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Nuclear Accident Dosimetry: Calculations and Comparison with Experimental Data

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ABSTRACT

Calculated results, carried out by the method of discrete ordinates, of the absorbed dose received by personnel at various locations relative to the source in a simulated reactor criticality accident are presented and compared with experimental data. The geometry of the room in which the "accident" took place is included approximately in the calculations. The calculated and experimental data are in good agreement in all cases.

I. INTRODUCTION

In a recent report, the results of a nuclear criticality accident simulated at the Health Physics Research Reactor at the Oak Ridge National Laboratory were presented.[1] The data presented in the report are the absorbed dose received by "personnel", portrayed by manikins, at various positions with respect to the reactor and the absorbed dose as determined by nuclear accident dosimeters. The absorbed dose data at the manikins were obtained from film badges and from the activation of the sodium in the saline-solution-filled manikins.

In this paper, calculated results are presented and compared with these experimental data. In the calculations a two-dimensional representation of the room enclosing the reactor was used and the neutron and photon fluences as a function of position were obtained by using the method of discrete ordinates.[2] The absorbed doses were obtained from these fluences by using available fluence-to-dose conversion factors.[3,4] In one case where one of the manikins was shielded by another of the manikins this shielding effect was taken into account in the computation of the fluences. In all cases, the calculated results agree well with the experimental data. For comparison purposes, calculated results are also given when the shield (= 20 g cm^{-2} thick) is composed of iron or concrete rather than water.

In Section II the calculational geometry is described and the details of the calculations are discussed. In Section III the results are presented and discussed.

II. METHOD OF CALCULATION AND GEOMETRY

The geometry used in the calculations is shown schematically in Fig. 1. The floor is simulated by the 60-cm-thick concrete slab and the roof is simulated by the 2-cm-thick iron slab. The sides of the building enclosing the reactor are very thin and were not included in the calculations. The composition and density of concrete was taken from Ref. 6. In the calculations the reactor is treated as a point source as indicated in the figure. Neutrons are assumed to be emitted isotropically from the point source with a measured energy spectrum.[5] The spectrum used is also shown in Fig. 1. The normalization of the source was taken to be $8.6 \cdot 10^{16}$ neutrons as measured during the experiment.

The manikins are not shown in Fig. 1 since they were not included in the calculation of the particle fluences, but rather the attenuation of the radiation in the manikins was taken into account very approximately by the use of fluence-to-dose conversion factors. In the one case when one of the manikins was shielded by another of the manikins the shielding was taken into account by the 20-cm-thick cylindrical shell of water shown in Fig. 1. This is a very approximate representation of the manikin, but is, as the results will show, satisfactory for the present purposes. To obtain the results for both the shielded and unshielded manikins, calculations were performed with the water cylindrical shell both present and absent.

For comparison purposes, calculated results have also been obtained when the water cylinder in Fig. 1 is replaced by iron and by concrete. In these calculations the iron shield (density = 7.87 g cm^{-3}) and the concrete shield (density = 2.3 g cm^{-3}) were taken to have a radial thickness

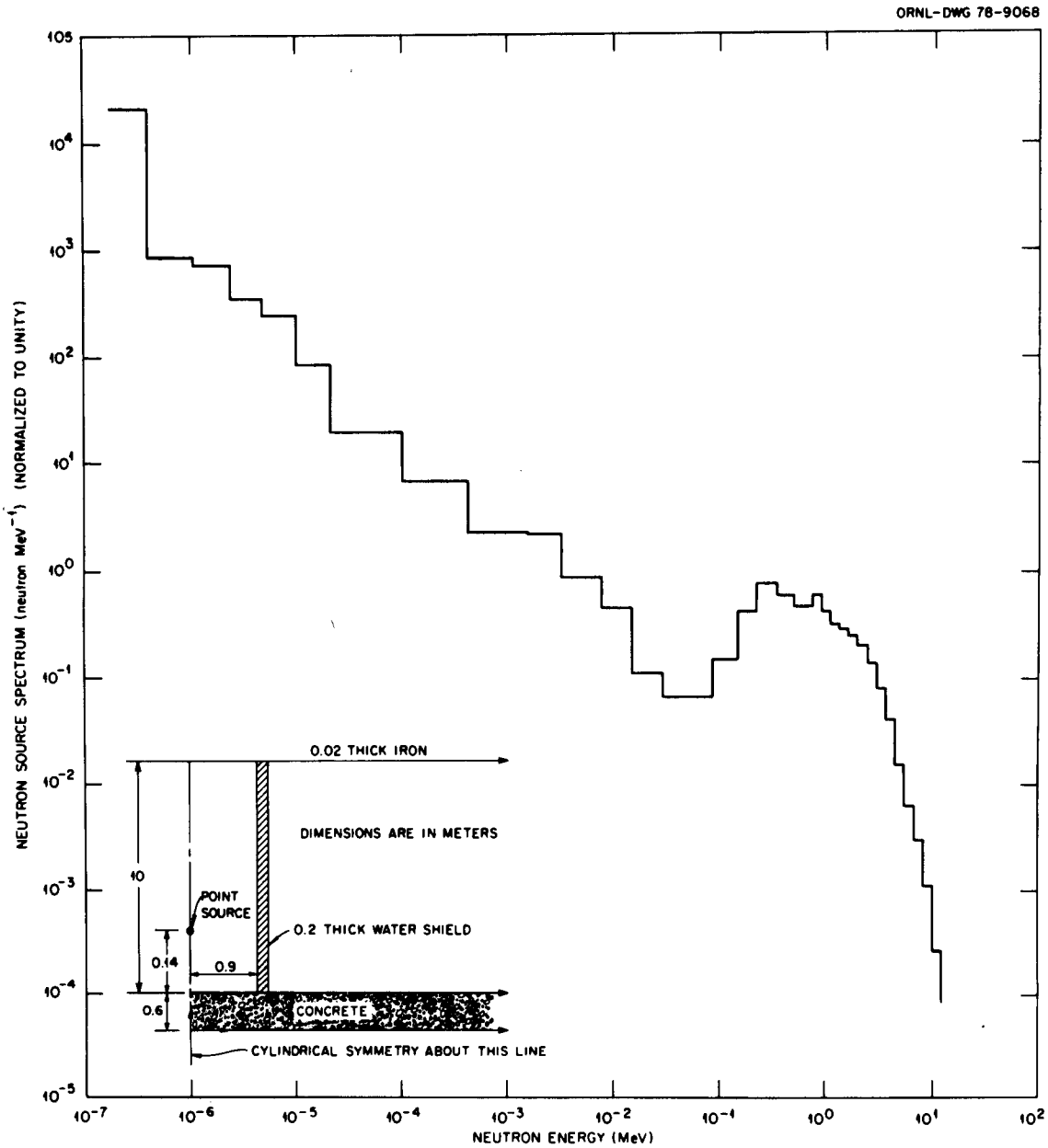

Fig. 1. Schematic diagram of calculational geometry and neutron source spectrum used in the calculations.

of 20 g cm^{-2} and to be in the same position (i.e., have the same radial centerline value) as the water shield shown in Fig. 1.

The calculations of the neutron and photon fluences were carried out with the two-dimensional discrete ordinates code DOT.[2] The auxiliary code GRTUNCL[7] was used to obtain a first collision source for use in DOT to avoid ray effects. The DOT calculations were carried out using an S_8 angular quadrature.

The neutron and photon cross section data were obtained from the DLC-37 compilation (based on ENDF/B-IV) of the Radiation Shielding Information Center of the Oak Ridge National Laboratory.[8] For use here, the 100-neutron, 21-photon group in DLC-37 were collapsed to 35-neutron and 21-photon groups. In all cases, the cross sections were represented by a P_3 expansion.

The neutron and photon fluence-to-absorbed dose conversion factors as a function of energy were taken from Refs. 3 and 4, respectively.

III. COMPARISON OF CALCULATED AND EXPERIMENTAL DATA

The experimental data are shown by the plotted points in Fig. 2. The various data points correspond to film badges and nuclear accident dosimeters analyzed by different groups in the Union Carbide Nuclear Division.[1] In the experiment, film badges were placed at shoulder height (1.4 m) both on the anterior and posterior of the unshielded manikins and at waist height (1.1 m) only on the anterior of the shielded manikin. The absorbed dose from the anterior and posterior film badges was given in Ref. 1 only after it had been corrected to centerline absorbed dose and these are the data shown in Fig. 2. The estimated error[1] on all of

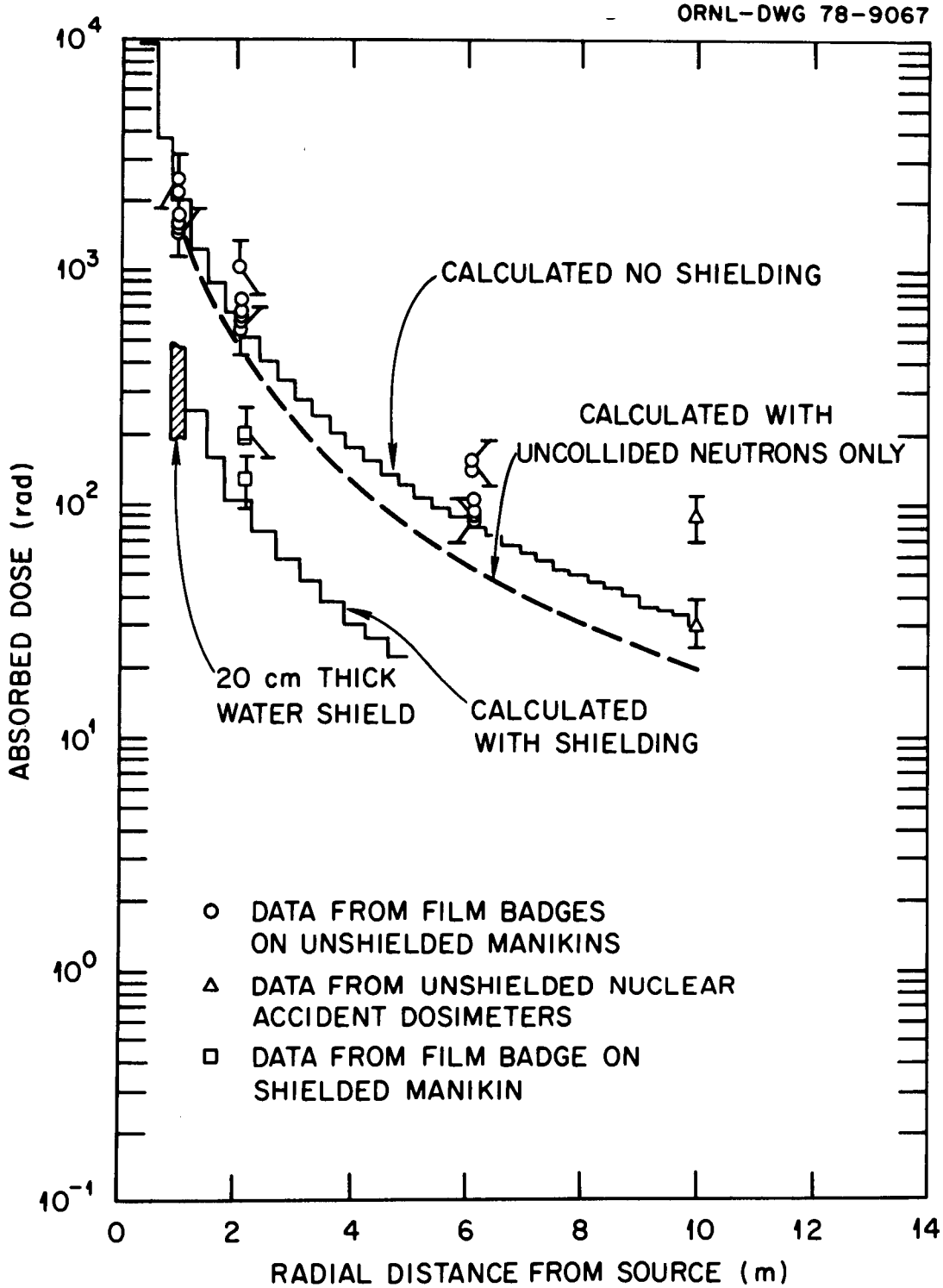

Fig. 2. Absorbed dose vs radial distance, r, (see Fig. 1) from the source for shields of iron, concrete, and water.

the data points is ± 25%. Error bars are shown in the figure only on the highest and lowest data points. The rather large spread in the data points is presumably due to the analysis methods used by the various groups.

The calculated results obtained by the method of discrete ordinates is shown by the histograms in Fig. 2. The upper histogram is obtained when the water shield is included. The two histograms are calculated at a distance above the floor corresponding approximately to the position at which the data were taken.

Also shown in the figure for comparison purposes - dashed line- is the calculated result if only the uncollided neutron fluence is considered. This dashed line is obtained from the equation

$$D_u(R) = \int_{\substack{\text{all}\\ \text{energy}}} \frac{1}{4\pi R^2} S(E)F(E)dE \quad , \qquad (1)$$

where

$D_u(R)$ = the absorbed dose due to uncollided neutrons at a spherical distance, R, from the source,

$S(E)$ = the number of neutrons per unit energy from the source,

$F(E)$ = the neutron fluence-to-absorbed dose conversion factor as a function of energy.

The upper histogram in Fig. 2 is in very good agreement with the experimental data considering the rather wide spread in the data. The difference between the upper histogram in Fig. 2 and the dashed line is a measure of the effect of the concrete floor and iron roof on the transport calculations. The difference between the histogram and the

dashed durve is not large, but it is clear that the effects of the floor and roof must be included in the transport calculation to obtain agreement between the calculated and experimental data.

The lower histogram in Fig. 2 is again in good agreement with the experimental data. Thus, the calculational procedure provides an accurate estimate of the absorbed dose when shielding is both present and absent. On the basis of this evidence, it seems fair to conclude that in accident situations such as that considered here a very good estimate of the absorbed dose to personnel can be obtained by calculations provided that the source spectrum and source normalization is known.

In Fig. 3 the absorbed dose behind shields of water, concrete, and iron are compared. The radial shield thickness was, in all cases, 20 g cm^{-2}. The results for the water shield are the same as those shown in Fig. 2. As the figure indicates the absorbed dose behind the iron and concrete shield are comparable but are considerably larger than the absorbed dose behind the water shield. This is presumably due to the fact that water contains a rather large amount of hydrogen and hydrogen is a very effective shield for low-energy neutrons. There are no experimental data with which to compare the iron and concrete results but the results in Fig. 2 do serve to indicate that the absorbed dose varies appreciably with shield material and that this effect can be taken into account in the calculations.

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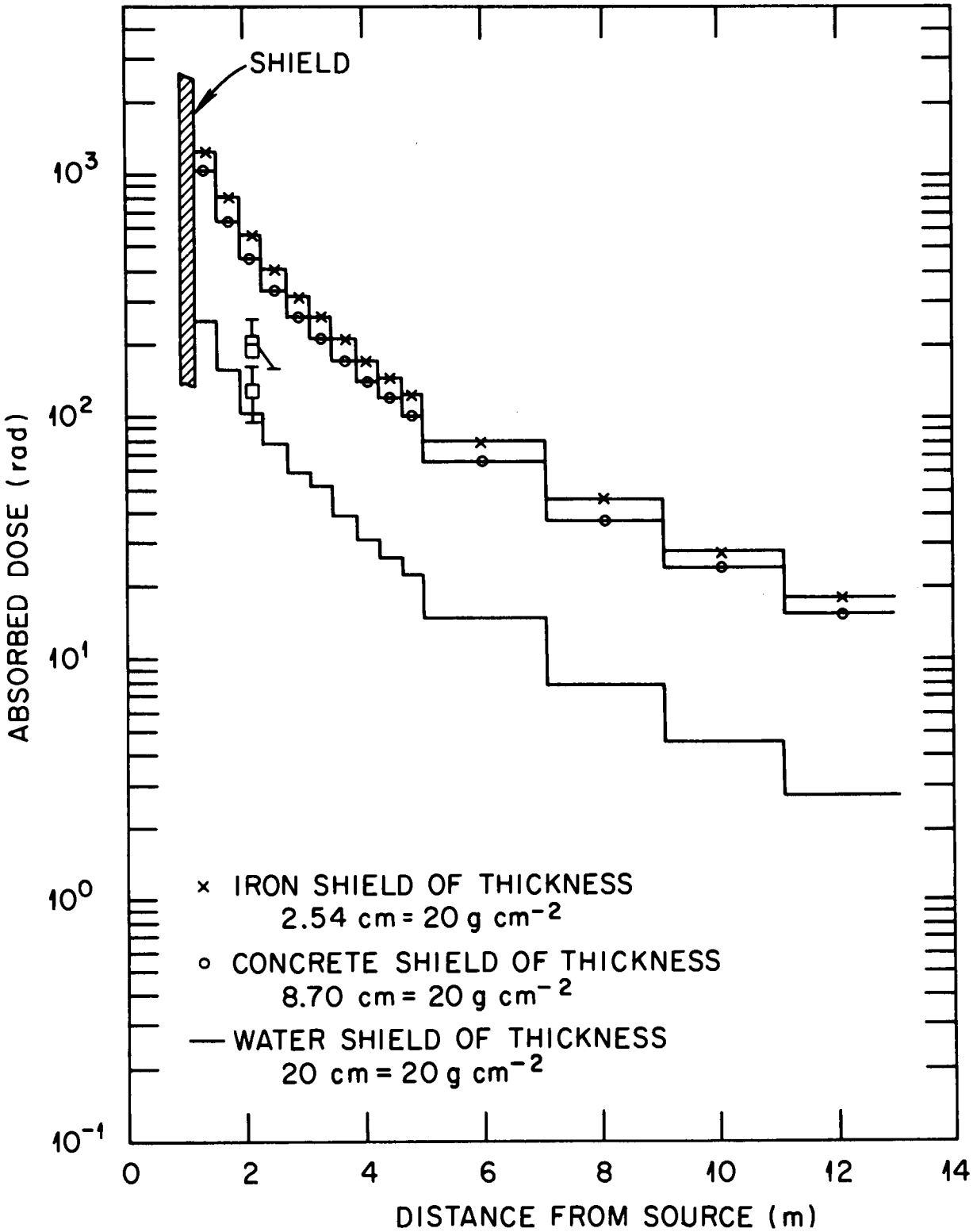

Fig. 3. Absorbed dose vs radial distance, r, (see Fig. 1) from the source.

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