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ATTENUATION BY WATER OF RADIATIONS FROM A SWIMMING POOL TYPE REACTOR

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Bulk Shielding Facility

Applied Nuclear Physics Division

ATTENUATION BY WATER OF RADIATIONS FROM A SWIMMING POOL TYPE REACTOR*

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Date Issued

SEP 7 1955

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* The gamma-ray data presented in this report supersedes the data published in ORNL-CF-54-6-165.

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ABSTRACT

The water attenuation of the gamma-ray dose rate and thermal-neutron flux from the Bulk Shielding Reactor (Swimming Pool Reactor) was measured out to distances of 770 and 470 cm, respectively. Beyond a distance of 200 cm from the reactor, the slopes of plots of the two measurements are similar; this results from the fact that essentially all neutrons beyond 200 cm are photoneutrons produced by gamma rays in the deuterium occurring naturally in the pool water. With the photoneutrons subtracted, the true thermal-neutron attenuation curve is exponential with a relaxation length that varies between 6 and 9 cm over a reduction of 10^{10}. From these measurements it was determined that for 1-megawatt operation the thermal-neutron leakage flux at the reactor surface (the reactor was loaded with 28 fuel elements containing a total of 3.6 kg of U^{235}, and was water-reflected) was 1.1×10^{13} neutrons/($cm^2 \cdot sec$); the gamma-ray dose rate at the reactor surface was 8×10^7 r/hr.

INTRODUCTION

A swimming pool type reactor depends upon the water in the reactor pool to act as the radiation shield. Other reactors utilizing the same type of core use water as the initial portion of the shield. Therefore, part of the basic design data needed for such reactors is the attenuation of the reactor radiations in water. Measurements of the attenuation of the gamma-ray dose rate and the thermal-neutron flux from the Bulk Shielding Reactor[1] (a swimming pool reactor) have now been made through greater thicknesses of water than would be required for any practical shield. It should be mentioned, however, that the results presented in this report pertain only to the direct penetration of radiation, no consideration being given to the possibility of activities transported through the water.

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1. W. M. Breazeale, "The New Bulk Shielding Facility at Oak Ridge National Laboratory," ORNL-991 (May 8, 1951). A later and more complete description of this reactor was contained in the pool-type reactor section of the Geneva Conference Report.

I. DESCRIPTION OF THE REACTOR

The ORNL Bulk Shielding Reactor (BSR) consists of an assembly of MTR-type[2] fuel elements that can be arranged in many critical configurations. The fuel is enriched uranium contained in aluminum-clad plates, several of which are supported in a rigid assembly to make a fuel element. The fuel plates are arranged in such a way as to allow free water flow between the individual plates.

The BSR, which can be operated at powers up to 1 megawatt, is controlled by means of two safety rods and one regulating rod. The elements through which the control rods move contain only half the normal number of fuel plates, and, therefore, only half the normal amount of fuel. The BSR is suspended in a 40 ft long by 20 ft wide by 20 ft deep pool of water which acts as the reactor coolant, moderator, shield, and reflector.

All radiation levels presented in this report are expressed in terms of the reactor power, which was determined by the usual method of mapping the thermal-neutron flux throughout the reactor. From the integrated thermal-neutron flux the fission rate was determined. The power was then derived from the known energy release per fission for this reactor.[3] The reactor power varied between 0.1 and 10^6 w.

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2. C. D. Smith, F. W. Droston, F. Kerze, Jr., "Production of Fuel Assemblies for the Materials Testing Reactor Mock-up Critical Experiments," ORNL-951 (April 9, 1951).

3. J. L. Meem, Nucleonics 12, No. 5, 62 (1954).

The particular reactor loading used for these measurements consisted of 28 fuel elements containing a total of 3.6 kg of U^{235} (Fig. 1).

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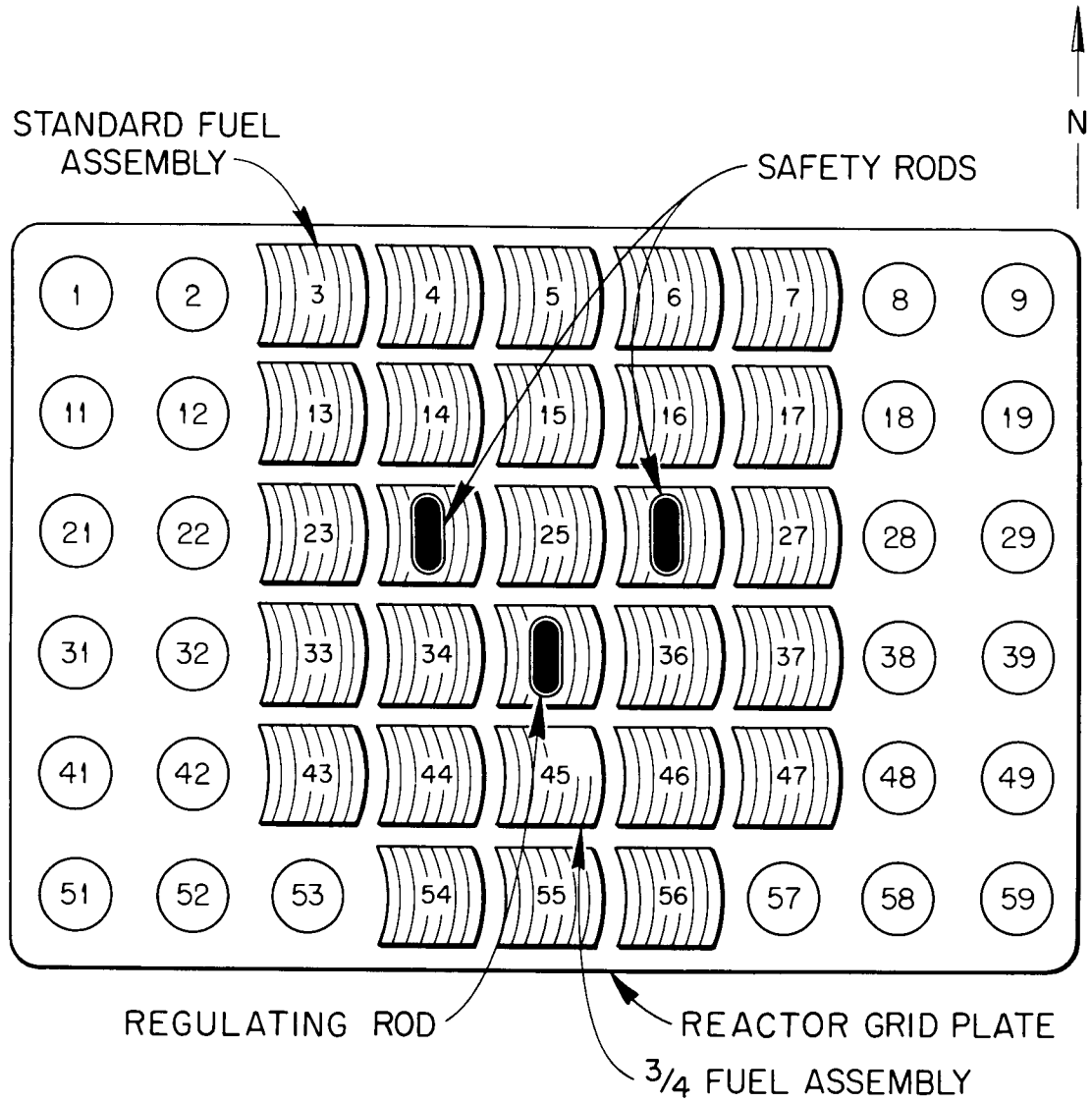

Fig. 1. Bulk Shielding Reactor Loading 33.

II. DESCRIPTION OF RADIATION DETECTORS

Gamma-Ray Detectors

Gamma-ray dose rate measurements were made with a set of ion chambers utilizing the Bragg-Gray principle. These chambers were designed by J. L. Meem and calibrated by L. H. Ballweg.[4] For such chambers the gamma-ray dose is proportional to the voltage produced by the ionization current flowing through a precision resistor. An electrometer[5] determines the null condition for which the voltage is just balanced by a known bucking voltage. The conversion from voltage to dose was taken from the report by Ballweg and Meem.[4] A check of the absolute calibration for one of the chambers was made with a radium source previously calibrated by the Bureau of Standards, and, for favorable geometries, agreement to within 3% was obtained. The center of detection calculated by Hungerford was used.[6] The measurements were made with three ion chambers of different sensitivities in order to cover a large dose range. The size of the instruments were: 50-cc volume, 10^{12}-ohm resistor; 50-cc volume, resistor variable from 10^{6} to 10^{10}-ohms; and 900-cc volume, 10^{12}-ohm resistor.

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4. L. H. Ballweg and J. L. Meem, "A Standard Gamma Ray Ionization Chamber for Shielding Measurements," ORNL-1028 (July 9, 1951).

5. F. M. Glass, "A Simple Low Drift Electrometer," ORNL-486 (Dec. 30, 1949).

6. H. E. Hungerford, "Center of Detection Calculations for Neutron Counters and Ion Chambers," ORNL-CF-51-5-177 (May 14, 1951).

Thermal-Neutron Detectors

For thermal-neutron flux measurements, three different detectors were utilized. Close to the reactor a 3-in.-dia single-plate U^{235} fission chamber was used; this instrument has little gamma-ray background response and the center of detection is well known. At greater distances where more sensitivity was required, the measurements were made with a large BF_3 proportional counter (12 in. long and 2 in. in diameter). The center of detection was again taken from Hungerford.[6] For both the chamber and counter a conventional amplifier and scaler were used.

Absolute calibration of the thermal-neutron flux was provided by exposure of indium foils with and without cadmium covers. The difference of such readings is proportional to the thermal-neutron flux. The foil calibrations were made in the ORNL Standard Pile with due correction being made for the change from a water to graphite medium.[7]

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7. E. D. Klema and R. Ritchie, "Preliminary Results on the Determination of Thermal Neutron Flux in Water," AECD-3174 (April 24, 1951); also see E. D. Klema, R. H. Ritchie, and G. McCammon, "Recalibration of the X-10 Standard Graphite Pile," ORNL-1398 (Oct. 17, 1952) (declassified Aug. 6, 1953).

III. EXPERIMENTAL RESULTS AND DISCUSSION

Gamma-Ray Measurements

For the gamma-ray attenuations, a new set of fuel elements was available so that the fission product gamma-ray activity was nil. As the distance between reactor and ion chamber was increased the reactor power was raised. At all points a check was made for residual gamma-ray activity by a background reading with the reactor at zero power.

The gamma-ray activity induced in the water was greatly reduced by the recent introduction of demineralized water into the BSF reactor pool. Previously, the process water, which was treated with sodium chromate, gave rise to appreciable induced activities. With the pure demineralized water, it became necessary to measure the background due to induced activities only at very large distances (greater than 500 cm). This background was measured with a 5-in.-thick, 4-ton lead slab between the reactor and detector. Any dose measured with the slab in place was due to activity in the water since penetration through the lead by gamma radiation was less than 0.3%. Rotation of the lead slab showed that no appreciable radiation scattered around the slab since the dose was essentially independent of the angle of rotation (up to 75 deg). An attempt to establish an optimum position for the lead between the detector and reactor led to little apparent variation in dose with position, so the slab was finally placed about halfway between the detector and reactor. The water activity backgrounds were less than 25% until the very last point, which had a 60% background.

The effects of structure around the reactor were small because of the large size of the BSR pool. The closest concrete was in the pool floor, 8 ft away from the reactor for most of the measurements and about 5 ft for the greatest separations. The pool walls, also of concrete, were 10 ft distant, while the surface of the pool water was 16 ft above the reactor.

Overlapping gamma-ray measurements were made with the three ion chambers as indicated by the different points in Fig. 2. Measurements with the other chambers were normalized to those with the 50-cc 10^{12}-ohm chamber, which was regarded as the standard instrument after calibration by the radium source.

Thermal-Neutron Measurements

Thermal-neutron flux measurements were made in the same manner as the gamma-ray dose rate measurements except that no background measurements were necessary. At all points the reactor power was adjusted so that no excessive gamma-ray fields were allowed to perturb the BF_3 proportional counter. The thermal-neutron flux data obtained was normalized to the indium foil measurements taken at the positions indicated in Fig. 3. The point at the reactor face (no water thickness) was extrapolated from the thermal-neutron flux in the reactor core as determined in the power calibrations. The error in the flux for this one point might be as large as 10%.

For the points of greatest attenuation, an additional uncertainty was introduced in the reactor power determination. At a 1-megawatt reactor power, possible deviations of 10 to 15% from the nominal reactor power give rise to the large errors shown in Figs. 2 and 3 at the largest attenuations.

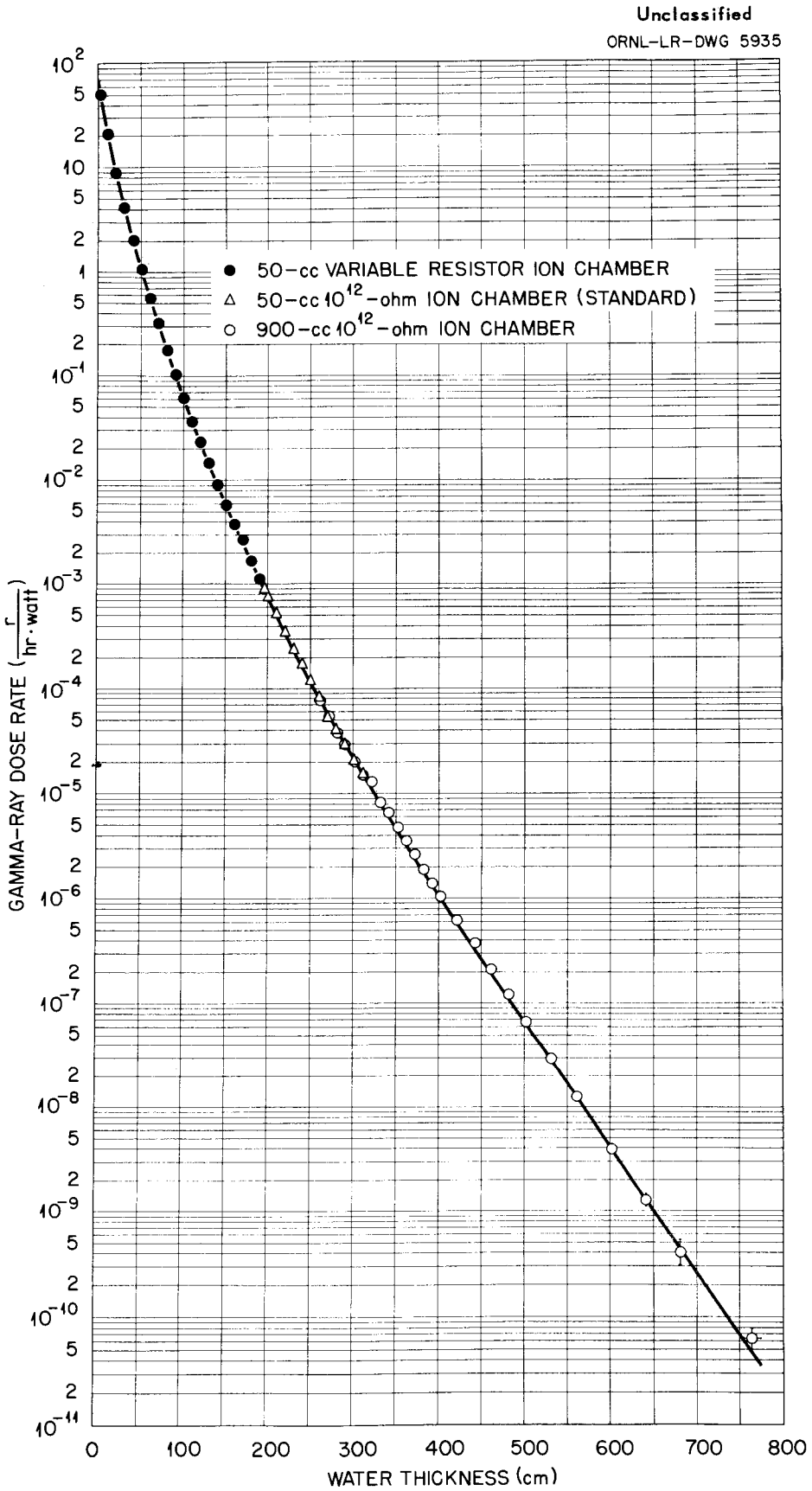

Fig. 2. Gamma-Ray Dose Rate as a Function of Distance from the Bulk Shielding Reactor.

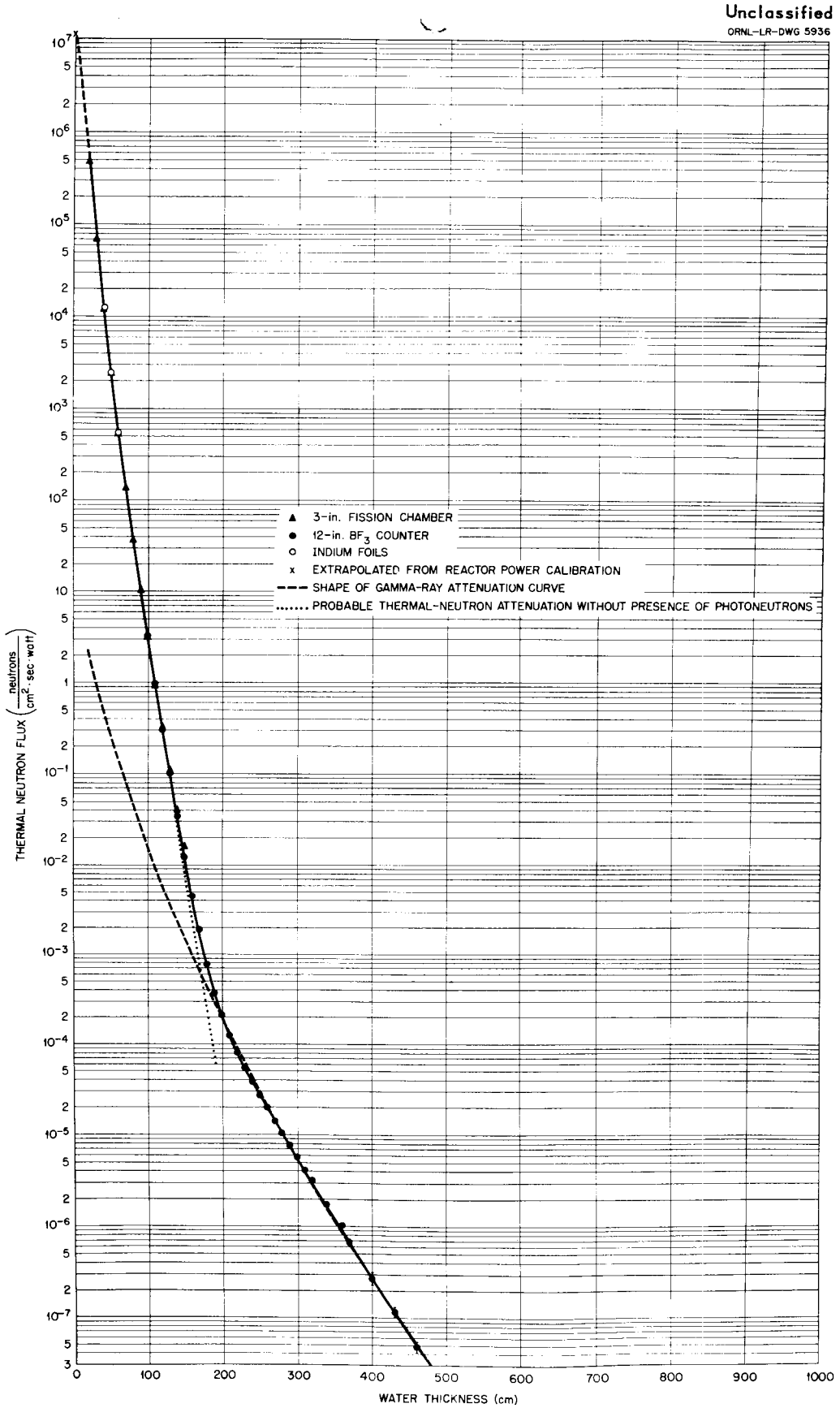

Fig. 3. Thermal-Neutron Flux as a Function of Distance from the Bulk Shielding Reactor.

Discussion

The curves shown in Figs. 2 and 3 may be used to determine the leakage flux at the reactor face. Thus at 1 megawatt the gamma-ray dose rate at the surface of the reactor is seen to be about 8×10^7 r/hr while the thermal-neutron flux is about 1.1×10^{13} neutrons/($cm^2 \cdot sec$).

The two curves are plotted on the same scale, even though the more penetrating gamma radiation extends to greater distances, so that the similarity between the curves beyond 200 cm may be observed. In Fig. 3, the dashed curve has the same shape as the gamma-ray dose rate attenuation curve from Fig. 2. The similarity in slopes is due to the fact that essentially all neutrons beyond 200 cm are photoneutrons produced by the gamma rays in the deuterium naturally occurring in the pool water. The difference curve, shown as a dotted line in Fig. 3, should correspond to the actual thermal-neutron attenuation. This attenuation is exponential with a relaxation length (reduction to 1/e) which varies between 6 and 9 cm over a reduction of 10^{10}.