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ATTENUATION PROPERTIES OF HANFORD PILE SHIELD MATERIALS

BY

W. L. BUNCH AND R. L. TOMLINSON

**PILE TECHNOLOGY SECTION
ENGINEERING DEPARTMENT**

SEPTEMBER 1, 1955

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ATTENUATION PROPERTIES OF
HANFORD PILE SHIELD MATERIALS

By

W. L. Bunch and R. L. Tomlinson

Pile Physics Unit
Pile Engineering Sub-Section

September 1, 1955

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ATTENUATION PROPERTIES OF
HANFORD PILE SHIELD MATERIALS

I. INTRODUCTION

An extended program of measurements has been carried out at Hanford to determine the attenuation characteristics of various concretes which might be used as inexpensive yet efficient shields for large, stationary, production piles. The original Hanford piles have biological shields consisting of laminations of iron and masonite which have performed satisfactorily; however, the high costs involved in machining and fabricating the components make the use of this type shield on a new pile undesirable. Concrete has been used successfully for some time to provide low cost shields; with the addition of a high density aggregate it can provide relatively inexpensive yet efficient shields for production reactors. The various concrete mixes whose attenuation characteristics were measured were devised as a result of a study by the Design Section to utilize local ores to minimize transportation costs. The concretes tested included iron-limonite, magnetite-limonite, and magnetite aggregates. Limonite is an iron ore having relatively high water content, whereas the iron ore magnetite has a greater density but lower water content. Iron punchings were used to increase the effective shield density in the more critical areas of the shield.

The objectives of the measurements have been twofold: first to provide the attenuation data so that shields could be designed utilizing the materials and mixes which had been used in the measurements, and second, to interpret the data so that other materials and mixes could be evaluated theoretically without the necessity of the expensive and lengthy experimental evaluation. Structural, radiation damage, and other physical properties of the concretes which are essential for design application were studied concurrently and are reported elsewhere.

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II. SUMMARY

The high density concretes tested were all found to have attenuation characteristics suitable for use in various parts of the KE and KW Pile Shields. Use of the concretes at the two K Piles is estimated to have saved eleven million dollars over the cost of iron-masonite laminated shields such as were used on the original piles. Based on the results of these attenuation data, it is possible to predict, within limits, the behavior of other type aggregate concretes. It has been shown that relatively low water content concretes have satisfactory shielding characteristics, although an evaluation of the lower limit was not made. The measurements show the feasibility of using the DR wells for bulk shielding measurements and also indicate the limitations of these facilities. A qualitative method of evaluating the entire neutron spectrum from foil flux traverse data is discussed. The following table summarizes the results of the study. These attenuation factors apply to a particular spectrum emanating from a graphite-uranium lattice which has been modified by a two-foot thick graphite reflector and an eleven-inch thick iron thermal shield. The neutron factors were obtained from fast neutron data and apply to the entire neutron spectrum only when an equilibrium condition exists which is governed by the fast component.

<u>Shield Material</u>	<u>Density gm/cc</u>	<u>Inches for Factor of Ten Attenuation</u>	
		<u>Neutrons</u>	<u>Gamma</u>
Iron-Masonite	4.27	6.37	6.97
Iron-Limonite	4.15 - 4.24	7.06	7.62
Magnetite-Limonite	3.23 - 3.40	8.06	9.52
Magnetite	3.29	8.63	10.26

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III. DETAILS

A. Background

Biological shields, as the name implies, are placed around piles to protect personnel from the radiations which result from the fission process. Of the several radiations which do result from fission only neutrons and photons are both sufficiently penetrating and biologically damaging to constitute a shielding problem. The heavy, charged fission fragments and the beta particles are readily stopped by the pile and reflector. Neutrinos on the other hand are so penetrating, i. e., have such a small interaction probability, that they are not biologically damaging. Thus in studying shield materials it is necessary to determine the attenuation characteristics for gamma rays and neutrons.

The gamma ray and neutron spectrum which results from the fission process in a pile is very complex when it reaches the shield. Gamma rays are not only a direct result of the fission process but also result from inelastic scattering of fast neutrons, capture of thermal neutrons, decay of radioactive nuclei, Bremsstrahlung, and annihilation radiation; Compton scattering complicates the spectrum even more. The energy spectrum of the prompt fission neutrons is fairly well known, as is the distribution of delayed neutrons from various fissionable elements; however, the spectrum which is incident on a shield is a function of the particular pile structure. It is the attenuation of this complex spectrum which must be evaluated.

Shields for mobile reactors must be optimized in terms of weight and size whereas biological shields for large, stationary, graphite moderated piles are optimized primarily on the basis of cost. Some consideration must also be given to the thickness because of problems which arise in regard to other pile components. The shield for stationary reactors needs to be inexpensive yet relatively efficient. The iron-masonite laminated shields used on the original Hanford piles are efficient,

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but the high initial cost of materials, machining, fabrication, and erection of this type of shield makes it very expensive.

Ordinary Portland cement concrete has been used in the past as an inexpensive shield for low-power piles even though its shielding efficiency is relatively low. With the addition of high density aggregates, such as iron and iron ore, it is possible to have a concrete which is a relatively efficient shield and is also relatively inexpensive. To minimize the shipping cost of the ores it is necessary to use ores which are mined close to the pile site. H. S. Davis of the Design Section selected several ores mined in the northern and western states and designed several structural concretes suitable for either conventional or "Prepakt" placement whose elemental composition and density met standards set up by the Pile Technology Section. An extended program of testing was undertaken before these concretes were used in actual pile shields; this report is concerned only with the radiation attenuation properties of these concretes.

Chronologically the first successful shielding measurements at Hanford were made by R. L. Dickeman in 1950. These measurements were taken in the imperforate Hanford iron-masonite laminated shield in order to provide a standard to which other facilities and measurements could be compared. The attenuation measurements of the concretes using the bulk shielding facility on the top of the DR Pile were begun in August of 1951 under the direction of R. L. Dickeman and J. A. Berberet. This facility permits flux detectors to be placed in slabs of 36" to 42" diameter *square* through a thickness of four feet. Sufficient early data were obtained to permit the use of iron-limonite concrete for the top shield of the C Pile; the C Pile side shields are of laminated iron and masonite. Since the construction of the C Pile the structural data have proven satisfactory⁽¹⁾

(1) HW-32456, "Troutdale Investigation of High Density Concrete," H. S. Davis, 7-16-54.

and the attenuation data have been completed permitting the use of the tested concretes in KE and KW piles.

B. Methods of Measurement

1. Gamma Detection

The gamma measurements were made primarily with ionization chambers and in part by radiographic film. Three different types of chambers were used: (1) 10 cc chambers having graphite walls, (2) 10 cc chambers having parallel plate magnesium walls, and (3) a 1000 cc chamber with a cylindrical bakelite shell. The chambers were all open to atmospheric air pressure, and readings were corrected for temperature and pressure as necessary. Either a Beckman RXC-2 or a Model V micro-micro ammeter was employed as the current measuring device.

The radiographic film used was 5" x 7" type K X-ray film enclosed in cardboard exposure holders. Very few film measurements were taken because of the limited range and versatility of this method. In every case when measurements were attempted a set of calibration films which had been exposed to a radium source were also processed. The sensitive spectral response of film makes the value of the absolute intensity meaningless, although some information was gained from the measured attenuation length (assuming a constant spectrum). Fortunately the film measurements agreed both in intensity and relaxation length with the chamber measurements.

The ionization chambers were also calibrated to a radium source (See Figure 1).⁽²⁾ Since the chambers are not as sensitive to spectral differences as are the film measurements, the intensity values recorded by the calibrated chambers should be reasonably accurate. No experimental determination was made of this spectral response.

The neutron sensitivity of the chambers was also checked by the use of radium-beryllium and polonium-boron sources. It was demonstrated

(2) HW-5355-T, "Personal Notes," W. L. Bunch, March 1953.

FIGURE 1
 CALIBRATION CURVES
 10 CC MAGNESIUM CHAMBER "B"
 NEUTRON SOURCE $1.42 \times 10^8 \text{ n/sec Po-B}$
 GAMMA SOURCE 1.0078gms RADIUM

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by the use of these sources, which are neutron-gamma and neutron only emitters respectively, that these chambers were quite insensitive to fast neutrons (See Figure 1). In order to eliminate as much as possible the effects of thermal neutrons the chamber containers were coated on the outside with a boron carbide solution which virtually eliminated induced activation of the chamber.

The chambers were placed in slots within the slabs as illustrated in Figures 2 and 3. Only one or two points were obtained each time; hence several months were required for a single traverse. The chamber on top of the well was subjected to much scattered background radiation and was therefore shielded by a thick lead cave. Except for an occasional case of insulator failure the chambers were trouble-free and very satisfactory.

2. Neutron Measurement Methods

To selectively measure representative portions of the neutron energy range, three different thin foil detectors have been employed: (1) bare gold foils which are highly sensitive to neutrons of thermal energies and also have an appreciable resonance cross section at about 5 electron volts, (2) gold foils of the same type described above but having been sealed in a cadmium capsule 20 mils in thickness; the cadmium effectively removes all possible activation below 0.5 ev but permits activation of the 5 ev resonance, and (3) sulfur either in the powdered form or in the form of compressed sulfur pellets, covered with a 20 mil thickness of cadmium to effectively eliminate the $S^{34}(n, \gamma)S^{35}$ and $S^{36}(n, \gamma)S^{37}$ reactions. This detector takes advantage of the $S^{32}(n, p)P^{32}$ threshold reaction which has been only minutely affected by the presence of the 20 mil thickness of cadmium. The threshold for this (n, p) reaction is about 1 Mev, the cross section increasing rapidly to a maximum at about 3 Mev. The gold and cadmium covered gold detectors have relatively high nuclear cross sections, whereas the cadmium sulfur detector has a cross

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FIGURE 2
PLACING GAMMA CHAMBER IN CONCRETE SLAB

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FIGURE 3
PLACING GAMMA CHAMBER IN IRON - MASONITE SLAB

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section of the order of one per cent of that of the first two, thus limiting its use to measurements of high intensities. This fast neutron detector has been satisfactorily employed in the inner half of the shield, permitting data extrapolation to greater shield thicknesses. Fast neutron dosimeters were not considered to be suitable for these measurements because of the high gamma background and the extended operation periods of the pile. Methods of foil preparation and beta counting techniques are presented in detail in references 3 and 4.

Detectors were placed in the test slabs in the shape of a cross as shown in Figures 4 and 5 in order to obtain both front to rear and side to side traverses of the test facility. These traverses provide an indication of the streaming present due to air cracks, to the steel test well liner, and to the difference in shielding effectiveness of the main pile shield from that of the shield slabs being tested. The 5 detector by 5 detector cross pattern gives a 9 point check on the center point, which is used in the final attenuation curves. To check the repeatability of the measurements, three measurements were made on each type of shielding material tested in the following order: (1) bare gold foils, (2) bare gold foils and cadmium covered gold foils, and (3) bare gold foils and cadmium covered sulfur foils. When cadmium covered foils were used with bare gold foils, the bare foils were placed no closer than four inches from cadmium covered detectors in order to prevent depression of the thermal flux in the region where it was being measured. A bare gold detector was used in all experiments as a power level monitor for use in normalizing all of the attenuation data for one shielding material. For a typical iron-masonite traverse there were 370 detector positions, whereas a typical concrete traverse consisted of 110 detector positions.

A great deal of care was exercised in handling and obtaining data from the various detectors. Rigid specifications were placed on detector purity, foil thickness, and foil diameter.

-
- (3) HW-33121, "Procedure for Radioactive Fire Propagation and Counting," R. L. Tomlinson, Jan. 1, 1955.
 - (4) HW-33122, "Two Techniques for Fast Neutron Detection Using the S^{32} (n, p) P^{32} Reaction," R. L. Tomlinson, Sept. 1, 1953.

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FIGURE 4
LOCATION OF NEUTRON FOIL SLOTS IN IRON - MASONITE SLAB

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FIGURE 5
LOCATION OF NEUTRON FOIL SLOTS IN CONCRETE SLAB

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C. Facilities

1. Bulk Shield Facility - DR Pile

The Shield Test Facilities at DR Pile consist of two large stepped wells located to the front of the vertical rod system in the top shield. These wells have six steps ranging from 36 inches square at the bottom to 42 inches square at the top. The wells are lined with one inch steel plates on four sides and a one-half-inch steel plate on the bottom. The purpose of this liner is to seal the facility entirely from the pile atmosphere and to prevent chipping of the biological shield surrounding the facility. A cross sectional diagram of one of the wells may be seen in Figure 6.

Due to the steel liner, the replica of the Hanford type shield used in this facility has one inch less masonite in the innermost lamination and one inch more of steel than does the normal biological shield; due to structural reasons it was necessary to use one-half inch of steel at the bottom of the innermost block, whereas the other one-half inch is attributable to the test well liner. Reducing the masonite thickness one inch was necessary to provide the alignment of succeeding laminations with the regular shielding of the pile. The wells are positioned 8 feet to the front of the pile centerline and 2-3/4 feet to either side of the centerline as shown in Figure 7. Access to the facilities is limited to pile outages which may be several weeks apart. Test slabs are removed from the facility by an electrically operated hoist. Special lifting apparatus has been designed to assist in this operation; the lifting mechanism for concrete shielding test slabs may be seen in Figure 8 and the lifting mechanism for iron-masonite slabs may be seen in Figure 3. The time consumed in removing the test slabs, making changes of detectors, and replacing the blocks into the wells is approximately 2 hours. Because of high activation of the test slabs in the lower part of the well, operation of the facility is seldom undertaken less than 12 hours after the reactor has been shut down. Since this radiation is of a reasonably short half-life (main contributor being Mn^{56}) safe levels are obtained after a reasonable waiting period. Little of the work has to be done remotely.

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FIGURE 6
CROSS SECTION OF BULK SHIELD FACILITY - DR PILE

FIGURE 7
PLAN VIEW OF TOP OF DR PILE SHOWING LOCATION OF
THE BULK SHIELD FACILITIES

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FIGURE 8
LIFTING MECHANISM FOR THE BULK SHIELD FACILITIES AT DR PILE

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While the pile is in operation streaming of neutrons through the vertical safety rod openings creates a considerable background which backscatters into the test wells. To eliminate this effect two paraffin blocks containing a 40 mil thickness of cadmium in the bottom have been fabricated. These blocks reduce the neutron scatter into the wells below a detectable level. Figure 9 shows the wells as they appear during pile operation with the paraffin slabs in place. The vertical safety rod openings are visible to the right of the test wells. The cylindrical hoppers concentrically located above each rod opening contain steel balls and are part of the pile safety system.

2. "A" Test Hole Step Plug Facility - D Pile

This facility consists of an unlined step plug in a liner-less stepped entry through the shield. It is located on the experimental side of the D Pile three feet nine inches to the front of the longitudinal centerline of the pile and one foot five inches below the vertical centerline. This particular step plug is of a sectional design with slots to allow neutron detectors to be placed at various positions throughout the entire step plug. The step plug matches the pile shield which is composed of alternate 3-3/4" iron and 4-1/2" masonite laminations. The entire plug is enclosed in the pile atmosphere by a gas seal at the face of the pile. Due to radiation hazards, removal of the step plug may be done only during a pile outage. Figure 10 is a diagram of the A test hole showing the 8-inch thermal shield and the successive iron-masonite laminations.

3. "Quickie" Irradiation Facility - F Pile

The E test hole at F Pile, a facility for charging and discharging small samples during operation, is located six feet eleven inches to the rear of the longitudinal centerline and one foot five inches below the vertical centerline. The hole consists of an aluminum thimble 150 inches long in which travels an aluminum rod with a graphite tip which contains eighteen 1-1/4" pockets spaced 1-3/4 inches apart from center to center (See Figure 11). Into these pockets various detectors are placed in one inch

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FIGURE 9
PARAFFIN SLABS TO ELIMINATE BACK SCATTERING INTO
THE BULK SHIELD FACILITIES AT DR PILE

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FIGURE 10
CROSS SECTION OF "A" TEST HOLE FACILITY - D PILE

CROSS SECTION OF IN-PILE PART

FIGURE 11.

QUICKY FACILITY - THE 'E' TEST HOLE AT F PILE

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long by 5/16-inch outside diameter capsules. The purpose of the facility is to obtain irradiations into the edge of the active zone and within the pile reflector.

The main advantage of this facility is that it is provided with exterior shielding for the push rod which permits charging and discharging during pile operation. This shielding entirely surrounds the aluminum rod except for a tapered slot on both top and bottom sides next to the face of the pile. When charging, samples are dropped into the upper slot one by one when the rod is in the proper position. Upon discharging, the rod is removed to the proper position and turned 180 degrees, allowing the samples to drop through the lower slot into a small portable cask. Further information on this facility may be found in HW-29125.⁽⁵⁾

D. Data

Table I presents the attenuation characteristics of the particular shielding materials tested for the Hanford pile flux spectrum as derived from least squares analyses of the data presented in the succeeding figures. These attenuation values are statistically accurate to $\pm 3\%$ with 90% confidence. Based on density variations the uncertainty in the content of any one elemental constituent appeared also to be $\pm 3\%$.

The flux traverse plots on the succeeding pages summarize all of the attenuation data obtained during the program to determine the shielding properties of high density concretes relative to those of iron-masonite. Each point on a neutron flux traverse plot from the bulk shield test facility is representative of a minimum of nine foils; repeat irradiations of bare gold foils were made for every shielding material tested. The gamma intensity is given in relative units normalized to unit intensity incident on the biological shield materials. The neutron intensity plots

(5) HW-29125, "Measurement of Neutron Flux in Graphite Reflector," R. L. Tomlinson, Sept. 1, 1953.

TABLE I
SUMMARY OF RESULTS

Shield Material	Calculated dry density gm/cm ³	Measured dry density gm/cm ³	Calculated Fe/H ratio	Calculated ratio of heavier elements to hydrogen	Relaxation lengths** cm		Abs coeff cm ⁻¹ **	
					n	g	n	g
Iron-Masonite	4.27	----	84	97	7.03	7.69	.142	.13
Iron-Limonite Conventional*	4.18	4.24	108	169	7.88	8.40	.127	.119
					7.69		.130	
Iron-Limonite Prepakt*	4.17	4.15	100	170	7.88	8.62	.127	.116
Magnetite-Limonite Conventional*	3.46	3.40	74	202	9.01	10.5	.111	.095
Magnetite-Limonite Prepakt*	3.29	3.23	55	142	8.93	10.5	.112	.095
					8.77		.114	
Magnetite Conventional*	3.35	3.29	123	271	9.52	11.2	.105	.091

*Prepakt refers to the method of placement in which aggregate is packed dry into the forms and the cement grout is pumped in to fill the voids and form the bond. Conventional refers to the more familiar method of mixing all of the constituents together before placement into the forms. In the case of the iron-limonite conventional and the magnetite-limonite prepakt, two independent sulfur foil transverse were taken and analyzed separately. The differences in these measurements was within the expected statistical uncertainty.

**These attenuation factors apply to the particular spectrum emanating from a graphite-uranium lattice which has been modified by a graphite reflector and iron thermal shield. The neutron factors were obtained by least squares fits of sulfur (n,p) data and apply to the entire neutron spectrum only in regions where an equilibrium condition exists.

FIGURE 12
NEUTRON ATTENUATION IN THE HANFORD IRON MASONITE SHIELD AS
MEASURED IN THE "A" TEST HOLE, D PILE

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FIGURE 13
NEUTRON ATTENUATION IN THE HANFORD IRON - MASONITE
SHIELD AS MEASURED IN THE DR PILE BULK SHIELDING TEST
FACILITIES

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FIGURE 14
NEUTRON ATTENUATION OF REFLECTOR, THERMAL SHIELD AND
IRON - MASONITE SHIELD

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FIGURE 15
NEUTRON ATTENUATION IN IRON - LIMONITE CONVENTIONALLY
PLACED CONCRETE AS MEASURED IN THE DR PILE BULK SHIELDING
FACILITY

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FIGURE 16
NEUTRON ATTENUATION IN IRON - LIMONITE "PRE PAKT" CONCRETE
AS MEASURED IN THE DR PILE BULK SHIELDING FACILITY

FIGURE 17
NEUTRON ATTENUATION IN MAGNETITE - LIMONITE CONVENTIONALLY
PLACED CONCRETE AS MEASURED IN THE DR PILE BULK SHIELDING
FACILITY

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FIGURE 18
NEUTRON ATTENUATION IN MAGNETITE - LIMONITE "PRE PAKT"
CONCRETE AS MEASURED IN THE DR PILE BULK
SHIELD TEST FACILITIES

FIGURE 19
NEUTRON ATTENUATION IN MAGNETITE CONVENTIONALLY PLACED
CONCRETE AS MEASURED IN DR PILE

FIGURE 20
TYPICAL BARE GOLD LATERAL TRAVERSES IN HEAVY AGGREGATE
CONCRETE - DR PILE BULK SHIELDING FACILITY

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FIGURE 21
GAMMA INTENSITY IN IRON MASONITE
DR PILE BULK SHIELDING FACILITY
11 1/2" IRON THERMAL SHIELD

FIGURE 22
GAMMA INTENSITY IN IRON LIMONITE CONCRETE
DR PILE BULK SHIELDING FACILITY
11" IRON THERMAL SHIELD

FIGURE 23
GAMMA INTENSITY IN MAGNETITE LIMONITE CONCRETE
DR PILE BULK SHIELDING FACILITY
11" IRON THERMAL SHIELD

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FIGURE 24
GAMMA INTENSITY IN MAGNETITE CONCRETE
DR PILE BULK SHIELD FACILITY
11" IRON THERMAL SHIELD

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FIGURE 25
RELATIVE GAMMA ATTENUATION OF TESTED MATERIALS

have been normalized to a single pile power level. In every case the abscissa is in inches from the interface between the graphite reflector and the iron thermal shield. The data for these plots are in the following notebooks:

HWN-1158
HW-2392-L
HW-3808-T
HW-4504-T
HW-4505-T
HW-5169-T
HW-5355-T
HW-6510-T
HW-6623-T

E. Validity of Measurements

This section discusses the accuracy of the measurements, the errors which are known to exist, and the approximations which have been made in an attempt to present an easy method of evaluating other similar shield materials without the necessity of such extensive tests.

1. Gamma Data Validity

The purpose of the gamma measurements is to make it possible to predict the dose rate penetrating through various shield thicknesses. Since all of the measurements were taken with the isotropic detector surrounded by shield material it is expected that a small boundary effect would have existed, making the measurements higher than they would have been had the detecting chamber not been backed by shielding material; this effect is expected to be small enough to neglect.

In order to be able to determine accurately the dose rate in the units of roentgens per hour it is necessary to have an air filled chamber with air equivalent walls. Data are reported in the literature on materials

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which approximate air equivalence;^(6, 7) in general it has been found that materials of low atomic number ($6 \leq z \leq 12$) approximate air equivalence over a rather wide energy range; hence the chamber walls used were composed of bakelite, magnesium, or carbon. The chamber walls were designed to be sufficiently thick to produce an equilibrium relationship between the photons and secondary electrons; no attempt was made to minimize wall attenuation. Because of the thick walls, the chambers were calibrated, using a 1 gram radium source. The calibration was found to be within 15 per cent of the calculated sensitivity of the chambers and in every case was linear over a rather wide range. As a check on the neutron insensitivity of the chambers they were also calibrated with a radium-beryllium (neutron and gamma) source and with a polonium-boron (neutron) source. In the particular flux fields in which these chambers were used for detecting gamma activity the neutron activity would have produced approximately 5% or less of the total readings.

The outside grounded covering of the chambers was painted with boron carbide solution in order to minimize activation of the chamber components. The chamber wall materials all possessed low capture cross sections and very little activation was encountered. The important chamber characteristic of spectral sensitivity was not experimentally evaluated. This spectral sensitivity is a result of the wall material exhibiting a different energy response than the air. If the gamma spectrum changes markedly as it penetrates the shield an error would be introduced into the relative readings; this error is not expected to be greater than 10 per cent.

An estimate of the accuracy of the absolute gamma dose rate measurements, based on the wall material limitations, unknown spectrum, and instrument drift, is ± 20 per cent. The relaxation lengths could be

(6) NP-1729, "Radiac Symposium," Sept. 14-16, 1949.

(7) NP-1341, "On the Usefulness of Different Radiation Detection Methods for Roentgen Measurement Purposes," Wolfgang Ramm, 9-30-49.

more exactly determined than this, the accuracy depending on the number of points which were obtained. Because of the time involved a minimum number of points were obtained for each material.

The data points were found to fit a simple exponential. The exact reason for this can not be stated because not enough information has been obtained. It can be postulated that the result is a combination of buildup, hardening, and gamma production (neutron capture) throughout the shield. Because the incident gamma spectrum and the distributed capture gamma sources are strongly dependent on the composition of the pile and the shield the extrapolation of these data to other types of piles and shields must be done with reservation.

It is concluded from calculations as well as the measurements that the high energy capture gammas emitted from the iron thermal shield are the predominant source of the gamma radiation attenuated by the biological shield.

2. Neutron Data Validity

As in the case of gamma radiations the desired goal of the neutron measurements was to be able to determine the effluent biological dose rate for various shield thicknesses. Because the relative biological effectiveness of neutrons is greater than that of photons, the determination of neutron dose rate by means of energy deposition is not presently feasible. Activation of small foils was determined to be the most practical neutron measuring device under the circumstances, even though the interpretation in terms of dose rate is somewhat nebulous.

The handling and counting of the foils is adequately described elsewhere. The errors introduced due to counting statistics, placement uncertainty, flux depression, introduction of perturbations, etc., are insignificant when compared to the interpretation in terms of dose and will not be considered.

To interpret the sulfur foil activity in terms of flux or dose it is necessary to establish a method of calibration. The interior of the pile provides a source of neutrons whose magnitude and spectrum can be calculated;^(8, 9) this information has been plotted in Figure 26. In order to convert this energy spectrum to dose rate it is necessary to utilize the existing knowledge of biological effect of neutrons as a function of energy⁽¹⁰⁾ which is plotted in Figure 27. The product of these two curves is plotted in Figure 28, showing the energy dependence of the neutron dose rate within the graphite of the active region of the pile. (These curves are all plotted as a function of vigor, which is defined as the logarithm of the ratio of the neutron energy to an arbitrary base energy. This term is used instead of the conventional lethargy units in order to provide a scale which increased with energy.) The area under the curve of Figure 28 represents the total dose rate; this linear plot clearly indicates the importance of the high energy neutrons in determining the dose rate. These curves can then be used to "calibrate" a cadmium covered sulfur foil in terms of dose rate.

It is not expected that the neutron spectrum will remain unperturbed in penetrating the reflector and shield. On the contrary, the lower energy groups are known to readjust violently in going from the reflector to the shield, but the contribution of these lower energy groups to the dose rate is so small as to be negligible. The important energy group for dose rate considerations is from 0.1 mev to 10 mev. In this energy interval there are two regions of interest, the energy group which activates the sulfur foil and the energy group which does not. In order to draw conclusions from the foil measurements it must be shown that the change, if any, in the ratio of neutrons in these two groups does not negate the use of the sulfur foil dose rate calibration. The gold and

(8) HW-28303, "Neutron Energy Spectra in a Hanford Pile," R. E. Heineman and J. W. Culvahouse, 6-5-53.

(9) HW-29135, "Measurements of Neutron Flux Spectra Insert Reactors," R. E. Heineman, 8-26-53.

(10) RH-1, "Reactor Handbook," Vol. 1, June 1953.

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FIGURE 26
NEUTRON FLUX SPECTRUM IN A HANFORD LATTICE

FIGURE 27
NEUTRON FLUX TO DOSE CONVERSION

FIGURE 28
RELATIVE NEUTRON DOSE RATE IN A HANFORD LATTICE

cadmium covered gold foils assist in this interpretation as shown in the following paragraphs.

Consider first those neutrons which are energetic enough to activate the sulfur foil. If hardening occurs, or if a high energy window exists in the shield: (1) the sulfur foil data plot would not be a simple exponential, and (2) the low energy detectors would come into equilibrium with a more penetrating component than that indicated by the limited number of available sulfur data points. Because this did not occur in any of the measurements, the use of the high energy fission spectrum tail seems justified.

Consider next the neutrons in the energy group not energetic enough to activate the sulfur foil. This group is attenuated at a rate governed by the shield cross section for this particular energy band, in which case there can be geometry buildup and also possible windows. Added to this energy band are those neutrons which are scattered in from the higher energy, sulfur-activating group. From the gold foil traverses it appears that no window exists in this intermediate energy band; for if it did, the thermal and 5 ev detectors would come into equilibrium with this longer relaxation length. However, this method of detection does not eliminate the possibility of buildup being important in this high intermediate energy region. Because the buildup relaxation length asymptotically approaches the relaxation length of the primary group, the low energy detectors would likewise exhibit this same relaxation length and would give no indication of the magnitude of the buildup. As a check for this type of leakage an integrator composed of lucite was used to thermalize the escaping neutron flux. From this information it was concluded that buildup in this high intermediate energy region was not sufficiently great to negate the use of the sulfur foil as a dose meter based on the calibration to the spectrum and intensity calculated to exist in the pile lattice.

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With the measurements that were obtained it is thus possible to interpret a leakage dose rate with some degree of confidence. Because of the approximations which must be made, however, there is an uncertainty of a factor of two or three between calculated and actual dose rates. The relaxation lengths are accurately determined with the knowledge that further thickness extrapolations, within limit, can be made accurately. Shield thickness must be adequate to protect against the actual dose rate; accuracy limits must therefore be treated conservatively rather than by "best guess" in the design of new shields.

3. Effects of Testing Facilities on Data Validity

The two features of the test facilities which must be considered are the edge effects and the source geometry. The A test hole at the D Pile has been used as a shield standard for iron-masonite data because of these factors: (1) this facility is located near the center of the far side face and the source can be considered to be an infinite plane source because of the large pile area it sees and the flattened distribution in this region; (2) the edge effects have been minimized simply by utilizing plugs which duplicated the shield structure and which fit with minimum clearance. Neither the plugs nor the stepped entry hole are lined. Because of the small dimensions and the laminar nature of the shield the experimental evaluation of edge effects is difficult to interpret but does indicate that the edge effects are negligible.

The DR test wells have a considerable edge effect. This arises both because of the steel liner which forms a gas seal between the wells and the pile atmosphere and because of the relatively large clearance which must be allowed to facilitate removal of the test slabs. It is possible to evaluate these edge effects by obtaining lateral flux traverses; however, in the outer regions the lateral traverses do not exhibit a clear minimum point (See Figure 20). The use of the centerline points in this outer region is actually conservative since it would result in a longer

relaxation length; any large deviation from the exponential attenuation determined at greater depths would have to be investigated further.

The location of the DR test wells at some distance from the center of the top shield made it necessary to determine the effect this had on the source geometry. It was noted that the radiation did not penetrate the wells normally; however, the correction for this angle penetration was calculated to be less than the error associated with the depth measurements and was not applied. The source was considered to be an infinite plane.

The E test facility at the F Pile has been discussed in a previous report⁽⁵⁾ in which streaming factors were evaluated; the calibration of this facility is estimated to be accurate to within ± 10 per cent.

F. K Pile Shield Design

The K Pile shield as finally constructed consists of ten-inch cast iron thermal shield blocks surrounded by 63 inches of concrete on the front and two sides, 45 inches of concrete on the rear face,^(11,12) and 80 inches of concrete on top as shown in Figures 29 through 32. At places where there is a large number of openings through the shield, namely the charge and discharge faces and the experimental side, iron-limonite concrete has been placed. The control rod face was made of lower density magnetite-limonite concrete since it is backed by a secondary shielding wall; during normal pile operation no one is allowed entry into the inner rod room due to the high beta and gamma activities from the control rods themselves. A similar situation exists on the rear face which is a restricted zone during pile operation. Here instead of magnetite-limonite concrete, higher density iron-limonite concrete was placed; but the thickness of the concrete was reduced from 63 inches to 45 inches, thus allowing shorter

(5) HW-29125, op. cit.

(11) HW-23193, "Pile Shielding Specifications," J. A. Berberet, 1-4-52.

(12) HWC-2463, "Preliminary Data on the Scope and Design of the Biological Shield for RDA-DC-3," H. S. Davis, 1-1-52.

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FIGURE 29
CONCRETE FILL ARRANGEMENT TOP SHIELD K PILE

FIGURE 30
 CONCRETE FILL ARRANGEMENT K PILE SHIELD SECTION "A-A"

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FIGURE 31
CONCRETE FILL ARRANGEMENT K PILE SHIELD SECTION "B-B"

FIGURE 32
CONCRETE FILL ARRANGEMENT TOP VIEW SIDE SHIELDS K PILE

process tubes to be used. All of the pile shield corners were made of magnetite-limonite concrete for two reasons: (1) the flux at a corner is only a fraction of that at the center of a face, and (2) radiation coming from the pile has to pass through a greater thickness of concrete in the corners than it does through the center of a face.

On the top shield a greater shield thickness is required to accommodate the ball hoppers which are part of the safety system for the pile. The hoppers and associated ducts create large volumes of inferior shielding material which must be compensated by a larger concrete thickness. Vertical safety rod openings which perforate the top also result in reduced shielding effectiveness. Inside the vertical safety rod and ball hopper pattern "Prepakt" iron-limonite has been placed to compensate for this large volume of voids. Surrounding this section lower density "Prepakt" magnetite-limonite concrete has been placed. In order to obtain a smooth top surface the top 10 inches of the entire top shield was made of magnetite-limonite concrete placed by conventional means.

Initially iron-limonite and magnetite-limonite concretes have strength properties approaching those of normal structural concrete⁽¹⁾ but the investigation of long range strength properties had not been undertaken at the time of pile construction. Therefore, as a precautionary measure the K Pile shield was designed in such a manner that the long range structural properties of the shield depended upon the steel framework rather than upon the concrete; large reinforced steel crates were constructed which would transmit the stresses accrued during pile operation and the shielding concrete was used only as a filler material. In 1950 a series of tests were run by the Design Section to determine the feasibility of conventionally placing high density concrete around the process tube entries in a shield crate mock-up using standard Hanford lattice spacings.⁽¹³⁾

(1) HW-32456, op. cit.

(13) HW-19563, "Construction Test High Density Concrete Shielding," L. L. Rice, 11-24-50.

These proved sufficiently promising that further mock-ups were made in 1952. ⁽¹⁴⁾ Experiments performed on these test blocks indicated that no large voids were present in either the conventional or the "Prepakt" placed concretes and that either method of placement could be used under properly controlled conditions in the construction of an all-concrete shield; however, it was decided by the Design Section that the uniformity of the concrete mixture on a construction scale could be more easily controlled using the "Prepakt" method. The choices of biological shield thicknesses and material configurations in the final design of the K pile shields were based upon the results of the attenuation measurements presented in this document.

IV. INTERPRETATION OF THE DATA

In addition to establishing the attenuation properties of the specific concrete used in C and K pile construction, a primary goal of the experiments has been to obtain information which might be used to predict the attenuation characteristics of other types of heavy aggregate concretes and thereby to eliminate the necessity of such extensive tests. Because all of the measurements were taken in the same facility and were subjected to the same incident spectra, the extrapolations could be expected to be accurate only for situations where similar spectra would be predicted. Results of these measurements even under these restrictions make it possible to predict the attenuation of neutrons and photons in heavy aggregate concretes to the accuracy needed for most designs. The purpose of the following discussion is to show qualitatively the physical reasoning behind the methods and to indicate the limitations.

A. Neutrons Shielding Effectiveness

Because of wide variations in elemental cross sections and in biological dose rates as functions of neutron energy, it is highly important that the factors determining the emergent spectrum be understood so that

(14) HW-31213, "Report on Development Work with High Density Concrete Performed in 1952 at the Hanford Atomic Products Operation," H. S. Davis, 3-23-54.

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the proper combination of shielding materials will be employed. The following discussion shows the manner in which these factors have been considered in evaluating the neutron attenuation information presented in this document. High energy neutrons are usually eliminated by being slowed down (thermalized) before being absorbed in a nucleus, since the probability for absorption type reactions at high incident neutron energy is low compared to the probability of scattering. The scatterings can be either elastic or inelastic.

1. Elastic Scattering

The energy loss suffered by the neutron due to elastic, or "billiard ball", scattering depends upon the relative masses of the neutron and the target nuclei and upon how direct is the hit. The minimum amount of energy which the scattered neutron can have, which corresponds to maximum energy loss, is given by the following relation:⁽¹⁵⁾

$$E_{\min} = \alpha E_0$$

$$\text{where } \alpha = \left(\frac{A - 1}{A + 1} \right)^2$$

A = target mass number.

To illustrate the importance of the target mass number let A = 1 (hydrogen) and A = 56 (iron). In the case of hydrogen all of the energy can be transferred from the incident neutron in a single collision and on the average half of the energy is transferred. In the case of iron, a maximum of about 7 per cent of the energy can be transferred so that the average neutron energy is virtually unchanged in a single collision. It can also be shown that an average of 18 collisions with hydrogen will thermalize a 2 Mev neutron whereas over 500 collisions with iron are required for the same process. The value of hydrogen as shield constituent for thermalizing neutrons is further enhanced by the 1/v scattering cross section which it

(15) "Nuclear Reactor Theory," S. Glasstone and M. C. Edlund, 1952, p. 140-141.

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processes. In a hydrogenous medium the thermalization process is rapidly convergent; that is, after each collision the neutron "sees" a bigger target, hence travels a shorter distance before each succeeding collision. Iron, on the other hand, does not have $1/v$ cross section, but rather exhibits large resonances and anti-resonances. In this case a neutron can be scattered into a "window" energy interval and penetrate much farther than it could as a higher energy neutron. This illustrates the important role which hydrogen plays in the shield in preventing intermediate neutron energy "windows" which would cause a gross reduction in biological shield attenuation efficiency.

A fission spectrum penetrating a shield composed primarily of hydrogen would undergo spectral hardening because of the $1/v$ cross section. Thus at each succeeding depth of penetration the most abundant energy group would be more energetic, hence more penetrating, and the relaxation length would be ever increasing. Therefore, hydrogen may be considered to have a high energy "window" as opposed to the resonance "windows" of some other elements.

2. Inelastic Scattering

In inelastic scattering, as in the case of elastic scattering, the incident neutron transfers a great deal of energy to the target nucleus, but the transferred energy in the inelastic scattering case takes the form of an excitation of the nucleus rather than of simple motion of the atom; the target nucleus decays to the ground state by the emission of gamma radiation. Inelastic scattering has a threshold energy corresponding to the lowest energy state to which the nucleus can be raised. In general, the inelastic scattering cross section increases with energy, as more levels are available for excitation, until a saturation value is reached. Inelastic scattering is therefore a means of removing high energy neutrons and preventing spectral hardening. An element with a relatively large inelastic scattering cross section will therefore minimize the probability of high energy "windows" existing in a shield.

3. Absorption

None of the known absorption cross sections are of significant magnitude at high energies, but most elements exhibit significant absorption cross sections at thermal energies. The most common absorption phenomenon is the $X(n, \gamma) X^*$ reaction, in which prompt high energy photons are emanated and the nucleus is left in an excited state. These high energy photons may contribute to the shielding problem; hence nuclei which react $X(n, \alpha) Y^*$ may be added to the shield to absorb the thermal neutrons. In general the elimination of thermal energy neutrons is not a problem, however.

4. Effective Removal Cross Section

Several years ago, Albert and Welton⁽¹⁶⁾ formulated a theory which postulated that each heavy element has an "effective removal cross section" for fission neutrons if an abundance of water (of the order of 80 per cent water by volume) is present. Concretes do not contain this much water; hence, in these experiments a different criterion had to be formulated for the necessary water content. The hydrogen content of a shield is considered to be sufficient if no intermediate or low energy windows exist. The existence of possible "windows" was monitored by comparing the "equilibrium" traverse slopes obtained from the gold and cadmium covered gold foils. All of the test shield materials did have sufficient hydrogen to prevent "windows", although the minimum essential content was not determined.

In the Albert-Welton method, an analytical expression is used for the hydrogen cross section, and hardening of the spectrum is taken into account. Hardening of the spectrum is an important feature in a shield which is predominantly hydrogen; however, in the shield materials tested, which contain a near minimum amount of hydrogen, hardening is suppressed by the inelastic cross section of the iron. The high energy neutron spectrum should not change drastically under these circumstances; it should therefore

(16) WAPD-15, "A Simplified Theory of Neutron Attenuation and Its Application to Reactor Shield Design," R. D. Albert and T. A. Welton, 11-30-50.

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be possible to assign a single effective removal cross section to hydrogen as well as to the heavier elements in the shield. The effective removal cross section for the heavy elements should be approximately the same as that obtained in the 80%-water measurements since the cross sections in general are not very energy sensitive in the 1-10 Mev region. Effective removal cross sections have not been measured for all of the elements used in the shields tested; in these cases approximate values were calculated. Table II compares the calculated and observed neutron attenuation efficiency and shows the composition of the slabs as estimated from the mix data and from chemical analyses of the ore constituents; the elemental mass absorption coefficients which have been used in the calculation are also listed. The mass absorption coefficient shown for hydrogen is that value which best fits the data based on a least squares analysis. The hydrogen effective removal cross section obtained by this method is 1.65 barns/atom, considerably larger than the 0.98 barns/atom reported elsewhere;⁽¹⁷⁾ the difference is attributed to the sensitivity of this cross section to the neutron energy spectrum. The value determined by these experiments is therefore applicable only in similar circumstances where there is sufficient iron present (or some other element with equivalent inelastic cross section properties) to prevent significant spectral hardening.

In order to determine the neutron absorption coefficient of other heavy aggregate concretes utilizing ores from other mines it is necessary to determine the abundance of each element in the shield, to apply the corresponding mass absorption coefficient based on the effective removal cross section, and to sum the results. If restricted to concretes of the same type used in the experiments (Fe - H-O) it appears that the absorption coefficient, μ (cm^{-1}), can be calculated to within ± 5 per cent. For shields of this type, and for neutron intensity reductions of the order of 10^9 , this

(17) WASH-152, "Reactor Shielding Information Meeting, Chicago, Nov. 12 and 13, 1953," pg. 66.

TABLE II

COMPARISON OF CALCULATED AND MEASURED NEUTRON ATTENUATION PROPERTIES

Mass Absorption Coefficient Element cm^2/gm	Iron-Magnetite		Iron-Limonite		Magnetite-Limonite		Magnetite-Limonite		Magnetite-Limonite		Magnetite Conventional gm/cm^3
	gm/cm^3	cm^{-1}	gm/cm^3	cm^{-1}	gm/cm^3	cm^{-1}	gm/cm^3	cm^{-1}	gm/cm^3	cm^{-1}	gm/cm^3
H	.043	.039	.028	.025	.023	.021	.029	.026	.015	.014	
C	.345	.014	--	--	--	--	--	--	--	--	
O	.310	.011	.806	.029	1.343	.048	1.306	.047	1.279	.046	
Mg-Al	--	--	.039	.001	.156	.005	.110	.003	.184	.005	
Si	--	--	.078	.002	.117	.003	.101	.003	.129	.004	
Ca	--	--	.253	.006	.170	.004	.143	.003	.220	.005	
Fe	3.568	.075	3.033	.064	1.593	.033	1.543	.032	1.460	.031	
	4.266		4.237		3.402		3.232		3.287		
μ calc		.139		.127		.114		.114		.105	
μ meas		.142		.130 and .127		.111		.114 and .112		.105	

*values obtained from least squares fit of the data

**values are taken from reference 17

***calculated values

uncertainty in absorption coefficient introduces an uncertainty of about a factor of three in the leakage rate. Thus, if a shield thickness were designed to reduce the escaping neutron flux to three mrem/hr, the actual shield should, with 90 per cent confidence, reduce the intensity to the range 1-9 mrem/hr. In other words the as-built shield could be about three inches too thick or too thin to exactly meet the design leakage estimate.

This uncertainty in neutron attenuation efficiency is in addition to other uncertainties which exist such as the exact source strength, the biological effectiveness of neutrons, the actual as-built shield composition (as opposed to design content of the concretes), the perturbations introduced by pile components penetrating the shield, and the admitted inaccuracy of a sulfur foil detector as a dosimeter. The combination of these latter uncertainties is in general more important in new reactor design than the uncertainty introduced by a neutron attenuation calculation of the type proposed.

B. Gamma Shielding Effectiveness

A simple, yet accurate, extrapolation of the gamma data information is not possible because the gamma source is a function of the neutron flux distribution which in turn is dependent upon the shield composition. A simple method for estimating the gamma absorption coefficient of a shield is illustrated in Figure 33 where the total absorption coefficient of each of the shield materials is plotted as a function of energy. It can be seen that the measured relaxation length in each case corresponds to that expected from simple attenuation in the 5 to 10 Mev energy interval, a region in which the absorption coefficient is varying rather slowly. If a thick iron thermal shield is to be employed, an approximate value for the gamma attenuation coefficient could be obtained from a similar type plot for the proposed biological shield. The accuracy limits would be comparable to those from a neutron calculation of the type proposed above. If no thick

FIGURE 33
CALCULATED GAMMA ABSORPTION COEFFICIENT
AS A FUNCTION OF ENERGY FOR THE
TEST SHIELDS

iron thermal shield were to be used, it is doubtful that the technique would be applicable. In such a case a detailed analysis of the sources and of the spatial and energy distributions of the sources would be necessary in order to estimate gamma leakage. However, there are two factors tending to simplify the gamma radiation aspect of shield design: (1) design of the shield for adequate fast neutron attenuation will very probably provide for adequate gamma absorption properties, and (2) because gamma radiation is more easily measured and corresponding biological dose rates are much better established, radiation zone working conditions can be defined much more exactly than in the case of higher energy neutrons for which large safety factors must be employed.

V. CONCLUSIONS

It is felt that the neutron relaxation length of proposed shields consisting predominantly of iron, hydrogen, and oxygen can be predicted accurately enough to minimize the need for further in-pile tests of the type just completed. Further tests are necessary to determine (1) the minimum necessary hydrogen content necessary in a concrete shield and (2) the feasibility of similar calculations for shields which do not contain iron as the predominant inelastic scattering element. Gamma relaxation lengths can be predicted adequately provided that a thick iron thermal shield is used; however, further experiments would have to be done in order to permit evaluation of the dependence of the gamma distribution on the type of thermal shield which is employed.

VI. ACKNOWLEDGEMENT

The authors wish to acknowledge the technical assistance, advice, and guidance of R. L. Dickeman, who initiated and supervised this attenuation program. The work of J. A. Berberet in development of the experimental techniques and in programming the experiments is also acknowledged.

The many laboratory personnel who prepared and counted the foils and assisted in other aspects of the experimental work deserve much credit for their part in the program.

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by LB

WLB:RLT:ag

APPENDIX I

DATA ON COMPOSITION

(From Mix Data Used in Fabricating Test Slabs)

Aggregate Placement Method Date Made	Iron-Limonite Conventional 5-10 to 5-16-51	Iron-Limonite Prepak 4-15 and 16-51	Magnetite-Limonite Conventional 11-4-52	Magnetite-Limonite Prepak 9-24-52	Magnetite Conventional 2-18 to 3-11-52
Composition (#/ft ³)					
Cement	34.8	--	24.4	--	31.3
Water	12.6	--	12.3	--	11.9
Limonite	68.6	62	46.6	67	--
Magnetite	--	--	139.7	87	170.5
Iron	149.2	143	--	--	--
Plastiment	0.6	--	.12	--	0.3
% Voids	--	43.9%	--	40.9%	--
Grout	--	63.2	--	59	--
-68-					
Grout Composition (#/ft ³)					
Cement	--	50.3	--	50.3	--
Water	--	28.8	--	28.8	--
Limonite	--	64.2	--	64.2	--
Intrusion Aid	--	0.7	--	0.7	--
Sp Gravity of Fresh Concrete	4.26	4.30	3.58	3.42	3.44

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ABSTRACT

The neutron and gamma attenuation properties are presented for the different types of high-density high iron-content concretes used in recent Hanford pile construction; attenuation properties of the laminated iron and masonite shields used on the original piles are also shown. The experimental facilities and techniques used to obtain the attenuation data are described, and the application of observed results to shield materials of similar composition is discussed.

END

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