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GENERAL REACTOR ENGINEERING RESEARCH

QUARTERLY PROGRESS REPORT FOR

PERIOD ENDING NOVEMBER 20, 1950

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REACTOR TECHNOLOGY DIVISION

Engineering Research - Section IV

GENERAL REACTOR ENGINEERING RESEARCH

QUARTERLY PROGRESS REPORT

for Period Ending November 20, 1950

Edited by

R. N. Lyon

DATE ISSUED APR 5 1951

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SUMMARY

Plastic shielding materials made of Unichrome 4032, a thermosetting plastic, were found to be usable up to 300°F.

Boral production was improved and the thermal conductivity was found to be 25 Btu/hr-ft^2-°F/ft at 200°F and 19.0 Btu/hr-ft^2-°F/ft at 500°F. A higher value reported in ORNL-629 is in error because of faulty equipment.

MO concrete tests to test the effect of clean and rusty punchings have been started, using samples from the same blend of magnesia. The cement and concrete were found to be weakened by an exposure to 4.4×10^{18} nvt in the X-10 Reactor.

Barytes concrete was found to have a thermal conductivity of 0.926 Btu/hr-ft^2-°F/ft at 122°F and 0.745 Btu/hr-ft^2-°F/ft at 482°F.

A limonite concrete mix for the new Bulk Shielding Facility was formulated, and 300 test blocks were cast by the Southern Cast Stone Company. These compared well in density and strength with blocks made from the same mix at ORNL. Work to formulate a boron-containing concrete for the HRE shield has started.

A concrete of density 3.1 containing portland cement with 1% boron and having a 5% water content has been formulated.

QM beryllium was found to have a thermal conductivity only slightly higher than that listed in the literature for beryllium of unknown purity.

Samples of Y-12 zirconium metal were found to have poor corrosion resistance to distilled water at 600°F and 1500 psi pressure.

Zirconium with 3% tin was found to have good corrosion resistance, and hafnium was found to have excellent resistance under the same conditions.

A technique for welding ½-in. zirconium plates developed for the Laboratory by Battelle was found to produce corrosion-resistant welds.

Preliminary data were obtained in a comparison of etched, alodized, and anodized 2S aluminum in filtered water and etched aluminum in dichromate-inhibited water.

Tantalum was found to show negligible corrosion in additional tests with RaLa process solutions at room temperature and at 70°C.

Preliminary data were obtained on thorium alloy corrosion, but final results will be delayed until a satisfactory defilming process can be developed.

An equation has been developed to enable calculation of the temperature distribution in the MTR mock-up when the cooling water is suddenly replaced by air.

Preliminary, unofficial conversations have been held with TVA personnel regarding the possible use of a 12-ft-diameter penstock for obtaining generalized velocity distribution profiles in turbulent flow.

The pile-irradiated plastics were found to have about the same electrical properties after irradiation as before, except for the saran and vinylite, which lost essentially all their electrical resistance.

Changes in the mechanical properties of irradiated plastics were found to continue for some time after removal from the X-10 Reactor. No difference was found in the effect of irradiation with or without oxygen present.

Preliminary data have been obtained on the activity induced in several steels and concretes which were irradiated in the X-10 Reactor.

1. SHIELDING MATERIALS

PLASTICS

Tungsten carbide sheets, in which the plastisol Unichrome 4032 was used as the binder, supported their own weight at temperatures above 300°F. Corresponding sheets made with Tygon paint as the binder began losing strength at temperatures below 200°F and disintegrated within one day when the temperature was increased from 200 to 300°F. The plastisol sheets became unusable after three days at 350°F. Below 300°F the plastisol sheets, when removed from the oven and cooled, regained their original hardness and stiffness.

For both types of plastic sheets the samples were suspended from one edge and heated in an oven at 50, 100, and 150°C on successive days and finally at 175°C until failure.

The rolling mill and associated furnaces for the fabrication of large, 56½- by 66½-in., sheets of the plastisol material are approximately 50% completed.

BORAL

Production. Boral sheets, 24 in. maximum width by 84 to 90 in. maximum length by ¼ in. thick, are now being rolled by the Metallurgy Division at X-10. Two or three ingots per day are being cast, framed, and clad at Y-12. Maximum rolling capacity at X-10 is two ingots per day. At the time of this writing it was expected that the MTR and Brookhaven requirements would be met by Jan. 1, 1951.

To eliminate some of the difficulties encountered in casting and rolling Boral ingots and to increase the workability of the rolled sheets in the various fabricating operations, it was necessary to reduce the B_4C content from 50% by weight to 35% by weight. Boral as now rolled contains 35% B_4C and 65% Al. Ingots are now mixed in a Tocco type induction furnace and cast into a graphite tray, 1 in. thick. The framed and cladded ingot is heated to approximately 1050°F prior to rolling and reduced from 1.5 to 0.5 in., reheated at 1050°F for 30 min, and then rolled to the desired thickness. Cold rolls are used and the rolls are continuously lubricated with kerosene.

Physical Properties. Recent determinations of the thermal conductivity, k, of Boral, 50% B_4C by weight, indicate that the thermal conductivity varies as follows:

Temperature (°F)	200	450	500
k (Btu/hr-ft^2-°F/ft)	25	19.2	19.0

These values are preliminary results and are being rechecked.

The thermal conductivity equipment was calibrated by determining k for commercially pure aluminum. The values obtained for aluminum checked within 5% the best accepted values for aluminum as reported in Bell Aircraft Corporation Report BAC-18.

The value of 86 Btu-ft per hr-°F, reported in ORNL-629 as the thermal conductivity of Boral that is 50% B_4C by weight, is in error, as it has since been demonstrated that the equipment used for the determinations was not operating properly.

Development Work. Methods are being studied by which B_4C and molten aluminum can be mixed together without the addition of aluminum powder. The process of adding small quantities of B_2O_3, blended into the aluminum with the B_4C, shows promise of causing the B_4C and Al to "wet," providing an easily mixed and less oxidized ingot. This study will be continued.

BERYLLIUM

Thermal conductivity of QM beryllium varies as follows:

Temperature (°F)	150	250	350	450	550
k (Btu/hr-ft^2-°F/ft)	94	91.3	90	88	88

These values are now being rechecked. They can be compared with the value for beryllium of unknown purity listed in the ASM *Metals Handbook* (1948) of 91.9 at 68°F.

SHIELDING CONCRETE

MO Concrete

Aging Tests. To evaluate the effect of rusty versus clean punchings more precisely, a test procedure was set up in which the variation in the magnesia

from batch to batch was kept to a minimum by blending four sacks of magnesia prior to mixing of the batches. Samples are stored in a controlled atmosphere with temperature 77°F and relative humidity 50% and in the uncontrolled laboratory atmosphere. Comparisons are still in progress and will be completed during the next quarter, at which time a full report will be issued.

Radiation Damage. MO cement continued to lose strength with a total irradiation of 4.4×10^{18} nvt in the X-10 Reactor. The results are reported in Table 1.

TABLE 1

Effect of Radiation on MO Cement

SAMPLE	WEEKS OF EXPOSURE	TOTAL nvt	COMPRESSIVE STRENGTH (psi)	
			EXPOSED	UNEXPOSED
Neat cement	4	2.06×10^{18}	2580	8435
Neat cement	8	4.4×10^{18}	2370	6500
Neat cement + 20% Fe	4	2.06×10^{18}	5000	8520

Data for irradiations of 1.21×10^{18} and 6.3×10^{17} nvt are reported in ORNL-836, p. 10.

Barytes Concrete

Physical Properties. The thermal conductivity of barytes concrete (density 3.5), using ¾-in. aggregate for the coarse material, has been determined for various temperatures. Samples were cured in wet sand for 28 days prior to the determination. The results were:

Temperature (°F)	122	212	257	302	392	482
k (Btu/hr-ft^2-°F/ft)	0.926	0.997	1.02	0.872	0.866	0.745

Special Concretes

New Bulk Shielding Facility. A mix has been formulated using portland cement, limonite, and barytes for the blocks to be used in the Bulk Shielding Facility. Approximately 300 blocks of three different sizes were cast by the Southern Cast Stone Company, Knoxville, Tenn., using the following mix, per cubic yard:

Barytes, coarse (¾ in.)	2210 lb
Barytes, fine (3/8 in.)	1050 lb
Limonite (crushed)	1220 lb
Portland cement, type I	610 lb
Water	46 gal

Test cylinders made at ORNL using the above mix had a density of 3.25 and a seven-day compressive strength of 3750 psi. Test cylinders made during the pouring of the blocks at Southern Cast Stone Company had an average density of 3.17, with the density varying from 3.05 to 3.22. The two-week compressive strength of the field cylinders averaged about 3600 psi. No difficulty was encountered by the contractor in pouring the various blocks, the predominant size being 2 by 2 by ½ ft.

Homogeneous Reactor Experiment. The specifications for the HRE shield call for a concrete with a density of 3.1 containing at least 8% water and 1% boron. Formulation of mixes to meet these requirements is in progress. Preliminary data indicate that a mix containing barytes, colemanite ($2CaO \cdot 3B_2O_3 \cdot 5H_2O$), and a cement phase of 50% Lumnite and 50% type III portland cement (High Early) will fulfill the shield specifications and have sufficient strength after 24 hr to be handled. The 24-hr strength of test cylinders was approximately 1300 psi. Work is still continuing, and, as soon as the formulation has definitely been decided upon, the physical properties of the concrete will be determined. Test blocks, 2 by 2 by 1 ft, are being cast for attenuation data experiments in the new Bulk Shielding Facility.

Lumnite is a calcium aluminate hydraulic cement. It is not a portland cement but, like portland cement, it sets and hardens when mixed with water. In less than 24 hr after being placed it can be subjected to the full load for which it is designed. Proper curing of Lumnite is essential and is accomplished

by spraying the surface of the concrete with water within 24 hr after mixing. The usual curing techniques for ordinary portland cement concretes are not satisfactory, and the peculiar technique needed for Lumnite would be rather difficult for a contractor to follow.

Usually portland cement is not added to Lumnite because it greatly accelerates the initial set, frequently causing flash-set. However, the addition of soluble boron (colemanite), which detrimentally affects the set of Lumnite, counterbalances the effect of the portland cement, giving a product which can be handled within a reasonable length of time before hardening is evidenced.

Boron-containing Concretes

Portland Cement. A mix has been developed containing 1% boron, type I portland cement, and barytes. The density of the resulting concrete is 3.1 and the water content is approximately 5%. The boron was added as an insoluble boron compound formed by fritting borax with silica. Soluble boron detrimentally affects the initial set and early strength of portland cement; therefore the boron was added as an insoluble boron compound formed by fritting borax with silica. By this method portland cement concretes can be made with varying boron contents. The frit used in the preliminary tests was supplied by the Chicago Vitreous Enamel Products Company, Cicero, Ill.

Lumnite. A mix has been formulated containing 1% boron (colemanite), Lumnite (cement phase), and barytes. The density of the concrete is 3.1. It developed 75% of its full strength (approximately 3500 psi) after being cured by water-fogging the samples for 24 hr. However, because of the rigid curing technique for Lumnite concretes, this formulation was abandoned in favor of the one containing Lumnite and portland cement so that accepted curing methods for ordinary concrete could be practiced.

2 AQUEOUS CORROSION

CORROSION TESTS ON Y-12 BOMB-PROCESS ZIRCONIUM AND HAFNIUM

Corrosion tests were conducted on samples of zirconium metal produced by the Y-12 Materials Section. The procedure for production of this zirconium is described in Report Y-595, *Preparation of Ductile Zirconium*. Distilled water at 600°F and 1500 psi pressure for one week was chosen as the corrosive medium to give the shortest testing period. To date, no sample has shown satisfactory corrosion resistance.

Data on tests from June 1950 to date are given in Table 2.

The one sample of hafnium metal which the group has tested, a piece of Y-12 hafnium button No. 5 in the as-cast state, has been in 600°F distilled water for 28 days and is in excellent condition. The weight change has been 1.4 mg/dm/day. It shows the typical dark iridescent film of good zirconium with the exception of one thin white streak. This test and the test on 3% Sn—97% Zr alloy are being continued.

ZIRCONIUM WELDING

During the past quarter a final report (BMI-T-37) was received from the Battelle Memorial Institute on the project of welding ½-in. zirconium plates. The results of this investigation indicate the following:

1. Satisfactory joints can be made using the helium-shielded tungsten-arc process in a controlled-atmosphere chamber. Only crystal bar zirconium is pure enough as filler metal to provide satisfactory welds.

2. All welds should be annealed at 780 to 790°C in helium and pickled for best strength and corrosion characteristics.

3. Such welds are corrosion resistant to uranyl sulfate at 250°C.

4. Static strength properties are comparable with the base plate; however, under tension impact the energy absorption is lowered approximately 30%.

TABLE 2

Corrosion of Y-12 Bomb Zirconium in 600°F Water

BUTTON NO.	CONDITION	WEIGHT CHANGE (mg/dm/day)	DESCRIPTION AFTER TESTING
11	Cold rolled and polished	Not determined	Heavy gray-white flaky corrosion product; sample exfoliated
11	Cold rolled and annealed	Not determined	Same
34	As-cast and polished	Not determined	Heavy gray-white corrosion product
45	As-cast and polished	Not determined	Heavy gray-white corrosion product
40	As-cast and etched	Not determined	Heavy gray-white corrosion product
40	As-cast and polished	Not determined	Heavy gray-white corrosion product
42	As-cast and polished	Not determined	Gray to white scale, flaky
42	As-cast and polished	Not determined	Gray to white adherent scale
39	As-cast and polished	Not determined	Gray to white flaky scale, peripheral cracking
39	As-cast and polished	Not determined	Mottled gray adherent scale
43	As-cast and polished	Not determined	Heavy gray-white deposit
37	As-cast and polished	Not determined	Heavy gray-white scale, metal badly cracked
37	As-cast and polished	Not determined	Heavy gray-white scale, metal badly cracked
38	As-cast and polished	Not determined	Heavy gray-white adherent scale
41	As-cast and polished	Not determined	Heavy gray-white adherent scale
44	As-cast and polished	Not determined	Heavy gray-white adherent scale
45	As-cast and polished	Not determined	Dull gray with white spots
46	As-cast and polished	Not determined	Oxide completely penetrated the sample
47	As-cast and polished	Not determined	Oxide completely penetrated the sample
49	As-cast and polished	+43.7	Brownish gray, white streaks
49	As-cast and polished	+48.4	Brownish gray, white streaks
49	As-cast and annealed	+48.0	Brownish gray, one white spot
50	As-cast and polished	+30.1	Greenish-white spotted scale
50	As-cast and annealed	-18.3	Greenish-white spotted scale
49	Cold rolled and etched	+11.0	Dull gray-green film
50	Cold rolled and etched	+26.4	Dull gray film
48	As-cast and polished	Not determined	Shiny gray film, white streaks
52	As-cast and polished	+67.8	Gray, white, and brown streaks
52	As-cast and annealed	+109.8	Gray white film
52	Cold rolled and etched	+39.2	Thick white scale
53	As-cast and polished	+63.3	Heavy green-gray scale
53	As-cast and annealed	+127.5	Coarse gray scale
53	Cold rolled and etched	+68.4	Thin green scale, partially flaked
54	As-cast and polished	+66.8	Banded green and gray scale
54	As-cast and annealed	+87.1	Banded green and gray scale
54	Cold rolled and etched	+35.4	Light gray film, shaded green on one end
55*	As-cast and etched	+5.1	Dark golden brown coating
55*	Cold rolled and etched	+6.1	Dark golden brown coating

* Nominal 3% tin alloy.

CORROSION REISSTANCE OF ALUMINUM ALLOYS IN FILTERED WATER

Tests on 2S aluminum in filtered water at room temperature are in their fourth month. Two samples were also tested from a plate made by hot rolling together a sheet of 2S and one of 72S. Results of the first 90 days are shown in Table 3. A new set of samples, specifically designed to be comparable to actual operating conditions, were started this past month. Sheets of 72S-clad 2S aluminum were brazed with 15% silicon alloy by C. D. Smith of the Metallurgy Division. These samples were sent to the American Chemical Paint Company, where they were treated with the hot dry alodizing process, since this is the planned treatment for the assemblies to go into the actual reactor. The group has been acting as consultants in setting up this process. Work is to be performed by shop personnel.

TABLE 3

Corrosion of Aluminum in Filtered Water at Room Temperature

MATERIAL	CONDITION	INHIBITOR	60-DAY EXPOSURE		90-DAY EXPOSURE	
			PIT COUNT PER SPECIMEN	WEIGHT CHANGE (mg/dm/month)	PIT COUNT PER SPECIMEN	WEIGHT CHANGE (mg/dm/month)
2S Al	Etched		8	+11.4	11	+13.0
2S Al	Etched		8	+15.1	12	+12.4
2S Al	Etched	60 ppm of Na_2CrO_4		+3.4		+1.3
2S Al	Etched	60 ppm of Na_2CrO_4		+2.2		+0.7
2S Al	Alodized			-0.9		-1.1
2S Al	Alodized		1	-1.2	5	+0.3
2S Al	Anodized			-12.5*		-13.6*
2S Al	Anodized			-4.2*		-17.7*
2S-72S Al	Etched		5	+23.4**	5	+58.8**
2S-72S Al	Etched		1	+287.3**	2	+37.4**

* Weight changes for these samples are essentially meaningless since the machined edges have been waxed, and this wax may chip. Surfaces are beginning to discolor.

** Weight gains are low owing to loss of corrosion product into the solution.

The tabulated data are not a comprehensive coverage of the corrosion picture. Samples exposed in inhibited water (60 ppm of Na_2CrO_4) are as bright after 90 days of exposure as at the start. All the other etched samples are discolored a dark gray and are badly pitted. Anodized samples between the 60- and 90-day period are beginning to darken. Of the alodized samples, one has not changed, but the other is showing minor pitting at the points of film breakdown.

ION-EXCHANGE RaLa PURIFICATION PROCESS CORROSION TESTS

Additional 30-day tests in the A-9 tank solutions were run during this period to check the effect of this solution on tantalum. The *p*H of the solution, whose composition was reported in the last quarterly report (ORNL-836), ranged from 6.5 to 6.85. The results are given in Table 4. There was no apparent change on the surface of any sample.

TABLE 4

Corrosion of Tantalum in RaLa Process Solution

TEMPERATURE	WEIGHT LOSS (mg/dm/month)	PENETRATION RATE (mils x 10^{-3}/year)
Room	1.6	1.1
Room	2.0	1.4
70°C	1.6	1.1
70°C	1.2	0.8

CORROSION OF THORIUM AND THORIUM ALLOYS

One set of samples of various thorium alloys has been tested for the Metallurgy Division for 30 days in distilled water at 93°C. The final results of the test will depend on the establishment of a satisfactory defilming-after-testing procedure. An investigation of various chemical processes, which are preferable because of their consistency, is being conducted.

Results on the as-removed weight changes and sample appearance are given in Table 5, together with the metallurgical history of the samples.

TABLE 5

History of Thorium and Thorium Alloy Samples

1. Ames thorium cold rolled 68%, annealed at 750°C for 1 hr in vacuum, and polished on 120-grit paper.

2. Ames thorium cold rolled 68%, annealed at 750°C for 1 hr in vacuum, cold rolled 25%, and polished on 120-grit paper.

3. Ames thorium remelted in vacuum, cast in zirconium oxide, cold rolled 68%, annealed at 750°C for 1 hr in vacuum, and polished on 120-grit paper.

4. Ames thorium remelted with 2% columbium in vacuum, cast in ZrO_2; cold rolled 68%, annealed at 750°C for 1 hr in vacuum, and polished on 120-grit paper.

5. Same as 4, but containing 4% columbium.

6. Same as 4, but containing 2% chromium.

7. Same as 4, but containing 4% chromium.

8. Same as 4, but containing 2% titanium.

9. Same as 4, but containing 2% zirconium.

10. Same as 4, but containing 4% zirconium.

TABLE 6

Corrosion Data on Thorium and Thorium Alloys Before Defilming

SAMPLE	WEIGHT CHANGE (mg/dm/month)	SAMPLE DESCRIPTION
1	-31.9	Coated with light gray powder
1	-26.6	Coated with light gray powder
2	-15.9	Coated with light gray powder, few brown specks, samples cracking
2	-23.7	Coated with light gray powder, few brown specks, samples cracking
3	-11.2	Adherent gray film
3	-7.1	Adherent gray film
4	+40.1	Heavy loose gray coating
4	+7.3	Heavy loose gray coating
5	+100.3	Very heavy spotty gray coating, few brown specks
5	+59.2	Very heavy spotty gray coating, few brown specks
6	-20.9	Shiny gray film
6	-10.3	Shiny gray film
7	-20.3	Shiny gray film, one surface shows a crack
7	-5.9	Shiny gray film, one surface shows a crack
8	+2926.0	Sample badly blistered, exfoliation parallel to the rolling plane
9	-22.9	Loose light gray coating
10	-51.1	Loose light gray coating, one pit shallow
10	-3.2	Loose light gray coating, one cluster of brown specks

3. HEAT TRANSFER AND FLUID FLOW STUDIES

MTR MOCK-UP EMERGENCY SHUTDOWN TEMPERATURES

A finite difference equation has been developed which represents the complex heat transfer—fluid flow mechanism for the MTR mock-up when all the water is suddenly replaced by air. The equation applies to the case in which air can circulate by natural convection through the reactor core, as well as to the case in which no natural convection takes place. A final report on this problem is nearing completion (ORNL-976).

GENERALIZED VELOCITY DISTRIBUTION FOR TURBULENT FLOW

The generalized velocity distribution expression in the turbulent core of a pipe was well established by the classic experiments of Nikuradse, but the same cannot be said for the buffer and laminar layers. A need therefore exists for more accurate knowledge of the fluid mechanics near a wall surface for computing heat transfer coefficients by the momentum transfer theory.

To obtain velocity data well into the laminar sublayer existing near a wall surface would require a large pipe and a method for measuring velocities down to 0.001 ft/sec. Unofficial conversations with Tennessee Valley Authority personnel revealed that a 12-ft-diameter penstock could possibly be made available for such experiments. A method of measuring low fluid velocities by visual observation of the motion of an immiscible slug of colored fluid injected into the main stream is under consideration, and it is planned to test this method in the laboratory for possible use in the 12-ft pipe.

This is a long range project and is to be sandwiched in between more urgent problems that arise.

4. PROPERTIES OF IRRADIATED MATERIALS

PLASTICS

Studies on the engineering properties of pile-irradiated plastics are now essentially complete, and a report (ORNL-928) covering this work will be issued during the coming period.

Preliminary results indicate no change in damage due to the presence or absence of oxygen during irradiation, but a few tests do show that changes initiated by radiation continue, at a slower rate, after removal from the reactor. The work to date has covered normal pile radiation. These studies will be continued with pure gamma radiation or beta and gamma radiation and also with fast neutrons in the absence of thermal neutrons. Equipment for subjecting materials to a high-intensity gamma or beta-gamma source is scheduled to be completed by February. The source will be Au^{198} with a gamma strength of about 10^5 r/hr. The fast-neutron studies will be made by inserting the specimens in the pile in a cadmium-lined container.

Results of tests on the mechanical properties of irradiated plastic were given in some detail in the last quarterly report (ORNL-836). Tests have now been completed on the electrical properties, i.e., volume resistivity, dielectric strength, and arc resistance. For most of these materials only a small change was observed in these properties even up to the point where almost complete breakdown occurred in mechanical strength. There were two notable exceptions. Saran (vinylidene chloride) and vinylite (vinyl chloride-acetate) lost nearly all their insulating value at a thermal nvt of 10^{17} (about one day in the ORNL pile). This is to be expected for these materials, however, because both break down during irradiation, with the liberation of copious carbon and hydrochloric acid. Detailed results on the changes in electrical properties will be given in the topical report.

The pile irradiations for these studies have been made by sealing the specimens in an air-filled container. Since it is known that many organic materials undergo more rapid decomposition in the presence of oxygen during irradiation, tests are underway to see if the presence of oxygen has any effect on the rate of deterioration of plastics in a radiation field. To date only

three plastics have been tested for the effect of the presence of oxygen in the container, and no accelerated deterioration has been detected. These materials were methyl methacrylate, cellulose acetate, and fluorothene. Specimens of each were sealed in cans containing pure helium and pure oxygen, placed in the pile together for the same length of time, and then conditioned and tested in the same manner.

Tests are also underway to determine the effect of aging of plastic materials after irradiation. Specimens of polyethylene, cellulose nitrate, and fluorothene are the only ones that have been tested so far. All three show that deterioration continues even after removal from the pile, but at a much slower rate than during irradiation. The tests were made by irradiating several specimens and testing at several time intervals (up to two months) after removal from the pile. Each was compared with a nonirradiated control specimen which was aged under the same conditions as the irradiated specimen (25°C and 50% relative humidity). No change was observed in the control specimens. This phenomenon has also been observed in gassing tests on less stable organic compounds (gas continued to be evolved after the material was removed from the radiation field). Apparently some energy is absorbed during irradiation to produce unstable molecules which slowly break down over a long period of time. Deterioration, after removal from the radiation field, has been shown to go on at a decreasing rate.

PILE-INDUCED RADIATION

The decay of several steels and concretes, after one month of irradiation in the ORNL reactor, has been recorded for 180 hr on the steels and for three months on the concretes. The data have not yet been analyzed, but the curves are presented in Figs. 1 to 12. Absorption curves are shown in Figs. 13 to 15 for the same materials.

This is part of a long list of materials of construction on which induced activity is being determined so that the radiation intensity to be encountered when equipment is removed from the pile for repairs or for other reasons may be predicted more accurately.

Long-time decay data will be presented as soon as they are accumulated, and decay curves will also be taken after 6 to 12 months of irradiation.

The gamma decay curves were taken on very thin specimens with an ionization chamber that was calibrated with Co^{60} standards (1.2-Mev gamma). The absorption curves were made by interposing sheets of lead between the source and a G-M tube.

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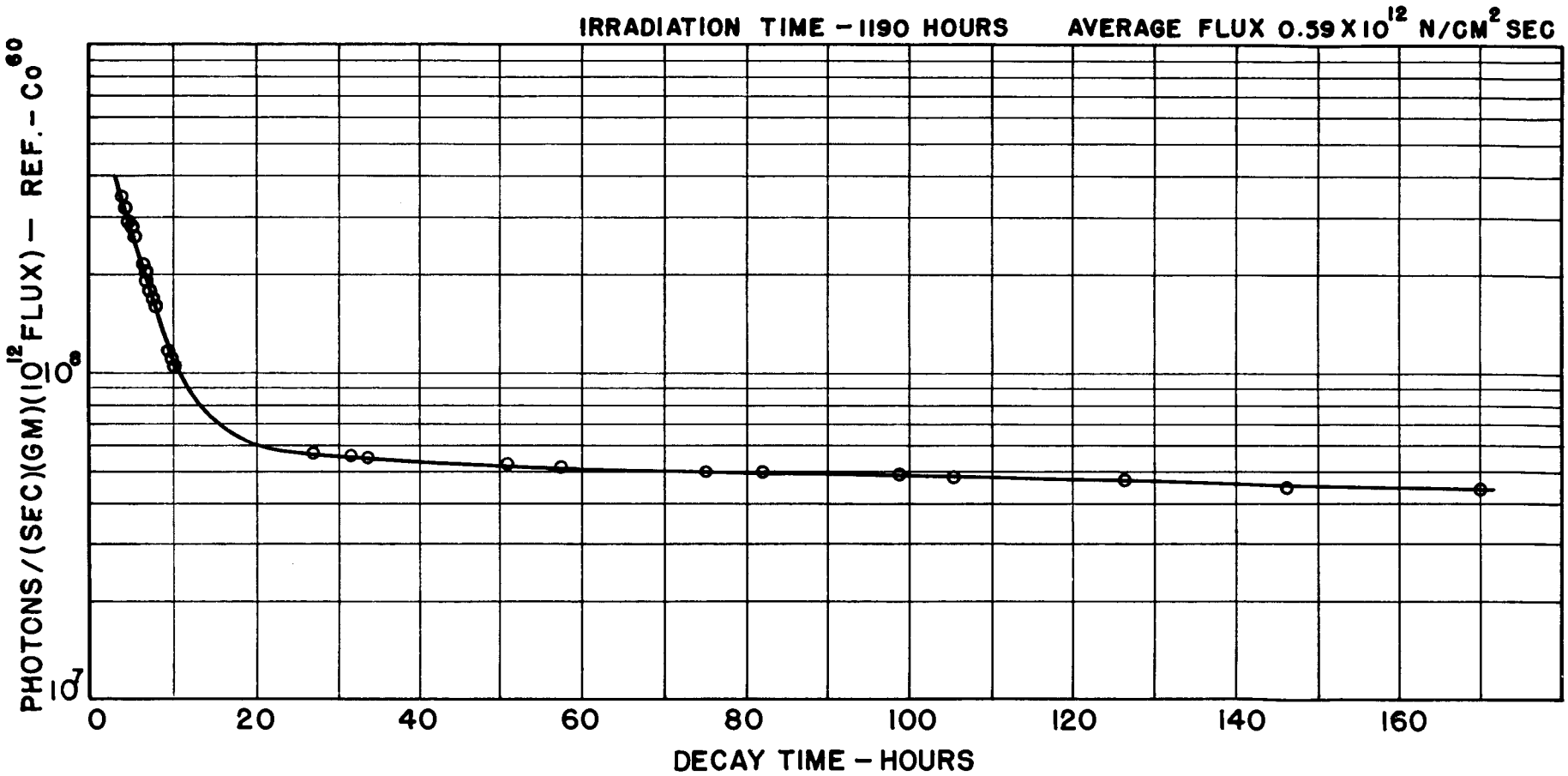

FIG. 1 GAMMA DECAY OF 304 STAINLESS STEEL

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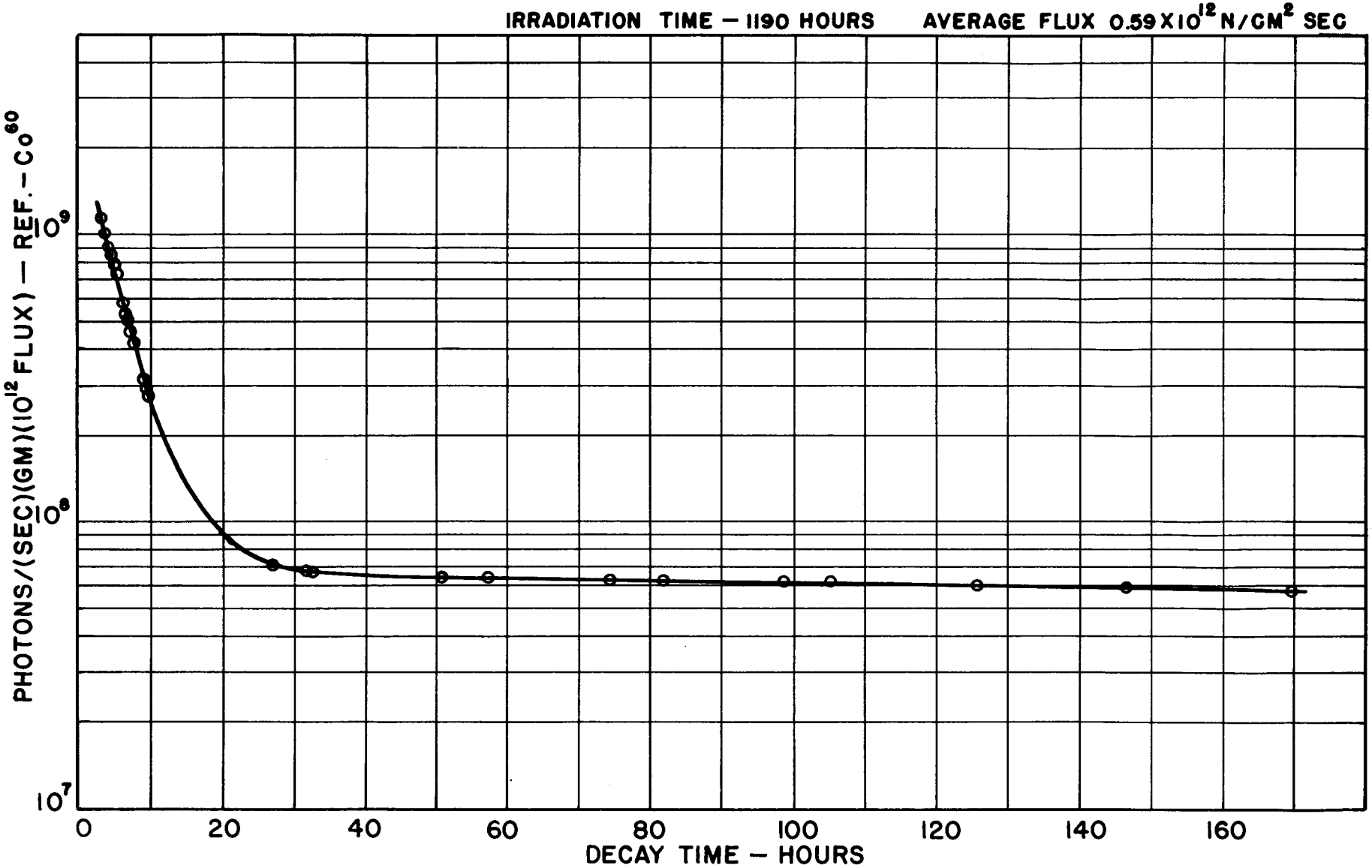

FIG. 2 GAMMA DECAY OF 310 STAINLESS STEEL

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DWG. 10698

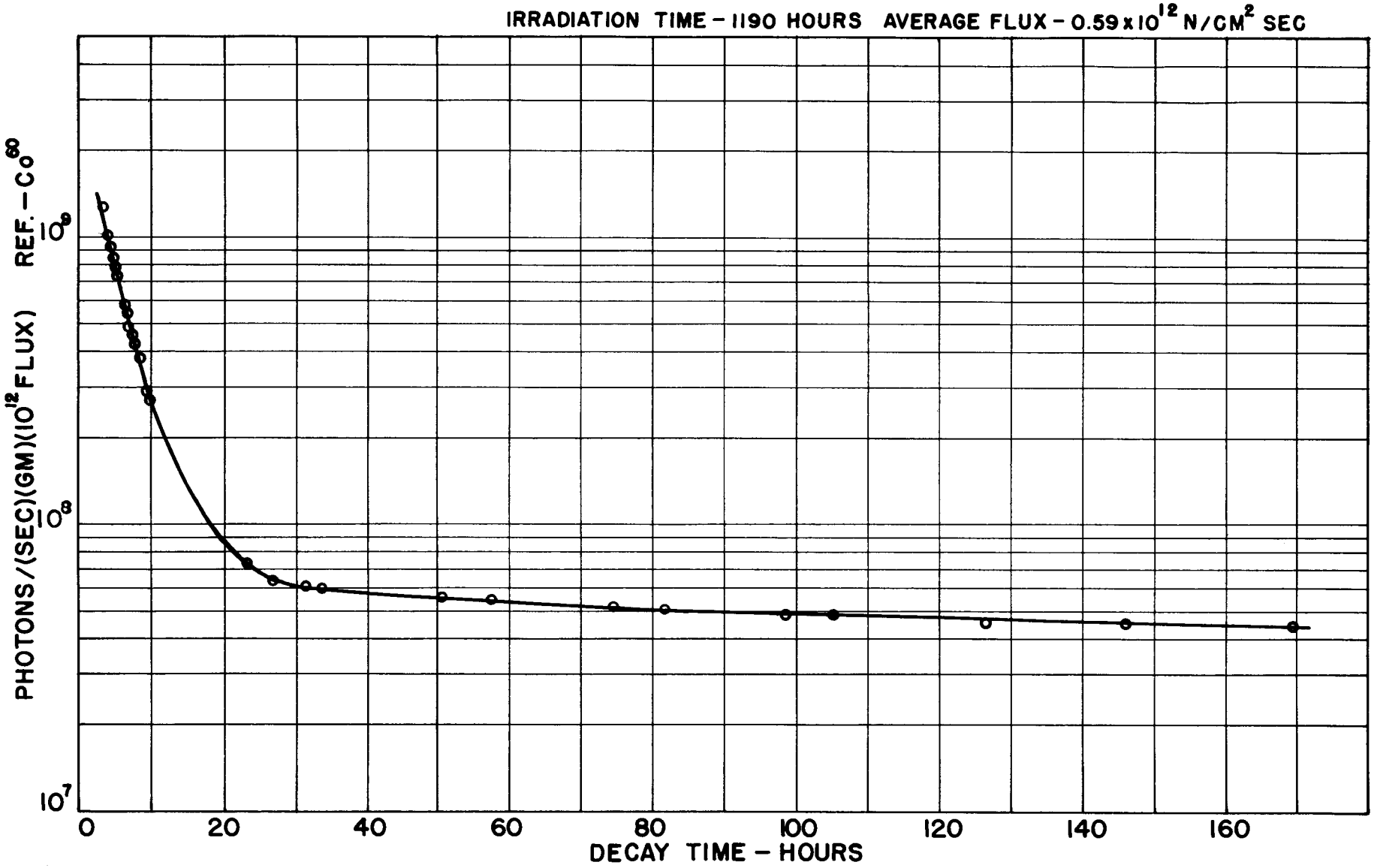

FIG. 3 GAMMA DECAY OF 316 STAINLESS STEEL

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DWG. 10699

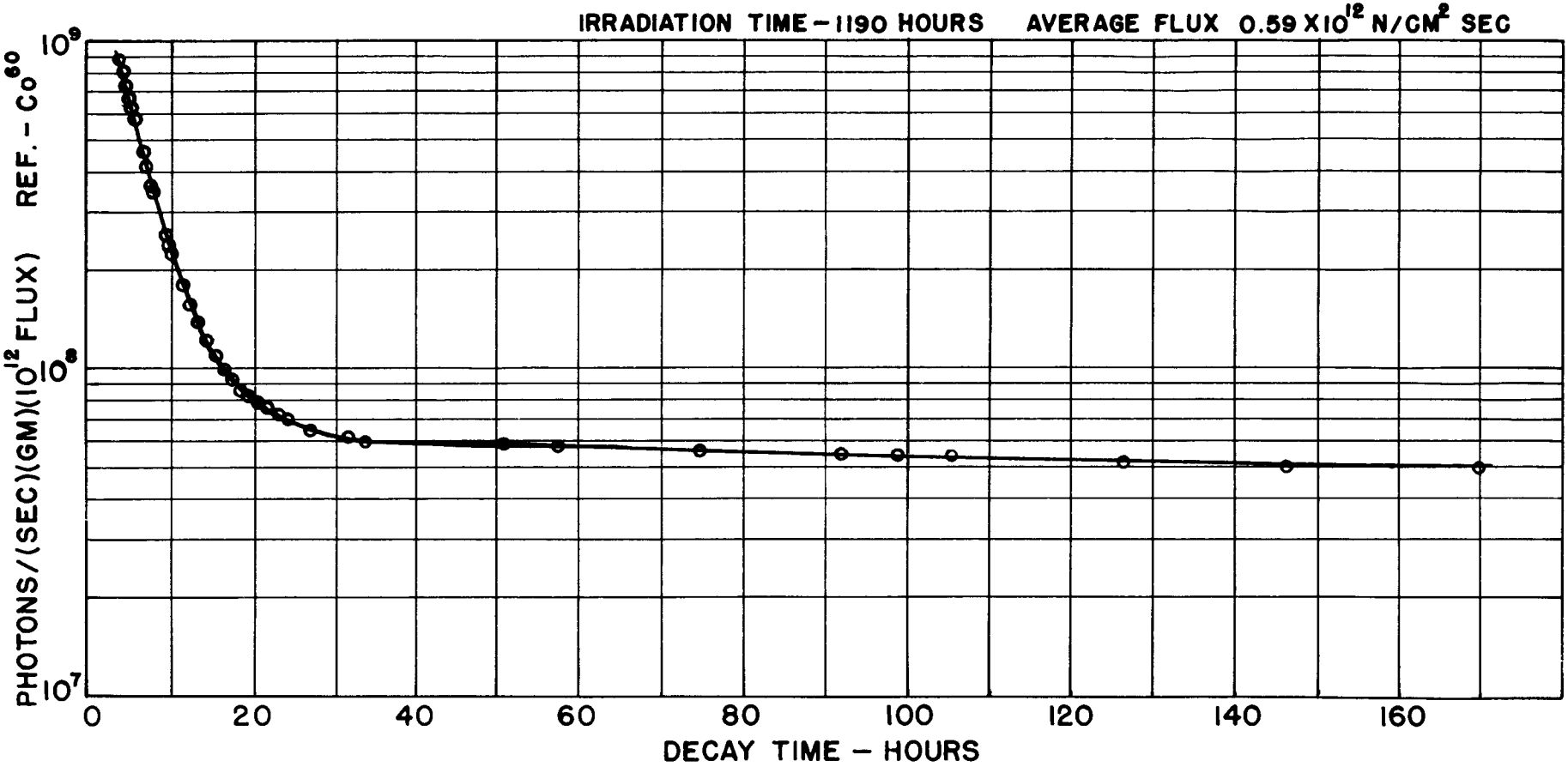

FIG. 4 GAMMA DECAY OF 347 STAINLESS STEEL

UNCLASSIFIED
DWG. 10700

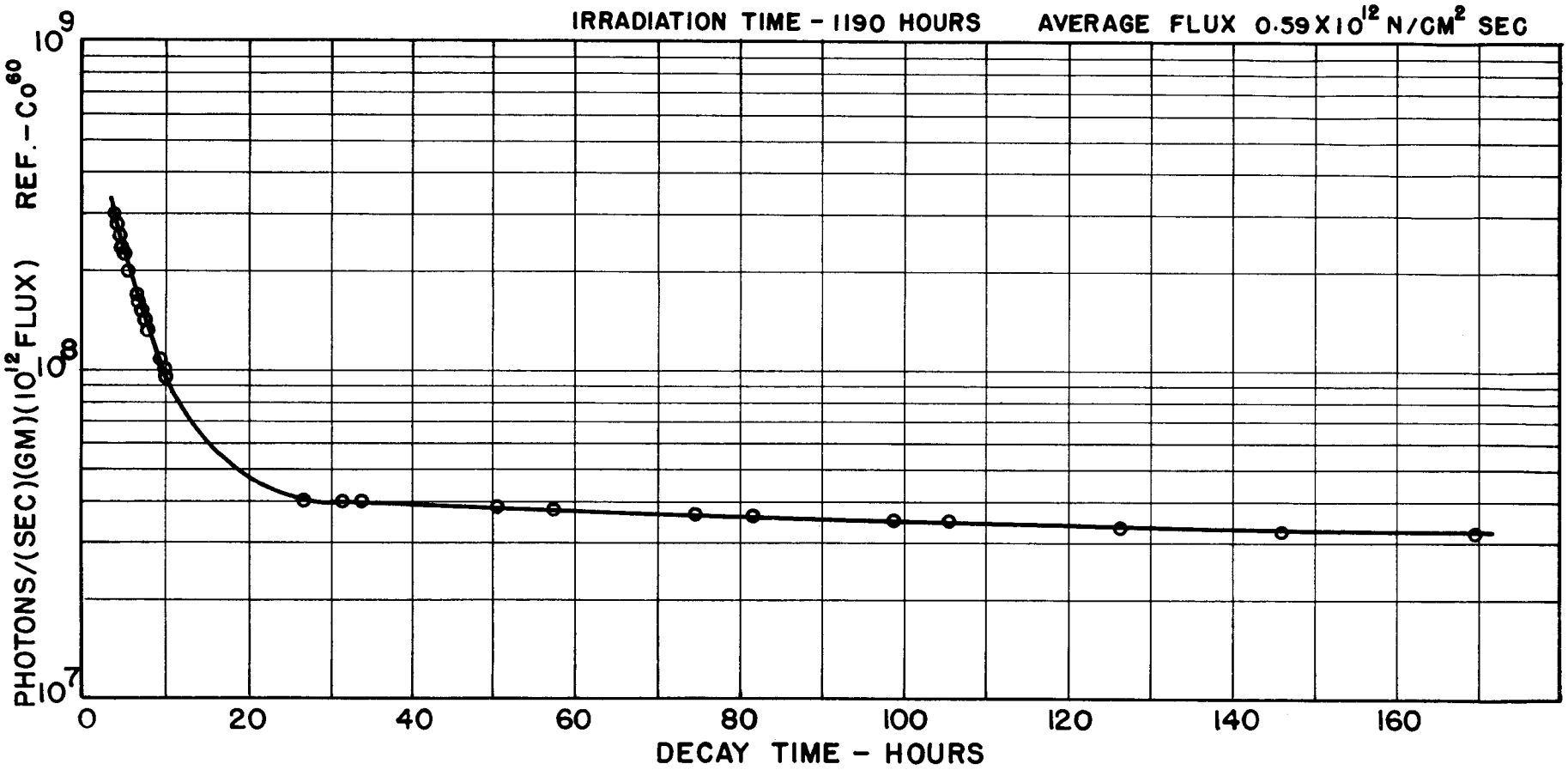

FIG. 5 GAMMA DECAY OF 405 STAINLESS STEEL

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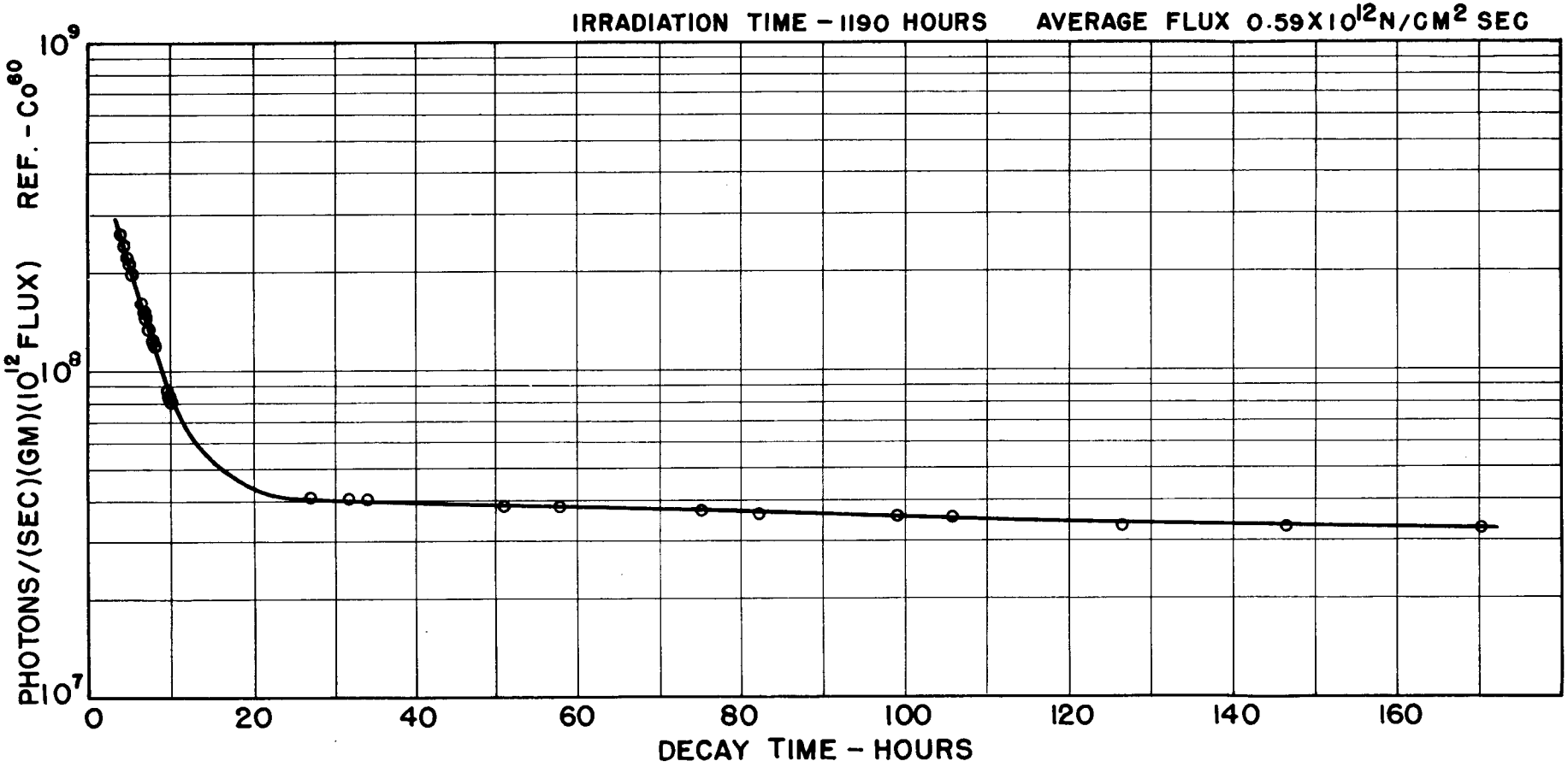

FIG. 6 GAMMA DECAY OF 430 STAINLESS STEEL

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DWG. 10702

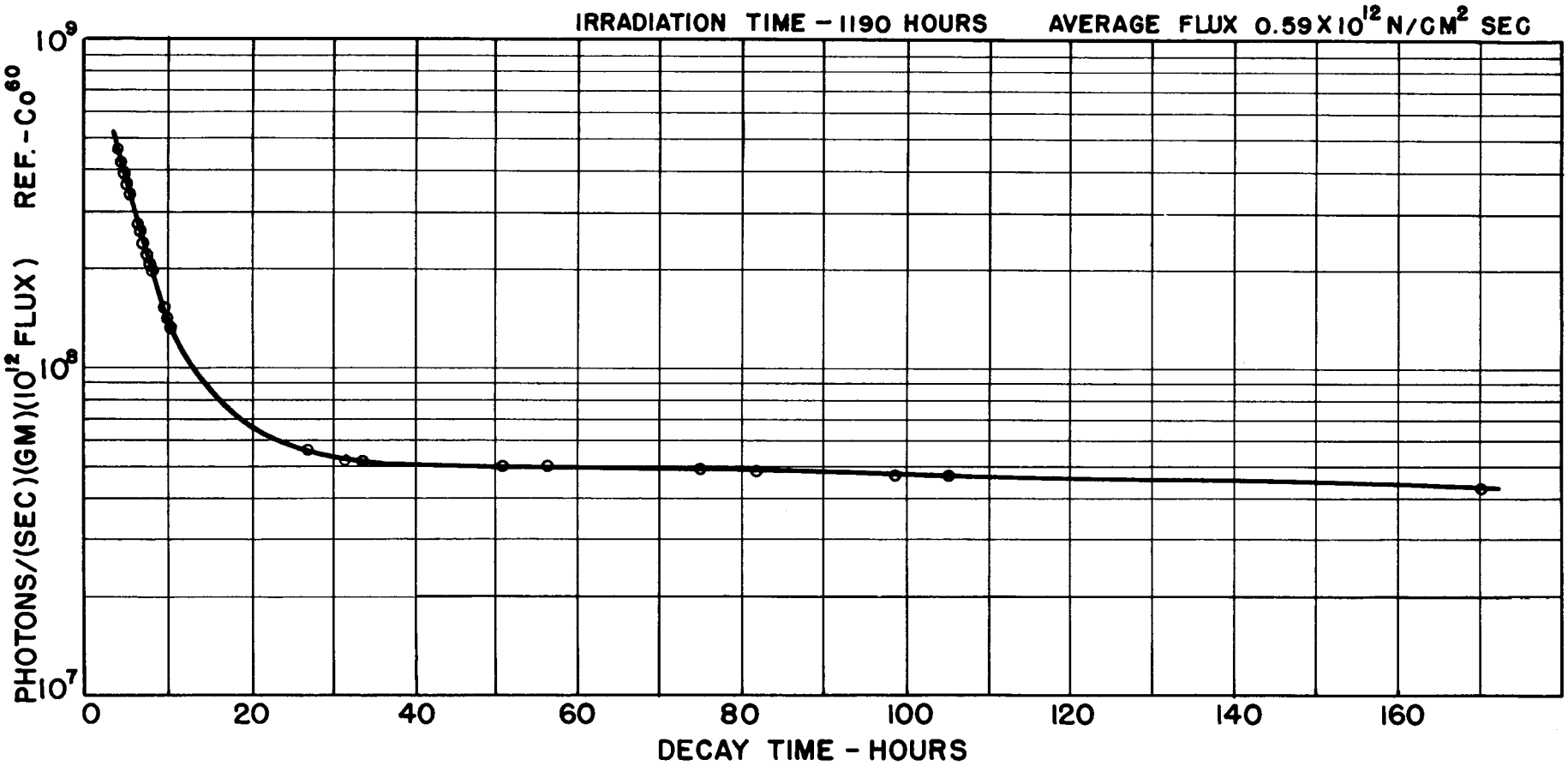

FIG. 7 GAMMA DECAY OF 445 STAINLESS STEEL

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DWG. 10703

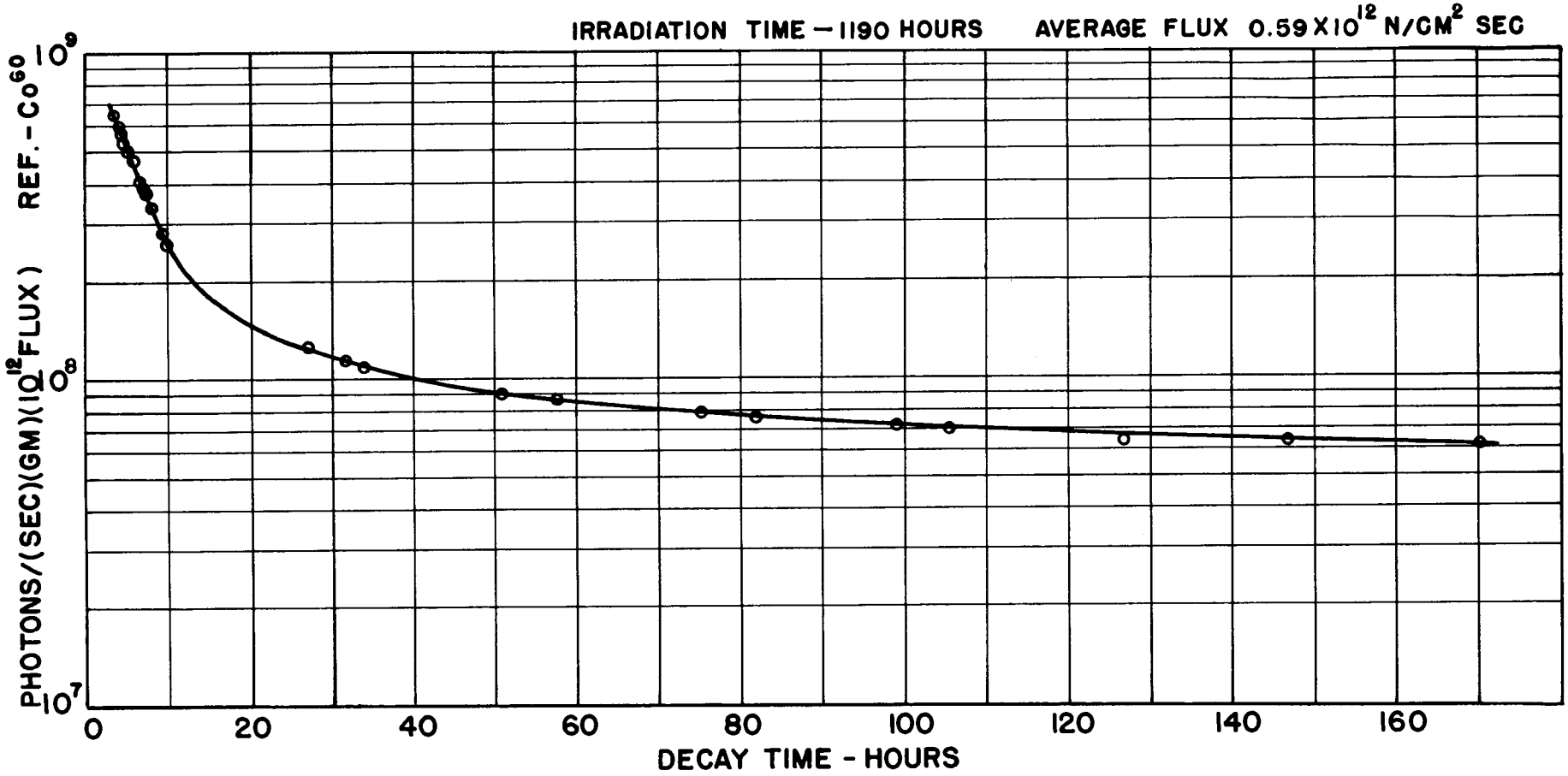

FIG. 8 GAMMA DECAY OF C20 STAINLESS STEEL

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DWG. 10704

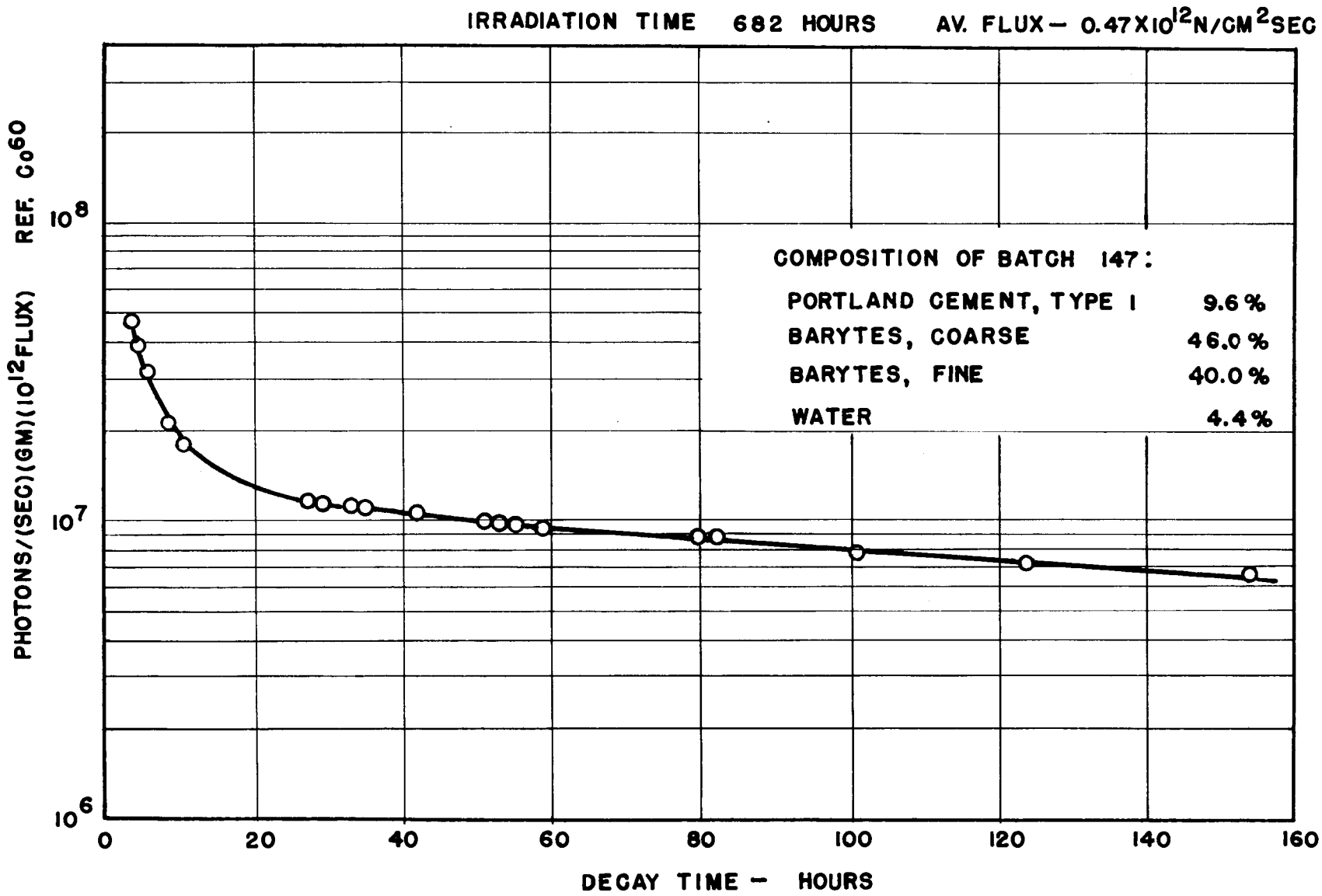

FIG. 9 SHORT TIME GAMMA DECAY OF BARYTES CONCRETE (BATCH 147)

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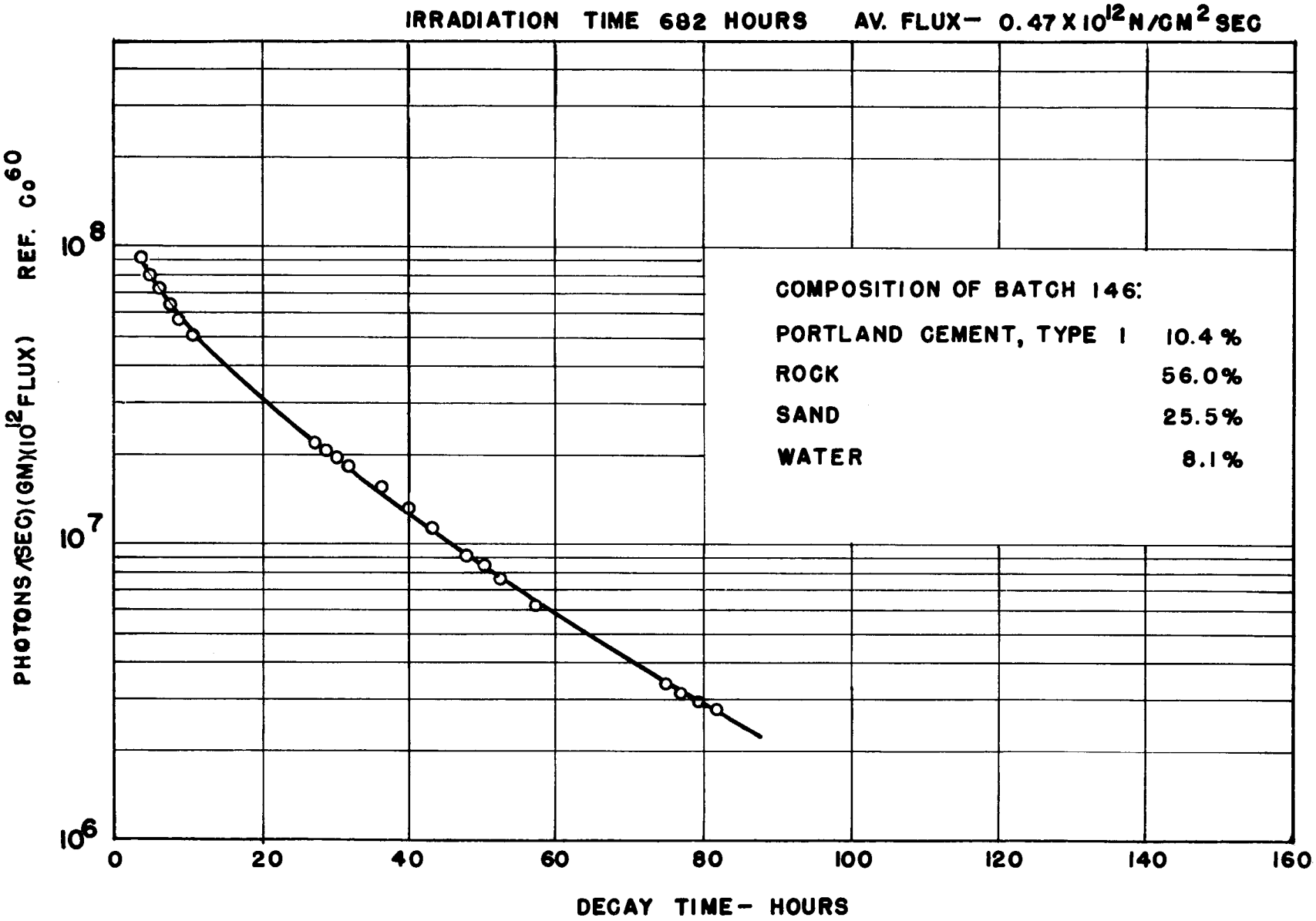

FIG.10 SHORT TIME GAMMA DECAY OF PORTLAND CONCRETE (BATCH 146)

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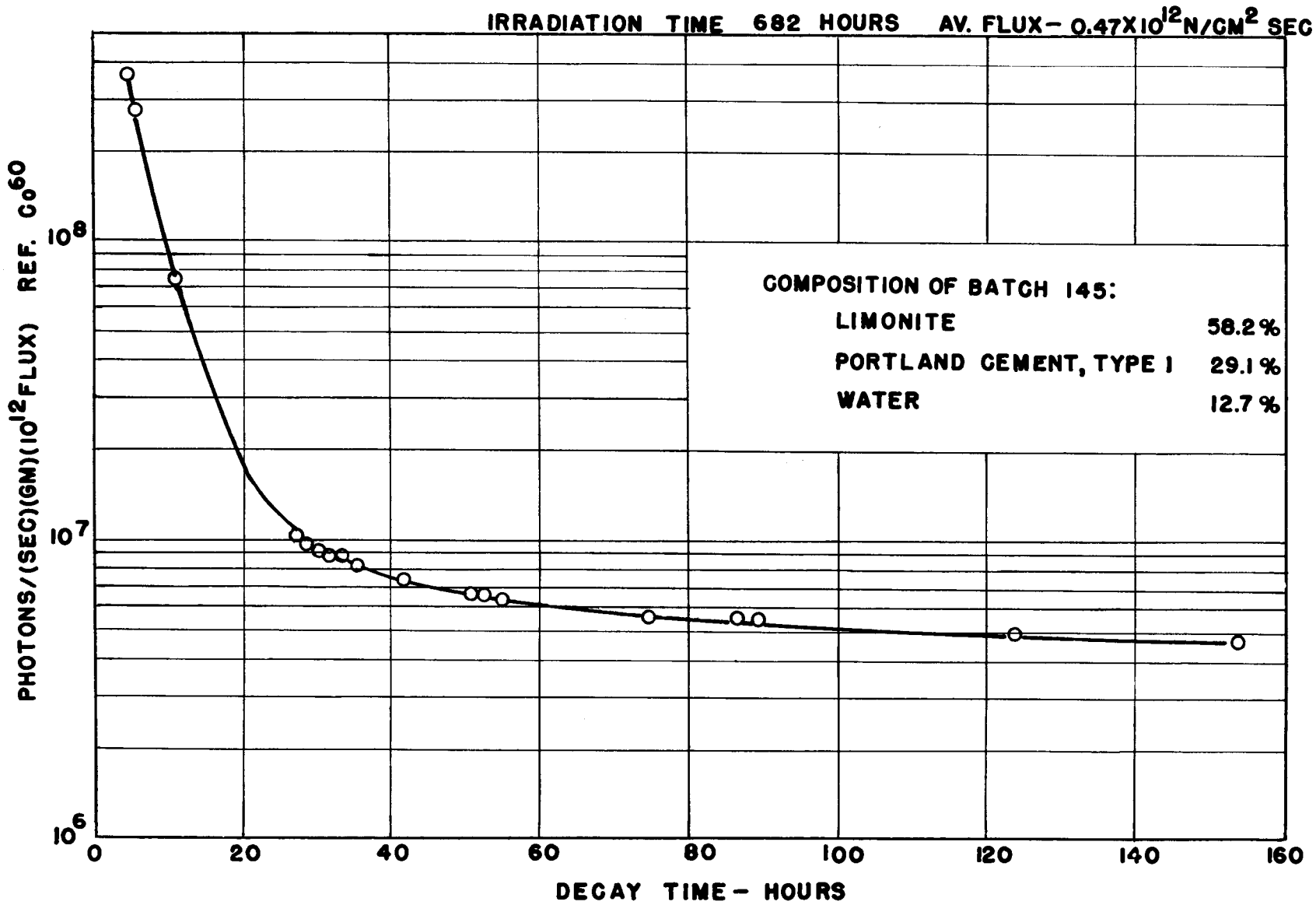

FIG. 11 SHORT TIME GAMMA DECAY OF BROOKHAVEN CEMENT (BATCH 145)

UNCLASSIFIED
DWG. 10707

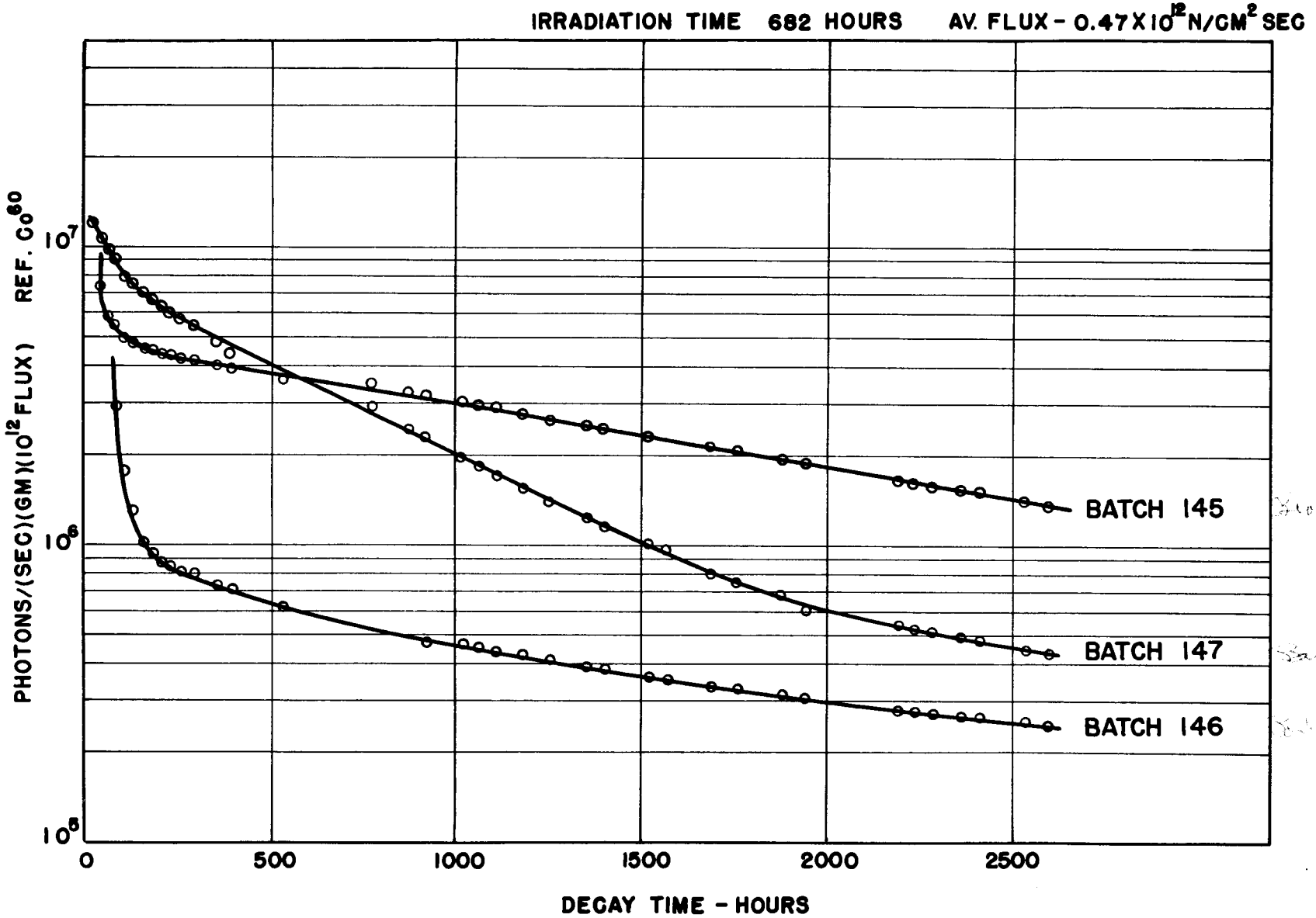

FIG. 12 LONG TIME DECAY OF CONCRETES

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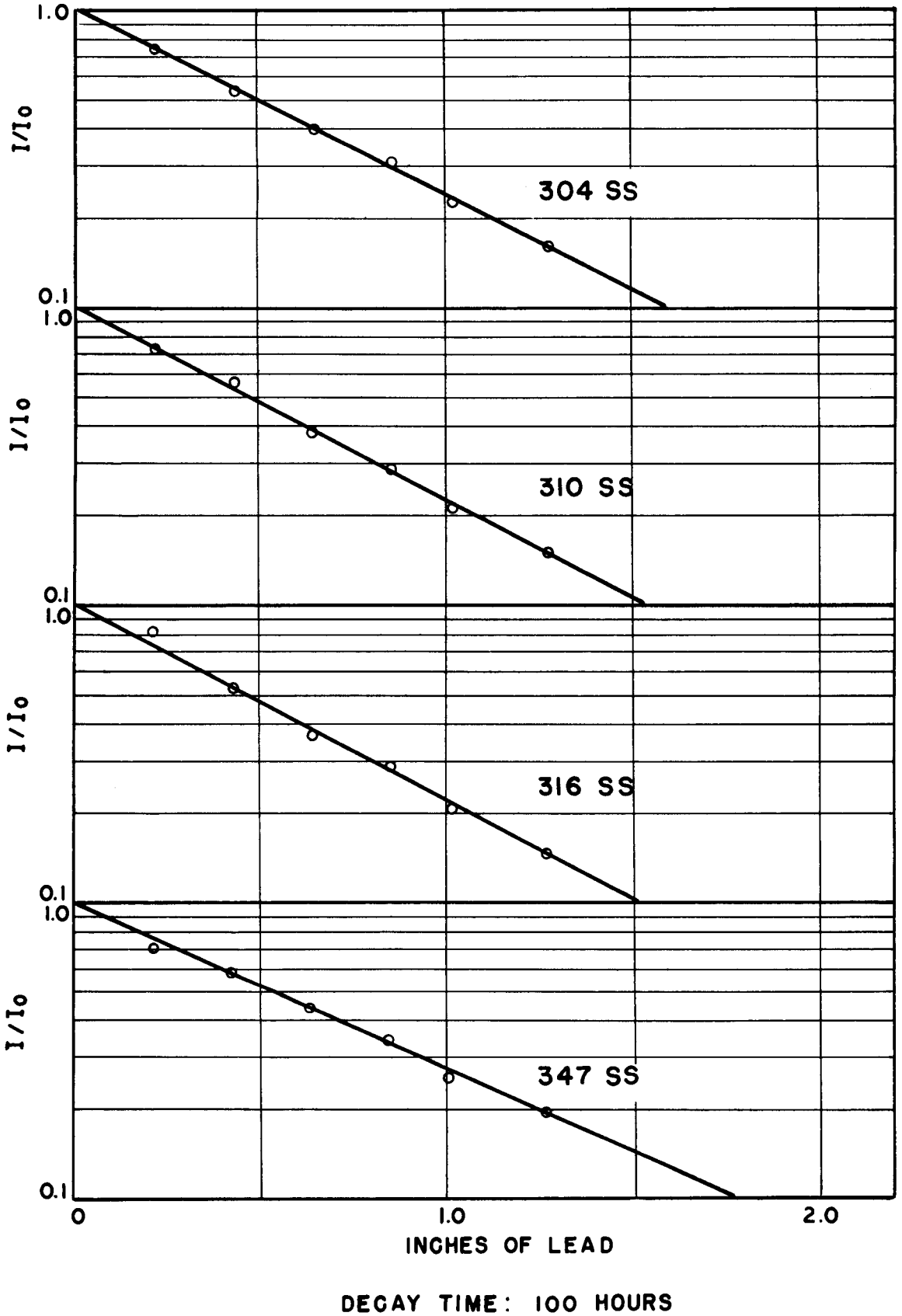

FIG. 13 ABSORPTION OF GAMMA RADIATION FROM STEELS

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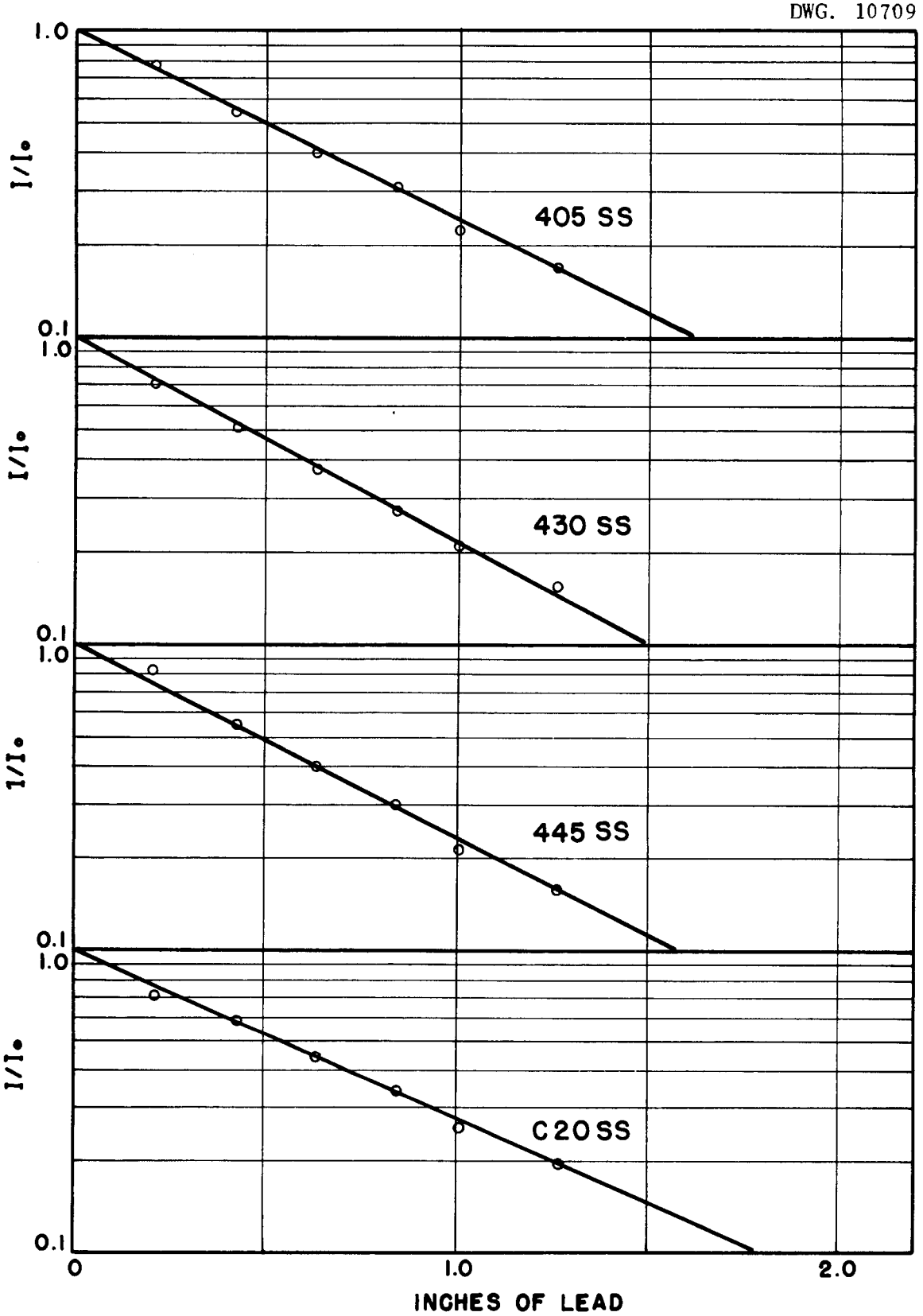
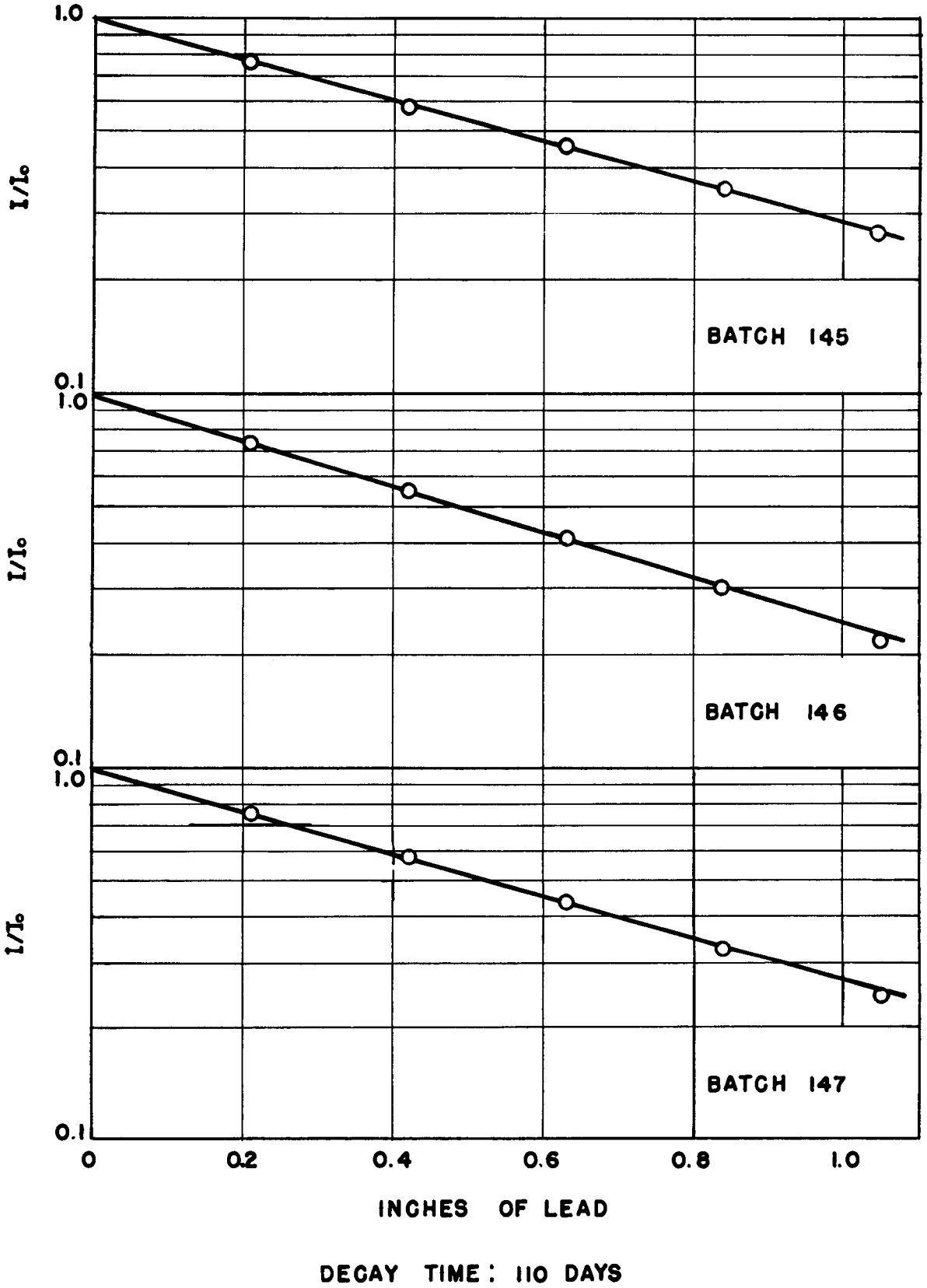


DECAY TIME: 100 HOURS

FIG. 14 ABSORPTION OF GAMMA RADIATION FROM STEELS

UNCLASSIFIED
DWG. 10710



DECAY TIME: 110 DAYS

FIG. 15 ABSORPTION OF GAMMA RADIATION FROM CONCRETES