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EFFECT OF WATER SOAKING ON SHIELDING PROPERTIES OF THE OAK RIDGE PILE SHIELD

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and
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PHYSICS DIVISION

EFFECT OF WATER SOAKING ON SHIELDING PROPERTIES OF THE OAK RIDGE PILE SHIELD

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Lt. S. W. W. Shor, U.S.N., and J. M. Cassidy

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A B S T R A C T

This experiment was originally designed to measure the intensity of neutron and gamma ray fluxes through the Oak Ridge pile shield in its present state. However, water used as a seal against streaming of neutrons soaked into the concrete of the shield. This resulted in a reduction of the effective neutron relaxation length by about 20%, as shown in Fig. 2. Inasmuch as there was already a large quantity of oxygen in the shielding material, the effect can probably be ascribed to an increase by a factor of 1.8 in hydrogen content.

The relaxation length for gamma rays was found to be 13.9 centimeters, (See Fig. 3) although the gamma rays produced in the shield by neutron scattering and capture cannot be distinguished from the primary gamma rays from the pile.

A number of auxiliary results are given, including information on size of experimental holes required to obtain data which are essentially those of the surrounding shielding material rather than of material in the hole, and discussion of the use of aluminum foils to detect neutrons above about 4.6 Mev.

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Description of Experiment

A uniform cross section 4-5/8" diameter hole was drilled with a core drill horizontally through the concrete of the shield at a point 3'4" to the left of and 1'3-5/8" above the center of the west side of the pile on July 27, 1948, as shown in Fig. 1. The hole was plugged at its inner end with a steel and rubber plug similar to an ordinary plumber's pipe plug, the inner end of the plug being 12" and the outer end 18" from the inner face of the shield. The hole was then filled with water and arrangements made to push into it cylinders of ordinary concrete 4-7/16" in diameter and 1' long, with foils attached to their forward ends. The concrete cylinders served to make the material in the hole as nearly like the surrounding shield as possible. They were cast of one part concrete, two parts sand and four parts gravel, which experience of the Oak Ridge shielding group has shown to be similar in its shielding properties to the Oak Ridge shield. The purpose of the water was to act as a seal to prevent streaming of neutrons past the concrete cylinders. In addition, a cypress stick 4-3/8" in diameter in the form of a half-cylinder was employed as a foil holder in order to determine the effect of the material in the hole on the results. The forward end of each concrete cylinder contained a recess 3"×2"×$\frac{1}{4}$" deep to act as a foil holder. The outer end of each cylinder—the end placed toward the outer face of the shield—contained a threaded steel pipe cap into which a rod could be screwed to pull the cylinder out of the hole. A sheet-metal trough 7'10" long, 12" deep and 8" wide was attached to the shield outside of the hole, its bottom 1" below the bottom of the hole, and this was kept filled to a depth of 10" by a continuous supply of demineralized water. No contamination of the water or excessive fluxes in the working area were experienced.

Description of the Shield

The Oak Ridge shield consists of one foot of ordinary concrete backed by five feet of barytes haydite concrete, followed by another foot of ordinary concrete. (See Fig. 1 and Appendix A) The layers are separated by a 1/8" layer of bituminous plastic. The inner face of the shield is 6 feet from the edge of the graphite pile cube, the intervening space being filled with air except for a 2" layer of cellamite placed 4" from the shield. Measurements in this test were made through the innermost six feet of the shield, the last foot having neutron and gamma fluxes which were lower than those outside of the shield in the working area.

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Evaluation of Experimental Data

1. It was impossible to place the plug as far into the shield as desired because water leaks occurred into the pile whenever the inner end of the plug was placed less than 12" from the inner face of the shield. Since the plug could be very thoroughly tigntened, these leaks were probably either through cracks in the ordinary concrete or through the bituminous layer separating the ordinary concrete from the barytes haydite concrete. The loss of measuring distance through the shield was not particularly serious. On the other hand the boundary conditions were considerably poorer than they would have been had the plug been at the extreme end of the hole.

2. The measurements of neutron flux with the concrete slugs, with wood, and simply with water in the hole gave very nearly the same results, indicating that the hole was of sufficiently small diameter so that the measurements were essentially of the attenuation in the concrete of the shield, not of the material in the hole. (See Figures 4 and 5) The measurements of gamma radiation were considerably more sensitive to the material in the hole than were those of neutrons, but even with gamma rays the intensity 3'9" from the inner face of the shield was only 2.5 times as great with just water in the hole as it was with concrete cylinders. (See Fig. 6) Gamma measurements were also taken with films, but these were erratic and apparently affected by neutron induced beta activity in the steel pipe caps in the cylinders, so that their steepness of attenuation approximated that of the neutron curves. They were discarded as not significant. It was found possible to take neutron traverses across the hole, and since these showed that the flux at the edge of the hole was about the same as that at the center, they provided additional evidence that the measurements were those of attenuation in the surrounding material. (See Fig. 7)

3. Although the experiment was originally designed to measure the neutron attenuation through the dry concrete of the Oak Ridge shield, it developed that the concrete was very absorbent and soaked up water for a considerable distance around the hole. Water appeared issuing from cracks in the shield ten or more feet from the hole on the level below. The water soaking materially changed the neutron attenuation of the shield, decreasing the relaxation length by about 20%. (See Fig. 2) Only one measurement was taken in actual dry concrete, and it was taken when the hole had been drilled part way through. Unfortunately, this measurement cannot be precisely evaluated

[redacted]

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because a 3/8" wide annular groove produced by the core drill extended beyond the location of the foil some $7\frac{1}{2}$" farther into the concrete. It is probable that this measurement corresponds to the flux at some point between the actual location of the foil and the point of farthest advance of the drill, and it has been so located on Fig. 8. The precise location of the "best point" was made by reference to an experiment performed with somewhat similar boundary conditions in hole #43 in the shield, in which streaming along a thin annular hole around a foil-carrying stick was of such importance that the flux increased only 50% with an advance of 6" into the shield*.

4. Although the measurements at the inner end of the hole are undoubtedly distorted by the presence of a foot of air in the hole, the measurements farther out are not likely to have been seriously affected by the poor boundary conditions at the inner end. In any case estimates, based on these measurements, of the total thickness of shield required to bring the neutron and gamma fluxes down to tolerance will be conservative.

5. Since the presence of the plug made impossible the direct determination of the neutron and gamma-ray flux at the inside end of the hole, it was necessary to obtain estimates of their value. Good measurements of both these quantities have been made at the center of the west face of the shield on its inner face[1], a point 3'7" from the center of the present hole. These values have been reduced to the inner end of the present hole by use of the formula

$$\phi = \text{constant} \times \cos \frac{\pi(r + .338)}{23.83}$$

determined empirically for the thermal flux across the face of the pile by Haydn Jones et al[2]. This gives a value for the present hole of .87 times the flux at the center, and since there is probably some diffusion of gamma rays and neutrons as they cross the air gap to the shield from a sinusoidal to a somewhat flatter distribution, the true value is probably somewhere between .87 and 1.00. Inasmuch as use of .87 will give a value of the fluxes at the inner face of the shield, which are either correct or very slightly low, estimates of shield thicknesses based upon attenuation from these values will

---

*Experiment (unpublished) performed by James D. Knox, July 30, 1946, in hole 43. Record in files of Operating Division, ORNL. Hole 43 is 4 ft. 8 in. from center of west face of pile.

[1]Gamma measurement: ORNL-51, Physics Division Report for the Quarter, March, April, May, 1948, p. 22. Neutron measurement: Unpublished data taken by Oak Ridge shielding group.

[2]Haydn Jones et al., CP 2602, p. 5.

[redacted]

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again be conservative. These data indicate an indium resonance neutron flux of 4.5×10^7 n/cm^2/sec., and a gamma flux of about 9000 R/hr.

6. The experimental hole was located not far from the large core hole at the center of the west face of the shield, and in order to minimize its effects the core hole was filled with blocks of ordinary concrete of the same one-two-four composition as the cylinders placed in the experimental hole. On the last day of the experiment this concrete was replaced by a block of MO concrete with a strong fission source at its inner end, of such intensity that the neutron flux throughout the MO was increased approximately tenfold. The MO concrete, whose shielding properties are considerably better than those of the Oak Ridge shield, extended only 48 inches, or a little over half way, into the shield, so that the large hole contained only air in the inner portion of its length. The effect of this change in boundary conditions is shown in Figures 9 and 10.

7. The most significant data obtained is undoubtedly the attenuation of epithermal neutrons as measured by cadmium covered indium foils. This is not subject to large perturbations due to the presence of small amounts of material with large thermal capture cross section, and so is much more representative of the actual value of the material as a neutron shield than is the bare indium activity. In this case the bare indium activity and the cadmium covered indium activity produced nearly identical curves, with an average cadmium ratio at the center of the hole of 10, so that measurements of bare indium activity gave approximately the same attenuation over most of the range; this can be described simply as a fortunate accident due to the essential homogeneity of the concrete.

By-Products of the Experiment

1. Measurement of gamma-ray attenuation through the open hole three hours after shut-down of the pile was made with an ion chamber. The results are shown in Fig. 11.

2. Attempts were made to measure the attenuation of high-energy neutrons in the shield by exposure of cadmium covered aluminum foils. Although it was possible to activate the 10.2 minute half-life and obtain results which were good statistically at a distance of $1\frac{1}{2}$' from the inner face of the shield, the results at a point $2\frac{1}{2}$' back were barely readable. These two points gave a relaxation length of 11.3 cm. which may or may not be significant. It is

[redacted]

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evident that readings may be obtained for this half-life through at least the first reduction in flux of 10^3, using an initial flux equal to that at the inner face of the X-shield. The reaction producing the 10.2 minute half-life in aluminum has a threshold of at least 2.5 mev, and with a fission distribution of energies corresponds to a mean energy of 4.6 mev[1]. It may be significant in determining the limiting relaxation length for neutrons at great distances from the source, since initially high energy neutrons undoubtedly predominate among those which survive to a great distance. Exposures of boron covered indium foils were also made in order to activate the 4.1 hour half-life in indium 115, and these gave two points (the second also with poor statistics) which indicated a relaxation length of 13.0 cm. for neutrons at some energy above 0.5 mev. Since the fission spectrum is not as accurately known at energies below 2 Mev as it is above, and since in a shield there is likely to be scattering into the energy region probably measured by this reaction as well as out of it, attenuation measurements made by this method are not likely to be as significant as those made with the aluminum foils. The range of measurement with 4.1 hr. indium in samples tested in the Oak Ridge pile will be about the same as with the aluminum foils, about 10^3 down from the initial flux. A rough estimate of the absolute flux of neutrons with energy above 4.6 Mev. at the point 1.5 ft. from the inner face of the shield has been made by two methods described in Appendix B. Method (a) gave 8.40×10^5 n/cm^2/sec, method (b) 1.24×10^6 n/cm^2/sec.

Recommendations

In order to obtain accurate data on the shielding characteristics of the Oak Ridge shield, it is suggested that a hole of very small diameter, perhaps $1\frac{1}{2}$", be drilled through from the outside with a drill of a type which does not produce a core and does not use water in its operation, and that measurements be taken with cadmium covered indium foils, films, and an ion chamber as the drilling progresses. In this way the errors caused by streaming down an annular groove, by poor boundary conditions, and by extraneous materials introduced in the hole should be eliminated.

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[1] Preliminary Collection of Data on (n,p) and (n,α) Cross-Sections, compiled by G. Haines (unpublished), and
CF-3574, N. Goldstein et al., Cross-Sections for Fission Neutrons, Threshold Reaction Cross-Sections, pp. 28-32.

[redacted]

[redacted]

APPENDIX A

Composition of the X Shield

The following description of the X shield is taken from memo C.F. #48-1-118, by T. Rockwell, dated December 23, 1947:

> "The Clinton shield consists of two feet of ordinary concrete, one on each side of five feet of special concrete. The 'X' identifies the latter.
>
> "X concrete is Portland cement, plus barytes for density and heavy metal content, plus Haydite for absorbed water. It is covered with a bituminous coat to prevent water evaporation, but there is some evidence that the coat may have cracked.
>
> Composition:

Cement (Portland, low heat)	16.3% by weight
Haydite	27.3
Barytes	46.4
H_2O	10.0
	100.0

> "As poured, the water was specified 12.4%. 10% seems to be a fair consensus for present content. If the water were only that bound to the cement, it would be only 3.3%."

Mr. V. L. DiRito made an analysis of the haydite section of the core sample from the hole in which the experiment was performed. His results showed the following average composition: (Iron content is higher than expected, probably because of iron rubbing into the sample from steel balls in the ball mill machine.)

	% by wt.	g/cm^3	mg-moles/cm^3
Fe	1.71	.0376	.674
H	.91	.0200	20.0
Ba	31.63	.696	5.08
Cu	7.00	.154	3.84
Al	3.53	.0777	2.88
Si	8.05	.177	6.30
S	7.76	.171	5.33
O	39.42	.867	54.2

Density = 2.2 g/cm^3

The average water content of the sample from the experimental hole was 6.3% by weight. Samples of dry haydite concrete were soaked in water to determine increase in water content. The average percentage by weight increased from 6.3% to 10.4% in 11.5 days, and leveled off to 10.9% after 38 days soaking. The hydrogen content increased in 11.5 days from .015 g/cm^3 to .027 g/cm^3.

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APPENDIX B

Measurements with Aluminum Threshold Detector

(1) The use of an aluminum threshold detector is based upon the reaction

$$Al^{27}\ (n, p)\ Mg^{27},$$

which has a threshold of 3.0±.2 Mev (Thesis of J. E. Hill) or 2.5 Mev (Gale Young). The cross section increases to a nearly constant value above 5 Mev, and decreases at some higher value. If we can assume a fission spectrum which decreases exponentially from above 2 Mev in accordance with the equation

$$(nv)_E = (nv)_{E_o}\ e^{-\frac{(E-E_o)}{1.6}}$$

(It seems to be very well established for fission neutrons, but is not necessarily correct for the neutrons incident upon the shield.) we may proceed to compute the flux above 4.6 Mev from aluminum threshold measurements.

(2) We shall assume such a spectrum.
σ_E is a function which looks like the figure below:

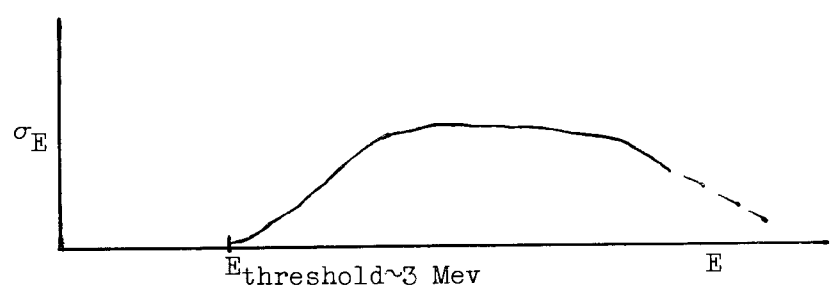

Fig. (a)

The rate of captures in the aluminum will be

$$\frac{dI}{dt} = N \int_{E_{threshold}}^{\infty} (nv)_E \sigma_E dE \quad .$$

[redacted]

[redacted]

The product $(nv)_E \sigma_E$ can be plotted as follows:

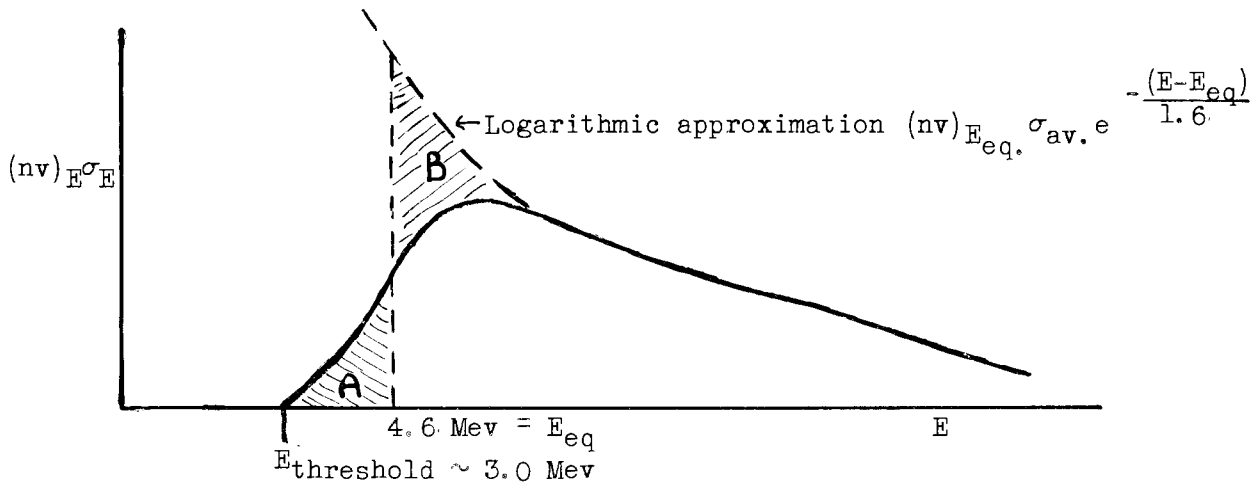

Fig. (b)

If, as shown in Fig. (a), the cross section approaches a constant value, the product σ nv will approach a logarithmic curve for values of E above 5 Mev. We may then make the substitution

$$\int_{E_{threshold}}^{\infty} \sigma_E \, (nv)_E \, dE = \sigma_{av.} \, (nv)_{E_{eq.}} \int_{E_{eq.}}^{\infty} e^{-\frac{E-E_{eq.}}{1.6}} \, dE$$

where $E_{equivalent}$ is such that on Fig. (b) area A = area B. If we make this substitution, then

$$\int_{E_{threshold}}^{\infty} \sigma_E \, (nv)_E \, dE = 1.6 \, \sigma_{av.} \, (nv)_{E_{eq.}} \quad ,$$

and we may note that the total flux above $E_{eq.}$ is equal to 1.6 times the specific flux per unit energy interval at $E_{eq.}$. For Al^{27} (n, p) Mg^{27} this has been worked out, and a value of 4.6 Mev obtained for $E_{equivalent}$, and a value of 2.8 millibarns for σ_{av}[1].

(3) With a knowledge of $\sigma_{av.}$ and $E_{eq.}$, we only need calibrate our foils so that a known counting rate corresponds to a known activity. This is

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[1] CF-3574 by N. Goldstein et al., pp. 28-32. A value of 6 mb was obtained from the data of L. Seren and W. Sturm in CP-455.

[redacted]

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possible with a counter of unknown efficiency by comparison either with

(a) the counting rate produced by a similar foil of magnesium activated in a known thermal flux, or

(b) with some possible error, by comparison with the counting rate produced by the same aluminum foil after activation of the 2.3 minute activity in a known thermal flux.

Method (a) is undoubtedly more accurate, since the thermal neutron reaction Mg^{26} (n, γ) Mg^{27} produces the same activity as Al^{27} (n, p) Mg^{27}.

Magnesium has an atomic weight of 24.32 and is 11.4% Mg^{26}. The atomic weight of aluminum is 26.97, and the element is 100% Al^{27}. Hence the two materials have a ratio of

$$\frac{\text{Atoms } Mg^{26}/\text{gram}}{\text{Atoms } Al^{27}/\text{gram}} = \frac{\left(\frac{.114}{24.32}\right)}{\left(\frac{1.0}{26.97}\right)} = .126$$

For the same nv, the activity ratio will be proportional to the ratio of total cross sections. We will compute the ratio of activity per unit weight, since aluminum and magnesium have the same absorption coefficient for β-particles, and slabs of the same thickness in gm/cm^2 (or of infinite thickness) will exhibit the same self-absorption. We will take the mean cross section of the aluminum to be 2.8 mb for Al^{27} (n, p) Mg^{27}, and the thermal cross section of magnesium[26] for Mg^{26} (n, γ) Mg^{27} to be 48 ± 10 mb[1]. Then we find that the ratio of counting rates for the same flux in nv will be

$$\frac{\text{Counting rate of Mg}}{\text{Counting rate of Al}} = \frac{\text{Atoms } Mg^{26}/\text{gm} \times \sigma_{Mg^{26}}}{\text{Atoms } Al^{27}/\text{gm} \times \sigma_{Al^{27}}} = \frac{.126 \times 48}{2.8}$$

$$= 2.16 \equiv C_{nv}.$$

We now find $(nv)_{th}/(A_s)_{Mg}$ for thermal neutrons by exposing the magnesium foil in a known thermal flux and then counting it. Then the ratio

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[1] CF-2376, Slow Neutron Activation Cross Sections by L. Seren et al., Table III.

[redacted]

[redacted]

$$\frac{(nv)_{E_{eq.}}}{(A_s)_{Al}} = C_{nv}\frac{(nv)_{th}}{(A_s)_{Mg}} = \frac{2.16 \times 12149}{31} = 845 \frac{\text{neutrons/cm}^2\text{/sec}}{\text{counts/min}}$$

We use this ratio thereafter to determine $(nv)_{E_{eq.}}$ from observed saturated activity of the aluminum 10.2 minute half-life.

In order to make the above comparison, the foils used must be either thick relative to the relaxation length for β-particles, or of the same weight in gm/cm^2. The relaxation length for β-particles of this activity in Al or Mg (as will be shown below) is between 0.066 and 0.100 gm/cm^2. Hence a foil of 1 gm/cm^2 will be virtually infinite in extent. (This is thicker than a foil, and is equivalent to the 3/16" slabs of aluminum which we used; they are 1.29 gm/cm^2.)

By this method the flux of neutrons of energy above 4.6 Mev at a point 1.5 ft. from the inner face is equal to 845 × 994 counts/min = 8.40×10^5 n/cm^2/sec.

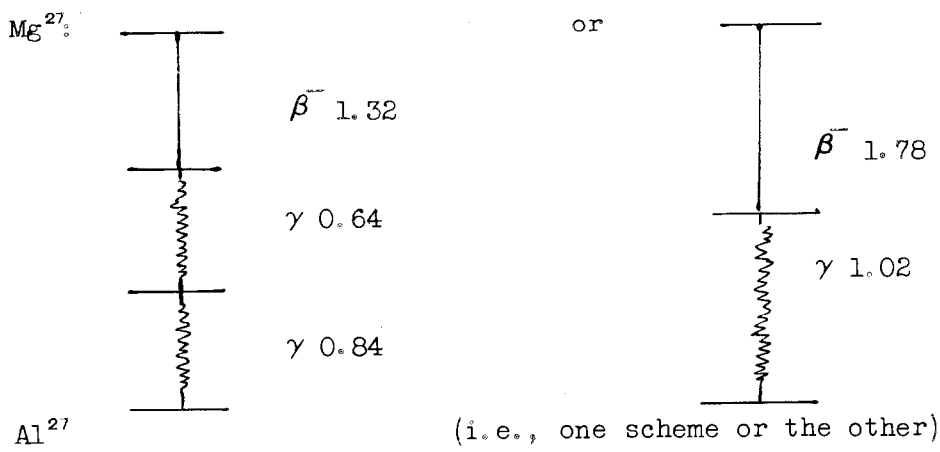
Method (b) is correct only if we can assume

(1) The Geiger-Mueller mica window counter counts nearly all β-particles, with very low efficiency for gammas.

(2) The efficiency of the counter for β-particles is constant over a wide range of energies.

(3) We know the energy of the β-particles produced in decay of both Al^{28} and Mg^{27}.

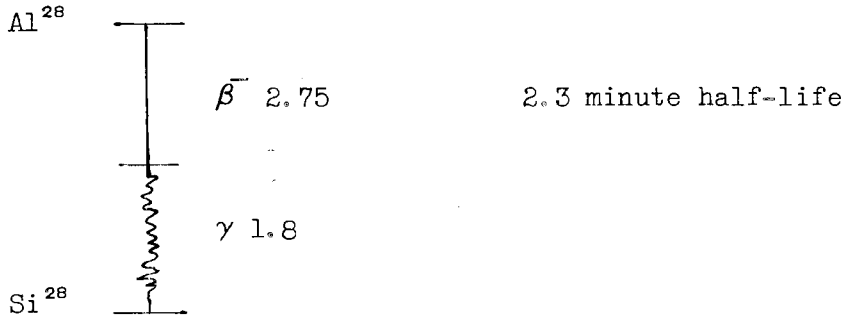
We must accept (1) and (2) at present; (3) can be found from Nucleonics for May, 1948, Part 2, p. 107. It shows the following decay schemes:



(i.e., one scheme or the other)

[redacted]

[redacted]



We do not know the ratio of frequency of the two decay schemes for Mg^{27}, so will assume that they are about such that the relaxation length of the electrons is the mean of the two; we can find the relaxation length for continuous β-ray spectra from the formula[1]

$$\lambda = \frac{.032}{.693} E_{max}^{1.33} .$$

This gives us values of λ of

E_{max}	gm/cm^2	
1.32	.066	mean value 0.083 gm/cm^2.
1.78	.100	
2.75	.180	

If we have equal activities (i.e., particles emitted per second) then the counting rates at the surface of an infinitely thick slab (several relaxation lengths are thick enough) will be proportional to the relaxation lengths for β-particles. This ratio is .180/.083 = 2.15 where the shorter half-life will give the stronger radiation. The number of particles decaying will be proportional to the capture cross section, so

$$\frac{(A_s)_{th}}{\sigma_{th}(nv)_{th}} = \frac{(A_s)_{fast}}{\sigma_{fast}(nv)_{fast}} \times 2.15$$

As a numerical example, in slot #1 of the standard pile[2] the thermal flux is 12149 neutrons/cm^2/sec.; the thermal cross section of Al^{27}

[1] Clark Goodman, ed., The Science and Engineering of Nuclear Power, p. 53.

[2] See CP-2804, Thermal Neutron Flux Distribution in a Standard Graphite Pile, by T. Arnette and H. Jones.

[redacted]

[redacted]

is 230 millibarns, and from slot #1 we obtained a saturated activity of 2760 counts/min. We assumed $\sigma_{(E_{eq.})_{av.}} = 2.8$ millibarns. We wish to find $(nv)_{fast}$ corresponding to $(A_s)_{fast} = 994$ counts/min.

$$(nv)_{fast} = \frac{(A_s)_{fast} \times \sigma_{th}}{(A_s)_{thermal} \times \sigma_{fast}} \times 2.15 \times (nv)_{thermal}$$

$$= \frac{994 \times 230}{2760 \times 2.8} \times 2.15 \times 12149 = 7.72 \times 10^5$$

The value of $(A_s)_{fast}$ was obtained at a point 1.5 feet from the inner face of the shield, so at this point we may say that

$$(nv)_{E \doteq 4.6 \text{ Mev}} = 7.72 \times 10^5 \ \frac{\text{neutrons}}{\text{cm}^2 \times \text{sec} \times \text{Mev interval}}$$

If we multiply this number by 1.6 we find the total flux of neutrons above 4.6 Mev:

$$(nv)_{E > 4.6 \text{ Mev}} = 7.72 \times 10^5 \times 1.6 = 1.24 \times 10^6 \ \frac{\text{neutrons}}{\text{cm}^2 \times \text{sec.}}$$

(4) In order to assure that the observed 10.2 minute activity in aluminum is not due to thermal capture in Mg^{26} impurity, the foil was exposed in the thermal column, which produces nearly pure thermal neutrons. The ratio of 10.2 minute to 2.3 minute activity was determined to be << .001, showing the magnesium impurity to be of negligible importance.

[redacted]

[redacted]

APPENDIX C

Calculation of $(nv)_{\text{Indium resonance}}$

To find $(nv)_{\text{Indium resonance}}$ from observations with cadmium covered indium foils, we must calibrate a cadmium covered foil in the standard pile. In the report describing the standard pile[1] it is possible to find the value of slowing-down density q for any position in the pile. Then we may obtain $(nv)_{\text{In res.}}$ from the formula

$$(nv)_{\text{In res.}} = \frac{q_{\text{In res.}}}{\xi \sigma_s N_s},$$

where ξ is the mean logarithmic energy decrement and $\sigma_s N_s$ the scattering cross section of the pile material. In our standard pile

$\xi = 0.0811$, $\sigma_s N_s = 0.76\ \text{cm}^{-1}$, and in slot #1, $q_{\text{In res.}} = 39.74$.

Then, in slot #1 $(nv)\text{In res.} = q/(.0811 \times .76) = 644.8$.

Now in slot #1 we obtained an A_s of 10,570 counts/min.

$$\frac{(nv)_{\text{In res.}}}{(A_s)_{\text{In res.}}} = \frac{644.8}{10,570} = .0610 \text{ for C counter, mica window,}$$

position #1, using the present tube. Hence to find $(nv)_{\text{In res.}}$ from $(A_s)_{\text{In res.}}$ for cadmium covered (.040" Cd) 25 cm^2 indium foils, $(A_s)_{\text{In res.}}$ is multiplied by .0610.

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[1] CP-2804, T. Arnette and H. Jones.

[redacted]

[redacted]

TABLE I

Thermal Neutron Measurements

Bare Indium Foils on Concrete Slugs

Run	Days of Soaking	Foil Position Feet from Inner Edge of Pile Shield	Saturated Activity 3800 KW
H	2.8 Days	1.5	5.10×10^{7}
		2.5	Not counted
		3.5	2.70×10^{4}
		4.5	741
I	3.5 Days	3.5	2.05×10^{4}
		4.5	936
		5.5	58.6
		6.5	10.3
P	6.75 Days	1.5	4.01×10^{7}
		2.5	3.24×10^{5}
		3.5	7.01×10^{3}
		4.5	303
		5.5	32.5

[redacted]

[redacted]

TABLE II

Epithermal Neutron Measurements

Cadmium Covered Indium Foils on Concrete Slugs

Run	Days of Soaking	Foil Position Feet from Inner Edge of Pile Shield	Saturated Activity 3800 KW
C	0.75 Days	1.5	2.41×10^{6}
		2.5	1.08×10^{5}
		3.5	3.83×10^{3}
		4.5	199
		5.5	23.6±4.0
F	2.5 Days	1.5	1.82×10^{6}
		2.5	5.31×10^{4}
		3.5	1817
		4.5	76.2
		5.5	6.9±1.5
J	3.7 Days	1.5	1.74×10^{6}
		2.5	4.68×10^{4}
		3.5	1.53×10^{3}
		4.5	67.5
W	11.5 Days	1.5	1.67×10^{6}
		2.5	2.98×10^{4}
		3.5	847
		4.5	42.2
		5.5	5.6
X	12.5 Days	1.5	1.68×10^{6}
		2.5	3.61×10^{4}
	Fission Source in Core Hole	3.5	2.50×10^{3}
		4.5	253
		5.5	20.7

To obtain (nv) Indium Resonance, multiply saturated activity by 0.0610.

[redacted]

[redacted]

TABLE III

Epithermal Neutron Measurements

Cadmium Covered Indium Foils on Wooden Rod

Run	Days of Soaking	Foil Distance From Inner Edge of Pile Shield	Saturated Activity 3800 KW
D	0.75 Days	18"	2.03×10^{6}
		29.5"	7.90×10^{4}
		41.4"	3.29×10^{3}
		53.1"	168
		65.0"	2.89
K	3.75 Days	1.5'	1.50×10^{6}
		2.5'	3.63×10^{4}
		3.5'	1.38×10^{3}
		4.5'	67.3

[redacted]

[redacted]

TABLE IV

Neutron Traverses Across Hole with
Bare and Cadmium Covered Indium Foils (1 cm^2)

Run	Days of Soaking	Type Meas.	Feet from Inner Edge of Pile Shield	Foil Position	Saturated Activity 3800 KW
E	1.7 Days	Wooden Rod Cd. Covered	3	Edge	6964
				Center	5066
				Edge	5719
		Bare	4	Edge	3290
				Center	3576
				Edge	3199
G	2.75 Days			Edge	2.00×10^4
		Cd Covered on Concrete Slugs	2.5	Center	1.724×10^4
				Edge	1.847×10^4
L	3.8 Days	Cd Covered on Wooden Rod	2.5	Right (Toward large core hole)	2.07×10^4
				Center	1.85×10^4
				Left	1.77×10^4

[redacted]

[redacted]

TABLE V

Threshold Detector Measurements of Neutrons

Cadmium Covered Aluminum on Concrete Slugs

Runs R and S	Ft. from Inner Edge of Pile Shield	Saturated Activity of 10.2 Min Half-Life at 3800 KW
7.7 Days Soaking	1.5	994
	2.5	66.2 (poor statistics)

Boron Covered Indium

Run M	Ft. from Inner Edge of Pile Shield	Saturated Activity of 4.1 Hr Half-Life at 3800 KW
4.5 Days Soaking	1.5	1070
	2.5	98

[redacted]

[redacted]

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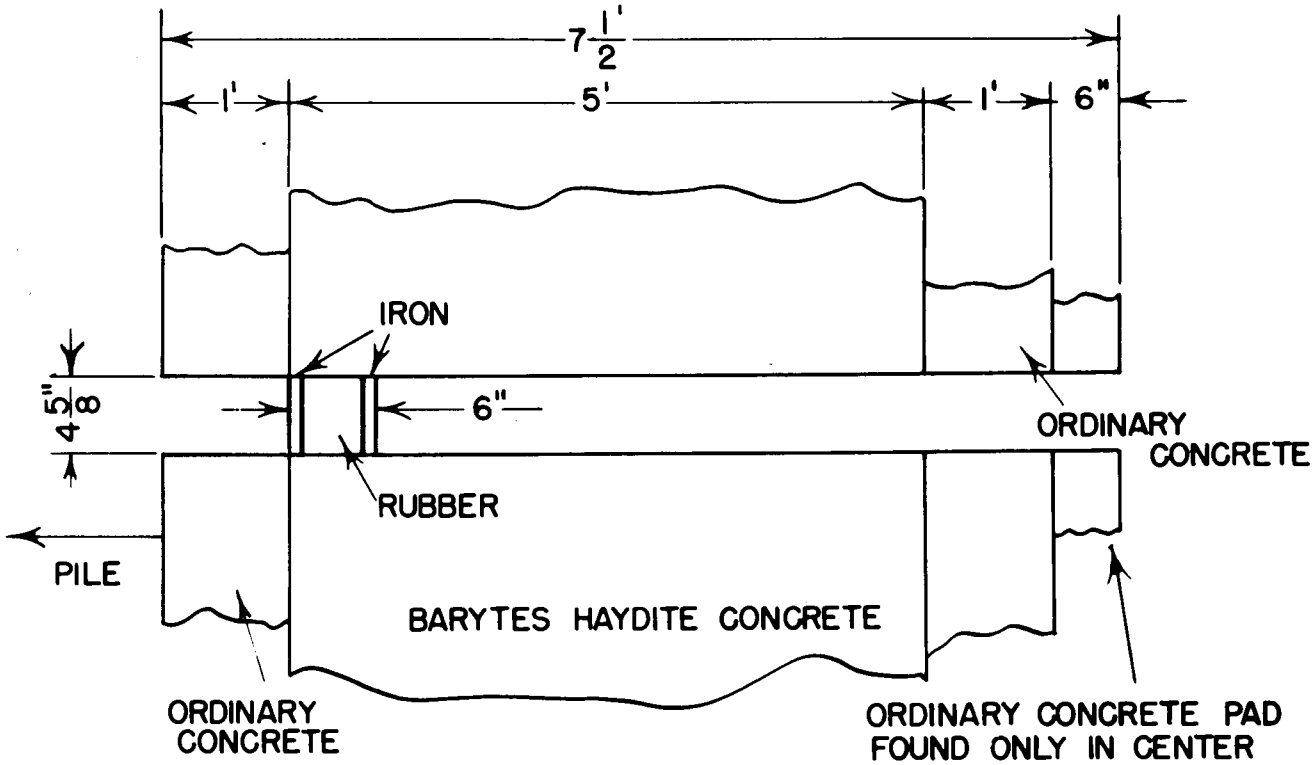
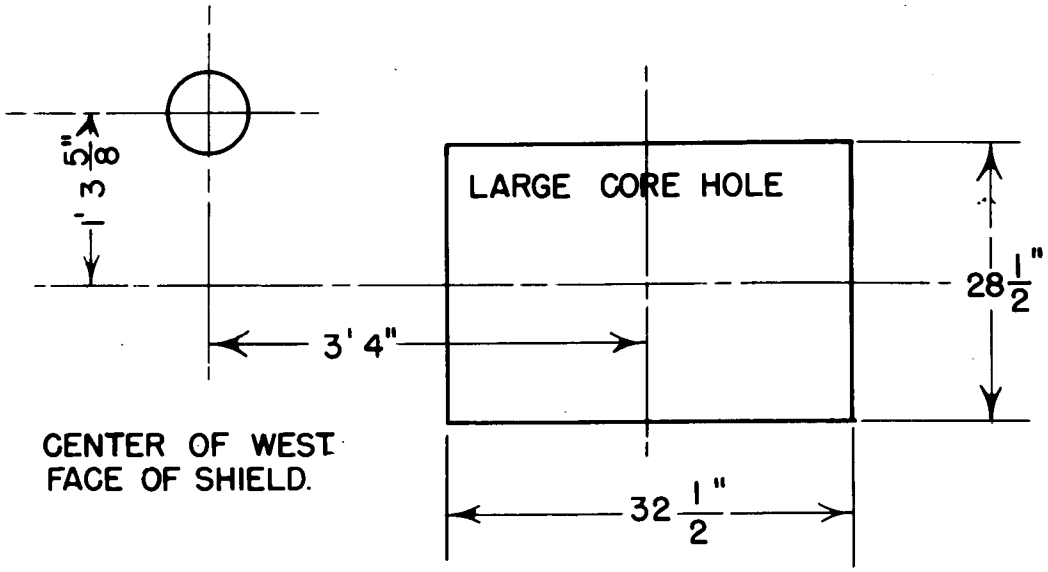


SECTION OF SHIELD
THROUGH SMALL CORE HOLE



LOCATION OF $4\frac{5}{8}$" HOLE RELATIVE TO CENTER OF WEST FACE OF SHIELD. LARGE CORE HOLE IS AT CENTER OF WEST FACE. VIEW LOOKING FROM EXTERIOR OF SHIELD.

FIG. 1

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FIG. 2

OAK RIDGE PILE SHIELD

MEAN NEUTRON RELAXATION LENGTH
VS. TIME OF SOAKING

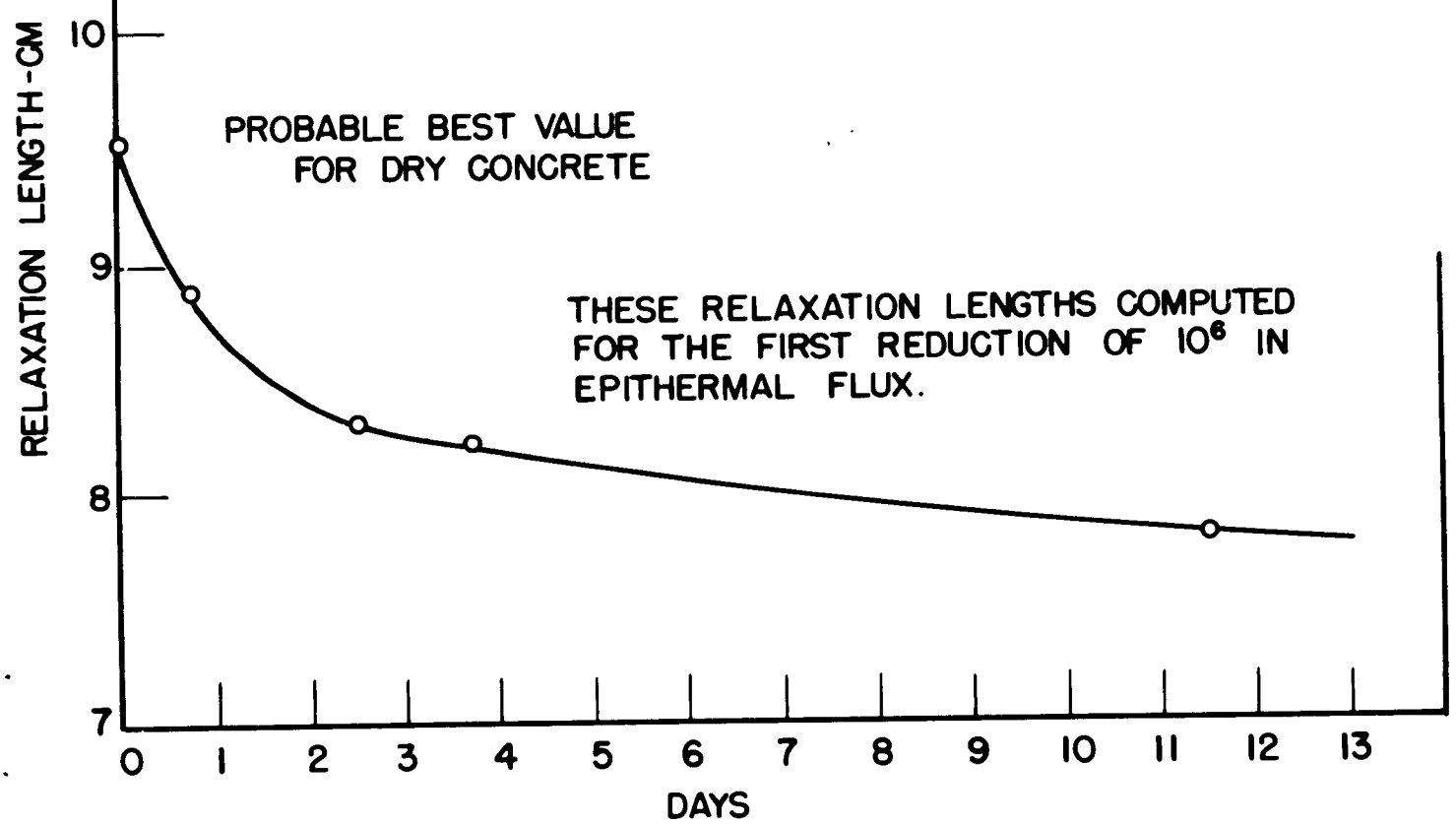

FIG. 3

GAMMA RAY ATTENUATION

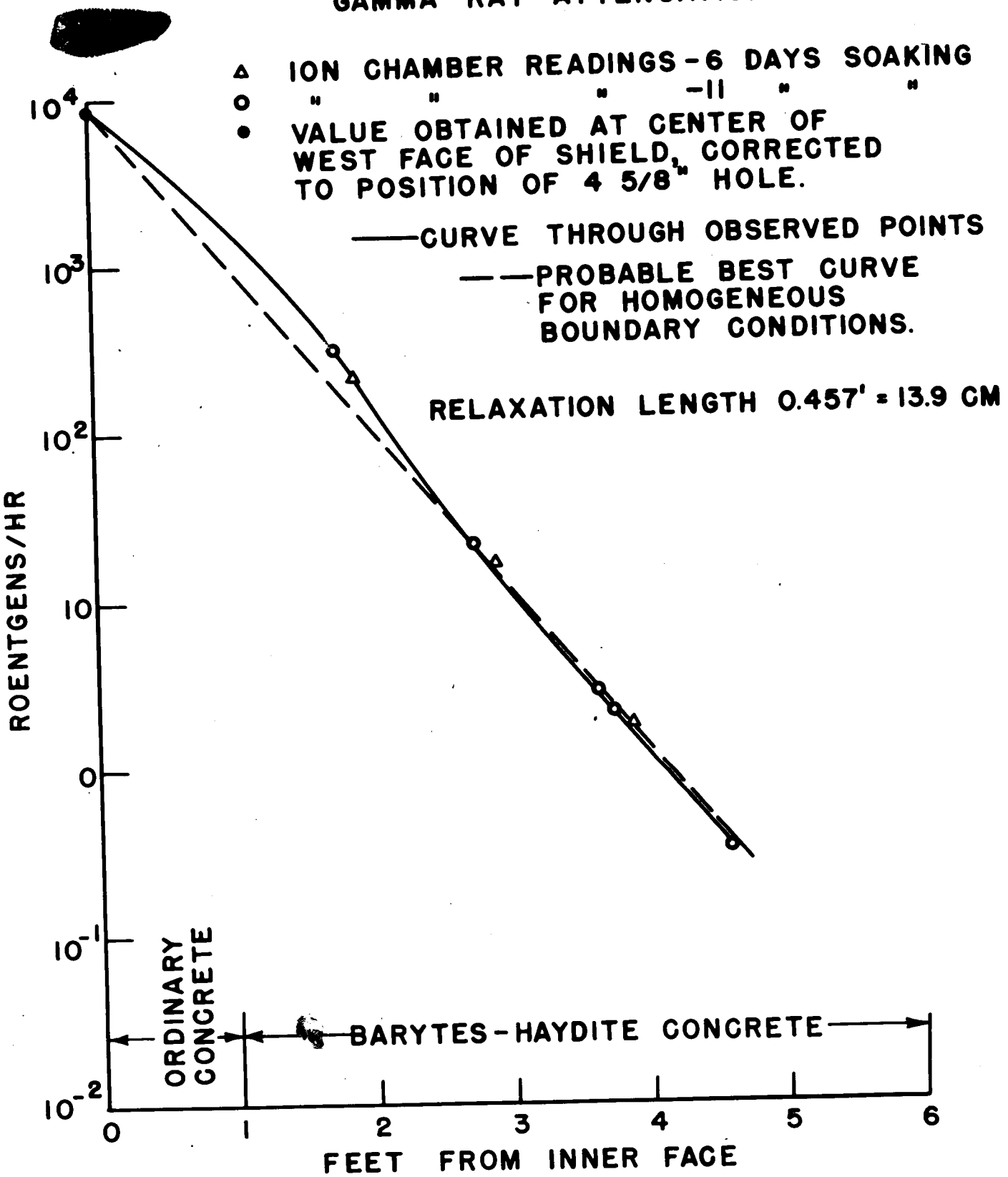

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FIG. 4

NEUTRON ATTENUATION AFTER 3.7 DAYS OF SOAKING (CADMIUM COVERED INDIUM)

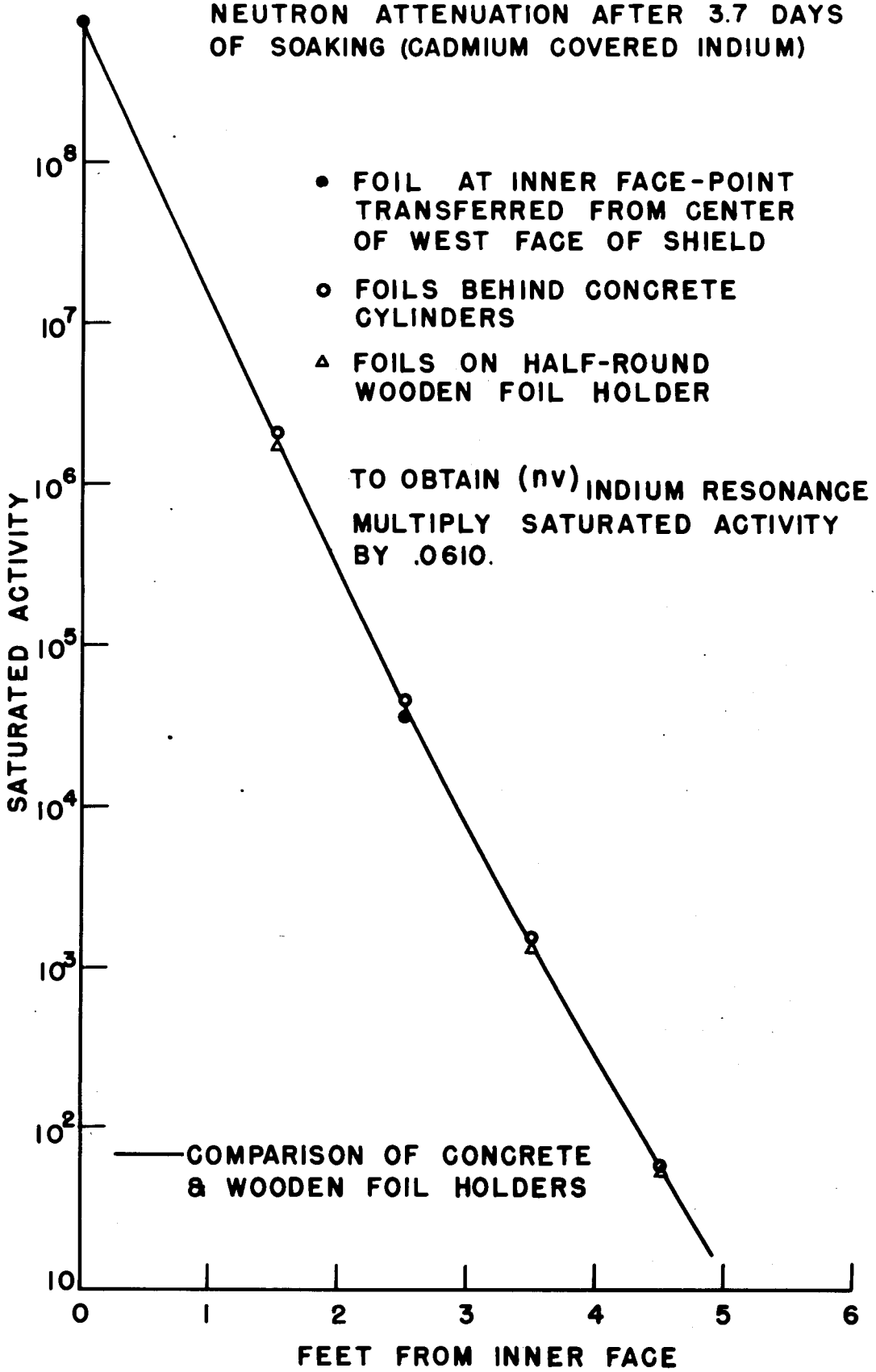

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FIG. 5

SLOW NEUTRON ATTENUATION

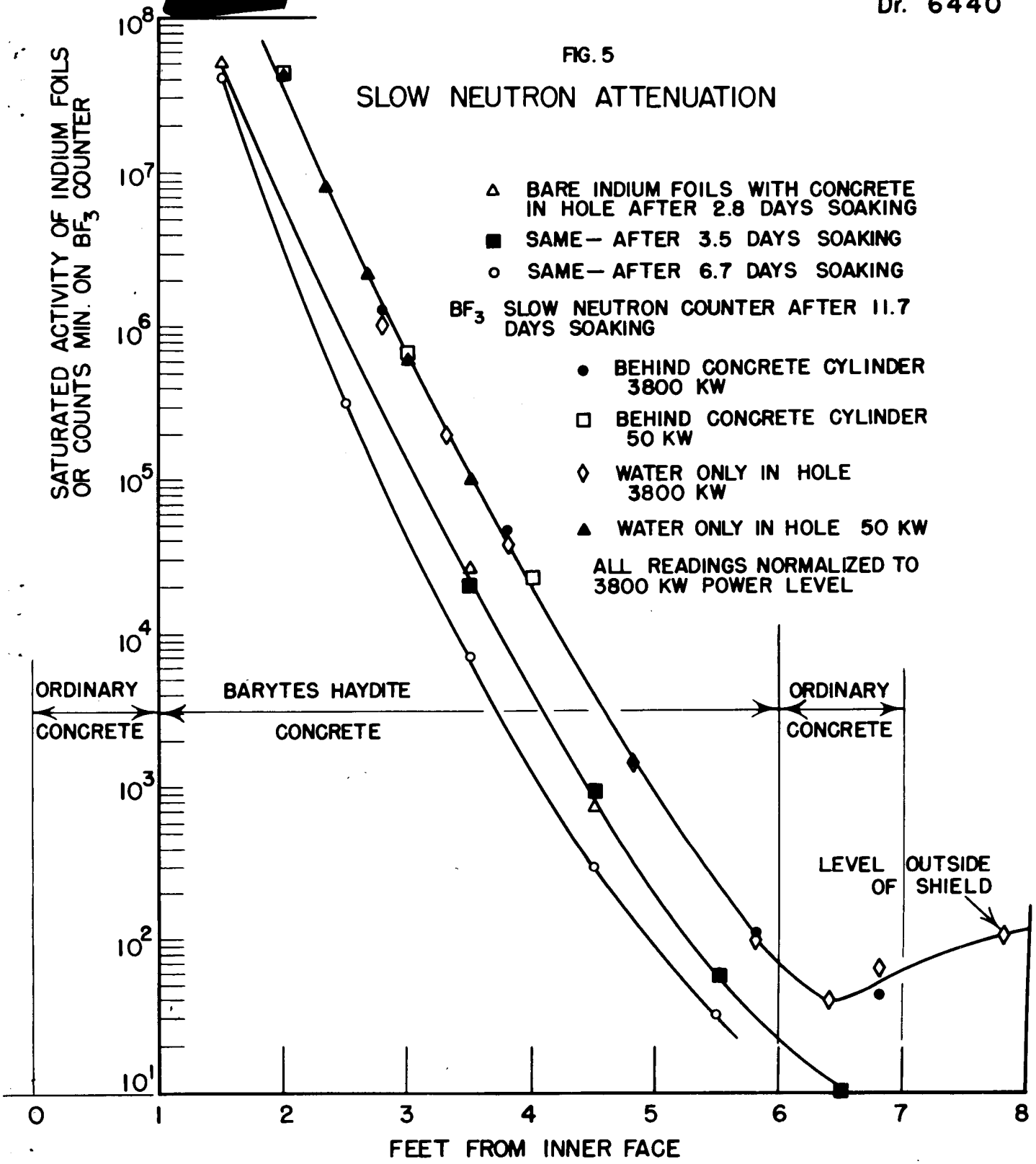
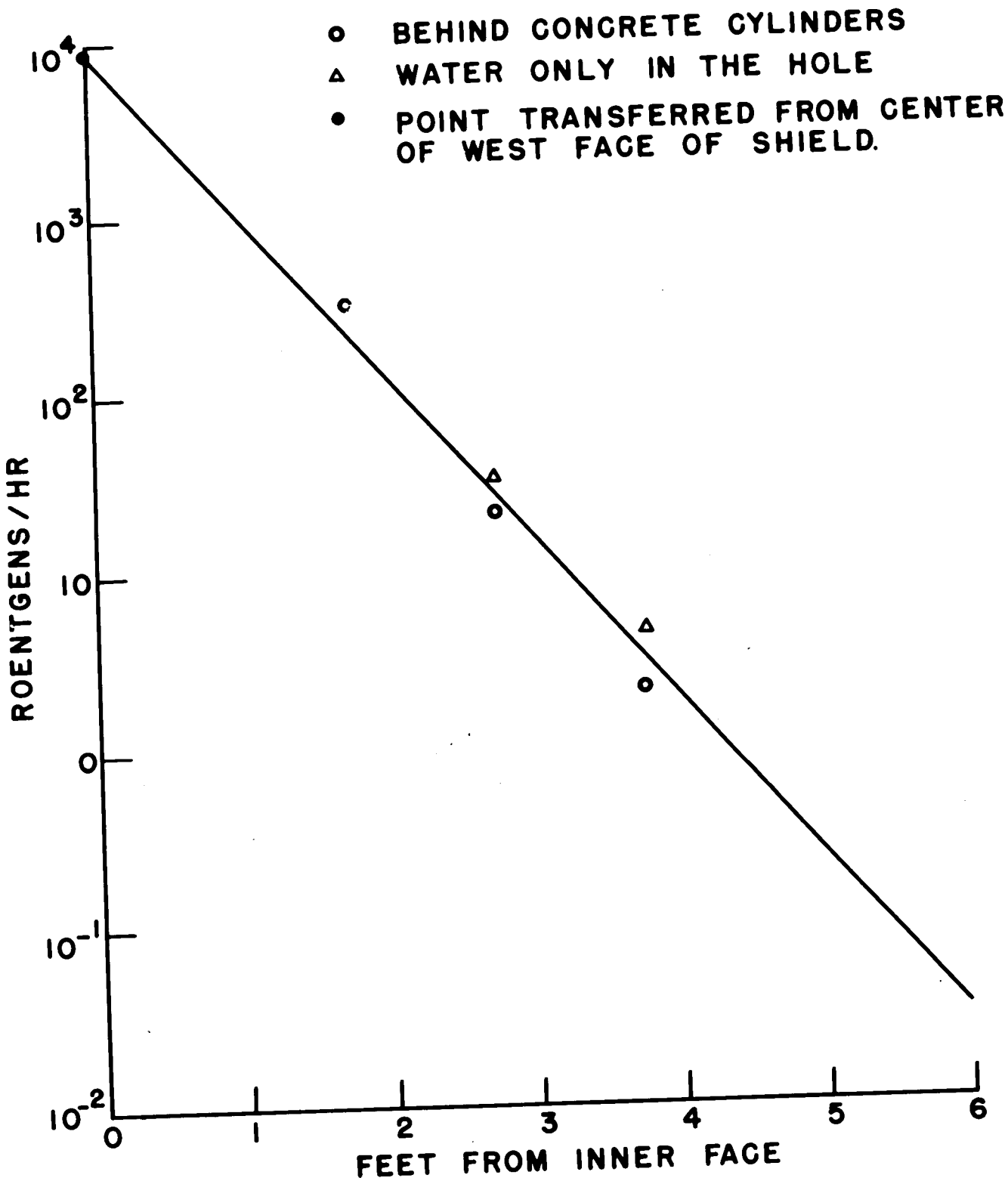


FIG. 6

GAMMA RAY ATTENUATION WITH WATER & WITH CONCRETE IN HOLE



[redacted]

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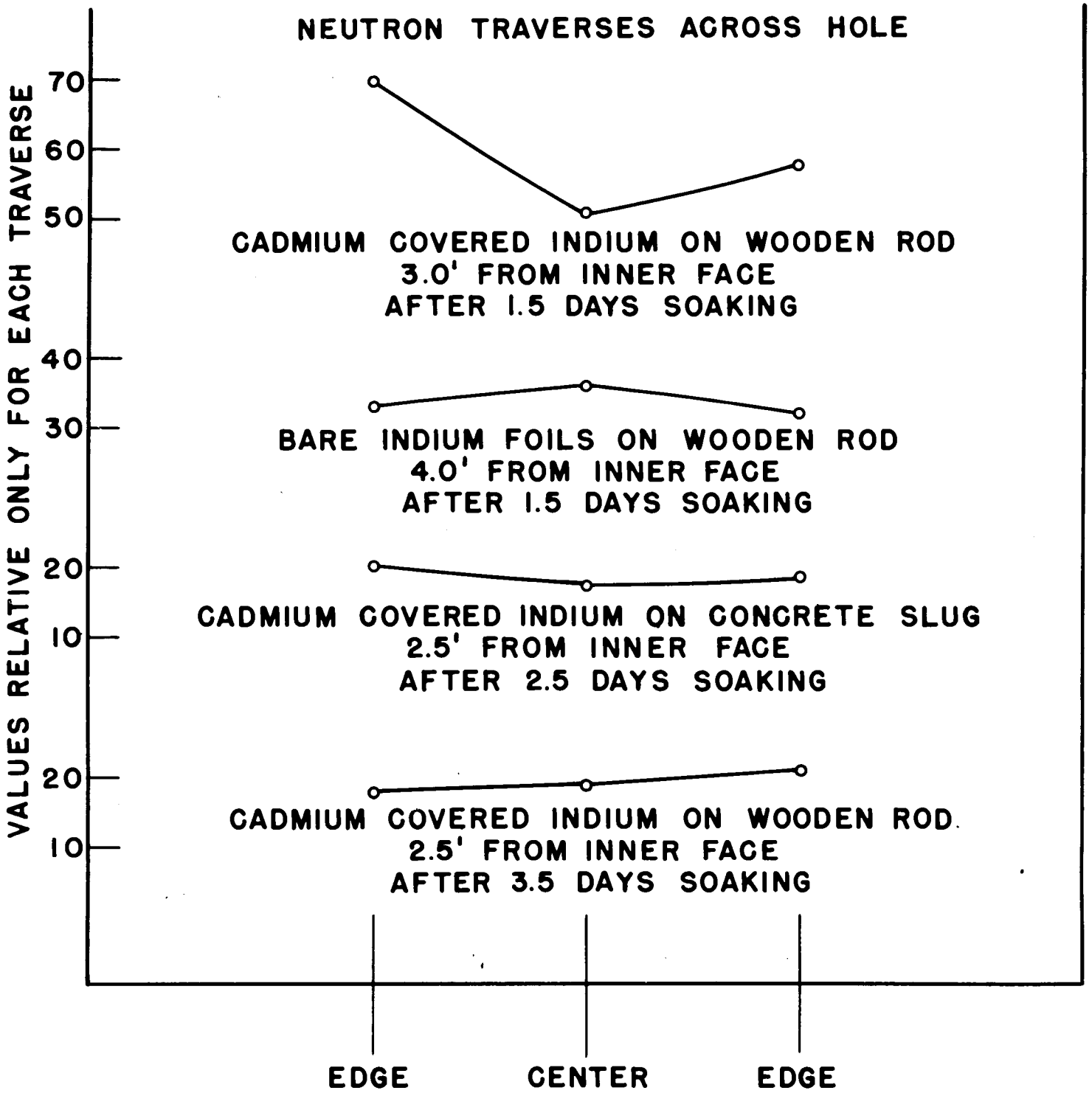
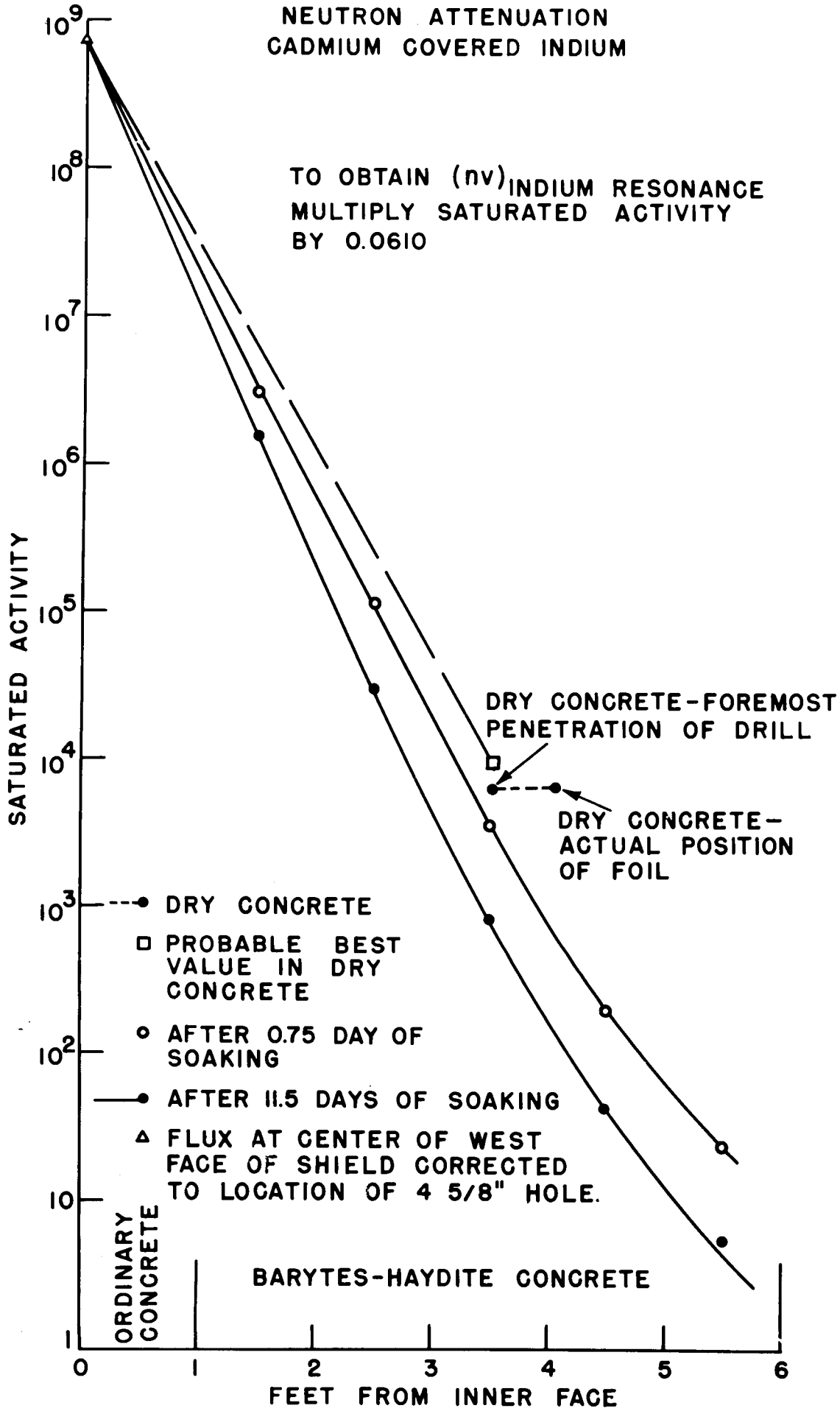
FIG. 7



[redacted]

FIG. 8

NEUTRON ATTENUATION
CADMIUM COVERED INDIUM



[redacted]

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FIG. 9

NEUTRON ATTENUATION IN $4\frac{5}{8}''$ HOLE

EFFECT OF FISSION SOURCE IN ADJACENT 2' X 2' CORE HOLE

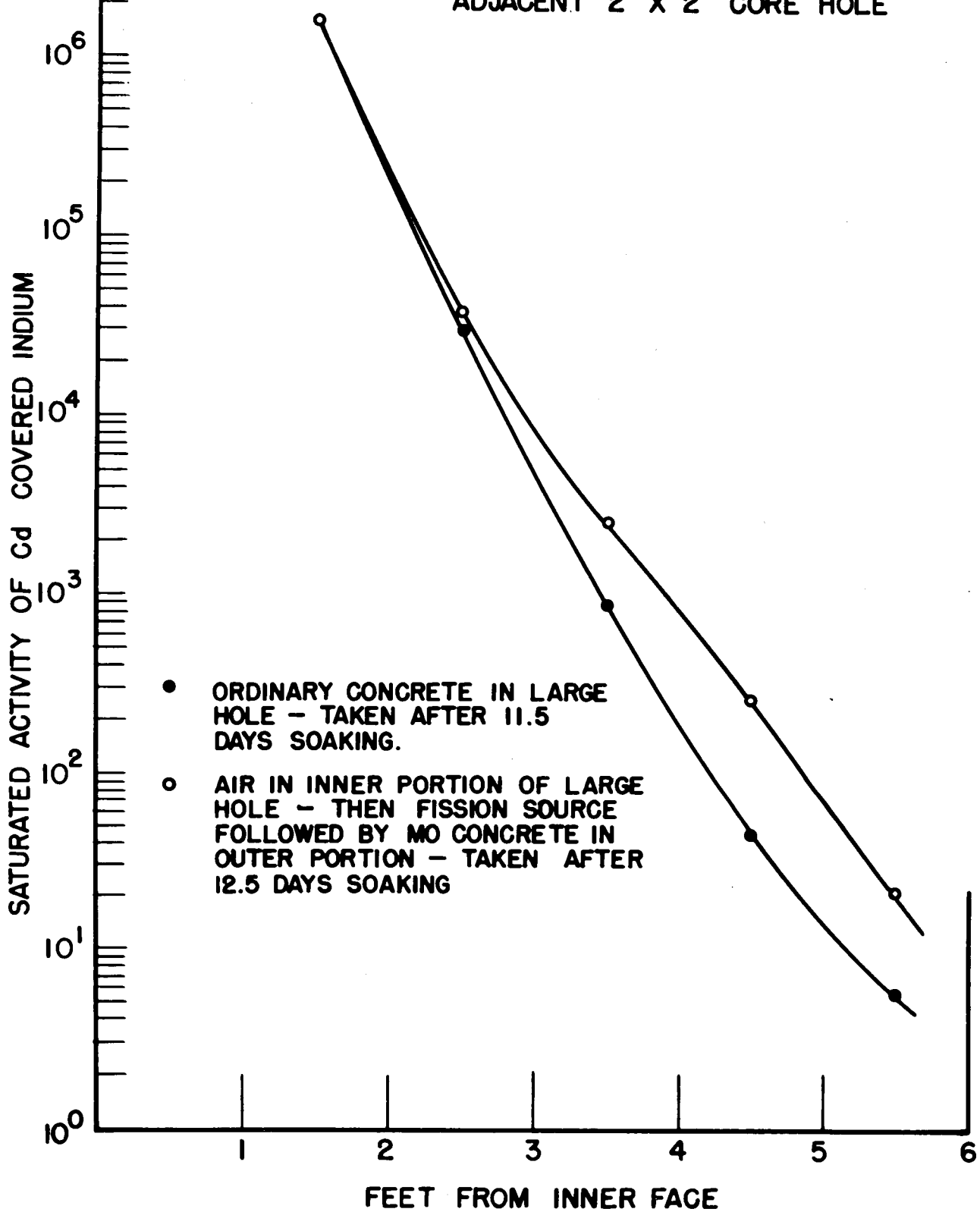

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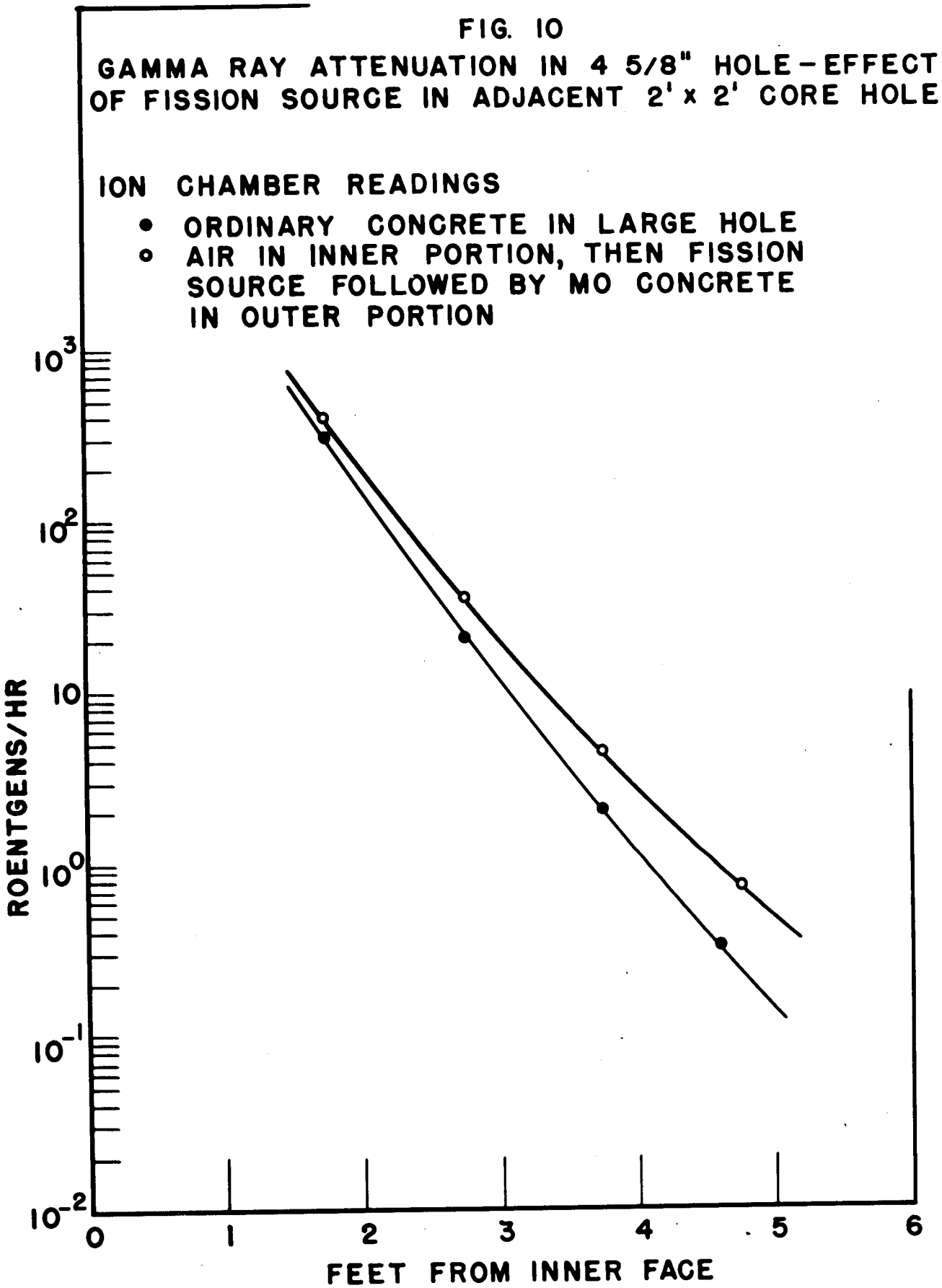
FIG. 10

GAMMA RAY ATTENUATION IN 4 5/8" HOLE-EFFECT OF FISSION SOURCE IN ADJACENT 2' x 2' CORE HOLE



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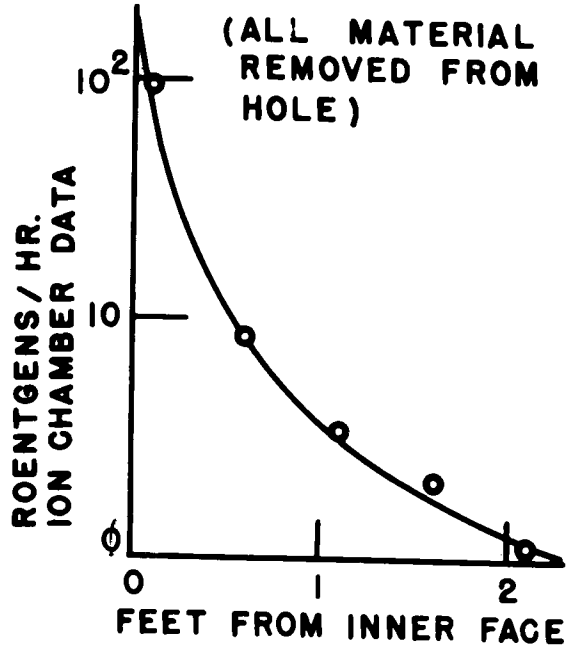
FIG. 11
GAMMA RAY ATTENUATION
THROUGH OPEN 4 5/8" HOLE
3 HR. AFTER PILE SHUTDOWN