Errata added p. 37

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THE NEW BULK SHIELDING FACILITY AT OAK RIDGE NATIONAL LABORATORY

AEC RESEARCH AND DEVELOPMENT REPORT

WILLIAM M. BREAZEALE

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PHYSICS DIVISION

THE NEW BULK SHIELDING FACILITY AT OAK RIDGE NATIONAL LABORATORY

William M. Breazeale

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By Authority Of AEC 11-29-54

By E. Goldberg Date 2.25.72

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FOREWORD

The cooperation of a large number of ORNL employees made possible the success of the work described in this report. Without detracting from the efforts of the others, mention should be made of a few who carried a large part of the responsibility for placing this facility in operation on the scheduled date.

The building and pool were designed by the ORNL Engineering Department under the supervision of Gibson Morris. Construction was accomplished by John A. Johnson and Sons, contractors. The reactor, reactor bridge and instrument bridge and associated equipment were also designed by the ORNL Engineering Department, under the direction of W. R. Gall, of the Reactor Technology Division, and C. W. Angel, and built in the ORNL Shops. The field engineer in charge of the mechanical installation was K. E. Jamison. The safety and control circuits were designed (or in some cases adapted from MTR designs) by T. E. Cole, and the installation was under the supervision of J. E. Owens. Procurement and, when necessary, design of the measuring equipment were the responsibility of J. L. Meem. Some of the equipment was adapted from C. E. Clifford's "Lid Tank". A great deal of the equipment was constructed in the ORNL Instrument Department under the direct supervision of F. W. Manning.

At present writing, the Facility is being operated by J. L. Meem and is part of the ORNL Shielding Group headed by E. P. Blizard.

[redacted]

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THE NEW BULK SHIELDING FACILITY AT ORNL

HISTORY

In the fall of 1949 it became evident that a shield testing facility with a much stronger neutron and gamma source than the one in the "Lid Tank" was necessary if experimental work on shields was to be prosecuted successfully. In addition, it was apparent that the facility should be much more flexible than the lid tank. Several designs were considered, and it was finally decided that a low power reactor operating in a pool of water would most nearly meet the requirements of a bulk shield testing facility as they were known at that time. Accordingly a proposal[1,2] was submitted to the Atomic Energy Commission, and shortly thereafter a directive to proceed with the work was received. The building, with standard plumbing and electrical equipment, and the pool were constructed under a lump sum bid by John A. Johnson and Sons, Inc., the low bidder. Bids were opened on May 1, 1950 and work began shortly before June 1st. One hundred and fifty days were allowed for this construction. Laboratory personnel moved into the building November 10th and began installation of the reactor and instrument bridges, although the contractor did not complete all details of his contract until January, 1951.

The reactor installation was completed on December 14th and the first critical experiment was successfully concluded on December 17th.

DESCRIPTION OF BUILDING

The facility is housed in a steel frame building with Q (corrugated metal) siding. The building is 77 ft. by 51 ft. overall, and houses the pool and reactor in a bay 77 ft. by 32 ft. and an office, two air conditioned instrument rooms, and a small shop. The ORNL construction drawings describing the building, pool, plumbing, electrical work, etc., are listed in Appendix I. The pool is 40 ft. long, 20 ft. wide, and 20 ft. deep, and in addition has a well 14 ft. by 14 ft. by 5 ft. deep centered 15 ft. from the north end. Fig. 1 is a three-dimensional sketch of the pool and reactor bridge, but without the instrument bridge, and

---

[1,2] A list of references will be found at the end of this report.

[redacted]

[redacted]

Figs. 2 and 3 are cross-sections of the pool. An aluminum gate 12 ft. by 21 ft. can be lowered between guides 10 ft. from the south end of the pool, blocking off this end and permitting the rest of the pool to be pumped dry. The gate was made by the Combustion Engineering - Superheater Corporation in Chattanooga, Tennessee. Fig. 4 is a view of this gate ready to be lowered into position.

The removable concrete blocks were made from the following mix suggested by Arnold Kitzes. Quantities given are for one cubic yard of mix.

2210 lbs. Barytes aggregate ($BaSO_4$) Coarse
1050 lbs. Barytes aggregate ($BaSO_4$) Fine
1220 lbs. Limonite ($2Fe_2O_3 \cdot 3H_2O$)
610 lbs. Portland Cement
46 gallons water

The blocks have a density of 3.2 and exhibit good neutron and gamma attenuating properties. Measurements to determine the exact attenuation characteristics of these blocks are scheduled. In addition to giving an adjustable floor level in the bottom of the pool, these blocks will be used to piece out shielding samples.

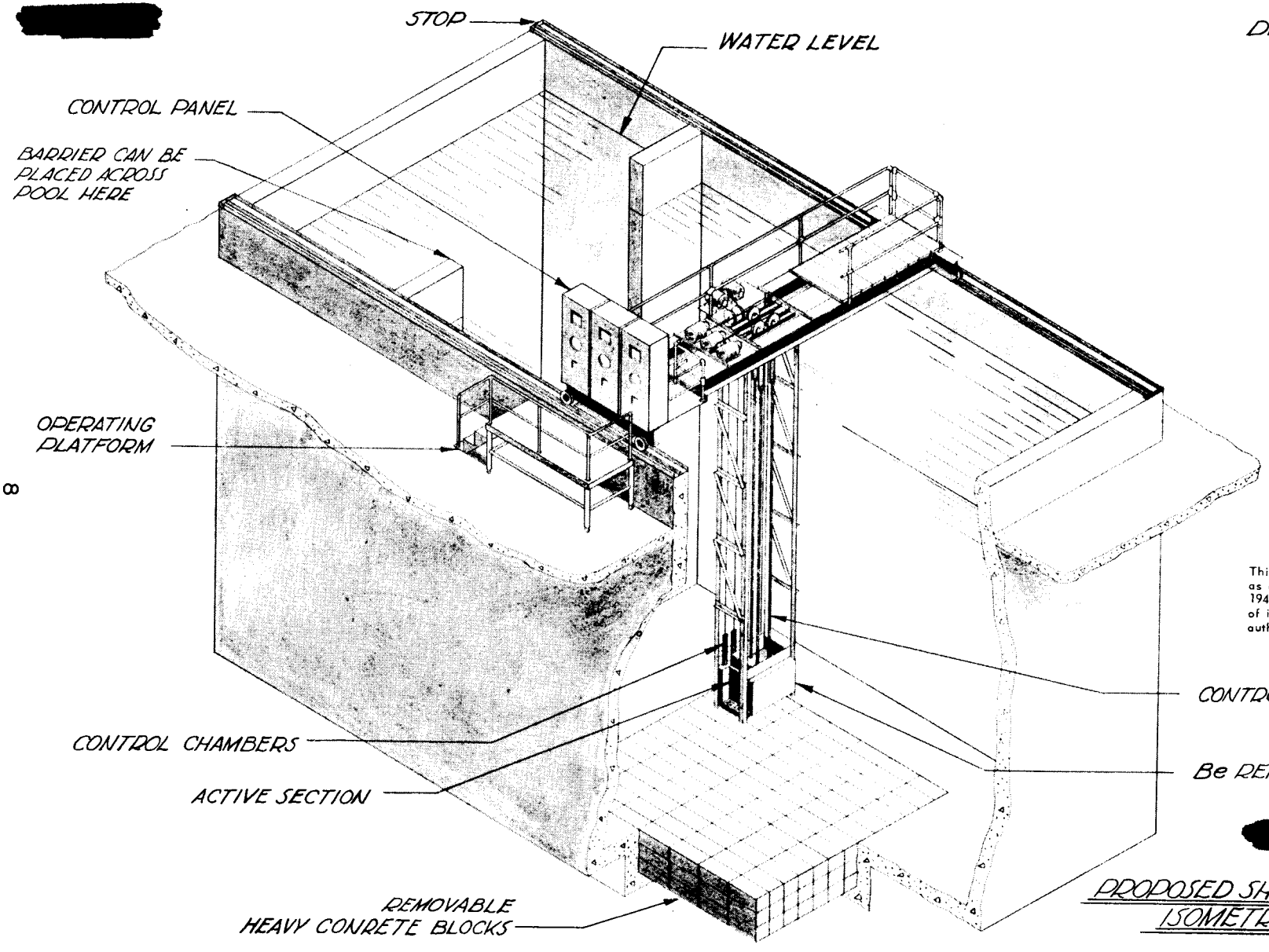
The reactor is suspended from a movable bridge by an aluminum frame so that the top of the active lattice is 16 ft. below the surface of the water. (A complete description of the reactor is given in the next section). It is possible to roll the bridge and reactor to the south end of the pool, replace the aluminum barrier and empty the rest of the pool. Personnel can then work on a shielding sample in the empty section and be protected from radiations from fission products in the fuel elements by the water behind the aluminum gate.

It was expected that this facility would be used to make attenuation measurements on bulk samples as well as to test mock-ups of practical shields. At the time it was designed, calculations indicated that the source should have sufficient strength to permit rough neutron and gamma ray spectroscopy through shields with overall attenuations up to 10^8. Such measurements require a leakage flux (fission spectrum) of about 10^{11} neutrons/cm^2/sec. which, with this type of reactor, corresponds to an operating power of some 10 kw. With this power, thermal neutron attenuations several magnitudes greater than 10^8 are measurable.

[redacted]

[redacted]

DWG. #8332



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PROPOSED SHIELDING FACILITY
ISOMETRIC LAYOUT

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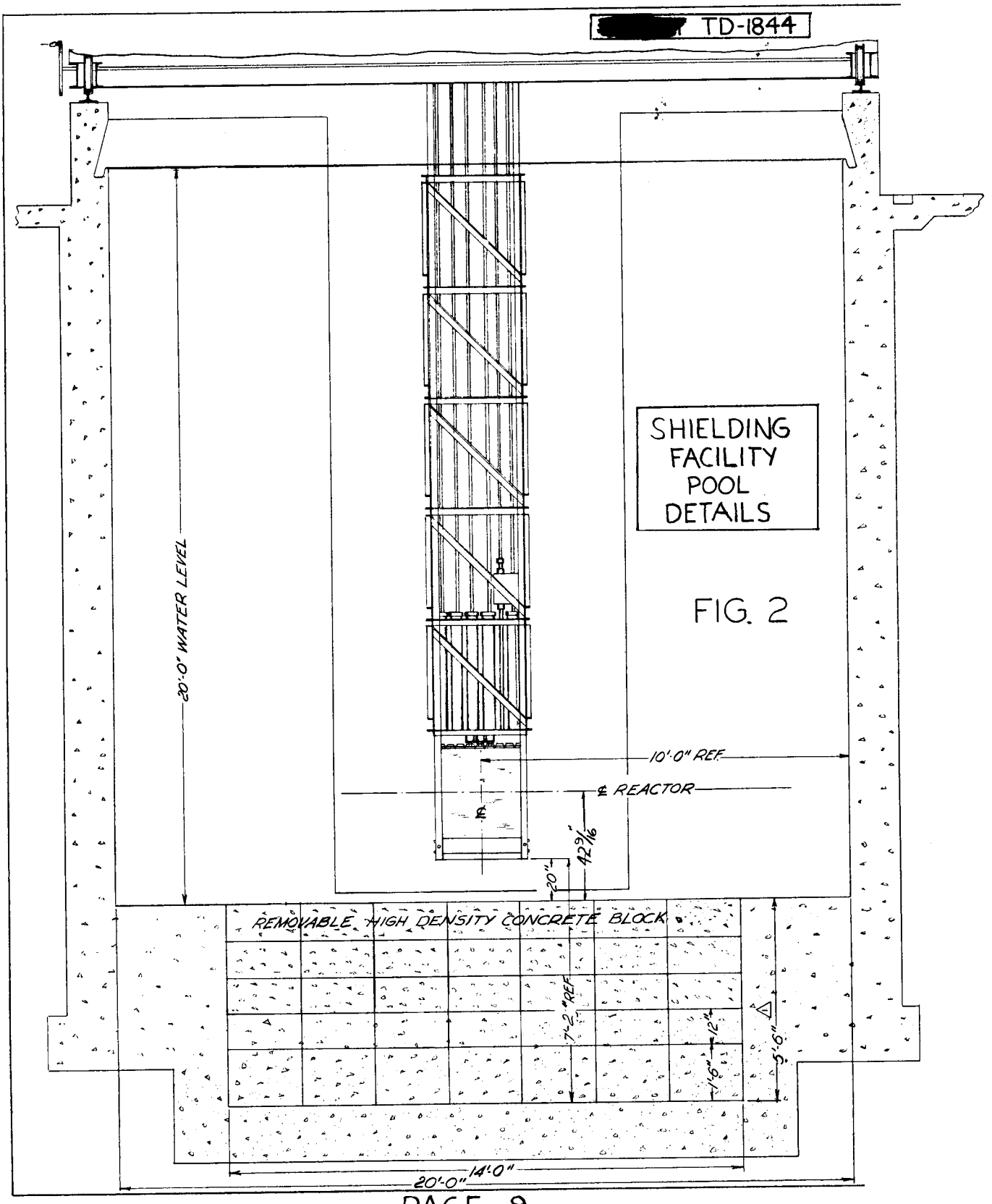
FIG. I

[redacted] TD-1844



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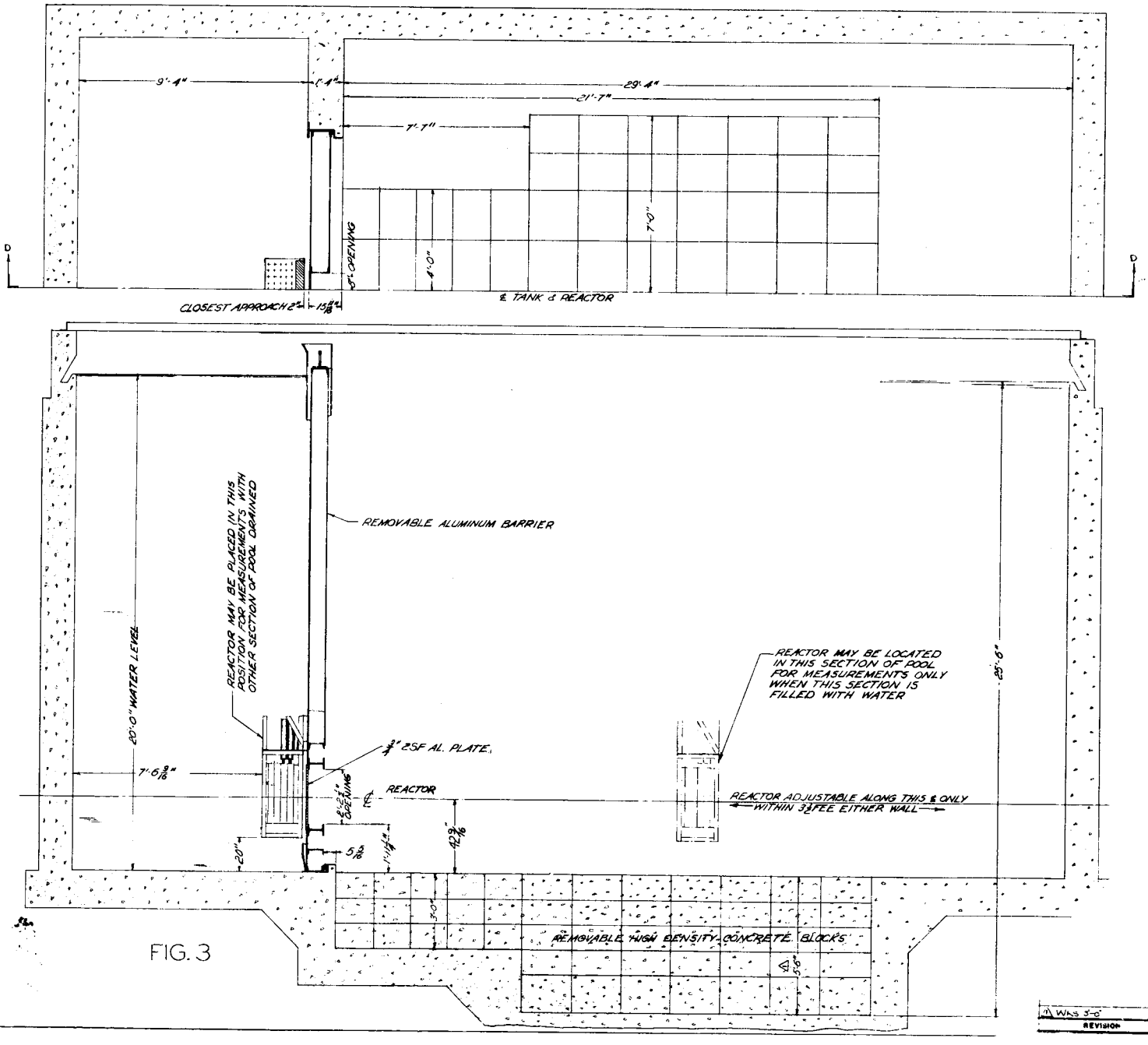

FIG. 3

NOTE

FOR ADDITIONAL DIMENSIONS & SECTIONS SEE DWG. TD-1844

[redacted]

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WAS 3'-0"	GILL 6-22-50	DRAWN BY GILL / TRACED BY	DATE 6-15-50 / DATE
REVISION	APPD. DATE		SCALE 1/2"=1'-0" / DRAWING NO. TD-1845 / REV. a

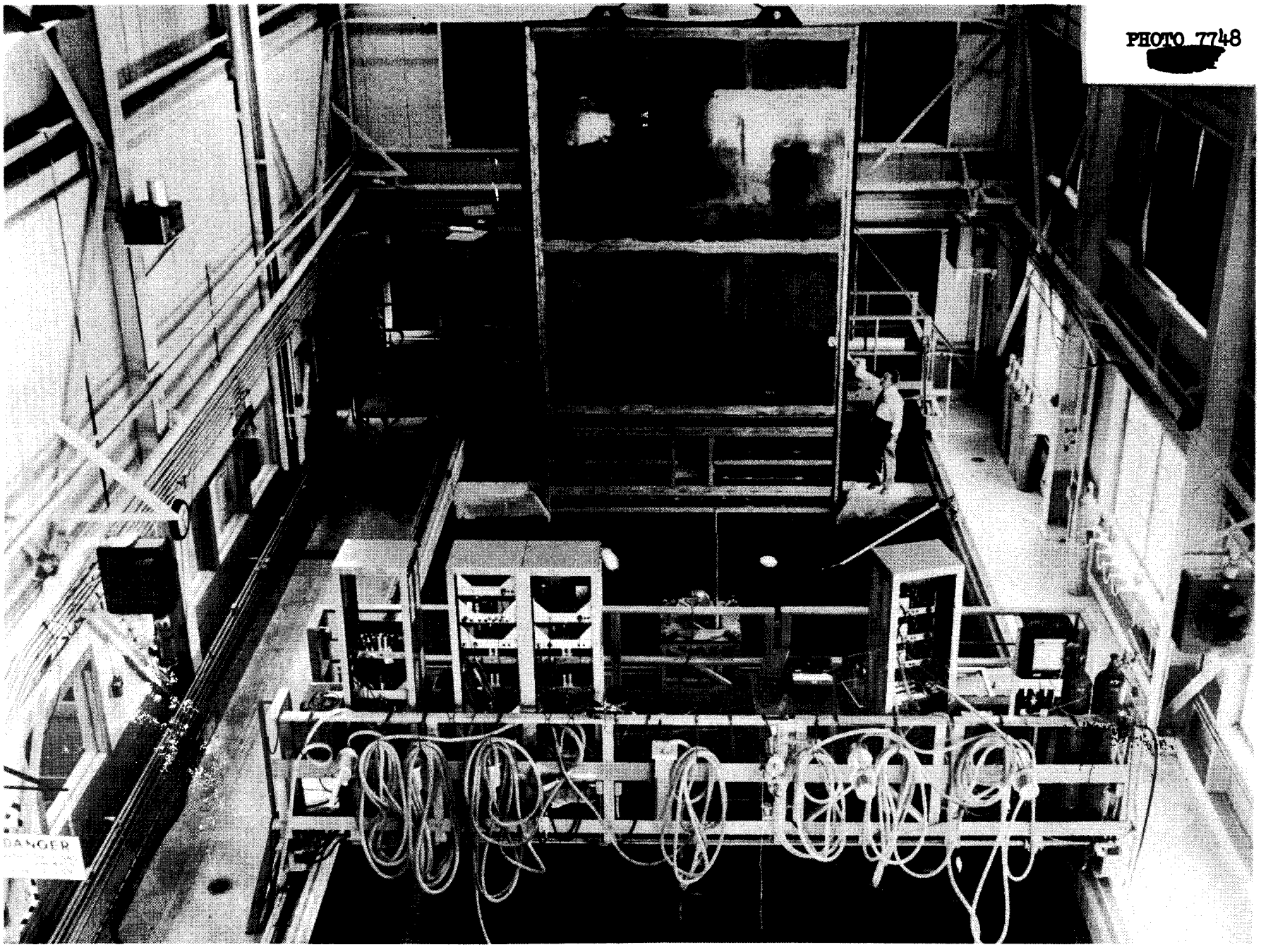

FIG. 4 ALUMINUM GATE READY TO BE LOWERED IN PLACE

The reactor Safeguard Committee found no fault with our plan of operation at 10 kw., and we expect to make all of our shielding measurements without exceeding this power*. Actually, the reactor, control equipment, etc., were designed for operation at power levels up to 100 kw., and the depth of water over the reactor is sufficient to protect personnel at this level. It is calculated that convection cooling will still be adequate at levels as high as 1000 kw., but operation at this power will require some modification of the reactor and controls and probably the addition of some 2 in. of lead shielding. One more cadmium-lead shim rod should be supplied before operating at megawatt levels to take care of xenon poisoning and depletion.

Operation of the reactor under water as a neutron and gamma source for bulk shielding measurements has several advantages. Beside the fact that the water supplies foolproof protection to the operating personnel, the following can be listed:

1) Absorption in water and human tissues is about the same. Thus the absorption of radiation in water directly behind the shield corresponds to the absorption in a person similarly located. Further, according to a calculation by W. S. Snyder of the ORNL Health Physics Division, the amount of radiation absorbed in the layers close to the shield is substantially independent of the material (air or water) more than 30 cm. from the shield.

2) Water surrounding the source permits extremely simple arrangements for metal-water shield tests.

3) The source is accessible through the water without the removal of heavy gamma shielding such as would be required for fission product radiations, etc.

4) The water cools the source in a simple and more than adequate fashion.

5) If the shielding surrounding the source becomes radioactive, it can be easily replaced (by emptying and refilling the pool) and thus the instrument background should always be low.

6) The water shields the instruments from outside radiation.

7) The source is easily moved toward or away from the shield sample. (The reactor is simply moved under water).

---

*Should it become desirable to raise the power above 10 kw., permission will be requested through the usual channels. Such eventuality may arise if much submarine work is undertaken, for this installation requires much more attenuation.

[redacted]

Among the disadvantages of operation under water are:

1) Instruments must be canned and located some distance from the support.

2) Liquid samples (except water) and solids which are hygroscopic must be canned.

3) Transmission at an air-shield interface is different from that at a water-shield interface, and the relation between the two must be determined, or at least confirmed, by experiments.

4) Measurements on a divided shield are difficult. This is because it is not possible to measure the scattering in the air between the shields directly. This will be discussed in detail in a later report.

It is possible in this facility to make measurements on shield samples in air. To accomplish this, the reactor is positioned behind and close to the aluminum gate as shown in Fig. 3. The rest of the pool (north of the gate) can be drained and the shielding sample assembled against the gate. The part of the pool south of the aluminum gate remains full of water which cools and moderates the reactor. In a test of this kind, either the shielding sample must furnish protection from neutrons and gammas for the operating personnel, or another shield must be constructed behind the sample.

A second set of fuel elements which is not intended for use at power levels greater than a watt or so has been fabricated. These will be used for low level measurements where it is necessary that there be no high gamma activity from fission products.

DESCRIPTION OF REACTOR AND POOL

A list of the drawings covering the reactor, reactor bridge, fuel elements, and controls is given in Appendix I.

The reactor is a water cooled and moderated, partially BeO reflected, thermal pile. The fuel elements are essentially the same as those designed for the Materials Testing Reactor.[3,4,5] This design was selected because laboratory personnel had experience in building and operating such a reactor (in the M.T.R. mock up); hence development time and expense would be saved. Further, in comparison with a homogeneous reactor, such as the water boiler, there is no problem of disposal of gaseous fission products. A photograph of a fuel element is shown in Fig. 5 and details in Dwgs. D-7205 and D-7206. It consists of 18 convex sandwiches, each 3 in. wide, 24 in. long and 0.060 in. thick. A complete assembly contains a little less than 140 grams of U^{235}. The sandwiches are 2S aluminum clad with 72S; the latter corrodes preferentially. Taking into account the structural members, the aluminum to water ratio is about 0.7. The square lower end boxes of the MTR elements are replaced in this design with round end boxes and the upper end boxes omitted.

[redacted]

PHOTO 7610

[redacted]

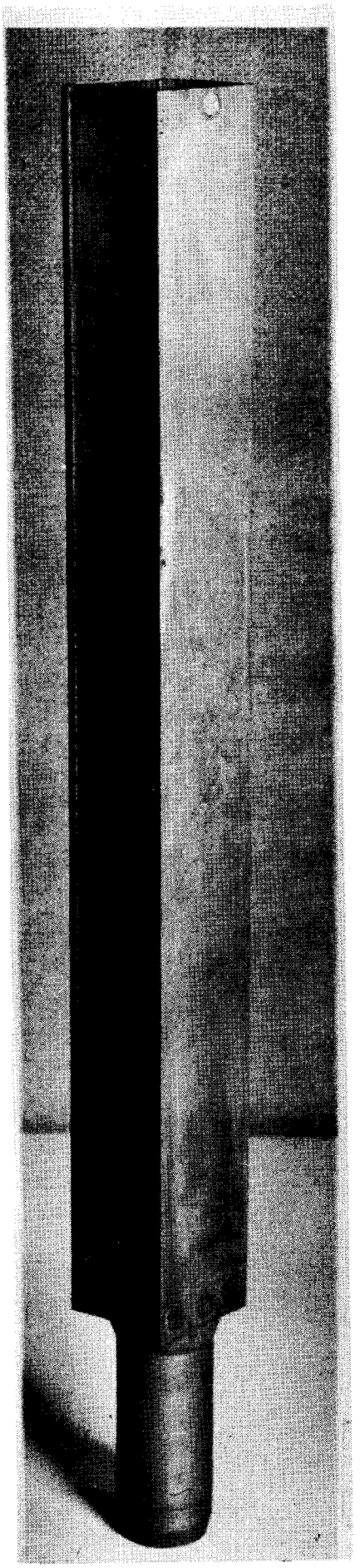

FIG. 5 FUEL ELEMENT

[redacted]

The fuel elements rest in an aluminum grid 28 in. by 19 in. by 5 in. deep. An aluminum can containing BeO blocks occupies 4 in. of the 19 in. dimension. The free part of the grid is then 27 in. by 15 in. allowing an array of 9 elements by 5 elements to be loaded. Actually criticality is reached with many fewer elements, but this grid design permits considerable variation in the loading pattern. Fig. 6 is a photograph of the grid and Fig. 7 is another view showing the flat aluminum can of BeO blocks (reflector) in position in the rear (south side) of the grid. Above the reflector the cylindrical aluminum cans containing the ion chamber can be seen. In front of the cans are the electromagnets which support the safety and control rods. Aluminum cans with the same outline as the fuel elements and filled with BeO blocks have been procured and can be placed on the other three vertical sides of the active lattice so that the reactor is partially beryllium reflected.

The BeO blocks were cold pressed in the Bureau of Mines laboratory in Norris, Tennessee. These blocks have an average density of 2.3 compared to a theoretical density of 3.0.

The reactor has two cadmium-lead safety rods, each equivalent to between 3 and 4 percent $\delta k/k$ (depending on the loading) and a cadmium-lead regulating rod equivalent to 0.8 percent $\delta k/k$. Fig. 8 is a photograph of one rod. These rods have a cross-section roughly 1-1/4" by 2-1/2" and travel inside special fuel elements which have half the plates removed. Removing the plates provides an aperture 1-1/2" by 3" to contain the rods. Above these three special fuel elements are cylindrical guides for the electromagnets and the magnet armatures, the latter being attached to the top of the rods. Fig. 9 shows these three special fuel elements in place together with the magnet guides. Fig. 10 shows the reactor completely loaded. The elements visible in front are the BeO filled reflector units. The fuel and BeO reflector units can be loaded or unloaded from the grid by an operator standing on the reactor bridge with the aid of a special tool (Dwg. D-7221). This tool has a handle about 20 ft. long, a hook at the lower end and an operating lever at the upper end which releases the hook. The fuel elements can be handled under water at all times and the operator protected from radiations from fission products. When not in use, the "hot" fuel elements are stored in a rack under 8 ft. of water.

The reactor, controls and control panels are all mounted on a bridge which spans the pool and move as a unit; the only connection to the rest of the building being through the power cables. The operator is located on a movable platform beside the pool rather than on the bridge. When in operation, the reactor bridge is locked to the rails with heavy clamps.

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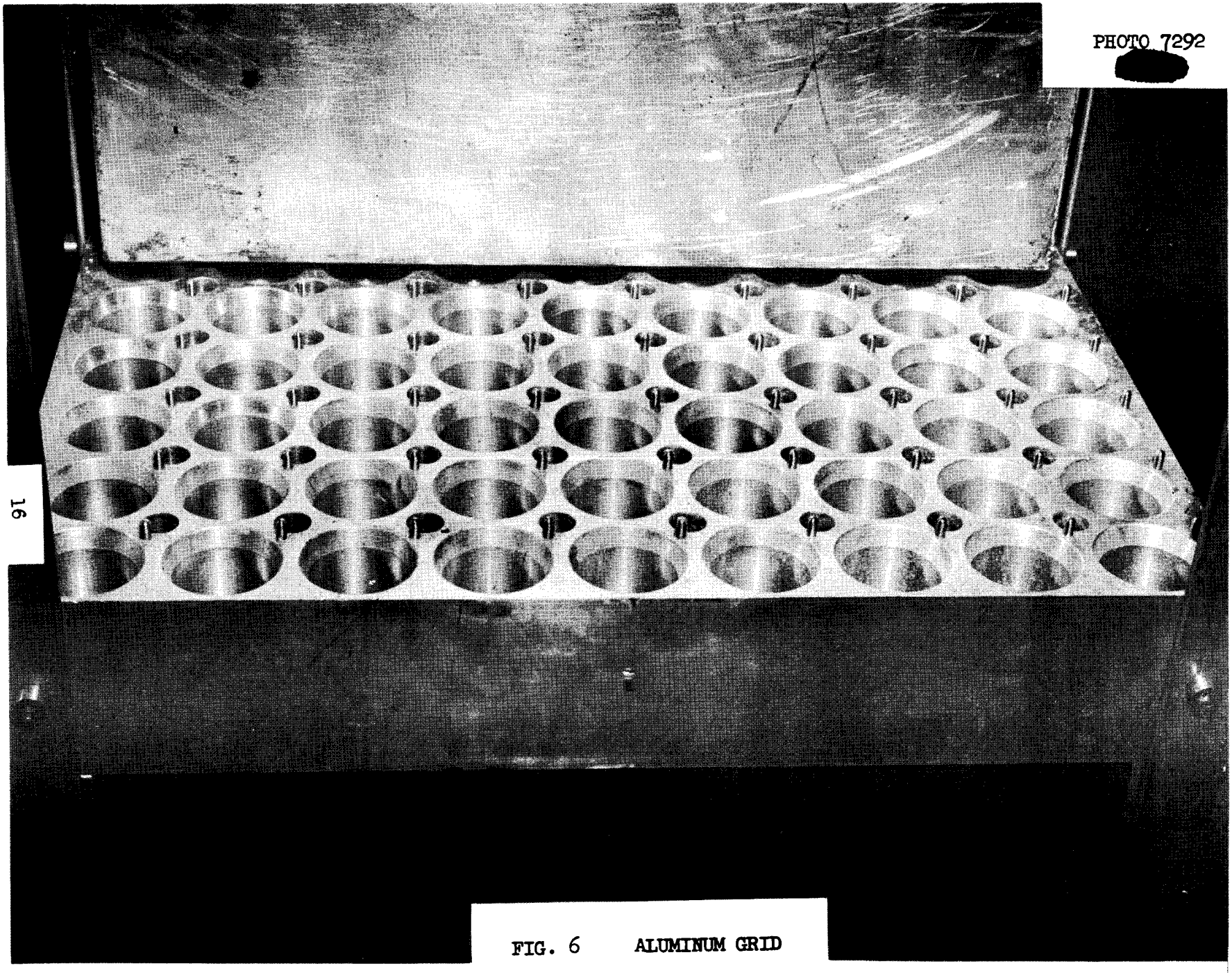
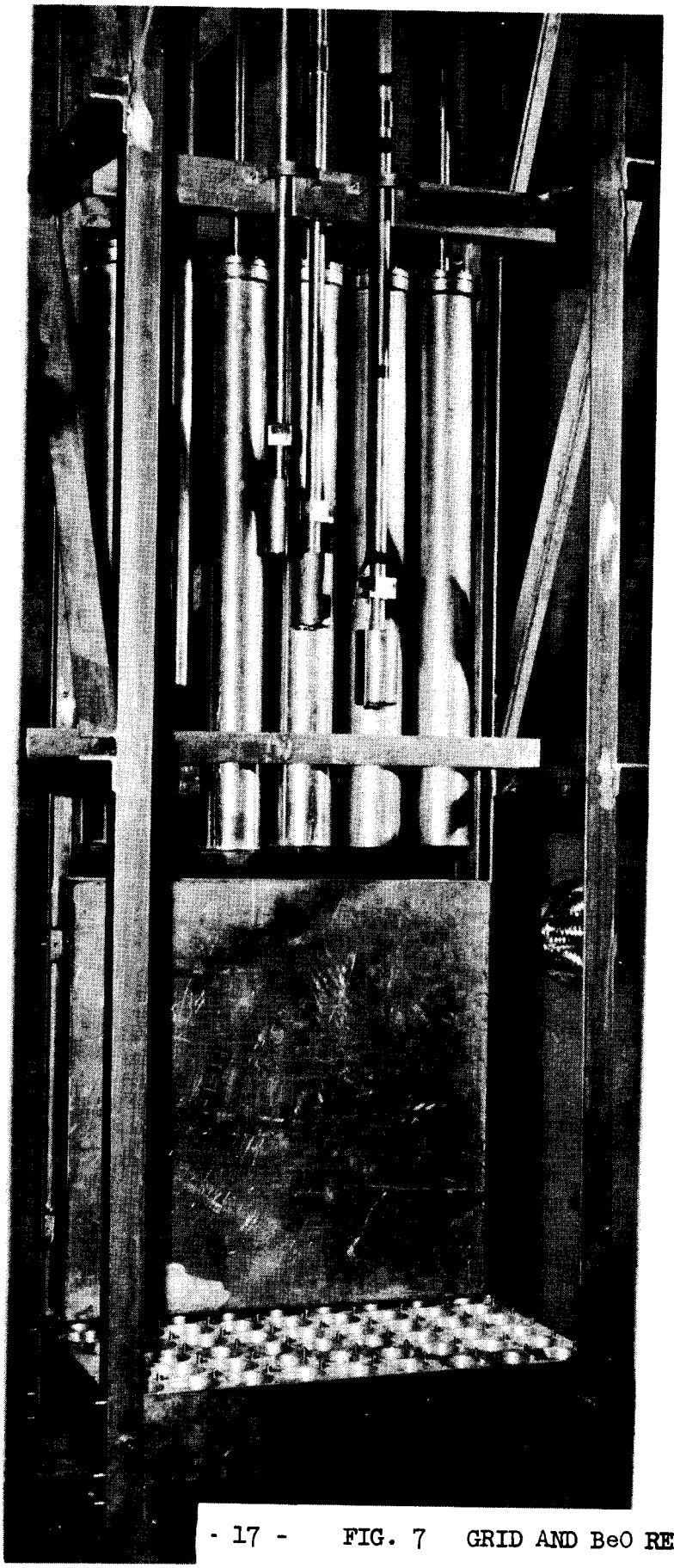


FIG. 6 ALUMINUM GRID

PHOTO 7290 [redacted]



- 17 - FIG. 7 GRID AND BeO REFLECTOR

PHOTO 7753 [redacted]

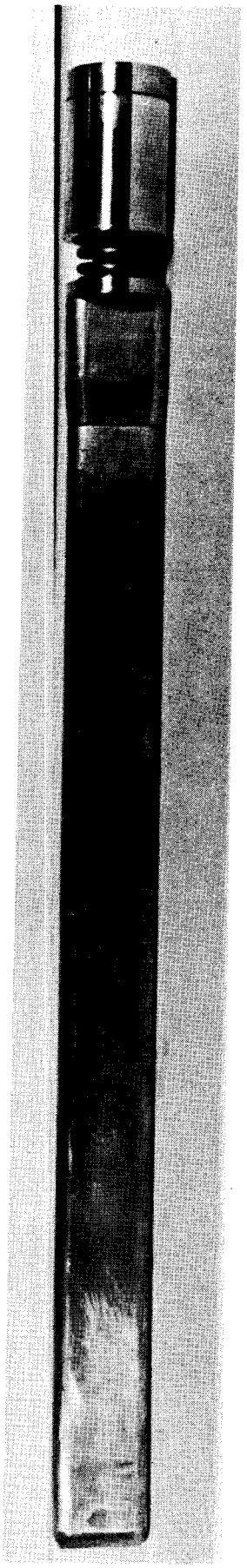


FIG. 8 SAFETY ROD

PHOTO 7415
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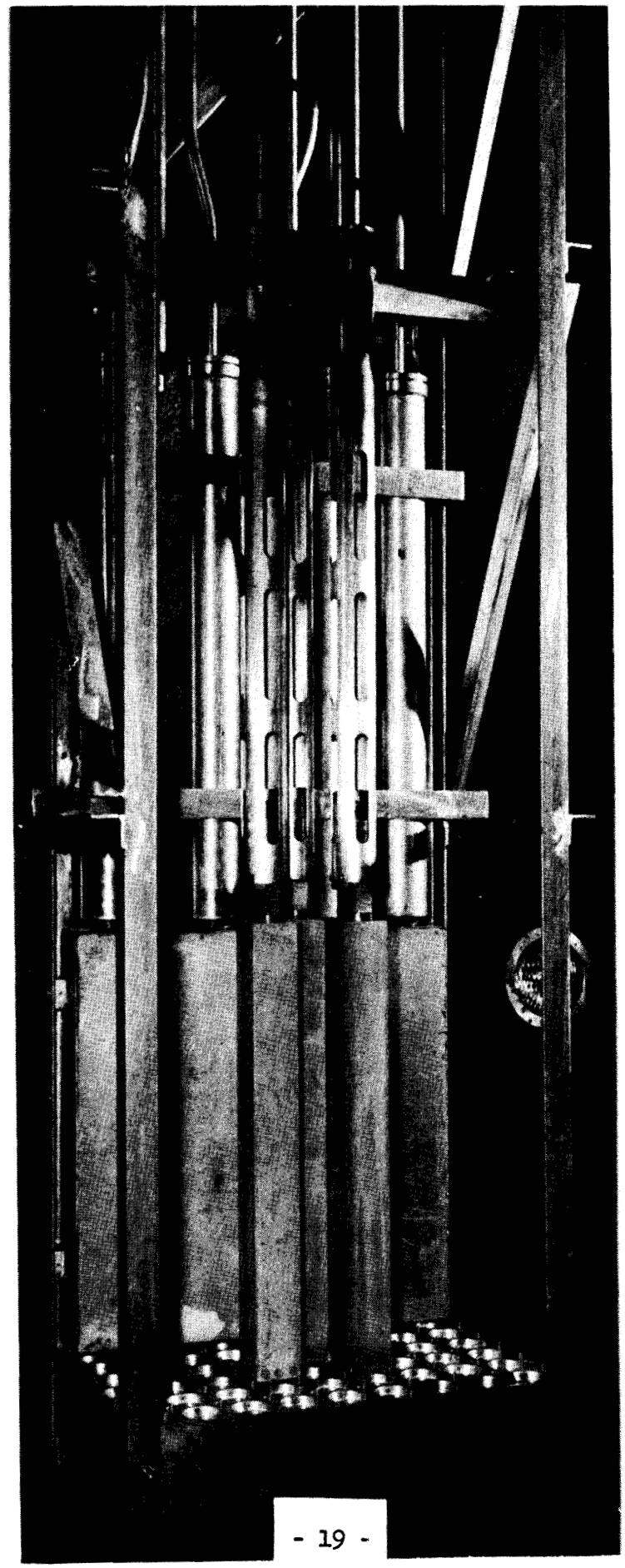

FIG. 9 REACTOR PARTIALLY LOADED

PHOTO 7427
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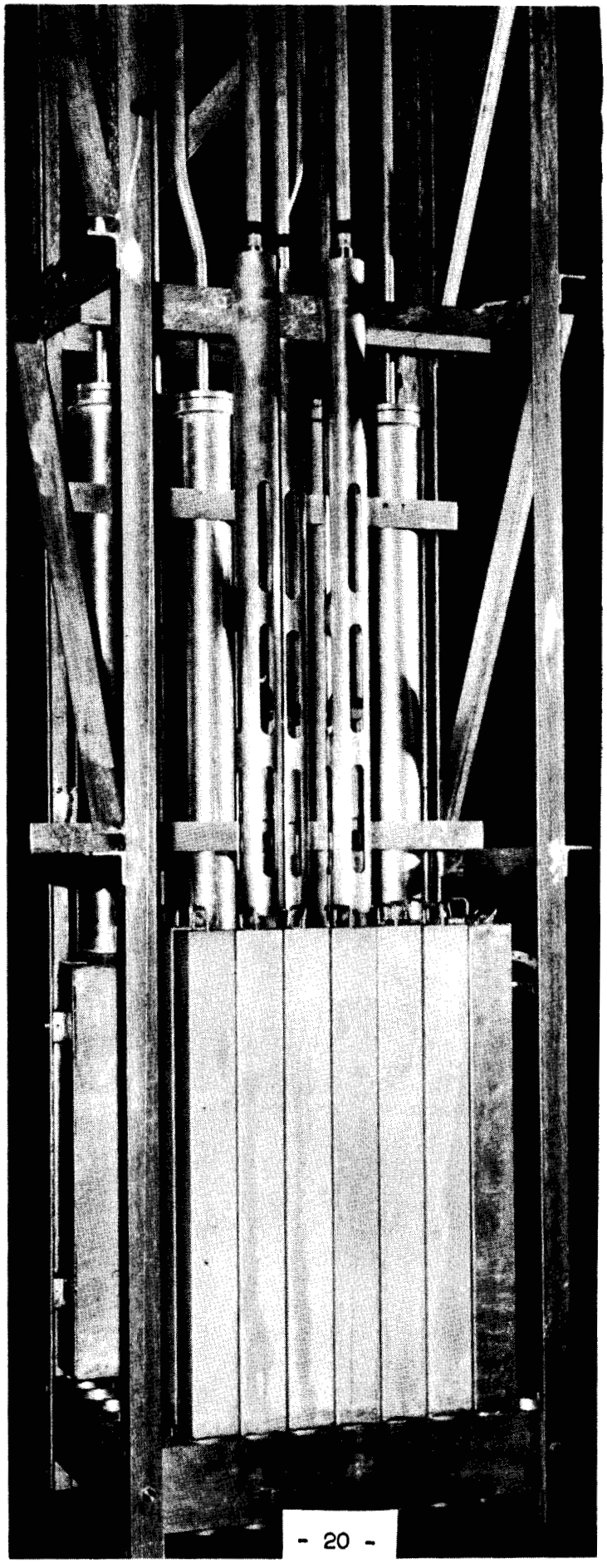

FIG.10 REACTOR COMPLETELY LOADED

As presently designed, the reactor moves only along a north-south center line in the pool. To have made it adjustable in the vertical or in the east-west directions would have considerably complicated the design. The removable concrete blocks previously described provide an adjustable floor level in steps of 6 in. and permit considerable latitude in the size of the shielding samples. Fig. 11 is a view of one end of the pool (looking north) with the reactor bridge and the operator's platform in the foreground and the instrument bridge in the background.

The pool is filled by gravity flow from the laboratory process water supply and is emptied by pumping. The outlet system has a "vacuum breaker" at its high point which insures that the pool will not empty by any syphon action. The water drains into White Oak Creek and thence into the Clinch River. It is expected that the pool will be drained and refilled before the background count rises to more than 10^{-3} microcurie per cc. and hence no health hazard in the creek or in the river will be created.

"The kinetic behavior of the reactor resulting from an instantaneous increase of 2% in the prompt reactivity was studied by M. C. Edlund and E. Greuling on the assumptions that the safety and control rods are jammed and the reactor is operating at 10 kw. at an average temperature of 20°C. The neutron flux and average power density increase exponentially with a period of five milliseconds from initial steady state values of approximately 10^{11} neutrons/cm^2/sec. and 0.06 cal/cm^3 on the basis of 12 fuel assemblies. The major part of the energy, in fact about 90% of the fission energy, is released in the U-Al alloy core. This energy is transferred by conduction through the Al cladding and by conduction and convection through the water.

The process can conveniently be described in terms of the following stages:

1) The temperature rises exponentially in the fuel plates until the temperature of the Al cladding at the surface of the water reaches the boiling point of the water. The temperature distribution in the fuel plates during this initial rise was obtained by assuming that there was little heat loss by thermal conduction to the water, since the power density in the core increases very rapidly. The temperature at the edge of the fuel plate reached the boiling point of water approximately 57 milliseconds after the $\frac{\delta k}{k}$ (effective) jump. At this time the maximum temperature in the core is approximately 170°C.

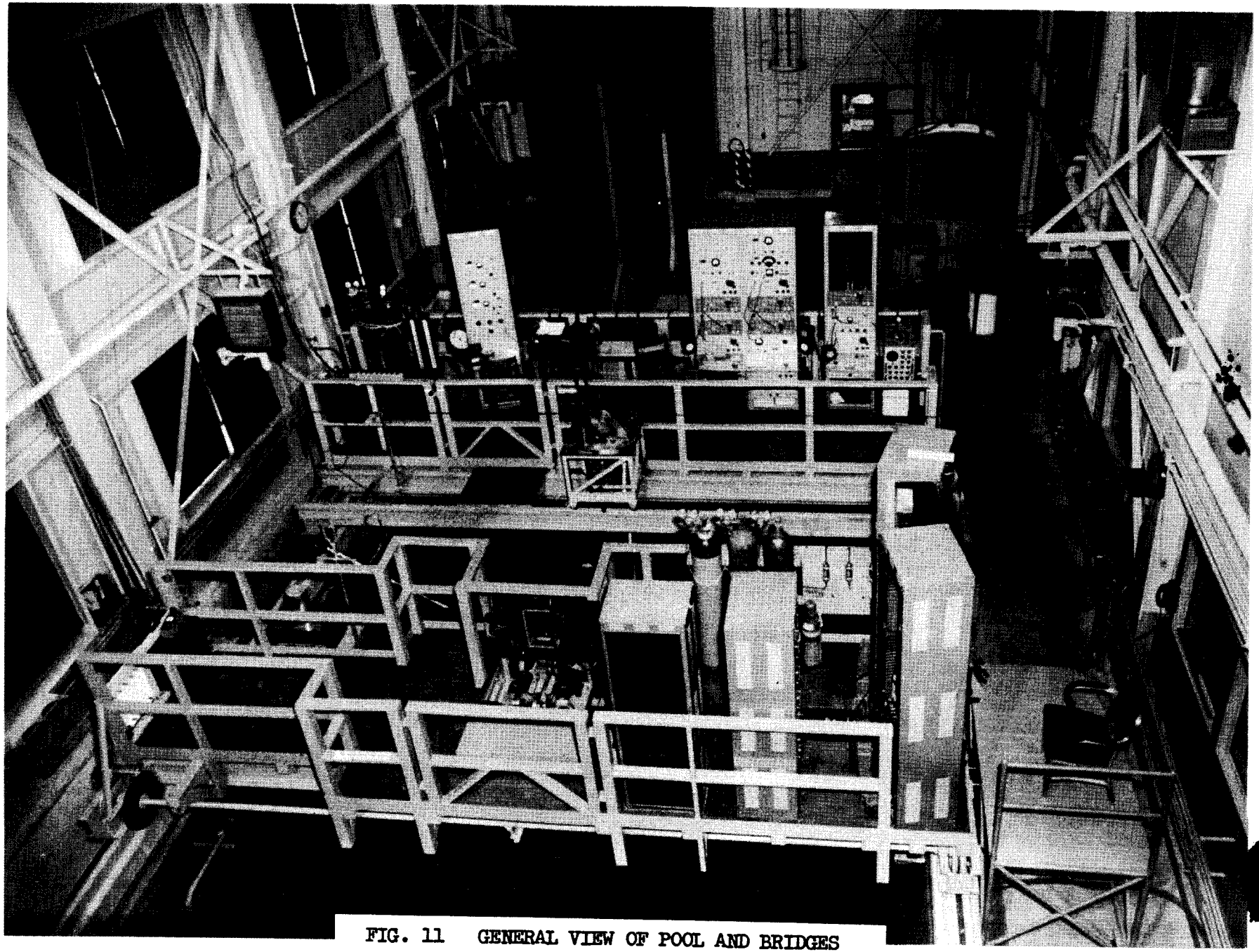

FIG. 11 GENERAL VIEW OF POOL AND BRIDGES

PHOTO 77149

[redacted]

2) At this point, the heat transfer by surface boiling of water becomes effective. It was assumed that the heat flux per cm^2 per second to the water is proportional* to $(T_{Al} - T_B)^{3.86}$, where T_{Al} is the temperature of the edge of the fuel plate and T_B is the boiling point of the water. Since the power density is now rising at a rapidly increasing rate, a surface layer of water is vaporized, driving out the water from between the fuel plates in the order of 4 milliseconds. The total heat produced in the core from the initial time is approximately 560 cal/cm^3. The maximum temperature in the core is approximately 500°C and at the edge of the fuel plate about 180°C.

3) Boiling of the water as it returns to the fuel elements continues until the plate temperatures are reduced to the boiling point of water. This process may require several seconds, depending on the exact nature of the cooling mechanism. After about 10 percent of the water has been driven out of the core, the system becomes sub-critical and the neutron flux falls sharply to a value equal to the fraction of delayed neutrons (0.76 percent) and then decays approximately as $e^{-0.9t}$, where t is the time in seconds.

   The power density in the core immediately preceding the exodus of the moderator is 1.13×10^5 cal cm^{-3} sec^{-1}. Thus, when the fuel plates have cooled sufficiently to stop boiling of the water, the system becomes super-critical again. This time, however, the initial power density in the core has increased to a value 340 times as great as the original steady state power density.

4) The reactor now runs only for about 2 milliseconds before the water is again ejected from the fuel elements. Since the fuel plates are now at a temperature corresponding to the boiling point of water, the time required to vaporize sufficient water to make the reactor sub-critical is much smaller.

After 0.25 seconds, the reactor begins the next cycle. The initial core power density is now 8.5 cal cm^{-3} sec^{-1}. In 6 milliseconds the water is again expelled. The reactor oscillates, damping down to a steady state oscillation in a few seconds. In the steady state the active time is approximately 0.28 seconds and the passive time about 0.25 seconds. The maximum power density in the core of the fuel plates is 20 cal cm^{-3} sec^{-1}, and the total energy liberated for this cycle is approximately 0.1 calories cm^{-2} of fuel plate or at a rate of 130 kw. for the entire assembly.

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* ANL-4268

[redacted]

[redacted]

The maximum power density and plate temperatures are 1.13×10^5 cal/cm^3/sec. and 500°C and occur 0.061 second after the $\delta k/k$ jump.

The phenomenon is consistent with the operation of any stable oscillating system. Since the pool contains more than 100,000 gallons of water, this is not a serious situation. It should be observed that because there are 16 ft. of water above the reactor, the oscillatory operation at 130 kw. average does not present a real radiation hazard. Our computations indicate that a person standing beside the pool would not receive much more than the safe daily dose in eight hours.

Since the calculations discussed above were based on a heat transfer mechanism which may not be entirely valid for this case, we cannot be positive that the aluminum fuel plates will not rise above 500°C. A relevant experiment has been performed recently by Dr. Untermeyer* at Argonne National Laboratory. A large electrical pulse was sent through a a piece of aluminum tubing 3/8" diameter, 3" long and 1 mm. wall thickness immersed in a water tank at room temperature. Ten, twenty, and thirty millisecond pulses resulting in heat fluxes of about 180 cal/cm^2-sec. failed to melt the aluminum, but vaporized and ejected the water from the interior of the tube.

Our calculations show that during runaway an average heat flux over the first cycle of approximately 100 cal/cm^2-sec. for sixty milliseconds is to be expected. However, Dr. Untermeyer's conditions are not identical with ours, and results of his experiment should not be taken as a positive indication that the fuel plates will not melt. If they rise above 640°C, some of the aluminum will melt and fission products will be released into the water."

If the excess in k over critical is less than prompt critical (such as might happen if the rods were raised and the safety circuits intentionally blocked), the reactor power level would rise to a point between 5 and 10 megawatts at which level the steam formed between the plates would drive the reactor sub-critical. Thereafter it would oscillate somewhat in the manner described above.

If, due to some accident, dangerous amounts of fission products should be released into the pool, the monitrons would give warning and personnel would leave the building. Gaseous fission products which did not dissolve in the water would be confined to the building except as they drift out through the ventilators in the roof. Operating schedules are so planned that with a maximum operating power level of 10 kw., the accumulated fission products do not exceed 7500 curies. Of this, some 3700 curies are gaseous, and this is the upper limit of gaseous activity which would be released in a complete catastrophe. The amount which did not dissolve in the pool water would be significantly less than this.

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*Private correspondence

[redacted]

CONTROL AND SAFETY CIRCUITS

Over the operating range this is essentially a constant temperature reactor. Hence, the temperature coefficient, while negative, does not vary sufficiently to stabilize the reactor, and a servo control is supplied to hold the power level constant. This servo system was designed and built at NEPA as part of a possible aircraft control system and is more than adequate for its present application. It is described in a NEPA Report entitled, "Final Report on an Amplifier and Actuator for Reactor Control", by Lyle Martin.

A block diagram of the circuits for controlling and observing the operation of the reactor is shown in Fig. 12. The first instrument is a 4 in. differential chamber* which supplies a Leeds and Northrup Model 2430D galvanometer. There are no electron tubes in this circuit and the galvanometer reading is always directly proportional to the chamber current. The range is about 2×10^4 and the relation between the galvanometer indication and the reactor flux depends on the location of the chamber. Another differential chamber is connected to a Leeds and Northrup Model 9836A electronic micro-micro ammeter. With the aid of shunts, the range is extended to 2×10^5. The micro-micro ammeter controls a Brown Recorder which in turn drives the servo amplifier. An extra slidewire has been added to this recorder, and the position of the contact on the slidewire determines the power level which the servo will seek in controlling the pile. A third differential chamber supplies a logarithmic amplifier. The output of this amplifier is differentiated to give a pile period indication. The output also controls a Brown Recorder from which a continuous record (without changing shunts) of the power level over a range of 10^6 is available. This is the so-called Log N indicator.

To provide indication during start-up, a U^{235} fission chamber, amplifier, scaler and register, and log count rate meter and recorder are provided. With the rods down, the source is of sufficient strength to supply about 3 counts per second. This circuit has a range of 10^4 (10,000 counts maximum) and both the micro-micro ammeter and the log N circuits are responding before this level is reached. An electric motor drive raises the fission chamber to keep it from being activated at high reactor powers.

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* This instrument contains two chambers, one of which is boron coated. Both respond to gammas but only the boron coated one to neutrons. The outputs are connected in opposition and thus, when the compensation is properly adjusted, the net output is a function only of the neutron flux. It is illustrated in Dwgs. Q-1045-1 and Q-1045-9. A report covering this chamber is forthcoming.

Unclassified
Dwg. #10821

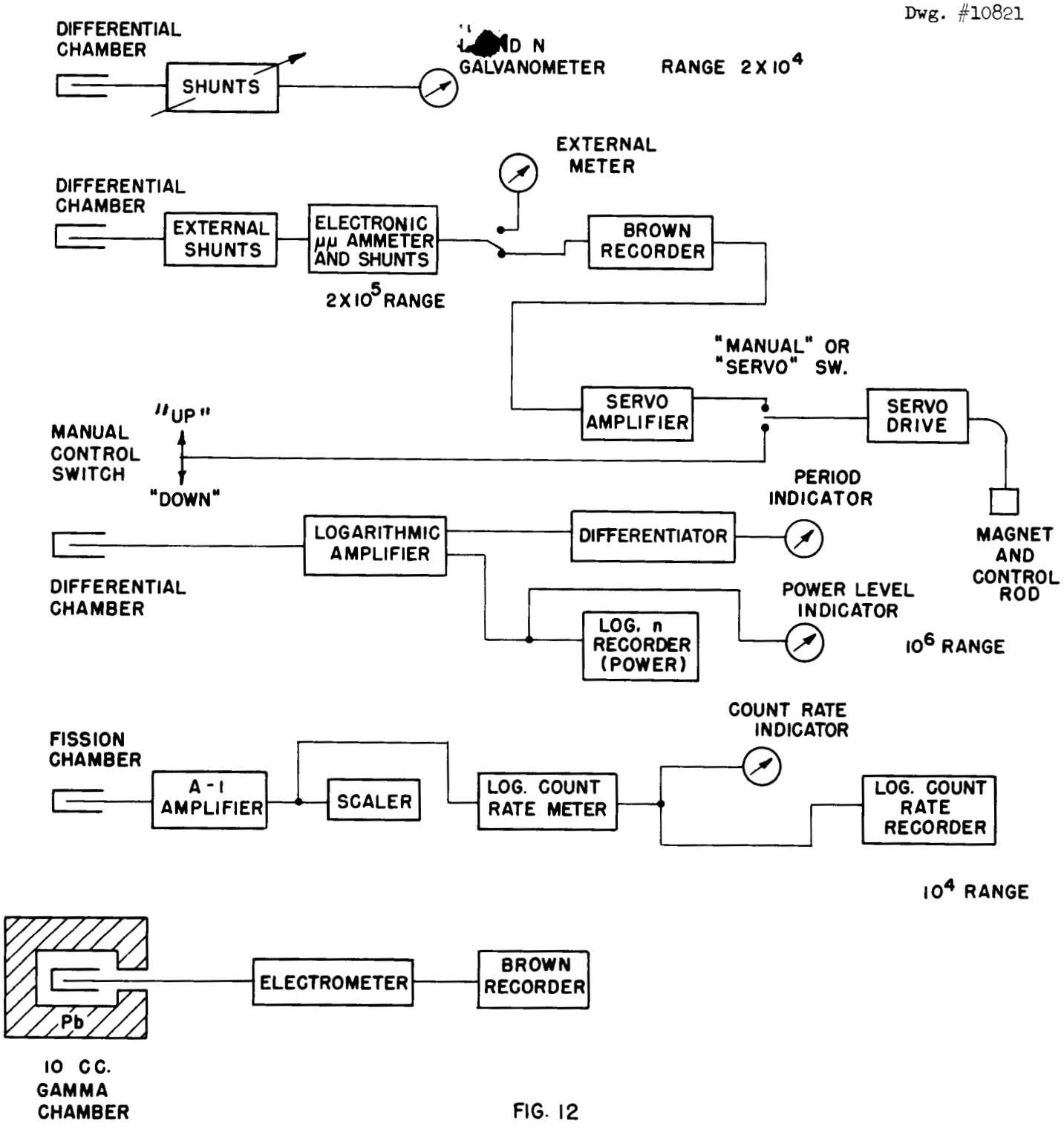

FIG. 12

CONTROL CIRCUITS

A 10 cc. graphite gamma chamber is located against the outer face of the large reflector in the back of the reactor. A lead half cylinder 3 in. thick shields the chamber from capture gammas in the water. Thus the chamber current is very nearly proportional to the pile gammas. The chamber is connected to an electrometer[7] and Brown recorder. Some scheme such as this is required to give an accurate measure of the gammas from the reactor since the gammas from fission products will be appreciable in relation to those from fissions when pile power is reduced after operation at high levels.

The philosophy of the safety system is that it should "fail safe", i.e., the safety rods must fall if the power is cut off or if major circuit trouble develops[6]. In furtherance of this idea, vacuum tubes are used throughout in preference to gas tubes or relays. The amplifiers are monitored at several points and lights warn the operator of an abnormal condition. The safety system is a modified copy of the one which has been extensively tested in the MTR Mock-up.

A block diagram of the system is shown in Fig. 13 and the titles and numbers of the drawings describing the individual components are listed in Appendix I. The "safety" chambers are 3 in. boron coated ionization chambers which supply a current proportional to the neutron plus gamma level. This current flows through a high resistance which is across the input of a preamplifier consisting of a single stage cathode follower. The output of the preamplifier feeds a DC amplifier (the "sigma" amplifier) which is the source of the signal for operating the safety circuits. As shown in the block diagram, both of these amplifiers are connected to a common point called the "sigma bus".

The electromagnets which support the safety and control rods draw exciting current from separate magnet amplifiers. The inputs to these amplifiers are controlled by the voltage on the "sigma bus"; as this voltage is increased, the current through the magnets decreases. Thus, the result of an increase in neutron flux is to reduce the magnet current and the circuit is adjusted to release the rods when the flux reaches a predetermined level*. Adjustment is obtained by changing the preamplifier input resistance or moving the safety chambers in the water away from the reactor. The speed of operation of the electronic part of the circuit is determined by the time constant of the chamber, cable and input resistor. Actually, most of the delay in operation is associated with the inductance of the electromagnets.

In this facility the reactor operator must associate himself closely with the progress of the measurements. This calls for a person of more than average alertness and ability and because of this, the circuits are not as thoroughly interlocked as those in the MTR. Nevertheless a certain amount of protection is provided. The safety rods are raised by 3-phase

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* Grounding the sigma bus will also cause the rods to drop.

Unclassified
Dwg. # 10822

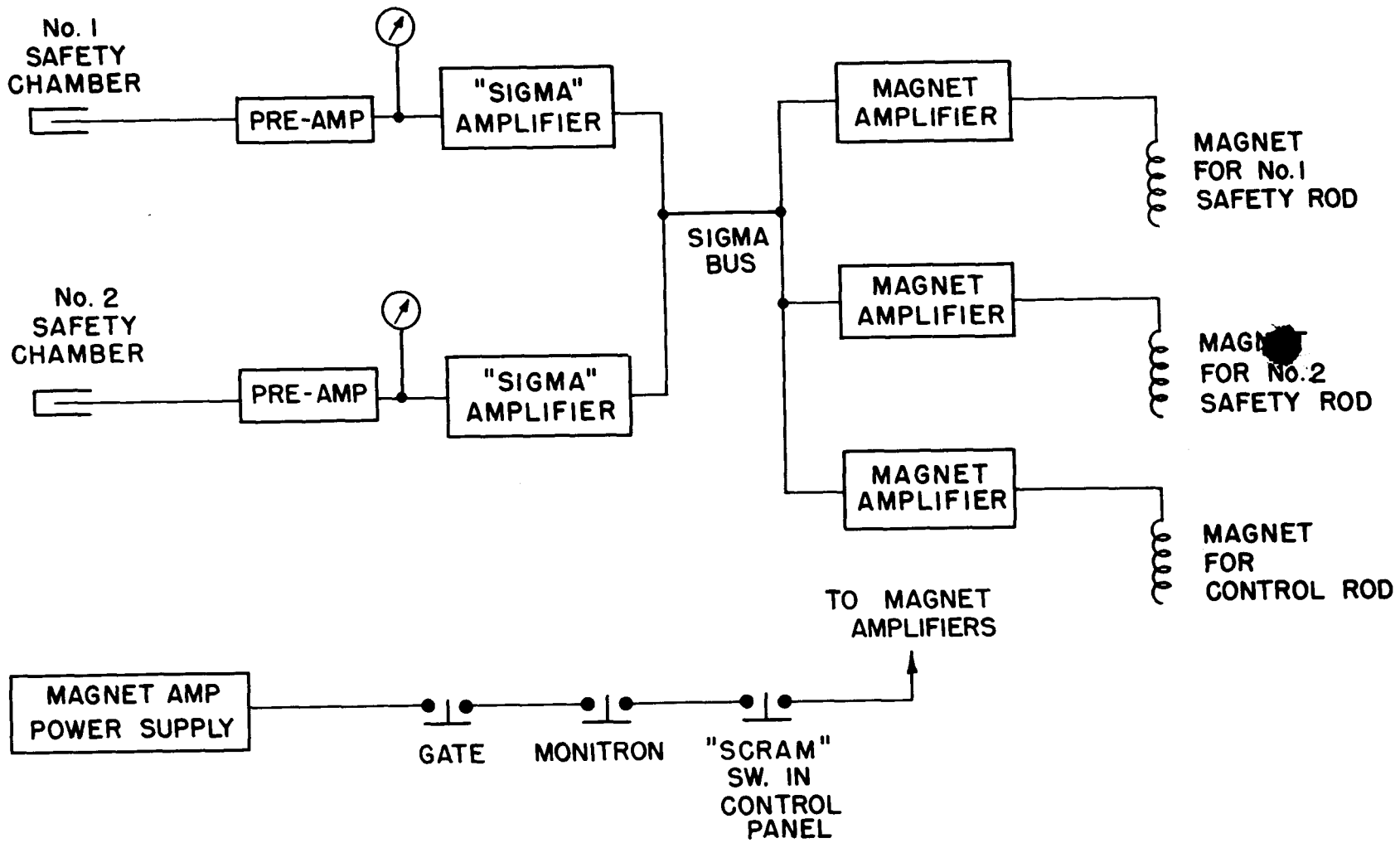


FIG. 13

SAFETY CIRCUITS

AC motors which have a definite speed and insure against too rapid withdrawal. It requires about three minutes to raise the rods completely. The safety rods cannot be raised unless the control rod is all the way in and the control rod cannot be operated until the safety rods are at least three fourths withdrawn. This last 6 in. of travel is the "shim range" and indicated by lights on the control panel. Instructions to the operator require that the reactor be operated only when the shim rods are withdrawn at least three fourths of the way. This insures that a negative δk of 6 to 8 percent is always obtainable by dropping the rods. Since fractional fuel assemblies are available, the reactor can be loaded to conform with this requirement. It is, of course, possible for the operator to disregard instructions deliberately and operate the reactor with a larger amount of possible excess δk, but the interlocking described will prevent him from doing so inadvertently. An interlock, operated by the pointer of the Brown recorder in the count rate circuit, keeps the operator from raising the safety rods unless the count rate is more than two per second. This prevents start-up unless the polonium-beryllium source and the fission chamber are in place. In addition, the magnet current will be interrupted if the gate to the reactor bridge is opened, if the monitron fixed on the bridge shows more than a few mr/hr of gamma rays emerging from the water, or if one of the "scram" buttons located on the control panel and on the walls near the four corners of the pool is operated. When the magnet current circuit is completed, "Reactor On" signs located near the pool and above all the entrances are lighted automatically.

A general view of the reactor bridge is shown in the foreground of Fig. 11. All the electronic equipment for operating and monitoring the reactor is contained in five relay racks which have been located on the bridge. Three of these, (Fig. 14) form the control panel. The other two located behind these three contain power supplies and the amplifiers for the control rod servo. The operating panel (the section with sloping front) has two switches for raising or lowering the safety rods, selsyn indicators to show their locations, a switch to raise or lower the fission chamber, a "manual" or "servo" switch to activate the automatic control system, and a switch to operate the control rod when the circuit is in the "manual" position. The "scram" button, in addition to dropping the rods, resets the safety circuit after it has been tripped by opening the gate on the bridge or operating one of the other scram buttons. Flowmeters are placed in the outgoing gas lines of all the chambers (except the fission chamber where the meter is in the ingoing line) and these enable the operator to determine whether or not the gas flow through the chambers is satisfactory. The flowmeters appear near the top of the center panel.

The electromagnets which support the safety and control rods are detailed in Dwg. D-7209. They are made from one piece of Armco soft iron in an iron clad design. The exciting coil has 4800 turns of #30 copper

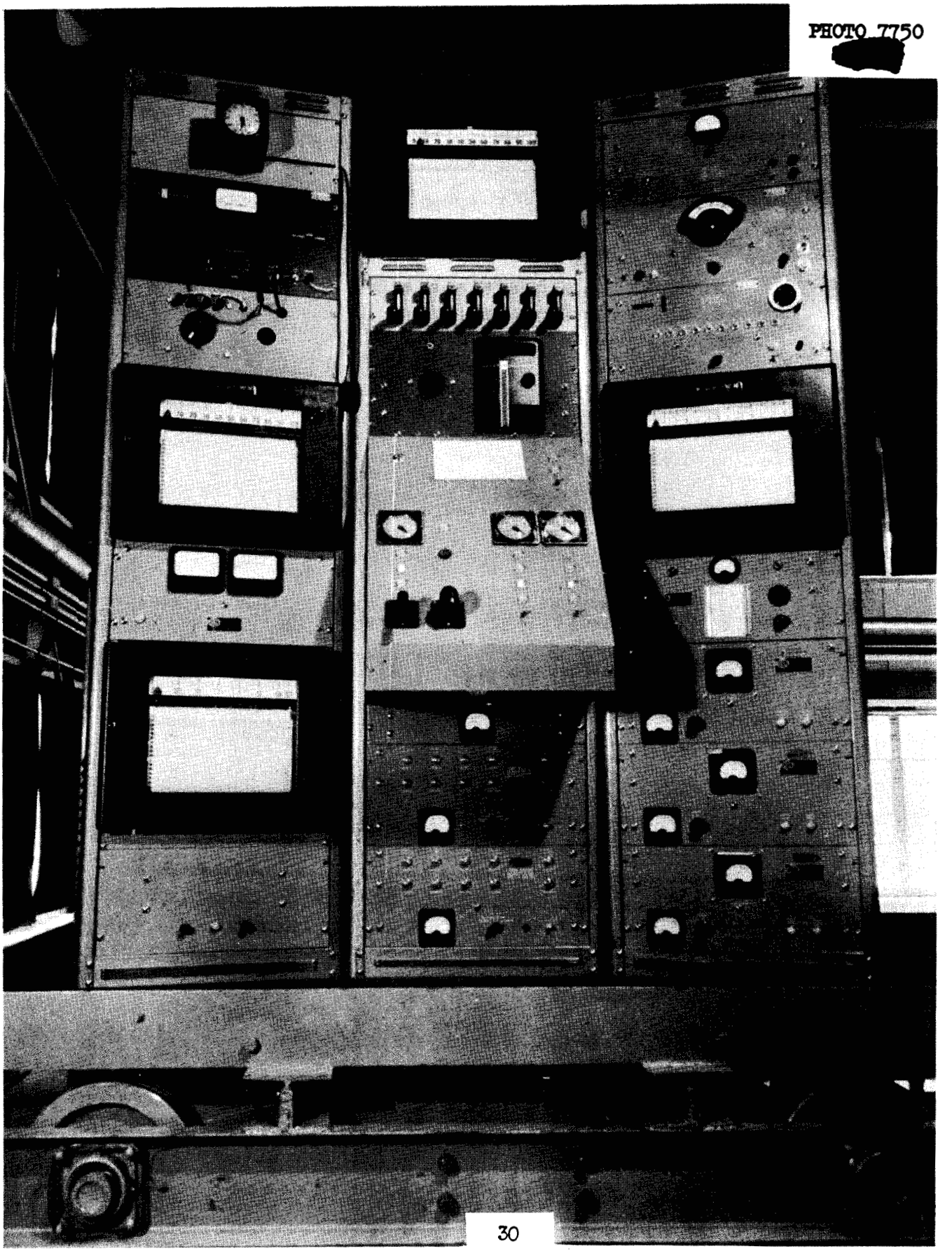


FIG. 14 REACTOR CONTROL PANEL

wire and is impregnated under vacuum after assembly in the magnet with a Irvington Varnish and Insulator Company's Harvel "Oil Stop". This forms a waterproof and shock absorbing bulk insulator around the coil. Tests in the ORNL graphite pile indicate that this material is stable under neutron and gamma ray radiations. As an added precaution against moisture, all joints are painted with glyptol.

The magnets were designed to support the rods with an air gap of 0.005 in. when 30 milliamperes exciting current flows in the windings. The air gap is obtained by crowning the face of the armature. This crowning also makes perfect alignment between magnet and armature faces unnecessary, and hence no universal joint is supplied with magnet or armature.

Tests indicate that the release time, with 50 percent more current than is required to hold the control rod, is of the order of 40 milliseconds. Since the exciting current is an inverse function of the reactor power level, the actual release time after the reactor passes the scram level is much less than this.

The release time can be sharply reduced by laminating the magnetic circuit. However, because it is intended that this particular reactor will be loaded in such a way that it will not reach a fast period, the extra expense and trouble required to laminate the magnetic parts is not justified.

INSTRUMENTS AND THE INSTRUMENT BRIDGE

An instrument bridge, somewhat similar to the reactor bridge, also spans the pool. One view of it is shown in Fig. 15. A cart travels on a pair of rails on one side of the bridge and a stainless steel framework is suspended from the cart and reaches nearly to the bottom of the pool. A carriage which slides on the framework and is raised or lowered with a winch (Fig. 16) carries the various chambers associated with the measurements. It is also possible to rotate the framework about a vertical axis. Thus the measuring instruments can be placed at any point in the pool and pointed in any direction in the horizontal plane. The bridge and cart can be clamped to the rails and the carriage to the framework. The last is accomplished with a spring-operated clamp which is released by a compressed air operated piston. In addition, by means of a pair of telescoping tubes, the framework can be clamped to the bottom of the pool to insure further rigidity.

A catalog of the various measuring instruments on hand February 1, 1951 when attenuation measurements were started follows. Most of these instruments are described by drawings listed in Appendix I.

PHOTO 7761 [redacted]

FIG. 15 INSTRUMENT BRIDGE

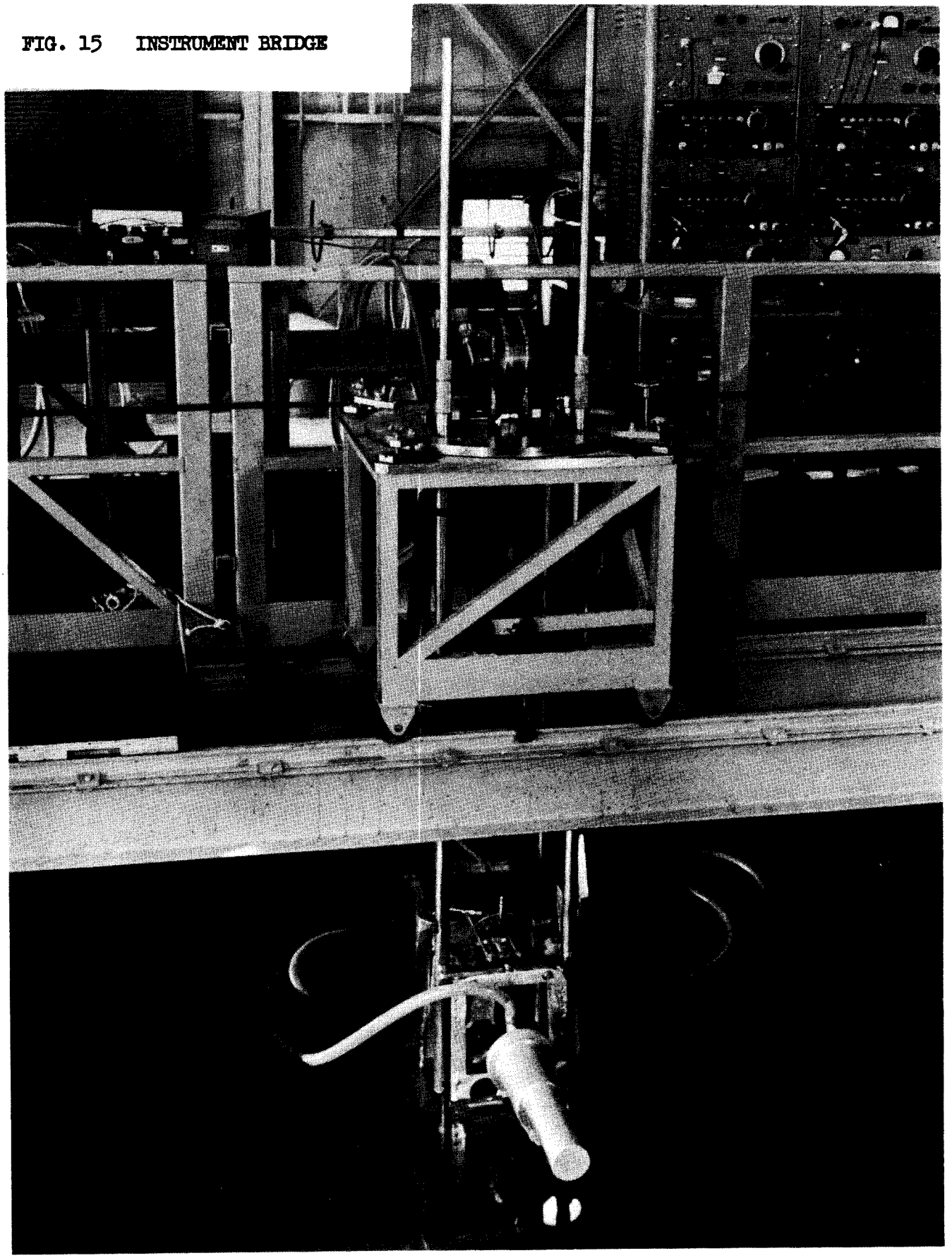
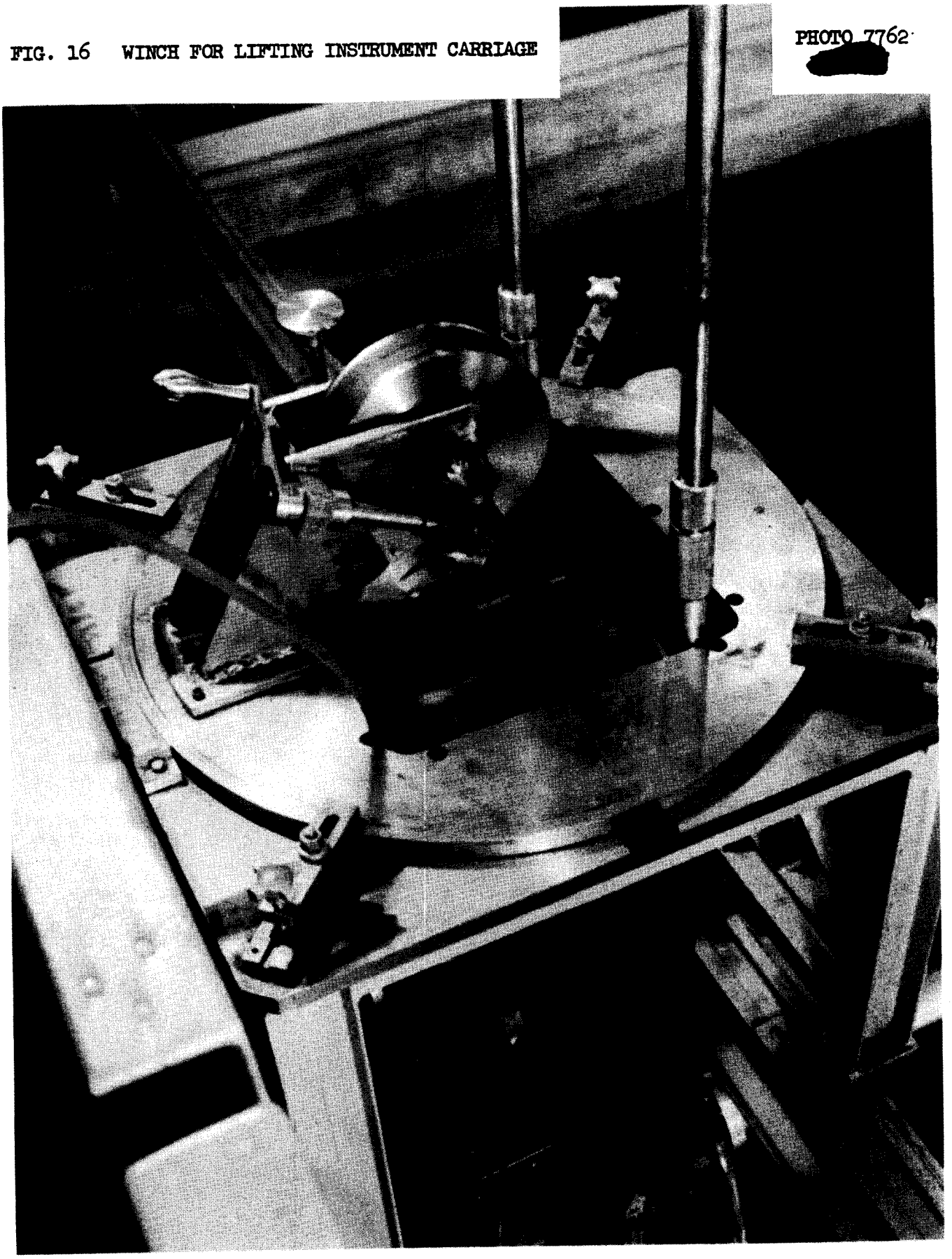


FIG. 16 WINCH FOR LIFTING INSTRUMENT CARRIAGE

PHOTO 7762 [redacted]



A. Thermal Neutron Detectors — Range

1/2 in. Fission Chamber (U^{235})	10^8 - 10^5 nv
3 in. Fission Chamber (U^{235})	5×10^5 - 10^3 nv
8 in. BF_3 Counter	5×10^5 - 10^2 nv
12-1/2 in. BF_3 Counter	10^3 - 10^{-2} nv
(Two of the 12-1/2 in. size are used in parallel to provide this sensitivity)	
Indium and Gold Foils	10^8 - 10^5 nv

B. Gamma Ray Detectors

50 cc. ion chamber 1000 cc. ion chamber	4000 - 10^{-3} r/hr (by changing resistors)
Proportional counter	5 - .05 r/hr
Geiger counter	.05 to .005 r/hr

C. Fast Neutron Detectors (Spectroscopy) — Range

U^{238} Fission Chamber (detects neutrons above 1 Mev)	
Recoil Proton Counter (developed by B. Gossick)	10^7 - 2×10^4 nv
Nuclear Plate Camera (developed at Los Alamos)	10^4 - 10^2 nv
Dosimeter	10^6 - 10^3 nv

This last instrument, developed at ORNL by G. S. Hurst of the Health Physics Division and Floyd Glass of the Instrument Division, has a response proportional to the neutron tissue dosage. The chamber is a proportional counter lined with a hydrogenous medium of thickness greater than the maximum recoil proton range. The chamber is filled with methane at 2 atmospheres pressure and its dimensions are comparable to the proton ranges.

The counter is connected to a circuit which effectively integrates the number of pulses (protons) times pulse height, while at the same time discarding gamma pulses. The rate of charge of a condenser depends on this integrated value. The condenser is discharged at a pre-determined voltage and each discharge operates a scaler and register. Thus, with chamber lining of the proper thickness, the reading of the register is proportional to the fast neutron dosage. A forthcoming report will describe this instrument in detail.

The necessary power supplies, scalers, A-1 amplifiers etc., were included as original equipment and are mounted in the instrument bridge. In most cases spares are provided to insure continuity of operation.

Under certain conditions, it is desirable to determine isodose curves (both neutron and gamma) in the water*. A servo system which will automatically move a chamber or counter along a predetermined isodose curve has been developed. A "Variplotter" (supplied by Electronics Associates, Inc. in Long Branch, N. J.) is connected to follow the movement of the chamber and record its path on a 30 in. by 30 in. sheet of paper. This makes the determination of isodose curves rapid and accurate. The servo system can, when necessary, be operated manually and in this way afford a quick and easy means of positioning a chamber in the vertical plane. The "Variplotter" automatically records the chamber position.

No trouble has been experienced in operating the measuring instruments and associated preamplifiers under water. Each instrument is enclosed in a watertight aluminum or polystyrene case which is supplied with compressed gas (CO_2, argon or dry air as appropriate) at 8 lb. psi above atmosphere. This keeps out the water should any leaks develop. The various conductors are gathered into a cable and pulled through a length of flexible Tygon tubing which is clamped to the case. An ionization chamber with its preamplifier and cable is shown in Fig. 17.

In order to measure distances under water, a sort of a "traveling microscope" has been provided (Fig. 18). This is a transit which has been fitted with a long aluminum barrel and an optically flat glass window at the lower end. This end is always beneath the surface of the water. Relative movements of the transit along the pool can be determined with the aid of the pointer and the scale which is fastened to the side of the pool. When the underwater lights are on, objects in the water are clearly visible.

Concurrently with attenuation measurements, development work is proceeding on instruments intended for neutron and gamma ray spectroscopy.

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* See Appendix II for further discussion of this.

PHOTO 7757 [redacted]

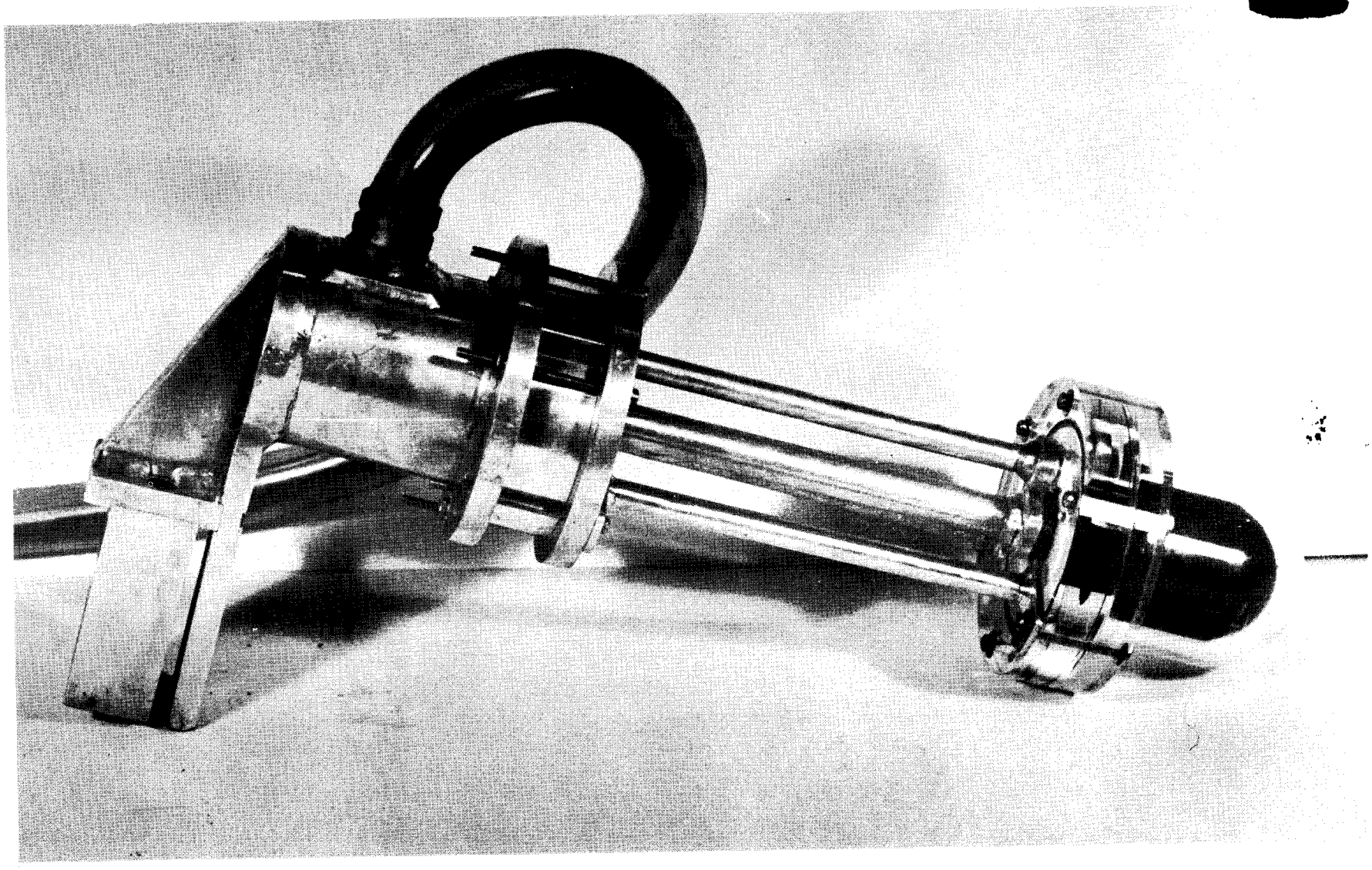


FIG. 17 IONIZATION CHAMBER AND PREAMPLIFIER FOR UNDERWATER USE

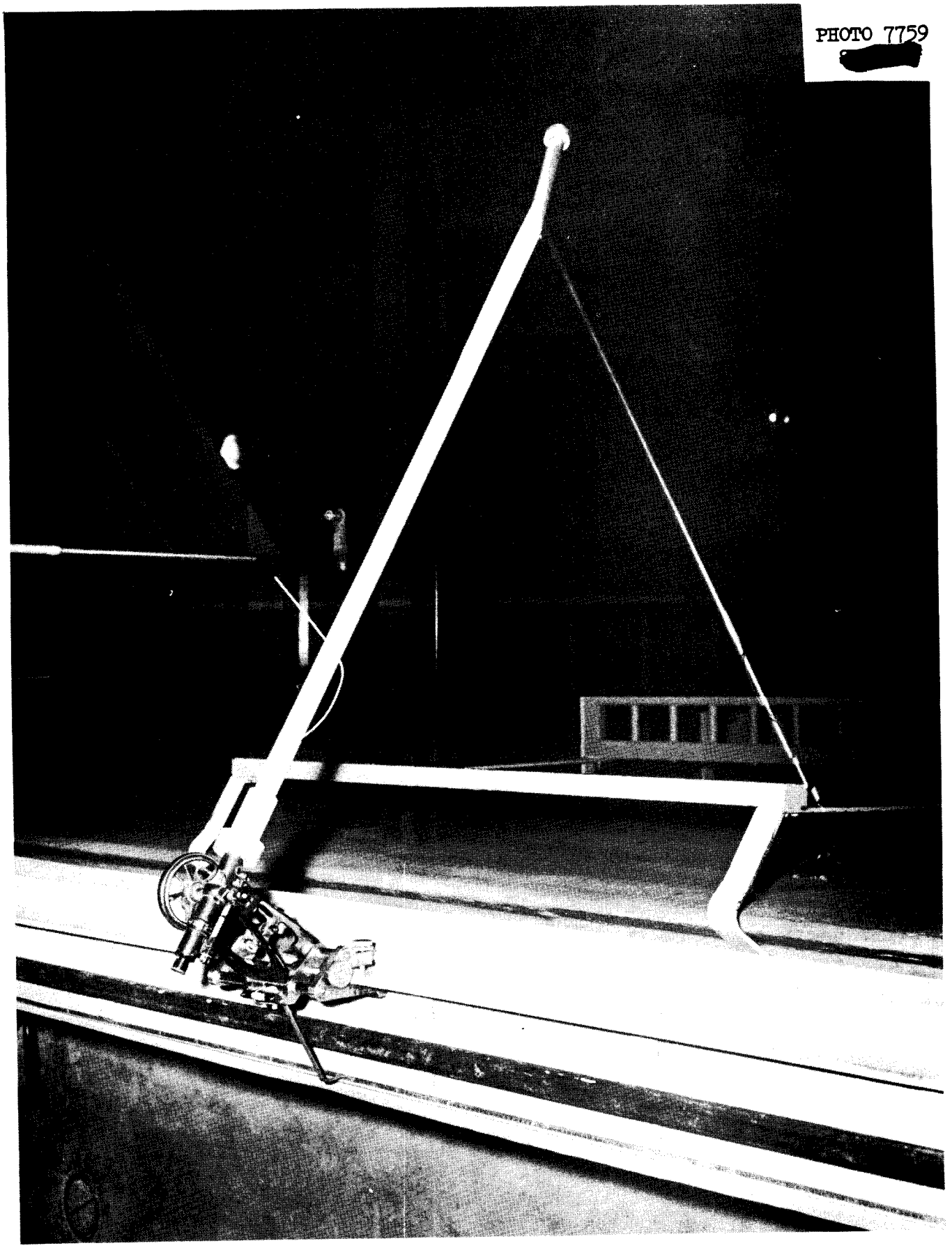


PHOTO 7759
[redacted]

[redacted]

When concluded, this work will be reported in the usual way. These instruments include:

a. Scintillation counter using sodium iodide crystals, for gamma ray spectroscopy.
b. Compton Recoil Counter using G-M tubes, for gamma ray spectroscopy.
c. He^3 counter for neutrons.

The He^3 counter makes use of the reaction:

$$_2He^3 + {}_0n^1 \rightarrow {}_1H^3 + {}_1H^1$$

The reaction is exothermal liberating 0.75 Mev of energy. Each pulse, less this energy, is proportional to the energy of the neutron.

CORROSION

At present the facility is being operated with the pool filled with ordinary process water. An average analysis is shown in Table I. The amount of chlorine varies with the state of the Clinch River.

Last fall Arnold Olsen initiated experiments to determine the amount of corrosion suffered by the fuel assemblies (2S and 72S aluminum) in process water and the effectiveness of the commercial methods of protecting aluminum from corrosion. It was quickly evident that the 72S corroded badly in the process water and that 2S aluminum corroded somewhat more slowly. Addition of sodium chromate (Na_2CrO_4) in a ratio of 50 parts per million by weight appeared to inhibit corrosion completely. Of the various commercial methods for treating aluminum, a procedure known as "alodizing" was selected for test. This is a hot dip process patented by the American Chemical Paint Company. In essence, it deposits a mixed aluminum oxide-chromate coating on the surface of the aluminum. Table II shows the results of the corrosion tests.

All the fuel elements were treated in our shops with the hot Alodine process but because the tests showed that this does not completely inhibit corrosion, the facility is now being operated with 50 parts per million of sodium chromate dissolved in the water. So far the elements show no signs of corrosion.

Computations indicate that the sodium or chromium activities will not be objectionable. Should corrosion set in or the sodium or chromium activities prove to be too great, demineralized water can be substituted for the process water. This, however, involves installation of additional deionizing equipment, Amercoating of the pool walls and all metal work, construction of a 125,000 gallon storage tank (to collect and store the deionized water), and extra piping. The probable cost is in the neighborhood of $25,000, and present operation with process water is directed toward saving this expense.

[redacted]

[redacted]

TABLE I

Filtered Water Analysis F[redacted] Five Samples Taken at One Month Intervals From August Through December 1950

Concentration of All Constituents in Sample	Units	Ave.	Max.	Min.
pH	pH	8.0	8.3	7.4
Ppm Alkalinity as $CaCO_3$	ppm	5.0	10.0	0.0
M.O. Alkalinity as $CaCO_3$	ppm	98.0	106.0	95.0
Specific Resistance	ohms at 10^3	6.92	8.26	4.95
Soap Hardness as $CaCO_3$ detn.	ppm	112.0	120.0	106.0
Soap Hardness as $CaCO_3$ calc.	ppm	101.0	117.0	90.0
Dissolved CO_2	ppm	3.0	6.0	0
Dissolved Solids	ppm	138.0	164.0	112.0
Non-Volatile Solids	ppm	80.0	131.0	35.0
SiO_2	ppm	5.3	9.2	2.7
Fe	ppm	$<.02$	0.08	0.01
Al	ppm	0.08	0.14	0.02
Cu	ppm	<0.02		
Ni	ppm	<0.02		
Cr	ppm	<0.02		
Ca	ppm	30.0	34.0	27.8
Mg	ppm	7.2	8.6	4.7
Na	ppm	9.6	14.0	6.6
SO_4	ppm	25.1	39.6	5.4
Cl	ppm	4.2	5.0	3.3
CO_3	ppm	58.2	63.6	56.4
HCO_3	ppm	118.8	129.3	114.7
NO_3	ppm	1.1	2.3	0.4
PO_4	ppm	0.8	1.7	0.2
F	ppm	0.02		

[redacted]

[redacted]

TABLE II

Material	Treatment	Inhibitor ppm	Length of Test Months	Ave.Wt. Change MDM*	Pit Frequency Pits/sq D	Max.Pit Depths Mils	Appearance
2S Al	None	60 $NaCrO_4$	6	-0.8	0.0	nil	Clean
2S Al	None	60 K_2CrO_4	2	-2.0	0.0	nil	Clean
Mock Assembly	Hot Alodine	None	3	-4.5	4	40	Corrosion product at braze joint
2S Al	Standard Alodine	None	6	-6.8	55	40	Corrosion product only at pits
2S Al	Anodized	None	6	-4.9	125	1	Grey
2S Al	None	None	5	-30.0	60	46	Grey-Brownish product at pits
2S-72S Al	None	None	5	-54.5	35	19	Very heavy corrosion product

MDM* Milligrams per square decimeter per month

Process water used in all tests. Specimen analysis in Table I.

[redacted]

[redacted]

HEALTH PHYSICS INSTRUMENTS

The following health physics instruments for personnel protection were supplied after consultation with the ORNL Health Physics Division.

3 Monitrons with boron coated chambers sensitive to gamma and thermal neutrons. One is mounted on the reactor bridge above the water and gives the dose rate to which the operating personnel is exposed. It is connected to the reactor circuits and will scram the reactor if the neutrons or gammas emerging from the water exceed a safe level. The second is mounted on the wall of the second floor offices and affords protection to personnel therein. When the pool is drained, the reactor is not "visible" to people in the first floor offices but can be "seen" from the second floor. The third monitron is available for location in any suspected area.

1 Counter-Scaler with recorder and immersion probe for monitoring the build up of induced activity in the water.

1 Mobile Counter-Scaler with beta-gamma probe ("tea cart") for verification of freedom from contamination of clothing, etc.

1 Beta-Gamma Portable survey meter ("cutie-pie") for quick determination of beta and gamma radiation levels.

2 Proteximeters for determining integrated dosage levels.

CRITICAL EXPERIMENTS

Two critical experiments were performed immediately after the reactor was put into operation. The loading for the first one is shown in Fig. 19. Three inches of BeO reflector were provided on the north, east and west sides and four inches on the south face. When critical, the active lattice consisted of 19 units arranged in a rectangle 4 units by 5 units with one corner missing and contained 2420 grams of U^{235}.

For the second experiment the BeO reflector was removed from the north, east and west faces. The new loading pattern is shown in Fig. 20. It required 3240 grams of U^{235} to make it critical with rods withdrawn. Earlier experiments with similar aluminum-water mixtures are covered in references 8-10.

[redacted]

[redacted]

DWG. 10527

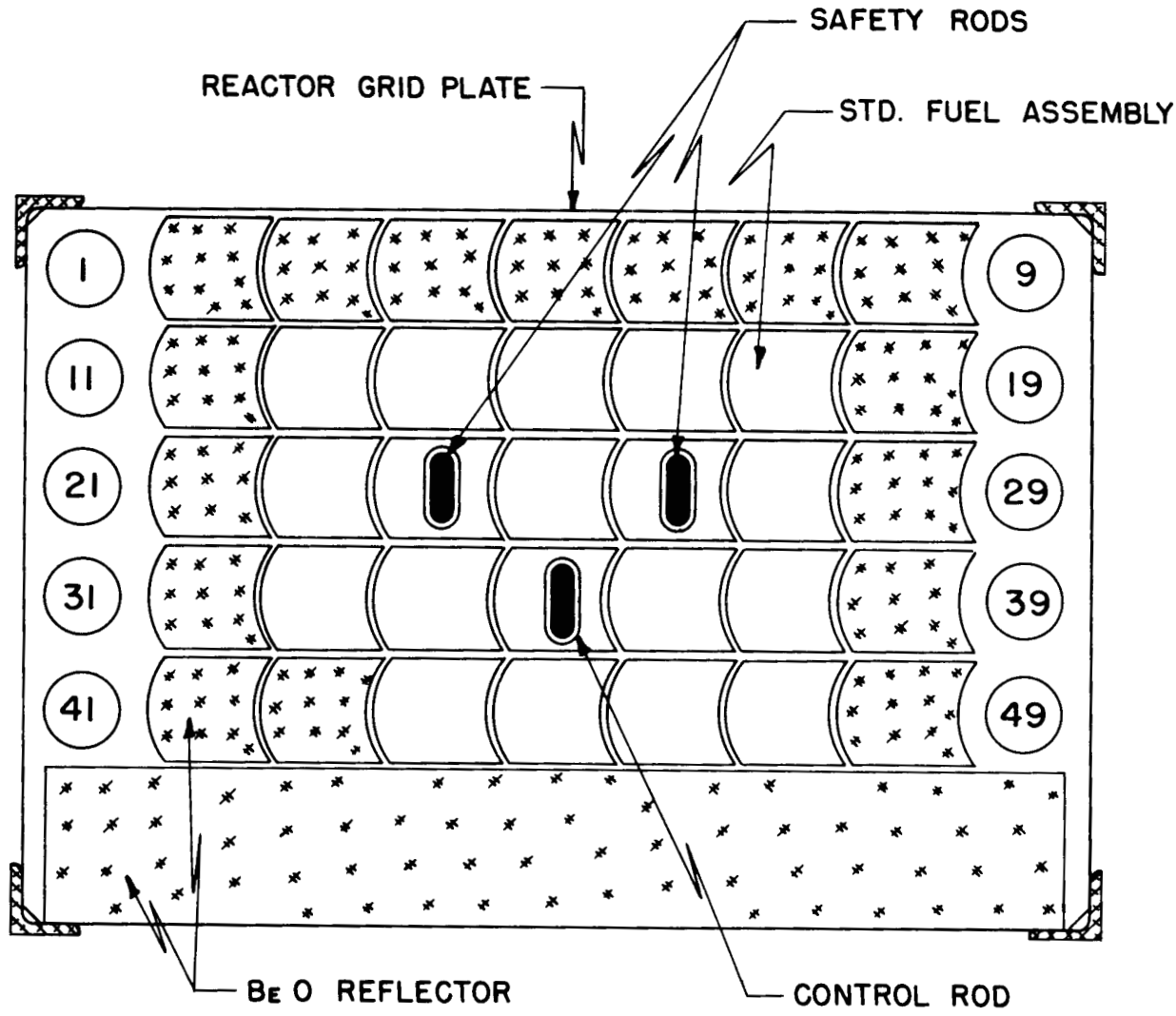


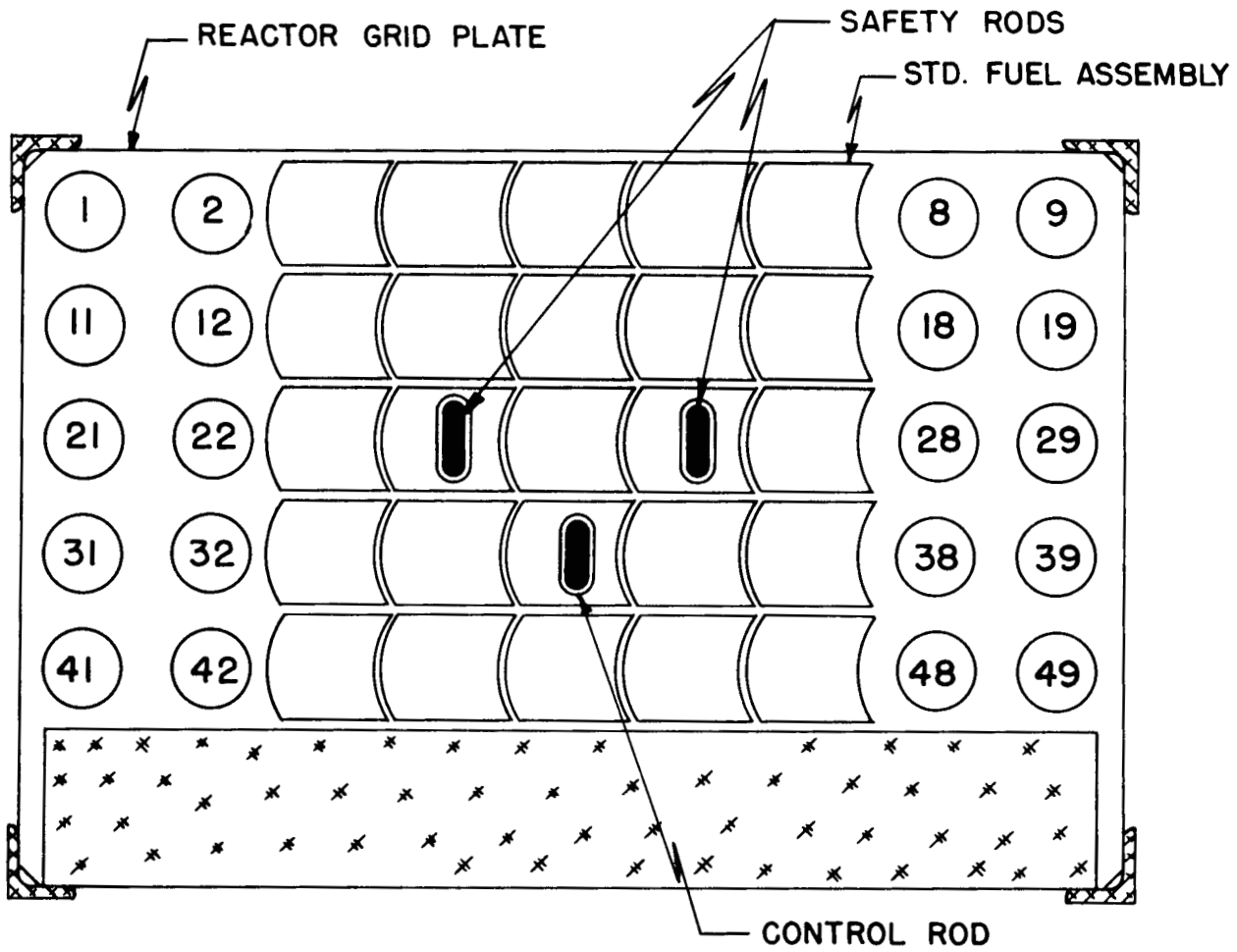
FIG. 19

FUEL ASSEMBLY ARRANGEMENT — TEST No. 1

CRITICAL EXPERIMENTS — BULK SHIELDING FACILITY REACTOR

[redacted]

DWG. 10528



NOTE: BeO REFLECTOR ONE SIDE
WATER REFLECTOR THREE SIDES

FIG. 20

FUEL ASSEMBLY ARRANGEMENT — TEST No. 2

CRITICAL EXPERIMENTS - BULK SHIELDING FACILITY REACTOR

Since the shielding sample located adjacent to the reactor will affect its reactivity, a critical experiment procedure will be followed when starting the reactor for the first time after any change in the arrangement of the samples. In general, the U^{235} requirement for criticality will lie between the values found in the two experiments mentioned above. However, the reactor will be loaded for any given test so that possible excess reactivity will always be as small as possible.

Changes in equipment for the facility, development of new instruments, and progress of the measurements will be reported in the Physics Division and ANP Division Quarterly Reports. Special reports will be issued when the material justifies them.

[redacted]

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4. Huffman, J. R. et.al., "Feasibility Report on Clinton High Flux Pile", MonT-433, Dec. 1, 1947.

5. Beall, S. E., "Procedure for the Erection of the Materials Testing Reactor Mock Up", ORNL-692, May 18, 1950.

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9. Mann, M. M., Martin, A. B., "Further Critical Experiments on A Small Reactor of Enriched U^{235} with Al-H_2O Moderator and Beryllium Reflector", ORNL-79, Sept. 16, 1948.

10. Mann, M. M., Martin, A. B., "Critical Experiments for the High Flux Reactor", ORNL-167, Oct. 22, 1948.

[redacted]

[redacted]

APPENDIX I

Appendix I contains lists of drawings under the following headings:

I. Building

   a. Civil

   b. Electrical

   c. Mechanical

   d. Structural

2. Reactor and Reactor Bridge

3. Instrument Bridge and Scanning Mechanism

4. Instruments

5. Reactor Control Circuits and Instruments

[redacted]

I. Building

a. Civil

D-7164	Plot Plan
D-7165	Manhole, Catch Basin and Septic Tank Details

b. Electrical

D-6947	Lighting Plan - First Floor
D-6948	Lighting Plan - Second Floor
D-6949	Power Plan - First Floor
D-6950	Power Plan - Second Floor
D-6951	Electrical, Section "B-B", Tank Room
D-6952	Power and Lighting - Pump House and Area Plan
D-7179	Transformer Station, Plan and Elevations

c. Mechanical

C-6928	Air Conditioning For Instrument Room, First Floor Instrument Room
D-7146	Sanitary Drainage
D-7147	Process and Cond. Drain. Plan
D-7148	Plumbing Details
D-7149	Piping at Tank
D-7150	Plan - Hot and Cold Water Supply
D-7155	Heating Plan
D-7156	Heating Sections and Details
D-7157	Outside Steam and Air
D-7158	Inside Piping, Sections and Details
D-7176	Potable and Process Water Supply
D-7186	Inside Air and Water Supply

d. Structural

D-6924	First Floor Plan
D-6925	Second Floor Plan
D-6926	Typical Cross-Section
D-6927	Elevations
D-7106	Foundation Plan
D-7107	Concrete Tank - Plan and Details, Sheet 1
D-7108	Roof Framing Plan
D-7109	Crane Runway Framing
D-7110	Steel Framing Elevations, Sheet 1
D-7111	Steel Framing Elevations, Sheet 2
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d. Structural (Cont'd)

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D-7118	Anchor Bolt Plan
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2. Reactor and Reactor Bridge

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D-7202	Grid Plate
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D-7210	Bridge Position Lock, Assembly and Details
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D-7213	Safety Rod Shock Absorber
D-7214	Reactor Bridge Frame
D-7215	Reactor Bridge Superstructure
D-7216	Motor Drive for Reactor Controls - Assembly
D-7217	Motor Drive for Reactor Controls - Detail Sheet #1
D-7218	Motor Drive for Reactor Controls - Detail Sheet #2
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A-7223	Moulded Block (BeO)
A-7224	Section of Drawn Tubing for Cans
C-7732	Concrete Block Details
D-8238	Power Plan - Reactor Bridge
D-8239	Power Details - Reactor Bridge Sheet 1
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3. Instrument Bridge and Scanning Mechanism

D-7601	Instrument Bridge Assembly
D-7216	Bridge Position Lock Assembly
D-7602	Instrument Bridge Superstructure
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3. Instrument Bridge and Scanning Mechanism (Cont'd)

D-8167	"R" Motor and Arm Details #2
D-8166	"R" Motor and Arm Details #1
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E-8162	Theta Motor Sub-Assembly
D-8161	Vertical Structure - Details - Sub-Assembly
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D-8155	Fixed Instrument Mounting - Assembly and Details
E-8154	Carriage - Detail Sheet #2
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E-8152	Carriage - Sub-Assembly
E-8151	Instrument Mounting Structure Assembly
D-8363	"R" Motor and Arm Detail Sheet #3
D-8365	Instrument Mount. Structure Theta Motor Sub-Assembly Details #3
D-8366	Inst. Mount. Str. Theta Motor Sub-Assembly Details #4
D-8367	Inst. Mount. Str. Theta Motor Sub-Assembly Details #5
D-8368	Inst. Mount. Str. Inst. Boom Mount Assembly and Details

4. Instruments

Q-369	Neutron Counter 1" O.D. - 8" Long
Q-804	Neutron Counter 2" O.D. - 12-1/2" Long
Q-846-4	Fission Chamber 1/2" Dia.
Q-846-5	Fission Chamber 1/2" Dia.
Q-1059	Fission Chamber 3" Dia.
Q-961	Proportional Counter (Brass Case 1" O.D. x 13-5/8")
Q-1058	50 cc. Ion Chamber
Q-1057	900 cc. Ion Chamber
Q-954	Watertight Housings
Q-1100-1 thru 6	Dosimeter (Fast Neutrons)

5. Reactor Control Circuits and Instruments

Q-1045-1	4 in. Compensated Ion Chamber
Q-975-1B	Neutron Chamber 3 in. PCP Model 2
Q-1092-1	Fission Chamber
Q-1058-1	Gamma Ion Chamber
Q-541 A-E	A1 Amplifier and A1A Preamplifier
Q-834-1	1024 Scaler (modified)
Q-751 A	Log Count Rate Meter

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5. Reactor Control Circuits and Instruments (Cont'd)

Q-826-1	Low Drift Electrometer
Q-995-1	Power Supply for Compensated Ion Chamber
Q-915-1	Log N Amplifier (Pile Period Meter)
Q-947-1	Sigma Amplifier and Safety Preamplifier (modified)
Q-889-1	Magnet Amplifier
Q-1054	Control System Circuits
NEPA-E-184	Servo Amplifier
NEPA-140173	Servo Motor System
NEPA-140174	" " "
NEPA-140175	" " "
NEPA-140176	" " "
NEPA-140178	" " "
NEPA-140213	" " "

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APPENDIX II

April 20, 1950

Memorandum

To: Dr. W. K. Ergen

From: E. P. Blizard

Subject: Geometric Transformations

The geometry of the shielding reactor can be approximated by a circular source of uniform isotropic surface strength imbedded in an infinite medium of isotropic attenuation properties. Deviations from this picture are:

1) The source is not infinitely thin and hence there is self-absorption and consequent collimation of the radiation.
2) The source is rectangular and not of exactly uniform surface strength. At some distance from the source, this deviation will not be noticeable.
3) In the case of laminated shields the medium does not possess isotropic attenuation properties. This effect is serious when the different laminae (e.g. Pb and H_2O) posses different attenuation properties and when the laminae are of thickness greater than about a relaxation length.

If we define a kernel G(R) to be the probability that a radiation will be detected by a given detector located a distance R from a point source, then the corresponding measurements at a distance Z from an infinite plane source (the detector is non-directional) is:

$$F(Z) = 2\pi \int_0^\infty G(R)\rho d\rho = 2\pi \int_Z^\infty G(R)\, RdR$$

$$R^2 = \rho^2 + Z^2$$

Measurements on the axis at a distance Z from a uniform disc source of radius "a" would yield

$$f(Z) = 2\pi \int_Z^{\sqrt{Z^2 + a^2}} G(R)\, RdR$$

Then

$$F(Z) = 2\pi \int_Z^{\sqrt{Z^2 + a^2}} G(R)\, RdR + 2\pi \int_{\sqrt{Z^2 + a^2}}^{\sqrt{Z^2 + 2a^2}} G(R)\, RdR + 2\pi \int_{\sqrt{Z^2 + 2a^2}}^{\sqrt{Z^2 + 3a^2}} G(R)\, RdR + \ldots$$

$$= f(Z) + f(\sqrt{Z^2 + a^2}) + f(\sqrt{Z^2 + 2a^2}) + \ldots$$

$$= \sum_{n=0}^{\infty} f(\sqrt{Z^2 + n\, a^2})^*$$

Thus by summing chosen ordinates on a measured curve one can obtain an infinite plane geometry attenuation which is then directly comparable to integral measurements over a constant Z plane.

In case the data does not extend sufficiently far to supply all non-negligible correction terms, it will often be expedient to extrapolate on an exponential. In this situation one uses:

$$F(Z) = \sum_{m=0}^{m=n-1} f(\sqrt{Z^2 + ma^2}) + F(Z')$$

where $Z' = \sqrt{Z^2 + na^2}$,

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* This result due to Hurwitz of G.E.

and we then make the representation that

$$F(Z') = f(Z')\, e^{Z'/\lambda} \sum_{\nu=0}^{\infty} e^{-\frac{\sqrt{Z'^2 + a^2}}{\lambda}}$$

The sum can be conveniently bracketed by an upper limit $Ue^{-Z'/\lambda}$ and a lower limit $Le^{-Z'/\lambda}$ as follows* :

$$\int_0^{\infty} e^{-\frac{\sqrt{Z'^2 + \nu a^2}}{\lambda}} d\nu < \sum_{\nu=0}^{\infty} e^{-\frac{\sqrt{Z'^2 + \nu a^2}}{\lambda}} < e^{-Z'/\lambda} + \int_0^{\infty} e^{-\frac{\sqrt{Z'^2 + a^2}}{\lambda}} d\nu$$

on evaluating the integrals, one finds:

$$L = \frac{2\lambda^2}{a^2}\left(\frac{Z'}{\lambda} + 1\right)$$

$$U = L + 1$$

Furthermore, the best value of $F(Z')$ is given by the average of the limits, and this in turn is seen to be a larger lower limit since the second derivative of the exponential is everywhere positive. Thus we find

$$F(Z') \gtrsim \frac{L + U}{2} f(Z')$$

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* Assistance of Drs. Murray and Householder is acknowledged for helpful suggestions here.

Given F(Z) we now are interested in finding G(R). Differentiating the following,

$$F(Z) = 2\pi \int_{Z}^{\infty} G(R)\, R dR$$

we find

$$G(Z) = - \frac{1}{2\pi Z} \frac{d F(Z)}{dZ}$$

For a "single collision" kernel,

$$G(R) = \frac{e^{-R/\ell}}{2\pi R^2} ,$$

$$F(Z) = - E i - \frac{Z}{\ell}$$

For linear buildup factor,

$$G(R) = \frac{e^{-R/\ell}}{2\pi R \ell}$$

and $\ell = \lambda$, i.e. the relaxation length exhibited in the plane geometry case is just the relaxation length of the penetrating component.

The above transformations allow the determination of the point-to-point or plane-to-point kernel by means of centerline measurements only. Where these transformations are valid, they save the work of mapping successive planes, Z = constant, and of integrating the measurement.

However, in the case of strongly unsymmetrical shields, such as shields with ducts, or in the case of sources (radioactive coolants) in the shield, the mapping will still be required. Automatic equipment

for this purpose is desirable. This equipment also can be used to check the validity of the above transformations prior to their extensive use in cases in which they are expected to be applicable.

/s/ E. P. Blizard
E. P. Blizard

EPB:lg