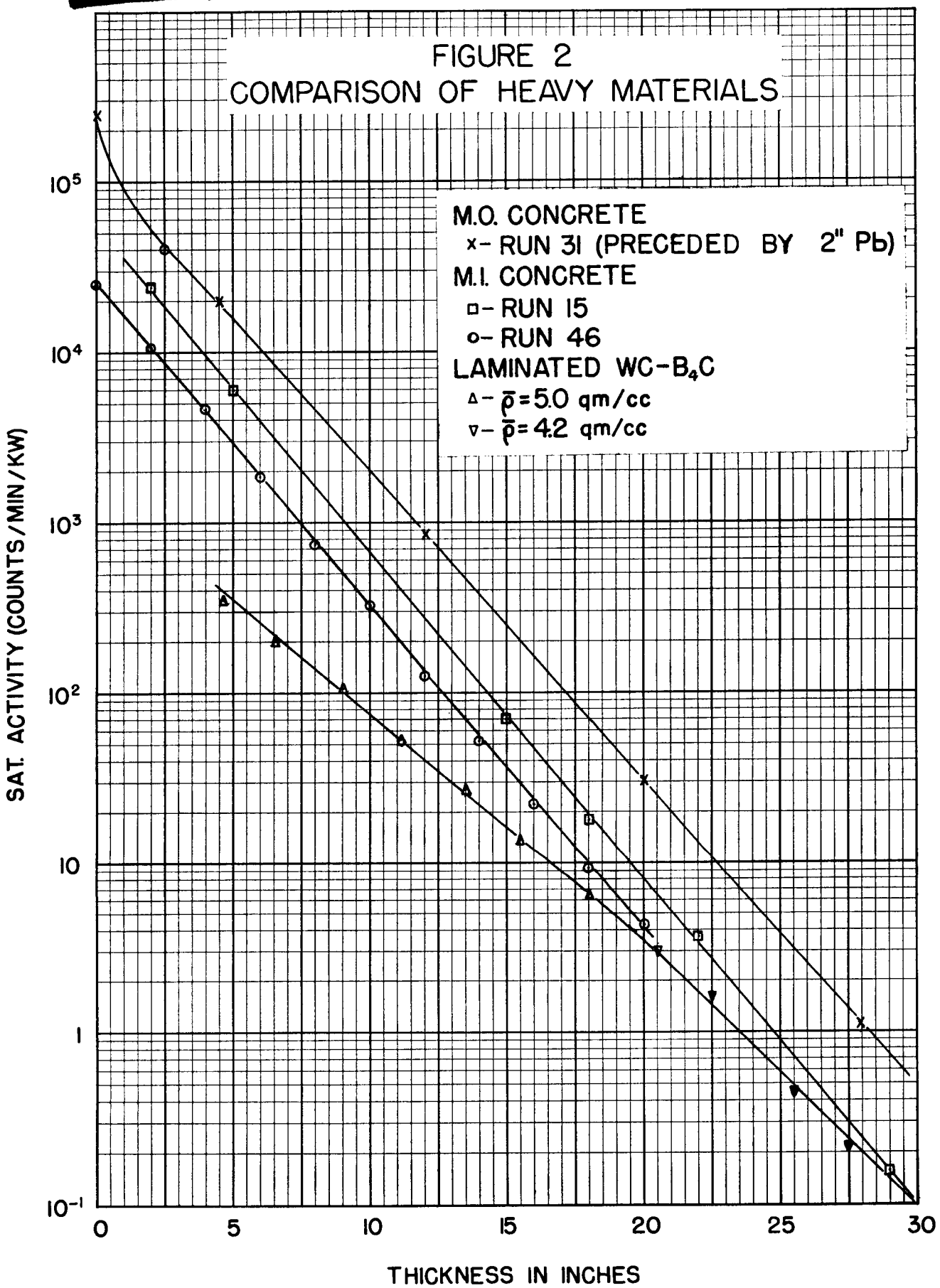

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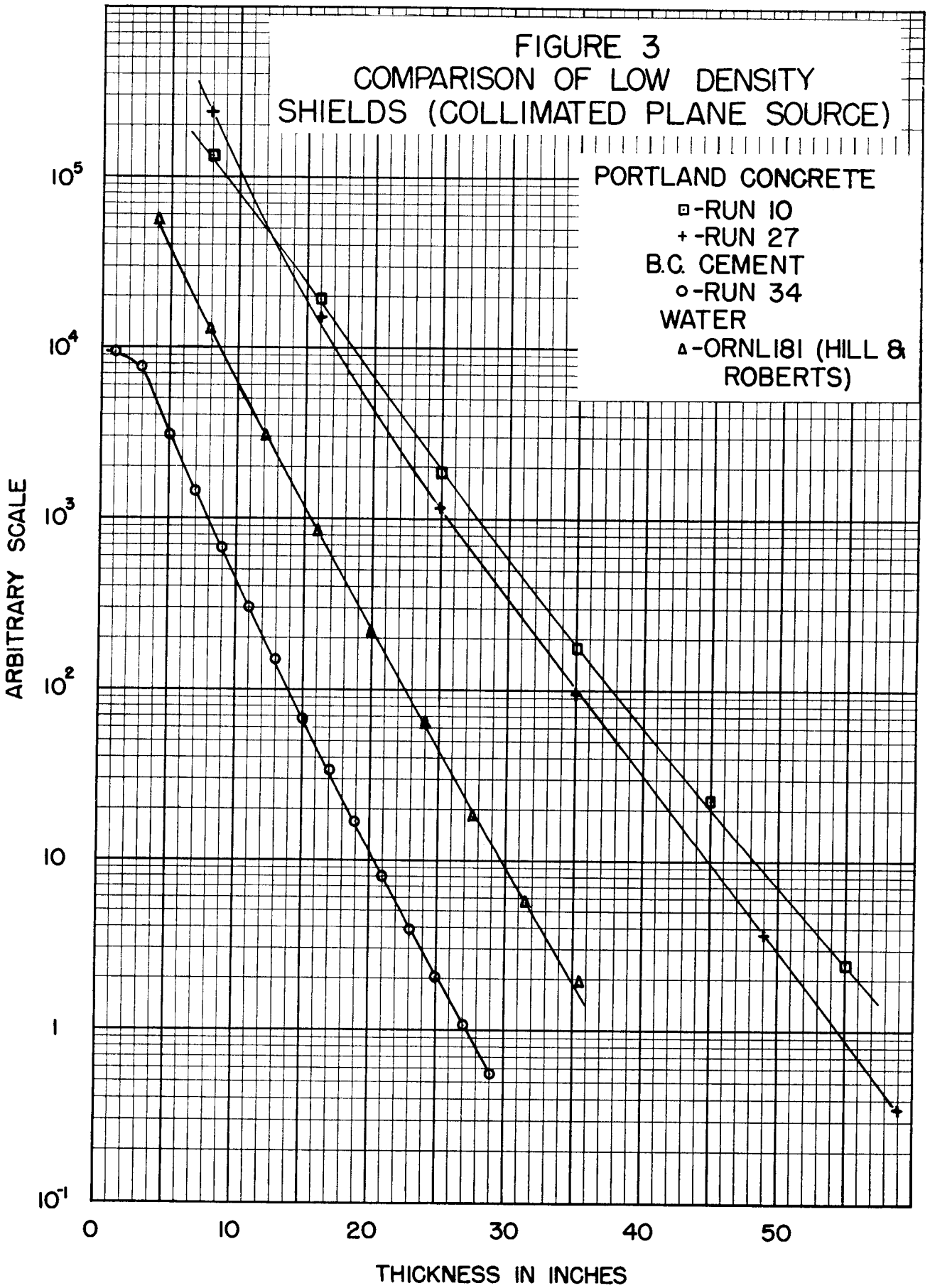

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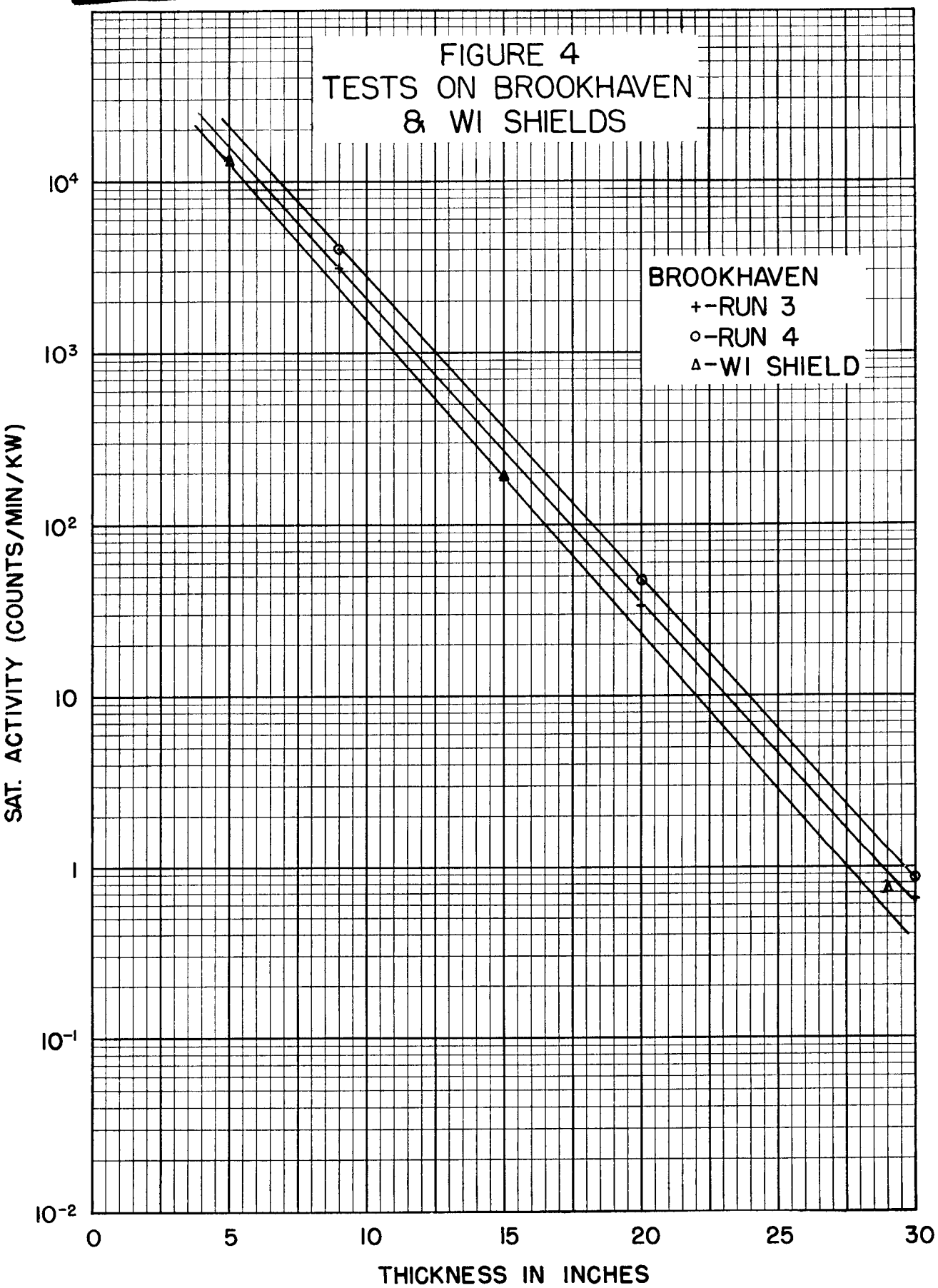

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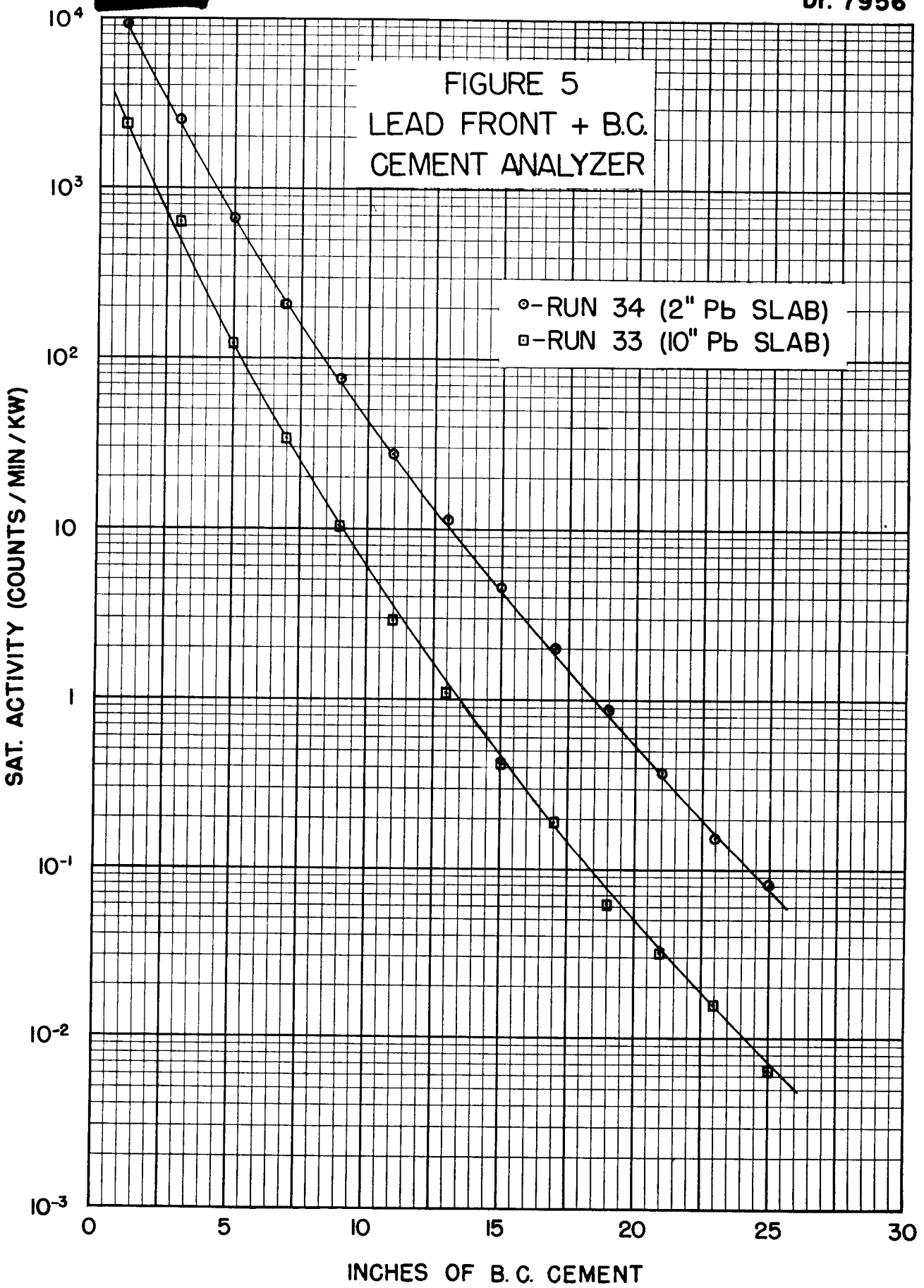

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Section IV

Numerical Methods

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IV. Numerical Methods

Shielding is not unique in having developed a statistical (numerical) phase as a counterpart to the analytical. The main trend in numerical techniques applicable to shielding has been toward the so-called Monte Carlo method. For computational simplicity, the method is altered by a sampling procedure. The one suggested by the workers of the Summer Shielding Session (Goertzel and Kahn) is quota sampling.

The discussion on Monte Carlo is limited here to a description of the ideas contained in references (1), (2), (3) and (4) since, quantitative results are lacking. A further limitation is imposed by the reports proper; that is, they are evolutionary in character and to a large extent given over to development of methodology. Thus, anything short of a complete and extensive treatment would be worthless. Consequently the presentation here is a brief outline of some of the concepts and contents of the above references. After this, a brief description of the matrix method will be given.

To supply a background and definition of the Monte Carlo method, a quotation from reference (1) is presented here.

"The Monte Carlo method is a random sampling process. To apply it to a shielding computation one might propose to sample life histories of particles by playing the following series of games. One considers a particle traversing the shield. First, he decides where the particle makes its collision. This is carried out by playing an appropriate game of chance. He then decides, again with a game of chance, whether the particle is absorbed on the collision or follows some appropriate alternative. The decision is then made,

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with a game of chance where the particle makes its next collision. This process is continued collision after collision, until either the particle escapes the shield or is absorbed. As a result of this series of games of chance one has obtained a life history of a neutron incident on the shield, selected at random from the population of all possible life histories. This process is repeated until the number of particles which have penetrated the shield is large enough to give a good estimate of the probability of penetration.

"The relative probable error of an estimate obtained in this manner is given by

$$.67 \left[(1-p)/N_p\right]^{\frac{1}{2}} \qquad (1)$$

where p is the probability of penetration and N is the number of particles histories, so that Np is the number of particles which have penetrated the shield. In a shield "p" is much less than one so that for a relative probable error of 10% in an estimate of the penetration one must trace enough histories so that almost 100 particles penetrate the shield. Thus one needs 100/p particles which for a "p" of 10^{-6} means handling 10^{8} particles. This is <u>prohibitively</u> large.

"The necessity for the introduction of the probable error as in the last paragraph arises from the statistical nature of the Monte Carlo method." The significance of the probable error and thus the meaning of the type of information obtained by Monte Carlo is now briefly discussed.

A Monte Carlo calculation using quota sampling yields a set of weights for particles, which correspond to a sample of particle histories selected at random from the population of all possible particle histories. The penetration

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probability, p, is the arithmetic mean of the weights of all histories and "m" the arithmetic mean of the sample weights is an estimate thereof. The reliability of this estimate is not known as it depends on the distribution function of the sample arithmetic means. However, it is known that for sufficiently large samples the distribution of sample means is normal.* If now the assumption of normality is made, one estimates the parameters of the normal distribution from the sample by the usual statistical procedures. Thus one uses the sample mean, m, and the probable error, d, estimated from the sample to approximately determine the normal distribution from which the sample was selected.

To put the computations on a practical basis, various modifications to the Monte Carlo technique have been proposed. The one to be discussed here is quota or importance sampling , the latter appears to be the most versatile since it places greatest emphasis on the most important regions in phase space where the importance of specific regions can be based upon the physics of the problem.

The importance of a particle is in the simplest case the probability that it will eventually penetrate the shield and this depends primarily upon the velocity and position of the particle just before a collision. This is the ideal importance of the particle at the particular time in its history and is thus the ideal importance of the corresponding point in phase space. Obviously, this is not known, but an estimate of the importance of the various parts of phase space can be made.

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* Unless the sampling method is "efficient" this is a very dubious assumption.

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This estimated importance function is the one used in the sampling method to modify each game of chance played in such manner that the particles are encouraged to travel into more important regions and to avoid the less important regions. The modification is suggested by the physical situation and is compensated for by giving a weight to each particle in accordance with the modification of the game of chance. Thus, each particle which penetrates the shield represents not one particle but that small fraction of a particle given by its weight.

It is to be noted that these modifications are not approximations that affect the expected value of the final result but rather a good choice of importance function and good technique have the effect of decreasing the number of particles which must be studied in order to obtain a given probable error.

In reference (2), Goertzel continues this exposition and introduces the general formulation of attenuation problems in terms of stochastic processes. The steps required to estimate statistically the necessary integrals are given. However, the method becomes inefficient when the weight assigned to capture at a given point (in phase space) is appreciable for such points that are large distances (geometrical) from the points where zero-th collision particle distribution function is appreciable. This is so since a small number of the points at which particles are absorbed will contribute to the statistical estimate of the capture distribution.

As a result of this, a sampling method is presented where proper weights can be found by quota sampling. Finally, the concept of importance function

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is defined and its relationship to the weight factor is indicated. That is, if $Q(\alpha)$, $Q(\beta)$ are the importance of a neutron at α, β, respectively where these are points of the phase space describing the problem then, the weight $W(\alpha, \beta)$ is defined by

$$W(\alpha, \beta) = Q(\alpha) / Q(\beta) \qquad (2)$$

It is to be noted that the ratio in equation (2) is only for the ideal case. In actual practice it is simply $f(x)/f^*(x)$ where $f(x)$ is the physical problem density function and $f^*(x)$ is the actual density used.

Reference (3) is a continuation of (2). Here the ideas are further discussed and most important a step-wise method is given. It is to be noted that the procedure is sufficiently detailed for one to perform a calculation independently. Further, the manner of computing the importance function for different types of collisions -- inelastic, elastic, and absorption are given. It should be noted that the importance function is defined by the exponential diffusion kernel (where exponential diffusion is a reasonably accurate solution of the problem).

In reference (4) Nelson describes a problem presently being solved by the Monte Carlo method. A plane monoenergetic source of 10 Mev neutrons is embedded in an infinite body of water and by tracing the life histories of 1200 neutrons an estimate is to be made of:

(1) efficiency of various techniques in sampling,

(2) the attenuation of several semi-infinite slabs with varying distributions of direction of source neutrons.

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Some inherent features of the problem are: (1) in considering the scattering process, greatest weight is placed upon those neutrons scattered forward; (2) when a neutron's energy falls below .014 Mev it is considered as captured; (3) although inelastic scattering is considered as an absorption process the neutron is retained but proper weight is used so that distribution of collision is corrected, and (4) after a neutron has made a certain number of collisions (5 to 10 say) it is subjected to a game of chance in which it has equal probabilities of survival and death. The surviving neutron's weight is doubled subsequent to its next collision, and the process is repeated, until all the neutrons are extinct.

We will now conclude this section with a brief statement regarding the matrix method.

A matrix method was devised by DeMarcus and Nelson (4) for solving problems related to the transmission of particles through matter.

The phase space is subdivided into a finite number of cells. To these are adjoined a few fictitious cells or "traps." Once in one of these traps, a particle remains in it and to some trap each particle ultimately finds its way. For example to the outer geometric face of the shield is adjoined a layer of traps. Similarly, at the "bottom" of the energy scale are found such traps.

These cells, real or fictitious are numbered i = 1, 2, 3, ---n. With each pair of cells is associated a number a_{ij} which is the probability that a particle in cell j will be in cell i directly after it has made one collision.

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The m x n matrix whose elements consist of these a_{ij}, is called A, the collision operator. For if the component V_i, i = 1, 2, 3 ...n, of the column vector V is the number of particles initially present in cell i, then

$$V^k = A^k V \tag{3}$$

gives the distribution after k collisions. The paper shows the existence (under conditions satisfied by shielding problems) of the limit of A^k as k becomes infinite. Moreover, it shows that the elements of the limit matrix can be computed by solving a set of linear, non-homogenous algebraic equations. Thus if

$$\lim_{k \to \infty} A^k = A^* \tag{4}$$

we have the steady state solution of the problem in $V^* = A^* V$ where V is the distribution of the incident particles and V^* is a record of their final disposition.

The difficulty of using this method is clearly the large number of cells which must be used and consequently the great size of the set of algebraic equations which must be subdivided into a finite number of cells.

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2. G. Goertzel, "Quota Sampling and Importance Functions in Stochastic Solution of Particle Problems" ORNL-434, December 1949 (Confidential)

3. G. Goertzel, "A Proposed Particle Attenuation Problem" ORNL-423 December 1949

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5. W. C. DeMarcus & M. L. Nelson, "Methods of Probabilities in Chains Applied to Particle Transmission Through Matter" NEPA report, June 1949.

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