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Physics and Mathematics

ANALYSIS OF BULK SHIELDING FACILITY

NEUTRON DOSIMETER DATA

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S. Podgor

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ABSTRACT

"Effective removal cross sections" are calculated for Pb, Fe and O from measurements of fast neutron dose in the water surrounding the BSF reactor. The values for Pb and Fe agree quite well with those previously determined from Lid Tank data, whereas that for O is somewhat lower.

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ANALYSIS OF BULK SHIELDING FACILITY NEUTRON DOSIMETER DATA

S. Podgor

I. Introduction

The new Bulk Shielding Facility has now measured the fast neutron attenuation of water as well as two metal-water combinations. In this report the fast neutron dosimeter data was used to determine the effective removal cross sections for oxygen as well as for the metals Fe and Pb.

II. Reactor and Experimental Arrangements

The active core of the Bulk Shielding Reactor occupies a rectangular parallelepiped, 15 in. by 15 in. by 24 in. The aluminum-encased water-cooled fuel elements are set into an aluminum matrix which is suspended in a pool of water. A complete description of the Bulk Shielding Facility is given in a separate report[1].

The attenuation measurements used in this analysis were performed with the fast neutron dosimeter developed by G. S. Hurst[2] of the ORNL Health Physics Division. This instrument measures the total ionization due to proton recoil of fast neutrons of energies greater than about 1/2 Mev.

Measurements were taken at various distances from the face of the reactor.

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1) Breazeale, W. M. - ORNL-991, "The New Bulk Shielding Facility at Oak Ridge National Laboratory" - May 8, 1951.

2) Hurst, G. S. - ORNL C.F. 51-4-122 - "A Proportional Counter Method of Measuring Fast Neutron Dose" - April 27, 1951.

III. Analytical Procedure

A. Source

In order to analyze these dosimeter measurements, it is necessary to know:

(1) the fast neutron dosage emerging from the reactor, and

(2) how the materials between the reactor and detector attenuate this dosage.

The first is based on measurements by Meem and Johnson[3] and preliminary calculations by H. E. Hungerford.[4]

In order to transform the reactor to an analytically more tractable form, an "equivalent thin source," in watts/cm^2, is calculated. For this purpose the power distribution in each row of elements is expressed in the form of a double cosine function,

$$P_n(x, y) = P_A + P_B \cos \frac{\pi x}{a} \cos \frac{\pi y}{b} ,$$

where the origin is taken at the center of the reactor.

$P_n(x, y)$ is the power density in the n^{th} row at any point (x,y) in the rectangular array of sides a and b, and P_A and P_B are constants

---

3) Meem, J. L. and Johnson, E. B. "Determination of Power of the Bulk Shielding Reactor," ORNL-1027.

4) Hungerford, H. E., personal communication.

for the row.

The distribution functions for the various rows differ only by the difference in the values of P_A and P_B. The equivalent thin source distribution for the face of the reactor is then obtained in the present analysis by integrating P_A and P_B over the Z direction taking the attenuation of the reactor itself into account.

From reference (4) we find the following values for the power distribution in the two rows of elements closest to the face of the reactor in which we are interested:

$$\left.\begin{aligned} P_{A_1} &= 3.92 \times 10^{-6} \text{ watts/cm}^3 \\ P_{B_1} &= 4.73 \times 10^{-6} \quad " \end{aligned}\right\} \text{ at } Z_1 = 3.8 \text{ cm.}$$

$$\left.\begin{aligned} P_{A_2} &= 4.34 \times 10^{-6} \quad " \\ P_{B_2} &= 5.24 \times 10^{-6} \quad " \end{aligned}\right\} \text{ at } Z_2 = 11.4 \text{ cm.}$$

where Z is zero at the reactor face and increases inward and power densities are listed for total power of one watt.

We assume a linear variation of P_A and P_B with Z and obtain:

$$P_A(Z) = (3.71 + .0553\ Z) \times 10^{-6}$$

$$P_B(Z) = (4.48 + .0671\ Z) \times 10^{-6}$$

The assumption is also made that the neutrons and therefore the power are attenuated exponentially. Then, for instance, P_A expressed as $P_{AO} + m_A Z$ integrates to:

$$L_A = \int_0^\infty P_A(Z)\, e^{-Z/\lambda_r}\, dZ = \int_0^\infty \left(P_{AO} + m_A Z \right) e^{-Z/\lambda_r}\, dZ$$

$$L_A = \lambda_r P_{AO} + m_A \lambda_r^2$$

Using infinity as the upper limit of integration introduces no serious error since the reactor is many relaxation lengths wide. The quantity λ_r is the relaxation length of the reactor for ($\sim$8 Mev) neutrons and it is found in the above report that: λ_r = 9.7 cm. Performing the substitutions we find that:

$$L_A = 4.12 \times 10^{-5} \text{ watts/cm}^2$$

$$L_B = 4.98 \times 10^{-5} \quad "$$

where L_B bears the same relation to P_B that L_A does to P_A. The distribution of power on the face of the reactor is then:

$$L(x,y) = L_A + L_B \cos \frac{\pi x}{a} \cos \frac{\pi y}{b} \quad , \qquad (1)$$

where a = 42 cm, b = 61 cm.

B. Attenuation by Shield Materials

To calculate the attenuation of this source of power or dosage, we shall use the "one collision" theory of shielding as developed by Albert and Welton.[5] According to that theory a neutron is considered as essentially lost from the shielding picture if it has made one collision with a nucleus. This is especially true for hydrogen where one collision usually degrades the neutron in energy significantly and after that it is more rapidly degraded and absorbed. On the other hand, those collisions with hydrogen which result in only slight degradation

5) Albert, R. D. and Welton, T. A. - WAPD-15, "A Simplified Theory of of Neutron Attenuation and Its Application to Reactor Shield Design," November 30, 1950.

in energy and therefore small angle change are accounted for by means of the so called "buildup" factor. This is the ratio of dosage due to the total flux to that due to the uncollided flux.

For heavy materials, inelastic scattering and significant angle elastic scattering are effective in eliminating neutrons. For these materials it has been found possible to postulate an "effective removal cross section" for neutrons which is independent of the energy over a fairly wide range. With this theory it has been possible to analyze[5), 6)] the Lid Tank thermal data of Blizard and Clifford and obtain removal cross sections for oxygen, Pb and Fe.

1. Water

T. A. Welton has used the theory to analyze[7)] the Lid Tank dosimeter water data and his method is here applied to the Bulk Shielding Facility data.

The hydrogen cross section is represented approximately as

$$\sigma(E) = \frac{\gamma}{E+\epsilon} \text{ cm}^{-1}$$

where E is the neutron energy in Mev and γ and ϵ are constants.

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6) Podgor, S. - ORNL-895, "Analysis of Lid Tank Neutron Data for Lead and Iron" January 23, 1951.

7) ANP-53, "Report of the Shielding Board" - page 120, October 16, 1950.

The fission spectrum is represented as:

$$N(E) = N_o e^{-\alpha E} (Mev)^{-1}$$

where N(E) is the number of neutrons per fission neutron per Mev energy range, and α is a constant. The values of the constants were taken from a paper [8] by Blizard and Welton, who give $\alpha = .75(Mev)^{-1}$ $\gamma = .735$ Mev/cm $\epsilon = 1.66$ Mev. These approximations are valid for the energy range 3 to 10 Mev, which is the important region for the attenuations of interest to this project. According to reference (7), then, the dosage from a point source will be attenuated in water as follows:

$$D(S) = \frac{A}{4\pi S^2} S^{1/4} e^{-\beta S - 2\sqrt{\alpha\gamma S}} \qquad (2)$$

where D(S) is the dosage at a distance S from the source, β is the oxygen cross section, and A is a constant which will be evaluated later.

It was thought that it would be too cumbersome to integrate analytically this point source function over the rectangular area of Equation (1). The source was therefore taken to be circular and of

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8) Blizard, E. P. and Welton, T. A. - "Shielding of Mobile Reactors – I" Reactor Science and Technology.

radius, $R = \sqrt{\frac{ab}{4}}$. It is believed that this is a good approximation, since the dosage is greatest in the center of the source in any case; and by making it circular, we are merely shifting elements of the source around the outside circumference and keeping them at approximately the same distance from the center.

The source is then taken to be:

$$L(r) = L_A + L_B \cos \frac{\pi R}{2R} \tag{3}$$

where L_A and L_B are the same as in equation (1), R = 25.3 cm, and r is the distance of any point from the center.

The source is considered to be isotropic.

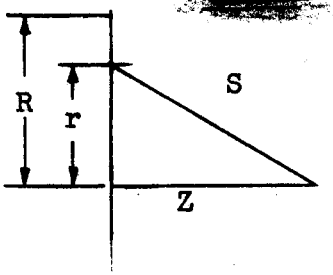
We integrate the point source of equation (2) over the circular source of equation (3), to obtain the dosage F(Z) at any centerline point Z out in the water.

$$F(Z) = \int_0^R \frac{A}{4\pi S^2} B(S,E)\, 2\pi r dr\, S^{1/4}\, e^{-\beta S - 2\sqrt{\alpha\gamma S}} \left(L_A + L_B \cos \frac{\pi r}{2R} \right). \tag{4}$$

where the only new quantity, B(S,E), is the buildup factor due to neutrons which have collided but continue with the primary beam. This is actually a function of both E and S. However, it is so slowly varying that it is accurate enough to consider it a function of Z only and to take it outside of the integral sign.

The following transformation is made:

$$r^2 = S^2 - Z^2 , \qquad rdr = SdS.$$



Then:

$$F(Z) = \frac{AB(Z)}{2} \int_Z^{\sqrt{R^2+Z^2}} dS\, S^{-3/4}\, e^{-\beta S}\, E^{-2\sqrt{\alpha\gamma S}} \left(L_A + L_B \cos \frac{\pi\sqrt{S^2 - Z^2}}{2R} \right)$$

We expand:

$$\cos \frac{\pi\sqrt{S^2 - Z^2}}{2R} \cong 1 - \frac{\pi^2(S^2 - Z^2)}{8R^2} \approx 1 - \frac{S^2 - Z^2}{R^2}$$

where the factor $\frac{\pi^2}{8}$ was omitted to make the expression vanish at $r = R$ or $S = \sqrt{R^2 + Z^2}$.

Then:

$$F(Z) = \frac{AB(Z)}{2} \int_Z^{\sqrt{R^2+Z^2}} dS\, S^{-3/4}\, e^{-\beta S}\, e^{-2\sqrt{\alpha\gamma S}} \left[L_A + L_B \left(1 - \frac{S^2 - Z^2}{R} \right) \right]$$

This can be integrated by the method used in reference (7).

Let $S = Z + t$, where $\underline{t}$ is small compared to $\underline{Z}$.

Then:

$$F(Z) = \frac{AB(Z)}{2} \int_0^{\sqrt{Z^2+R^2} - Z} dt\, (Z+t)^{-3/4}\, e^{-\beta(Z+t)}\, e^{-2\sqrt{\alpha\gamma(Z+t)}} \left[C - D(Z+t)^2 \right]$$

where $C = L_A + L_B\left(1 + \frac{Z^2}{R^2}\right)$, $D = \frac{L_B}{R^2}$

Using the binomial expansion,

$$F(Z) = \frac{AB(Z)}{2} \int_0^{\sqrt{Z^2+R^2}-Z} dt e^{-\beta Z} e^{-\beta t} e^{-2\sqrt{\alpha\gamma Z}} \left(1 + \frac{t}{2Z}\right) \left[CZ^{-3/4}\left(1 - 3/4\ \frac{t}{Z}\right) - DZ^{5/4}\left(1 + \frac{5t}{4Z}\right)\right]$$

$$F(Z) = \frac{AB(Z)}{2} Z^{-3/4} e^{-\beta Z} e^{-2\sqrt{\alpha\gamma Z}} \int_0^{\sqrt{Z^2+R^2}-Z} dt e^{-t\left(\beta+\sqrt{\frac{\alpha\gamma}{Z}}\right)} \left[\left(C-DZ^2\right) - \frac{t}{4Z}\left(3C+5DZ^2\right)\right]$$

$$F(Z) = \frac{AB(Z)}{2} \lambda Z^{-3/4} e^{-\beta Z} e^{-2\sqrt{\alpha\gamma Z}} \left\{\left(1 - e^{-\frac{\Delta}{\lambda}}\right)\left(C-DZ^2\right) - \frac{\lambda}{4Z}\left[1 - \left(1+\frac{\Delta}{\lambda}\right) e^{-\frac{\Delta}{\lambda}}\right]\left(3C + 5DZ^2\right)\right\}$$

where $\Delta = \sqrt{Z^2+R^2} - Z$, $\lambda = \frac{1}{\beta + \sqrt{\frac{\alpha\gamma}{Z}}}$

using equations (4)

$$C = L_A + DR^2 + DZ^2 .$$

then

$$F(Z) = \frac{AB(Z)}{2} \lambda Z^{-3/4} e^{-\beta Z} e^{-2\sqrt{\alpha\gamma Z}} \left\{\left(1 - e^{-\frac{\Delta}{\lambda}}\right)\left(L_A+DR^2\right) - \frac{\lambda}{4Z}\left[1 - \left(1+\frac{\Delta}{\lambda}\right) e^{-\frac{\Delta}{\lambda}}\right]\left[3(L_A + DR^2) + 8DZ^2\right]\right\}$$

Equation (5)

Equation (5) then gives the dosage for the assumed source of Equation (3) at any Z along the centerline axis in the water.

Everything has been fixed in equation (5) except A and β. The buildup factor B(Z) has been given in preliminary form in reference (9). It is given in revised form in Appendix I and Figure III. A is the normalization constant which is derived analytically in Appendix II. The quantity β is the oxygen removal cross section in cm^{-1} which it is proposed to determine by fitting to the experimental data. However the data[10] for H_2O only goes up to $\sim$ 70 cm from the source, which is the region where this analysis is expected to give the poorest agreement with experiment. It was decided instead to fit to the Pb and H_2O data of the same experiment[10] which uses only three inches of lead and extends from Z = 46 cm to 166 cm.

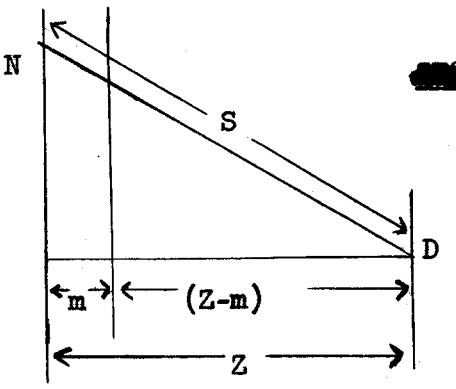
2. Metal and Water

In order to take the metal into account, it is necessary to modify equation (5) somewhat. A point source of neutrons will be attenuated by a slab of metal followed by water in the following manner:

---

9) ANP-60, ANP Quarterly Progress Report for Period Ending March 10, 1951, page 169.

10) Cochran, R. G. and H. E. Hungerford, ORNL-C.F. 51-5-61 - "Fast Neutron Dosimeter Measurements for Experiment No. 1 in Bulk Shielding Facility - May 7, 1951.



$$P(S) = \frac{\left(S\,\frac{Z-m}{Z}\right)^{1/4}}{4\pi S^2}\, e^{-S\left[\beta\,\frac{Z-m}{Z} + \frac{\sigma m}{Z}\right]}\, e^{-2\sqrt{\alpha\gamma}\,\sqrt{S\,\frac{Z-m}{Z}}} \qquad (6)$$

Where P(S) is the dosage at the detector, D, at a distance S from the neutron source N, the perpendicular distance of D from the farther edge of the metal slab is Z, the metal thickness is m, and its neutron removal cross section is σ cm^{-1}.

When the new point source of equation (6) is integrated over the above circular source of equation (3), we obtain the dosage at any point Z as attenuated by the metal slab and water.

This comes out to be:

$$F(Z) = \frac{AB(Z)\lambda}{2Z}\,(Z-m)^{1/4}\, e^{-\left[\beta(Z-m) + \sigma m\right]}\, e^{-2\sqrt{\alpha\gamma(Z-m)}}$$
$$\left\{\left(1 - e^{-\frac{\Delta}{\lambda}}\right)\left(L_A + DR^2\right) - \frac{\lambda}{4Z}\left[1 - \left(1 + \frac{\Delta}{\lambda}\right)e^{-\frac{\Delta}{\lambda}}\right]\right.$$
$$\left.\left[3\left(L_A + DR^2\right) + 8DZ^2\right]\right\} \qquad (7)$$

where $\lambda = \dfrac{Z}{\beta(Z-m) + \sigma m + \sqrt{\alpha\gamma(Z-m)}}$

$$\triangle = \sqrt{R^2 + Z^2} - Z$$

and everything else has been defined.

IV. Application to Experimental Data

A. Pb and H_2O

Equation (7) was fitted to the Pb-H_2O data[10] of Cochran and Hungerford. Everything is determined in that equation except A, β, σ. For B(Z) it is considered a good approximation to use the buildup factor for water. For σ (Pb), the value (0.112 cm^{-1} or 3.6 barns) obtained in the analysis[6] of Clifford's Lid Tank data is used. It is somewhat lower than that obtained by Albert and Welton[5] (0.118 cm^{-1}), but the small amount of lead involved makes the difference unimportant. The oxygen cross section β is then determined to make the analytical curve agree in shape with the experimental data. It is then normalized by means of the quantity A. The value of β comes out to be 0.0274 cm^{-1} or 0.82 barn, and $A = 4.094 \times 10^5$. The graphical comparison of analytical fit with experimental results is shown in Figure I.

B. Fe and H_2O

The same procedure is applied to the Fe-H_2O data[11] of Cochran and Hungerford. However in this case we use the β and A found from the Pb-H_2O data. The thickness of iron is 18 slabs of 7/8" each or 40 cm. The σ (Fe) that is needed to fit the data is .167 cm^{-1} or 1.98 barns. This agrees with the value of 2.0 barns found in the analyses[5), 6)] of the Lid Tank data. The comparison between analytical fit and experimental values is shown in Figure I.

In this case also, the same buildup factor is used as for water. This seems to be satisfactory far enough out in the water. However, close to the metal, there is a buildup of $\sim$1 Mev neutrons caused by the iron. This is the energy region where the absorption cross section of the Fe has become low and the inelastic cross section has not yet become significant. The same effect has appeared in the Lid Tank data. The present analysis does not include this effect.

C. H_2O

When equation (5) is now applied to the water data of reference (9), using the values of A and β found above, the comparison between analysis and experiment is shown in Figure II. As indicated above,

---

11) Cochran, R. G. and Hungerford, H. E. - ORNL C.F. 51-5-73, "Fast Neutron Dosimeter Measurements for Experiment 2," - May 11, 1951.

the comparison was not expected to be as good in this case as in the metal and water cases. The approximations used for the hydrogen cross section and the fission spectrum, and method of integration are all less valid for the lower energies and lower Z values at which the data was taken.

V. Discussion

The only result that is significantly different from those obtained in previous analyses is the value of the oxygen removal cross section. The value found here is 0.82 barn. The only other comparable figure is that obtained by Albert and Welton[5] and that is 0.91 barn. Their analysis is based on thermal data, whereas the present one is made on dosimeter data. The value obtained in the present report depends on the neutron buildup factor used. If a more rapidly increasing buildup factor is assumed, then a larger oxygen cross section is needed to compensate for it.

The results for Pb and Fe are in good agreement with those obtained from the Lid Tank data. However the relatively small amount of lead involved in this experiment does not permit a complete comparison for that metal.

If we consider the question of agreement in absolute values, we find in Appendix II, that the experimental results are greater than the

calculated values by a factor of ~ 2.9. The following considerations would tend to make the analytical values too low:

(1) The circular source was perhaps made somewhat small. To make this area agree with the area of the actual source, we need

$$r = \sqrt{\frac{ab}{\pi}} \quad \text{instead of} \quad r = \sqrt{\frac{ab}{4}}$$

(2) The use of only two terms of the cosine expansion is an underestimation of the value.

(3) A redetermination of the neutron leakage by the experimental group indicates an increase of about 10% over the values used in the present analysis.

A very crude calculation indicates that these items might account for as much as a factor of 2.3. The calculated values would then be only 20% lower than the experimental results.

VI. Acknowledgements

The writer is very pleased to acknowledge many interesting and informative discussions with E. P. Blizard and T. A. Welton during the course of this investigation, as well as the very able assistance of Phyllis Brown with the calculations.

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Encl: Figures I, II, III.

Appendix I

Neutron Buildup in Water

The buildup in neutron dose at any point is defined as the ratio of the total neutron dose at that point to the uncollided dose there, or in symbols:

$$B = \frac{D_t}{D_o} = 1 + \frac{D_c}{D_o}$$

where B is the buildup, D_t - the total dose, D_c - the collided dose, D_o - the uncollided dose.

Calculations of the buildup in water were reported in preliminary form in an ANP Division Quarterly Report[9]. The present results given in Figure III are a revised and corrected version. They give the buildup for two values of the oxygen cross section:

$$\sigma = 0.7 \text{ barn} \quad \text{and} \quad \sigma = 1.0 \text{ barn.}$$

The first value was the one used in the calculations of the present report.

- - - -

Calculation of Normalization Constant

This calculations is based on the one made in reference (7)

$$N(E) = 1.8\, e^{-.75E}, \quad \alpha = .75$$

3×10^{10} = no. of fissions per joule

$$.67 \times 10^5 \frac{\text{neutrons}}{\text{cm}^2 - \text{sec}} = 1 \frac{\text{rep}}{\text{hour}}$$

N(E) is the number of neutrons emitted per fission neutron per Mev energy range.

$\nu = 2.5$, number of neutrons per fission

$$A_{calc.} = \frac{3 \times 10^{10} \times 1.8 \times 2.5}{.67 \times 10^5} e^{\alpha\epsilon} \sqrt{\pi}\, \alpha^{-3/4}\, \gamma^{1/4}$$

where $\epsilon = 1.66$, $\alpha = .75$, $\gamma = .735$

$A_{calc.}$ = 1.425×10^5 millirep/hr/watt - calculated value

$A_{exp.}$ = 4.094×10^5 millirep/hr/watt - value needed to fit experimental data

$$\frac{A_{exp.}}{A_{calc.}} = \frac{4.094 \times 10^5}{1.425 \times 10^5} = 2.873$$

In other words the experimental results are greater, in absolute value, than the calculated values by a factor of ~ 2.9 .

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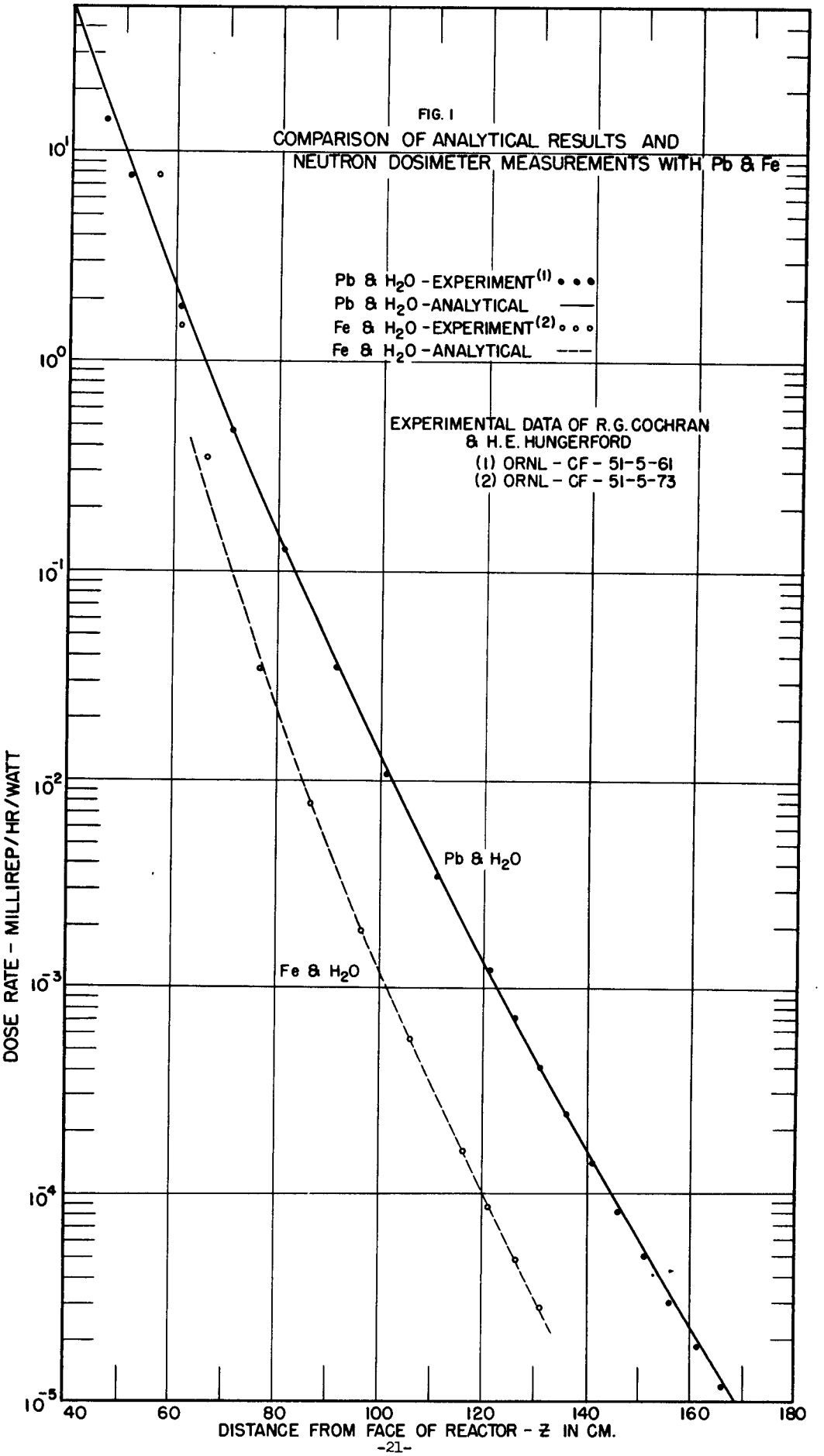
FIG. 1
COMPARISON OF ANALYTICAL RESULTS AND
NEUTRON DOSIMETER MEASUREMENTS WITH Pb & Fe
Pb & H2O - EXPERIMENT (1)
Pb & H2O - ANALYTICAL
Fe & H2O - EXPERIMENT (2)
Fe & H2O - ANALYTICAL
EXPERIMENTAL DATA OF R.G. COCHRAN
& H.E. HUNGERFORD
(1) ORNL - CF - 51-5-61
(2) ORNL - CF - 51-5-73
Pb & H2O
Fe & H2O
DOSE RATE - MILLIREP/HR/WATT
10^1
10^0
10^-1
10^-2
10^-3
10^-4
10^-5
40
60
80
100
120
140
160
180
DISTANCE FROM FACE OF REACTOR - Z IN CM.

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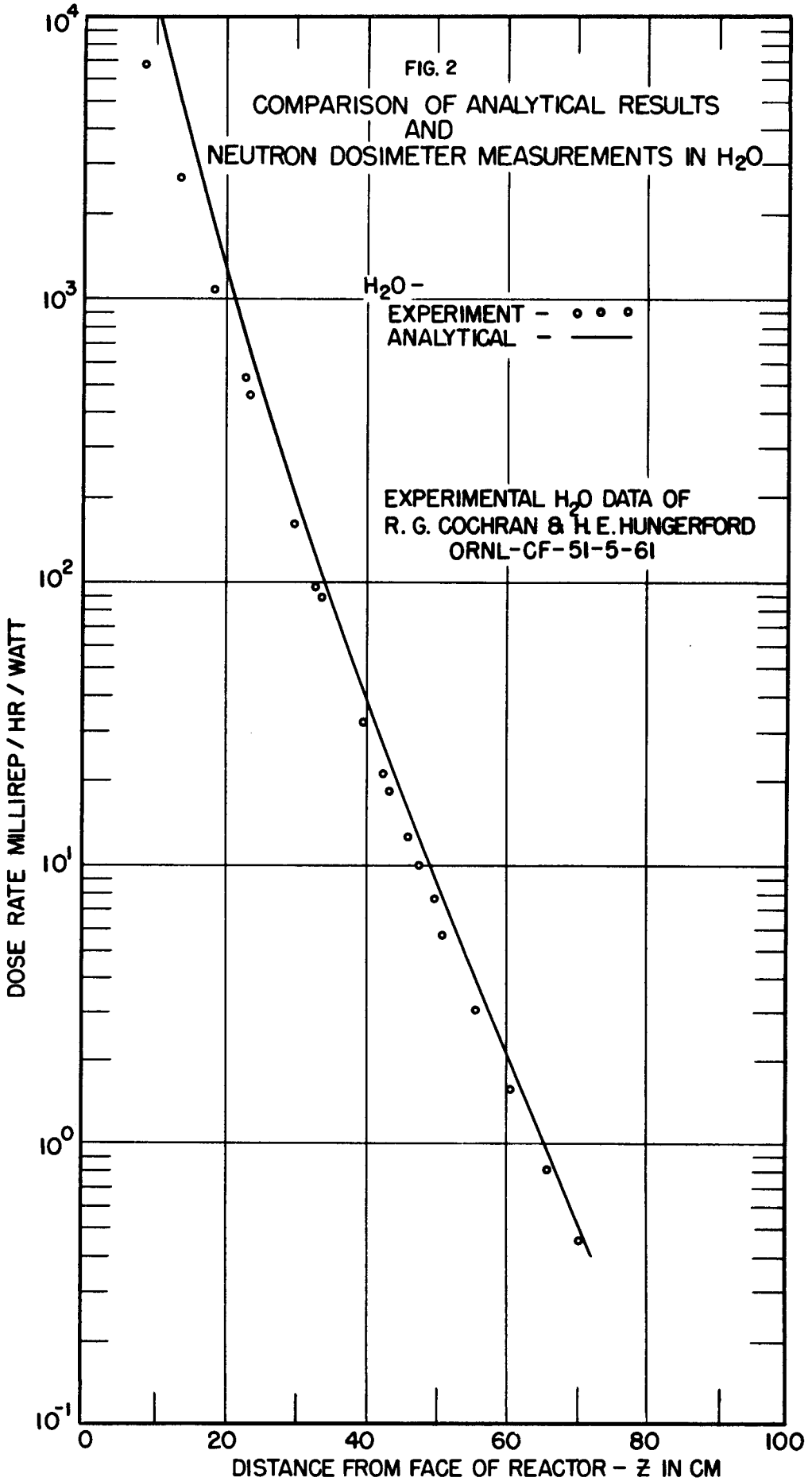
FIG. 2

COMPARISON OF ANALYTICAL RESULTS AND NEUTRON DOSIMETER MEASUREMENTS IN H_2O



[redacted] DWG. 12767

FIG. 3