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MATERIALS TESTING REACTOR PROJECT
PROGRESS REPORT

FOR THE PERIOD ENDING NOVEMBER 30, 1949

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Contract No. W-7405, eng 26

MATERIALS TESTING REACTOR PROJECT

PROGRESS REPORT

for the Period Ending November 30, 1949

M. M. Mann, Project Engineer

J. A. Lane, Associate Project Engineer

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SUMMARY

The fundamental design and development work on this reactor, which has been carried out at ORNL during the past three and one-half years, has been reported from time to time in the progress reports of the several divisions engaged on the project. With the advent of a design contractor (Chemical Plants Division of the Blaw-Knox Construction Company) for the MTR, and with the accelerated schedule requested by AEC, it has been decided that progress reports are best made up by project in order to present a properly integrated picture of activity.

The MTR project organization described previously in other reports, from both ORNL and ANL, consists of a Steering Committee composed of Stuart McLain, ANL, Chairman; J. R. Huffman, ANL Project Engineer; M. M. Mann, ORNL Project Engineer; A. M. Weinberg, ORNL Research Director; and W. H. Zinn, ANL Director.

There are five classes of project activity in the Oak Ridge National Laboratory, and the organization is set up as follows:

Project Engineer—M. M. Mann

Associate Project Engineer—J. A. Lane

Active Working Groups

Analysis	J. A. Lane
Design	R. M. Jones
Experimental	M. M. Mann, C. E. Winters
Metallurgy	J. H. Frye, F. Kerze
Control System	W. H. Jordan

It may be of interest to note that the project organization is rather independent of permanent or normal laboratory organization. War experience at several large laboratories showed the project type of organization, which cuts across group and divisional lines, to be flexible and effective. At ORNL there are about forty persons, from Technical, Metallurgy, and Physics Divisions, now engaged on the MTR project.

During the past quarter the Blaw-Knox Company has actively entered into the design of the reactor under contract with the Idaho Falls office of AEC. Early in October a number of Blaw-Knox design engineers were sent to Oak Ridge and have contributed to the project in the design, analysis, and control sections.

At present those Blaw-Knox engineers in the Design Section have returned

to Pittsburgh, and those working on the control system make periodic visits to Oak Ridge.

In the Analysis Section work is proceeding on an accelerated basis.

Subjects under study have been: induced activity in graphite, activity in pile cooling water due to exposure of core material, argon activity in reactor cooling air, heat production and temperature distribution in thermal shields, water flow in the reflector and reactor. A thorough analysis of shielding is in progress. Results of some of these studies are included in this report.

In the Design Section the principal effort has been on the completion of a series of design reports for use by the contractor. Some of the information in these reports has already been transmitted in semi-final form both verbally and in drawings. The fundamental design of the reactor structure, which includes the concrete shield, the thermal shield, the graphite reflector, and experimental facilities, is essentially complete. ORNL drawings DRP-61 through 87 contain this design.

Work is proceeding on revision of certain mechanical items in the control system to improve ease of manufacture and operation as shown to be required by Mock-Up tests.

The ORNL Control Group has been re-organized and now contains a number of people on loan from NEPA. The original group which worked out the philosophy of the control system and most of the critical circuits during 1946 and 1947 was decimated by resignations and transfers during 1948. Only three persons of the original thirteen remain on the project.

Considerable time and effort has gone into explaining the proposed control scheme to the Blaw-Knox engineers.

Detailed work is now proceeding on instruments, safety, circuits, and control room layout requirements. It was recently agreed in a meeting between ORNL and Blaw-Knox representatives that ORNL will provide production prototypes of the special instruments needed. These instruments will be constructed by the ORNL Instrument Division under the supervision of members of the Control Group. It is possible that ORNL will have to supply a complete set of special instruments for the MTR.

In the Experimental Section work is drawing to a close on the hydraulic measurements in the Mock-Up, and preparations for critical experiments in the Mock-Up are rapidly nearing completion. Remote handling tools and accessories have been developed.

Thermal cycling experiments were conducted on an MTR fuel assembly. No appreciable change in dimensions was noted after a total of about 3900 cycles in the temperature range 17° C - 100° C.

More work, both theoretical and experimental, has been done on thermal stresses in beryllium. Results indicate that the present design of the beryllium reflector and its cooling will enjoy a safety factor of the order 4 in thermal stress. Corrosion work has been done on both extruded beryllium and the hot pressed or QM material. From results available to date, no great difference is noted in extruded and pressed material. It is hoped that specifications for corrosion acceptance tests can be prepared in the very near future.

In the Metallurgy Section techniques of laboratory melting and casting have been substantially improved.

Additional extrusion experience in the Adrian, Michigan plant confirms the feasibility of consistently producing sound 3¾ in. round-cornered square extrusions for the MTR, subject to the availability of sound 8-inch billets. Recent billets received from Reading containing small additions of aluminum were sound and extruded satisfactorily. These aluminum-containing extrusions are now undergoing mechanical and corrosion tests.

Studies have been made to compare grain structure, mechanical properties, and corrosion behavior of beryllium metal extruded at different temperatures, and annealed at different temperatures, together with various types of hot-pressed beryllium.

Twenty fuel assemblies, containing enriched uranium, are being made by the Metallurgy Division for use in the criticality experiments to be performed in the MTR Mock-Up at ORNL.

A brief summary of the present status of the chemical processing plant design is given in Section VI.

MTR ANALYSIS GROUP

ORGANIZATION

Recent expansion of the analysis group has been necessary to meet the commitments of a tighter design schedule for the MTR. This has been possible through the cooperation of the Blaw-Knox Company by the addition of four engineers on a loan basis. The present organization of the group and distribution of effort is as follows: J. A. Lane, Special calculations; R. B. Briggs, Heat transfer, fluid flow; V. Zora, Blaw-Knox; S. Kasper, Blaw-Knox; N. F. Lansing, Shielding; G. Thornton, NEPA; J. Pryor, Blaw-Knox; J. Mulrine, Blaw-Knox.

INDUCED ACTIVITY IN GRAPHITE

Ref.: ORNL, Central Files Number 49-10-183

The activity of the graphite at the edge of the thermal column in the MTR has been estimated from data obtained at ORNL on the induced activities of GBF and CS graphite. Since the thermal flux at this point in the reflector will be about 5×10^{12} neutrons/sq cm (sec), the saturation activity of GBF graphite will be about 6×10^{5} γ's/cc (sec) and CS (or AGOT) graphite, 1.3×10^{6} γ's/cc (sec). This means that the radiation intensity inside the thermal column after removal of the graphite will be 15 r/hr for CS graphite and 7.5 r/hr for GBF graphite. A three inch lead shield or equivalent will be required for the protection of personnel.

ACTIVITY IN PILE COOLING WATER DUE TO EXPOSURE OF CORE MATERIAL

Ref.: ORNL, Central Files Number 49-10-186

The formation of pits and blisters resulting in the exposure of core material to the pile cooling water causes an increase in the activity of the water above that due to N^{16}, Na^{23}, and Al^{27} due to the escape of fission recoils from the fuel plates. The magnitude of this activity per sq cm of ex-

posed surface on each fuel plate is 2.1×10^{-8} watts/cc, allowing 2 minutes decay time. In Fig. 1-1, a comparision of the activity due to fission recoils with the background activity is shown for various decay times.

The data in Fig. 1-1 may also be used to estimate the importance of picture framing the fuel assembly plates to avoid exposing the edge of the U-Al core.

If one assumes that the entire edge of the core is exposed to the water flow, the increased activity in the water due to fission recoil, allowing 4 seconds decay, will be 2.2×10^{-8} watts/cc, compared to 7×10^{-7} watts/cc for the N^{16} activity. The energy of the fission product radiation, however, is not as high as that for the N^{16} and an additional 1½ inches of MO concrete only are required for shielding the exit water lines.

This means that the edge cladding is not extremely important from the standpoint of activity of the process water which opens the possibility of simpler methods of fabricating fuel assemblies. This should be considered in the study of remote methods of manufacture in the metallurgical plant.

ARGON ACTIVITY IN REACTOR COOLING AIR

Ref.: ORNL, Central Files Number 49-11-180

For operation of the reactor at 30,000 kw, the induced activity in the air due to A^{41} will amount to 1500 curies per day. Assuming a total air flow of 50,000 cfm up the stack, the activity at the top of the stack will be 7.5×10^{-10} curies/cc. Experience at Oak Ridge indicates that with a 250 ft stack, a minimum dilution factor of 1000 can be expected. This means that the radiation intensity at ground level due to argon activity will be of the order of 1 mr/hr, unless extremely unfavorable inversion conditions exist. A meteorological study of the site should determine this.

MTR SHIELDING SURVEY

A series of reports summarizing the MTR shielding calculations is being prepared. Report #1 of these reviews earlier calculations and establishes the basis for future work. The mechanism of gamma ray attenuation is discussed with particular reference to absorption coefficients, build-up factors and a

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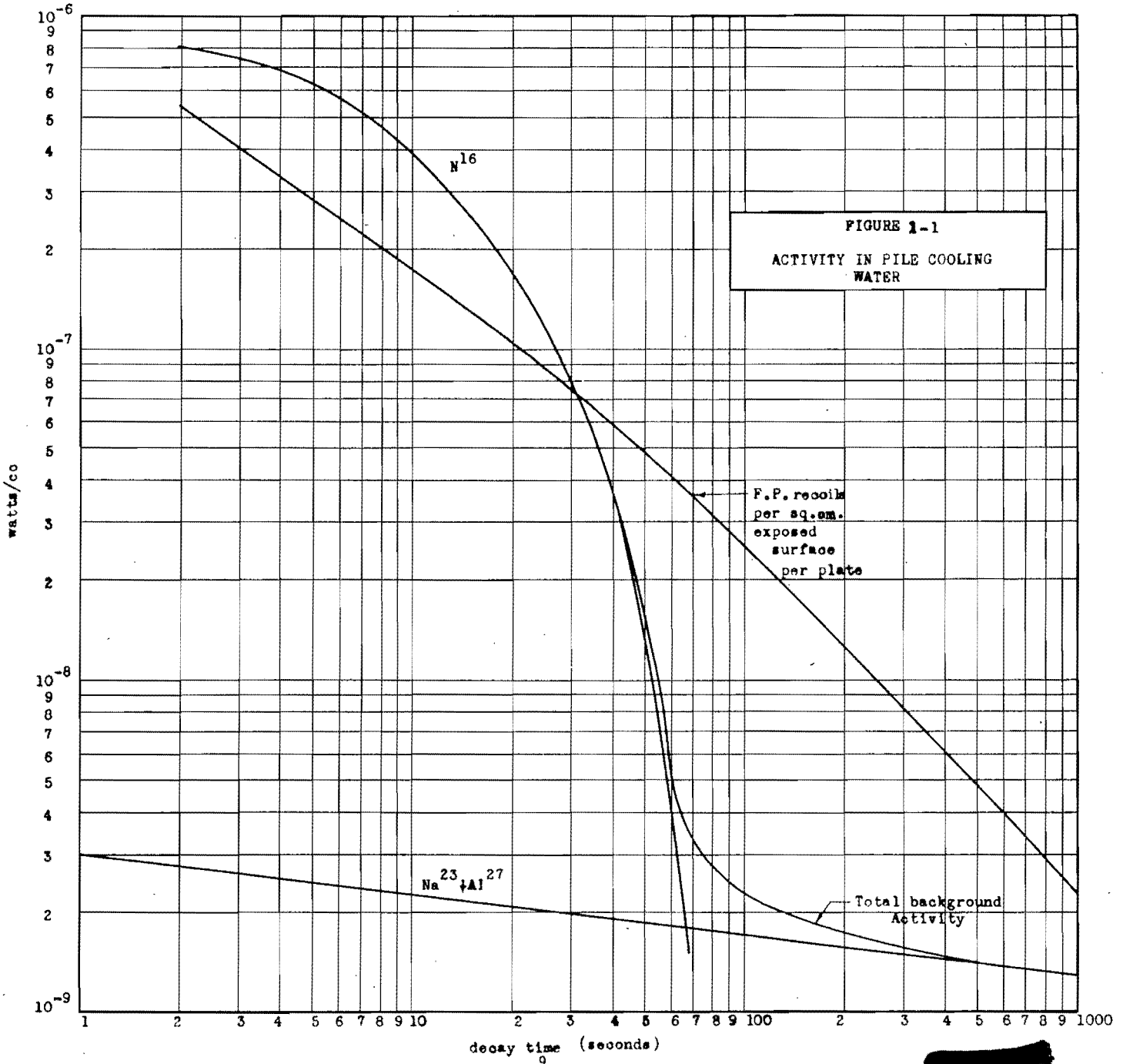

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comparison of experimental and theoretical values. This report will be issued in the next quarter.

Report #2 of the series (Central Files Number 49-10-184) covers the source spectrum of the MTR, the radiation level below the bottom plug, the capture gamma ray source in the upper and lower grids, the capture gammas in the beryllium reflector, the radiation leakage around the bottom plug and, the radiation leakage through monitor tubes.

Report #3 derives equations for the activity of delayed neutrons, photo neutrons, gamma rays, and beta rays after shutdown of the reactor. These data are important for the design of beam hole shielding after removal of plugs. This report will be issued in the next quarter.

HEAT PRODUCTION AND TEMPERATURE DISTRIBUTION IN THERMAL SHIELDS

The problem of estimating the rate at which heat is produced in the steel thermal shield is that of determining (1) the neutron flux and the gamma flux at the face of the steel, (gamma sources are reactor core, beryllium reflector, and graphite), (2) the number and position of formation of gamma rays produced by the capture of neutrons in the steel, and (3) the heat produced as a function of position, in the steel, by gamma ray absorption from all sources. An estimate of the neutron flux at various positions on the surface of the steel is being obtained by assuming that the active lattice and part of the beryllium constitute a spherical source of thermal neutrons of 50 cm radius. The thermal flux at the surface of the sphere is assigned a value of 2.5×10^{14} n/(cm^2) (sec), for 30,000 kw operation, based upon data presented in ORNL 167. Neutrons leaving the surface of the sphere are assumed to reach the thermal shield by the shortest path through moderator. The flux at the surface is said to be equal to the uniform flux at the surface of a thick steel shell in a geometry where the spherical reactor is surrounded by shells of beryllium, aluminum, graphite balls, solid graphite and steel of the thickness obtained from the shortest path. Neutron fluxes estimated in this manner are thought to be conservative near corners and reasonable near the center of the shields. A neutron flux map of the bottom thermal shield is shown in Fig. 1-2. Those for the top and side thermal shields are in various stages of completion. Values from Fig. 1-2 must be multiplied by 2.5×10^{14} to give actual neutron fluxes.

Calculation of the gamma flux at the thermal shield requires a knowledge of the spatial neutron flux in the reflector that is not available at the present time. Considerations based upon the extrapolation of curves from Fig. 3

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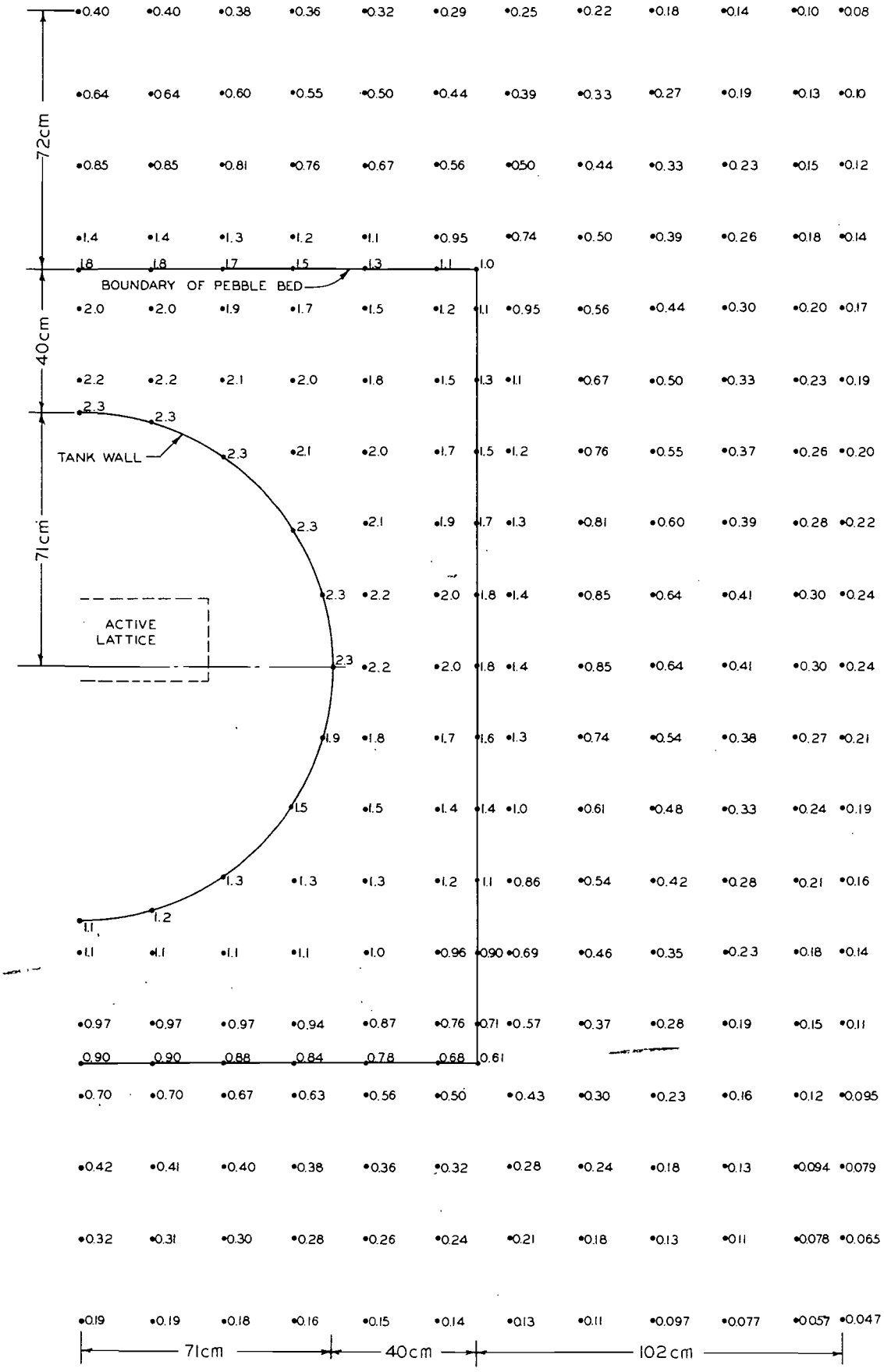


FIG. 1-2

NEUTRON FLUX AT SURFACE OF BOTTOM THERMAL SHIELD

NUMBERS ON PLOT ARE $10^5 \times$ nv FOR FLUX OF 1 NEUTRON/CM^2 SEC. AT THE SURFACE OF THE SPHERICAL REACTOR. $2.3 = 2.3 \times 10^{-9}$ NEUTRON/CM^2 SEC. NORMAL GRID SPACING IS 20 Cm.

[redacted]

of the "Materials Testing Reactor Design Data Sheets", CF 49-10-232, show the heat produced in the steel by pile and reflector gammas to be about the same as that produced by the neutron capture gammas. Again it is believed that the basic data are conservative so it is assumed, for the present, that the two sources give rise to equal amounts of heat in the steel.

Integration of Fig. 1-3, a curve of the heat production rate as a function of plate thickness, shows that a flux of one neutron/cm^2 sec will result in heat being produced in the steel at a rate of 3×10^{-13} watts/cm^2 if the steel is several inches thick and 8 Mev of gamma rays are produced per capture. Using this constant and Fig. 1-2 as the basis, total heat produced in the bottom thermal shield was found to be 32 kw for operation at 60,000 kw. It is to be expected that the total heat produced in the entire thermal shield will be of the order of 100 kw, when calculated from the completed neutron plots.

Equations have been derived for use in approximating the temperatures in the thermal shield plates. It was found that a single bottom thermal shield plate would operate with a maximum temperature of 500 to 600° F, an intolerable condition. Various arrangements of split shields are being calculated to arrive at more reasonable conditions.

WATER FLOW IN REFLECTOR AND REACTOR

Hydraulic testing of the "Mock-Up" showed the water flow through reflector to be somewhat greater than expected. It is probable that this resulted from a lack of published literature on the design of the water flow through the reflector. However, a calculation was made to determine whether the total flow could be estimated upon the basis of observed pressure conditions.

A listing was made of all of the holes and cracks in the beryllium reflector, including that part in the lattice, and the flow through the openings was calculated for the pressures observed. Upon this basis it was calculated that the total flow through the beryllium reflector should be ~ 1900 gpm and that through and around the beryllium in the active lattice should be ~ 700 gpm for an overall ΔP of 35 psi. These values are in good agreement with estimates by S. E. Beall of the actual flow through the reflector for an overall ΔP of 33 psi. Since the calculations agree with the measurements it is believed that modifications can be made in the design of the reflector without requiring subsequent large scale flow tests.

Assistance was given Section 2 of the Technical Division in interpreting

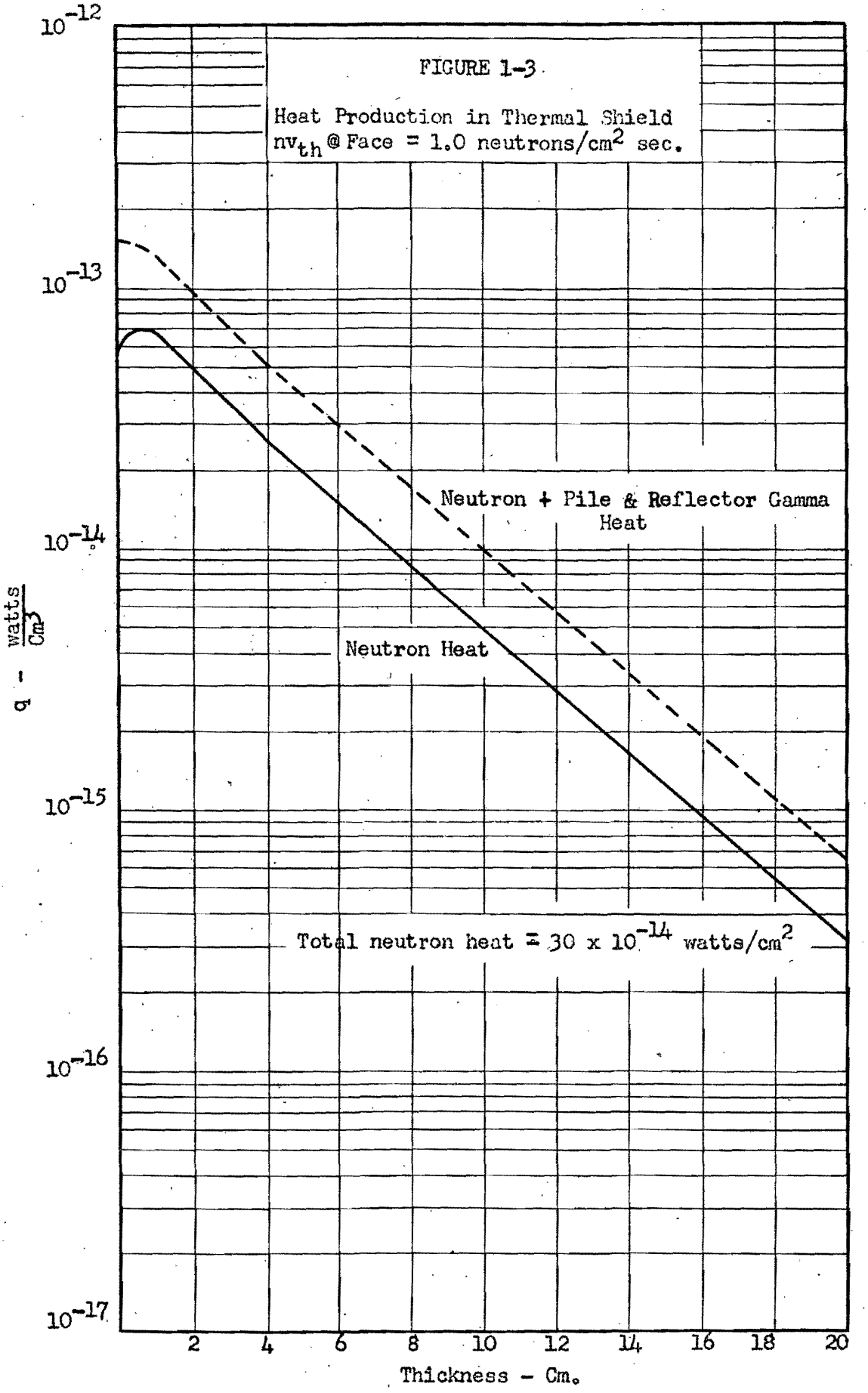


FIGURE 1-3
Heat Production in Thermal Shield
nv_{th} @ Face = 1.0 neutrons/cm^2 sec.
Neutron + Pile & Reflector Gamma Heat
Neutron Heat
Total neutron heat = 30 x 10^{-14} watts/cm^2
10^{-12}
10^{-13}
10^{-14}
10^{-15}
10^{-16}
10^{-17}
q - watts/Cm^3
2
4
6
8
10
12
14
16
18
20
Thickness - Cm.

data obtained in measuring the flow through the active assemblies. Results of those tests are published in the section of this report dealing with the Mock-Up.

THERMAL STRESSES IN BERYLLIUM

Maximum thermal stresses to be expected in the beryllium reflector were estimated taking into consideration the heat production rates observed in the critical experiments reported in ORNL 167 plus the heat liberated by the thermalizing of fast neutrons. It was found that the maximum stress in the present design is a surface stress that should not exceed 12,000 lb/in^2 for operation at 30,000 kw. This is a factor of 3 or 4 less than the stress observed to produce damage in experiments reported by H. C. Savage of the Technical Division.

Assistance was given Section II of the Technical Division in interpreting results of their thermal stress tests.

DESIGN PROGRESS

DESIGN CONTRACTOR

During this quarter the Blaw-Knox Company has actively entered into the design of the reactor under contract with AEC through the Idaho Office. On October 6, five Blaw-Knox design engineers were sent to Oak Ridge and have since contributed significantly to the progress of ORNL design drawings while at the same time becoming familiar with the problem. These men have made drawings of the reactor tank, the thermal column, HG-9 experimental hole plug, thermal column lead door, regulating and control rod position indicators and magnetic coupling details, and reactor structure layouts. They are working as members of the ORNL design group in so far as is possible under existing Blaw-Knox policies.

Blaw-Knox has submitted for approval first drafts of flow diagrams of the "Reactor Process" and "Experimental Facilities", as well as those for the Air System and other items more in the scope of Argonne National Laboratory responsibility. Those on the reactor and experimental facilities have been carefully reviewed and corrections recommended.

Present indications are that Blaw-Knox will do all detail design external to the reactor tank, but all details inside and including the reactor tank and the complete system of controls will be the responsibility of design groups at ORNL.

DESIGN REPORTS

Efforts of the ORNL design group have been concentrated since July 1 toward issuance of a series of design reports for transmitting complete design data to the Blaw-Knox Company. Titles and present status of these reports are given in Table 2-1. Much of the information for these reports has already been transmitted in semi-final form to Blaw-Knox for use in a preliminary cost estimate. It is expected that the complete series will be issued by March 1, 1950.

TABLE 2-1

Status of MTR Design Reports

TITLE	STATUS
1. Reactor Cooling Water Monitoring System.	Rough draft partially complete.
2. Thermal Column	Rough draft.
3. Vertical Hydraulics	Preliminary study only.
4. Instrument Holes	Rough draft.
5. Tank Sections A & E	In reproduction.
6. Graphite Reflector	Issued October 6, 1949, CF-49-10-88.
Addendum to Graphite Report	Issued November 17, 1949, CF-49-11-176.
7. Air System (Internal)	In reproduction.
8. Thermal Shield	Indefinite pending calculations now in progress.
9. Structural Steel (for Reactor structure)	Rough draft.
10. Neutron Crystal Spectrometer	In preliminary study.
11. Shielding Facility	Rough draft.
12. Reactor Control Rod Mechanisms	Rough draft.
13. Reactor Experimental Facilities	Drawings complete but report not yet started.
14. Reactor Structure	Drawings complete, but unchecked.
15. Design Data Sheets	Issued October 18, 1949, CF-49-10-232.
16. Mock-Up Design Report	In reproduction, CF-49-10-185.

REACTOR STRUCTURE

In order to show in detail the most recent conception of the reactor structure, a new set of drawings, DRP-61 through 87, has been made.

Based on experimental hole plug designs submitted by Argonne National Laboratory, details of hole liners have been completed for all horizontal and down beam holes and the horizontal through hole. Each liner is capable of adapting the use of removable "radiation valves".

New drawings also indicate steel plates imbedded in vertical faces of the structure for the purpose of tapping mounting holes at any future time when it may be desired to hang apparatus, pipes, etc. there. Removable steel balconies are shown supported on the face of the shield for access to down beam holes. The arrangement of these balconies is shown in Fig. 2-1.

EXPERIMENTAL AND SERVICE FACILITIES

Six 4-inch square horizontal holes have been added in the thermal column to increase the utility of that facility.

Fifteen 2-inch diameter vertical holes have been spotted in the west portion of the concrete shield to permit radiation and temperature readings in the shield.

Entrances to the offset horizontal beam holes are to be recessed in the concrete and 8-inch diameter holes connect these recesses to the basement for use in experiments where shielded lines to the facility from the basement are required.

These are the only major changes in the experimental facilities. A new list of the facilities and their locations has been made (DRP-64) and issued.

With the aid of ORNL research people, a survey of desirable services for experimental facilities has been made, and it is planned that the following will be available at or near each experimental facility:

1. Liquid Supply:	Cooling water (dimineralized)
	Raw water
	High pressure water
2. Liquid Drains:	Hot drains
	Warm drains
	Cold drains

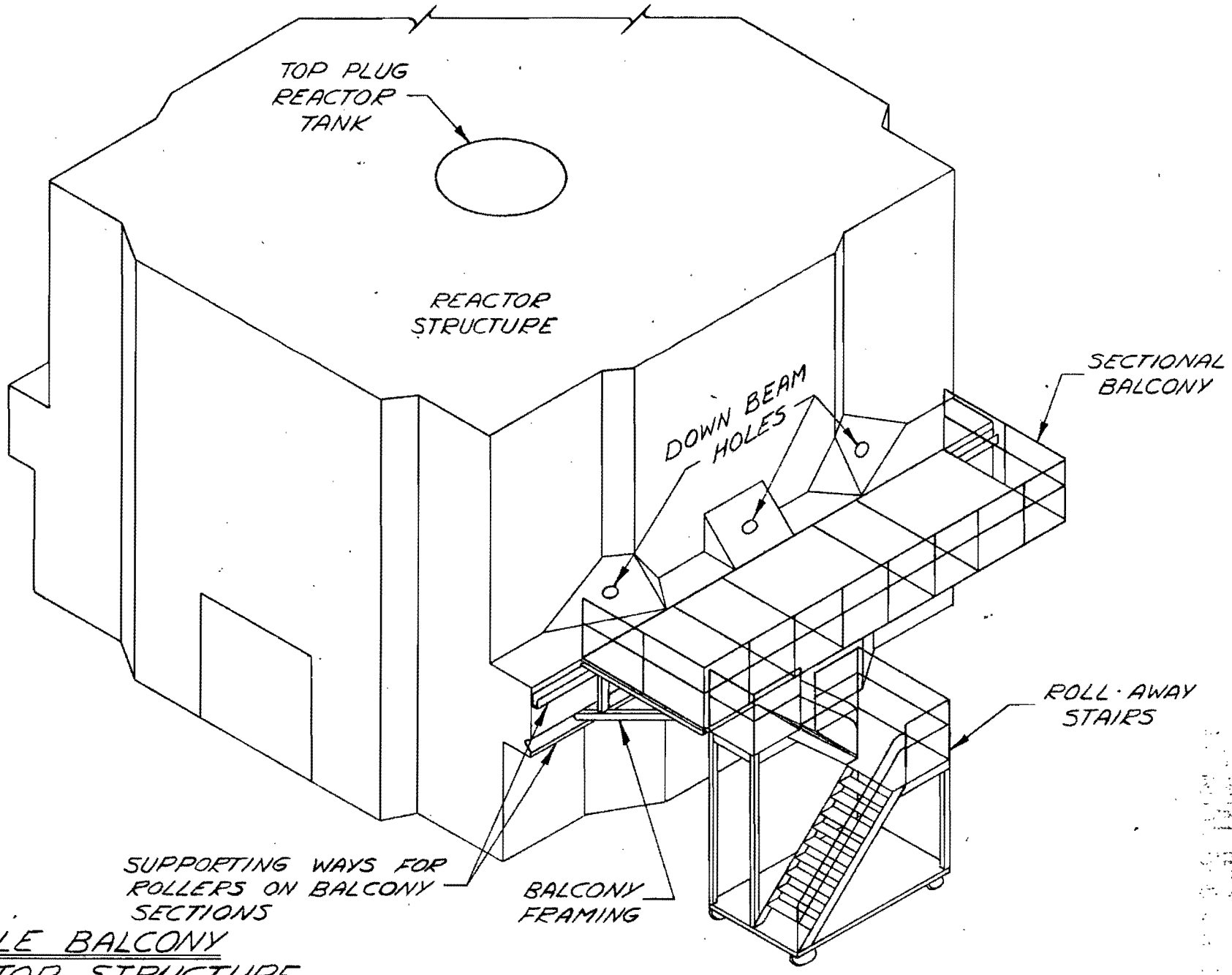


FIG. 2-1
REMOVABLE BALCONY
ON REACTOR STRUCTURE

DWG. No. 8047

3.	Gases and Vapors:	Air
		Carbon dioxide
		Propane
		Steam
4.	Vacuum:	Hot exhaust
5.	Electrical	110-volts a-c
		110-volts a-c regulated
		220-volts a-c
		440-volts a-c
		200 - 300 volts d-c
6.	Control:	Open conduit
		Telephone

THERMAL SHIELD

Completion of the thermal shield design report has been delayed for lack of complete information regarding temperature patterns within the steel plates during operation. Calculations are now being made to determine operating temperatures as accurately as possible so that a tentative design proposal may be included in the design report. Figure 2-2 shows the bottom construction of the thermal shield presently proposed. It is hoped that approximately 70 percent of heat generation in the bottom shield due to neutron and gamma-ray absorption will occur in the 2-inch plate which is cooled on both sides. This should result in the bottom 4-inch plate being at a relatively low temperature. For this reason the support structure for the graphite stands on this lower plate. A similar arrangement is being considered for the top plate of the thermal shield.

GRAPHITE REFLECTOR

The design report on the graphite reflector has been issued and followed by addendum No. 1 which covers design of the pebble discharge system, graphite radiation tests at ORNL, thermal warping of solid graphite, graphite pebble production at Y-12, and design of the pebble support.

Figure 2-3 shows some details of one of the two pebble discharge chutes

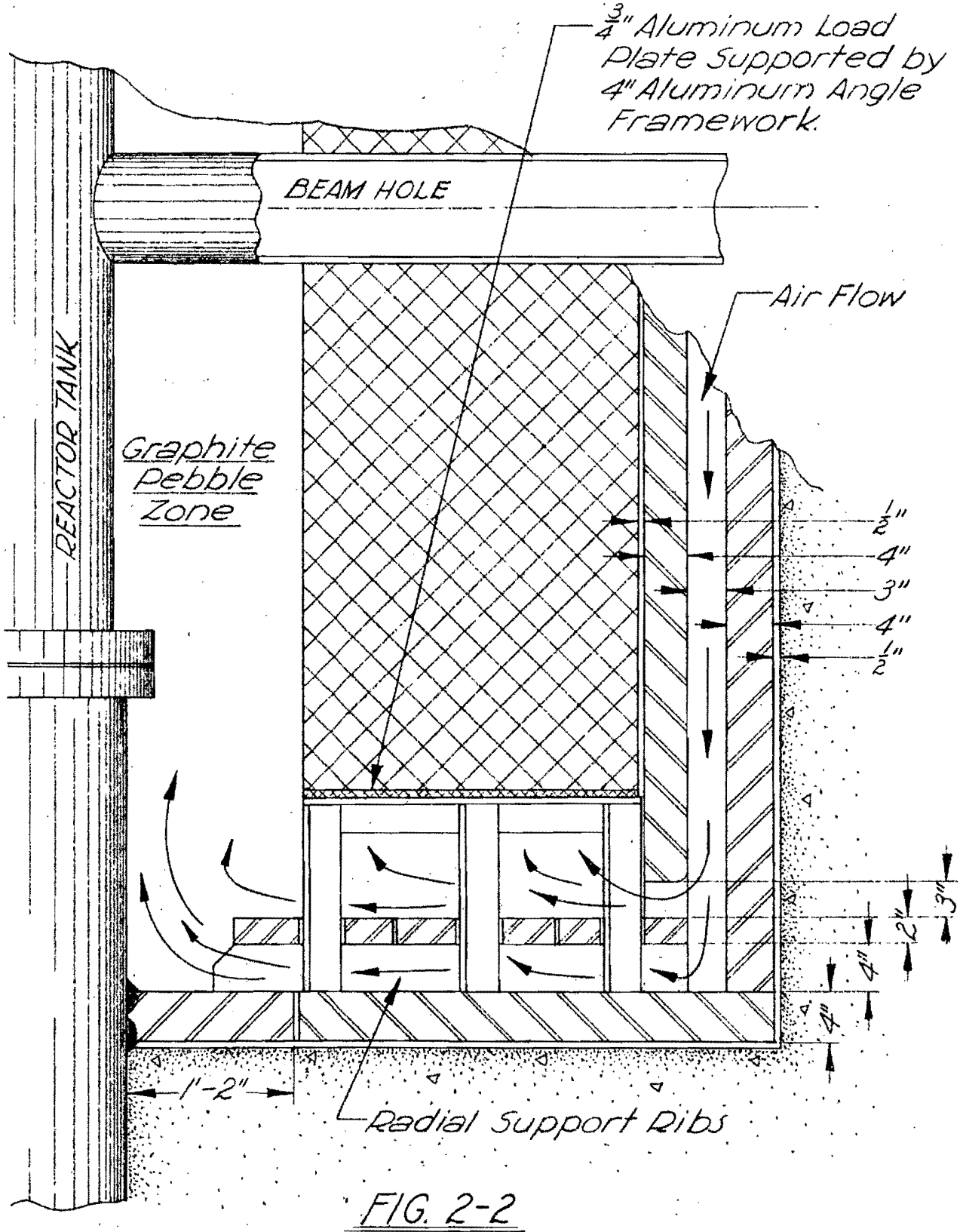
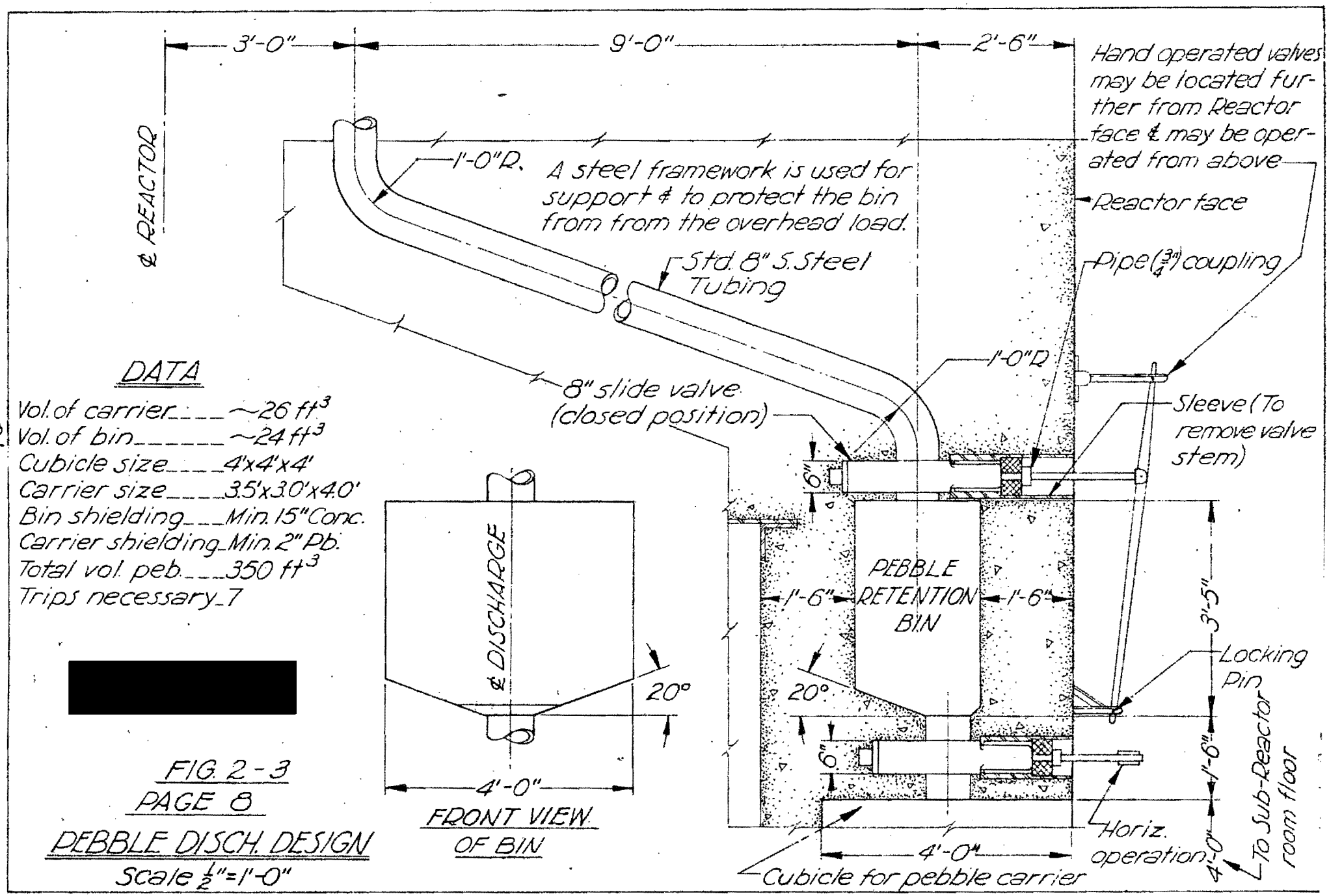


FIG. 2-2

LOWER THERMAL SHIELD COOLING PROPOSAL AND GRAPHITE SUPPORT



DATA

Vol. of carrier ~26 ft³
Vol. of bin ~24 ft³
Cubicle size 4'x4'x4'
Carrier size 3.5'x3.0'x4.0'
Bin shielding Min. 15" Conc.
Carrier shielding Min. 2" Pb.
Total vol. peb. 350 ft³
Trips necessary 7

[redacted]

FIG. 2-3
PAGE 8
PEBBLE DISCH. DESIGN
Scale ½"=1'-0"

by which it will be possible to drain all pebbles from the pebble zone of the graphite reflector. The 8-inch tube shown in the figure connects to the support plate for the pebbles inside the thermal shield. Figure 2-4 gives some details of a shielded carrier suggested for catching the pebbles and removing them.

From results of recent induced activity experiments it now appears likely that AGOT graphite will be used for the graphite reflector. Present designs are based on use of AGOT.

Cost data from production of AGOT pebbles at Y-12 for the critical experiment indicate a total cost of 3.96¢ per pebble.

THERMAL COLUMN AND HG-9 PLUG

The addition of the HG-9 plug and liner to the subject matter has delayed the design report on the thermal column, but is now (Nov. 21) essentially complete and will be issued in a few days. The following drawings have been prepared:

DRP-82	Thermal Column Assembly
DRP-83	Graphite Arrangement
DRP-84	HG-plug Sect. "A"
DRP-85	HG-Plug Sect. "B"
DRP-86	HG-Plug Sect. "C"

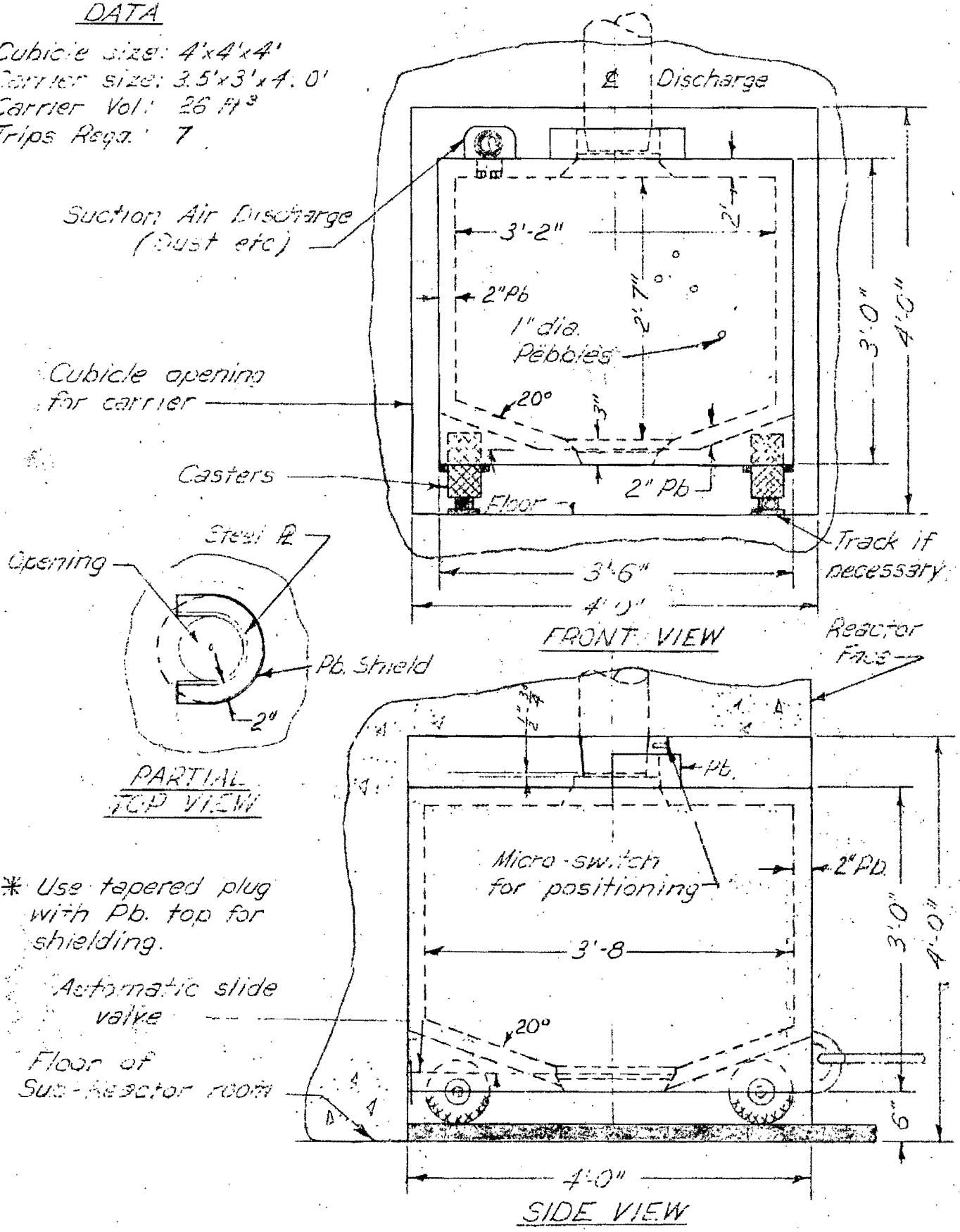
The present concept of the design is shown in Fig. 2-5. On this, the graphite filler and the HG-9 plug were omitted for clarity of the details of construction. T. H. Thomas, aided by R. B. Briggs, has calculated air cooling requirements for the HG-9 plug where it enters the pebble zone. These calculations are included in the forthcoming report. The design of the thermal column with respect to shielding adequacy is as yet unconfirmed by calculations.

SHIELDING FACILITY

A design conception has been completed on the experimental shielding facility. The report on this design is now in rough draft. Drawings DRP-73, 74, and 75 are layouts of the completed conception. It consists essentially of a "lid-tank" type of facility similar to that now in use at ORNL. The plan

DATA

Cubicle size: 4'x4'x4'
Carrier size: 3.5'x3'x4.0'
Carrier Vol.: 26 ft³
Trips Reqd.: 7



* Use tapered plug with Pb. top for shielding.

Automatic slide valve

Floor of Sub-Reactor room

FIG. 2-4

SUGGESTED PEBBLE CARRIER

Scale ¾" = 1"

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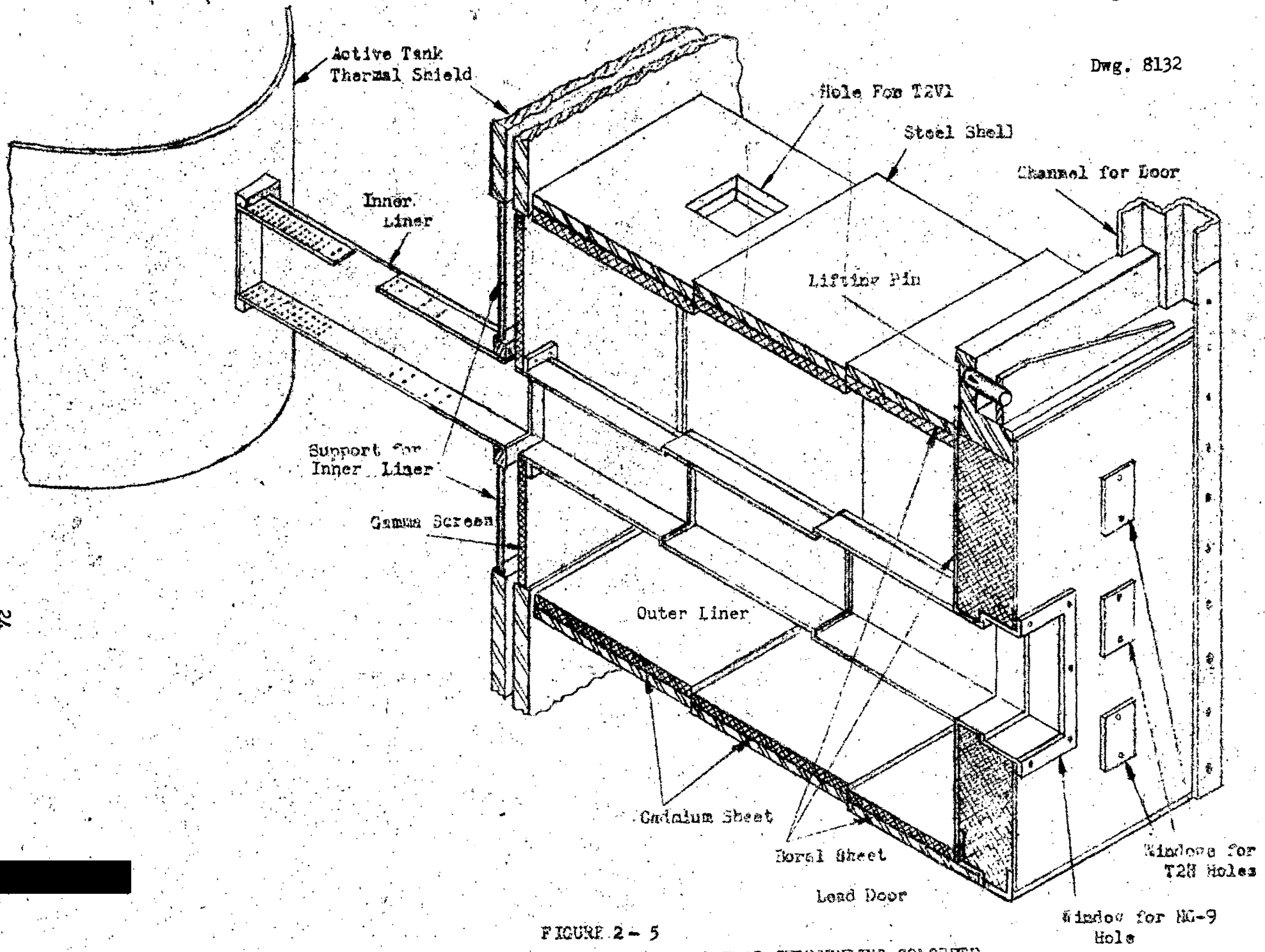

FIGURE 2 - 5

THERMAL COLUMN- CUTAWAY VIEW WITHOUT GRAPHITE FILLER OF SURROUNDING CONCRETE

[redacted]

is to use an open space in the reactor shield formerly intended as a thermal column, coupled with an external water tank so that the entire space may be flooded with water for removal and replacement of shield slabs during operation of the reactor. The outer end of the tank contains a deep well for storage of extra hot specimens during cooling. A remotely controlled carrier will move the slabs into position inside the hole where a U-235 converter plate will be previously placed for generation of fast neutrons. The converter plate also will be remotely removable through the water tank after slabs are withdrawn.

REACTOR TANK

Detail drawings of all components of the reactor tank have been completed by Blaw-Knox engineers working with D. J. Mallon. Stress analyses of the tanks and the flange joints have been made and tabulated. A design report on tank sections A and E is being issued. These are the non-removable portions of the tank and must be included in design of the reactor structure. It is now planned to build the upper section (A) with a 5/8 inch thick wall and the lower section (E) with a 2-inch thick wall. These thicknesses appeared satisfactory in recent analyses of the shielding requirements.

No other significant changes are contemplated for other parts of the tank. All bolting and flanges are to be made stronger than those in the Mock-Up.

CONTROL RODS AND DRIVES

During the past three months a general program of clarification of detail design of the reactor controls has been in progress. Some of the more important items which have been considered are outlined below.

The assembly drawing of the magnetic coupling for the shim-rod has been revised to incorporate those features which provide improvement in the manufacture and operation as determined by tests on the Mock-Up. The upper part has been redesigned to include a quick-release mechanism for releasing the magnet from the drive rod to facilitate its handling after irradiation. The quick-release mechanism may be operated remotely by means of a socket wrench

with a handle of suitable length. The major portion of the redesign of this unit was accomplished by Mr. Otto Vogelsang of the Blaw-Knox Company and Dunlap Scott of NEPA.

Assemblies and details of the redesigned position indicators for both the shim rod and the regulating rod are nearing completion.

A design study of the regulating rod shock absorbers is now in process by Mr. Vogelsang with a view toward making the regulating rod locking mechanism accessible without the necessity of removing the top shock absorber.

A large number of new assembly drawings are being made for the purpose of clarifying the location and operation of the various parts.

A new assembly is being made of the Top Plug Platform which will show all the component parts of the controls in their proper relationships and will include all revisions to date.

COOLING WATER MONITORING SYSTEM

In October, T. H. Thomas made a trip to the Hanford Engineering Works to get first hand information on the present practices and equipment used in their monitoring system for the pile water. The detailed results of this trip are covered in a memo (CF-49-11-272). This information will be of considerable value in the future development of the MTR monitoring system, especially with respect to activity monitoring.

The subject matter of the design report on the monitoring system has been broadened to include the entire water system, both internal and external. This will necessitate rewriting of the existing report which is in the rough draft stage awaiting data on the monitor tube from the Mock-Up. Since these data have been delayed, it is intended to complete the report for the present and issue subsequent information as addenda.

CONTROL SYSTEM

A summary of the development of the MTR controls as of April, 1948, appeared in CNL-35. The philosophy of the control system had been worked out, and most of the critical circuits had been tested. Many of the more straightforward items had not been designed and the construction details had scarcely begun.

We are now attempting to complete the design of the MTR Control System. This has not been easy for most of the engineers and physicists that started the development have left the project and the remaining ones shifted to other jobs. The original group consisted of the following people:

H. W. Newson, Leader (Resigned)
P. R. Bell (Now serving as a consultant)
John Trimmer (Resigned)
Bert Frey (Resigned)
S. N. MacNeille (Resigned)
T. E. Cole (Present)
Henry Straus (Resigned)
Fontaine Armistead (Resigned)
L. W. Mead (Resigned)
E. P. Epler (Consultant)
David Cardwell (Transferred)
Joseph Gundlach (Present)
W. H. Jordan (Present)

The reactor control group now consists of the following persons:

W. H. Jordan, Leader (Loan from Physics)
T. E. Cole
E. P. Epler (Soon to be available)
J. E. Owens (Loan from Instrument Division)
Gorman Nelson (Loan from NEPA)
Lyle Martin (Loan from NEPA)
E. E. St. John (Loan from NEPA)
J. M. Groover (Technician, Instrument Division)
O. S. Swift (Part-time, loan from Westinghouse)

Some additional help is available on a part-time basis from P. R. Bell and Joseph Gundlach of the Physics Division. It is anticipated that Edward Bettis from NEPA will soon be loaned to the group.

These people are now working on the following projects:

(1) *Providing Instrumentation for the Pile Mock-Up.* Mr. Owens is responsible for this phase of the work. St. John, Cole, and Groover are working

on other safety circuits. It was found that the original design was not entirely adequate; necessary modifications are being made and the circuits tested.

The servo system is being installed on the Mock-up under the direction of Nelson. He is modifying the circuit somewhat to insure reliable and stable operation. Actually, the servo has demonstrated its ability to control the pile simulator, but the speed is not as fast as hoped for. One of the chief reasons for this is not being able to find a satisfactory tachometer - this in spite of considerable development work by Weston Instrument Company.

A safety ionization chamber (Drawing Q-940-1) has been designed by Mr. Abele of the Instrument Division, after considerable consultation with Bell, St. John, and Jordan. These chambers are 2 in. in diameter and designed to operate on the neutron flux streaming through the holes in the reactor shield. Two of these chambers are nearly complete. Two more will be needed, but we are waiting until some flux calculations by Lansing are finished. We are seeking an answer to the question as to where the chambers must be placed in the hole to keep the time lag small and how large must the chamber be to generate 100 μ-amp of current.

(2) Working with Blaw-Knox on the Design and Layout of the MTR Controls. Considerable time and effort has gone into explaining the proposed control scheme to Messrs. Bates and Driskell of the Blaw-Knox Company. They have submitted proposals and requested information which have required the attention of the entire group.

(3) Continuing the Development of MTR Controls. Much remains to be done.

a. The circuits must be proven on the pile mock-up and any necessary modifications made.

b. Detailed layouts of the shim safety circuits must be made and reliable components chosen.

c. Detailed plans for the control and instrument room must be completed; close cooperation with Blaw-Knox and Argonne will be required.

d. Provision for the placement of instruments and cables at the pile structure must be made.

e. Plugs for the instrument holes must be designed.

f. A mechanism for moving the fission chamber must be designed.

g. Annunciator circuits must be provided to warn the operator of failure of any instrument.

h. A mimic board to inform the operator of the position of the important interlocks must be worked out in detail.

Much of the work outlined in this section will be done by the contractor, but we must keep in very close touch. Epler and Martin will watch their phase closely.

(4) *Agreement that ORNL Would Provide Prototypes of the Special Instruments Needed.* These instruments will be constructed by the Instrument Division under the supervision of the members of the Control Group.

(5) *Further Considerations of the Safety Circuits.* We have recently realized that the safety chambers may be subject to considerable shading by the shim rods. Since the shim rods vary in position, the flux at any chamber may vary by perhaps a factor of two while the pile is operating at a fixed power level. Present thinking is to set the fast safety circuits at twice operating level. Slower safety circuits which are not sensitive to shading would then be set to "scram" the pile at some 1.1 times operating level. For this purpose one could use water power, water temperature, N^{16} activity of the exit water. Cole is looking into the possibility of using boron thermopiles.

EXPERIMENTAL

STATUS OF MTR MOCK-UP

Conversion of the Mock-Up for use in criticality experiments is 50% complete. Experiments are scheduled to begin January 1, 1950. Fluid flow and handling tool studies have been completed and final reports are being written. Control-system mechanical testing is complete except for tests on redesigned components.

CRITICAL ASSEMBLY

In September it was decided to prepare the MTR Mock-Up for use with a critical mass of fuel. Experiments are planned at an operating level of 1 to 100 watts (approximately 10^9 neutrons/cm^2 sec). Several objectives of these experiments have been enumerated by M. M. Mann: (1) Performance of control system; (2) Neutron and gamma flux measurements in water above and below reactor; (3) Neutron and gamma flux measurements throughout graphite; (4) Fluxes in experimental holes; (5) Determination of temperature coefficient; (6) Changes in reactivity caused by bubbles and vibration due to water flow.

The present Mock-Up structure is being altered with the general aim of providing facilities for these measurements under circumstances which will closely approximate operating conditions in the MTR. More specifically, materials and major dimensions specified for the MTR are being duplicated in almost two quadrants of the Mock-Up. Photographs 4-1 and 4-2 show the work approximately 50% completed. The major alterations in progress are:

1. *Reflector inside Reactor Tank.* The aluminum reflector which was installed for hydraulic testing will be removed and partially replaced by 2500 lbs of beryllium metal (old powder process) now stored at the laboratory.

2. *Graphite.* The graphite pebble zone immediately adjacent to the reactor tank will be partially (40%) filled with 300,000 one-inch diameter graphite balls being made for use in the MTR installation.

AGOT graphite blocks obtained from surplus stock are being stacked outside the pebble zone in three quadrants surrounding the tank. Forty one-inch diameter vertical holes through the graphite are being provided for flux measurements.

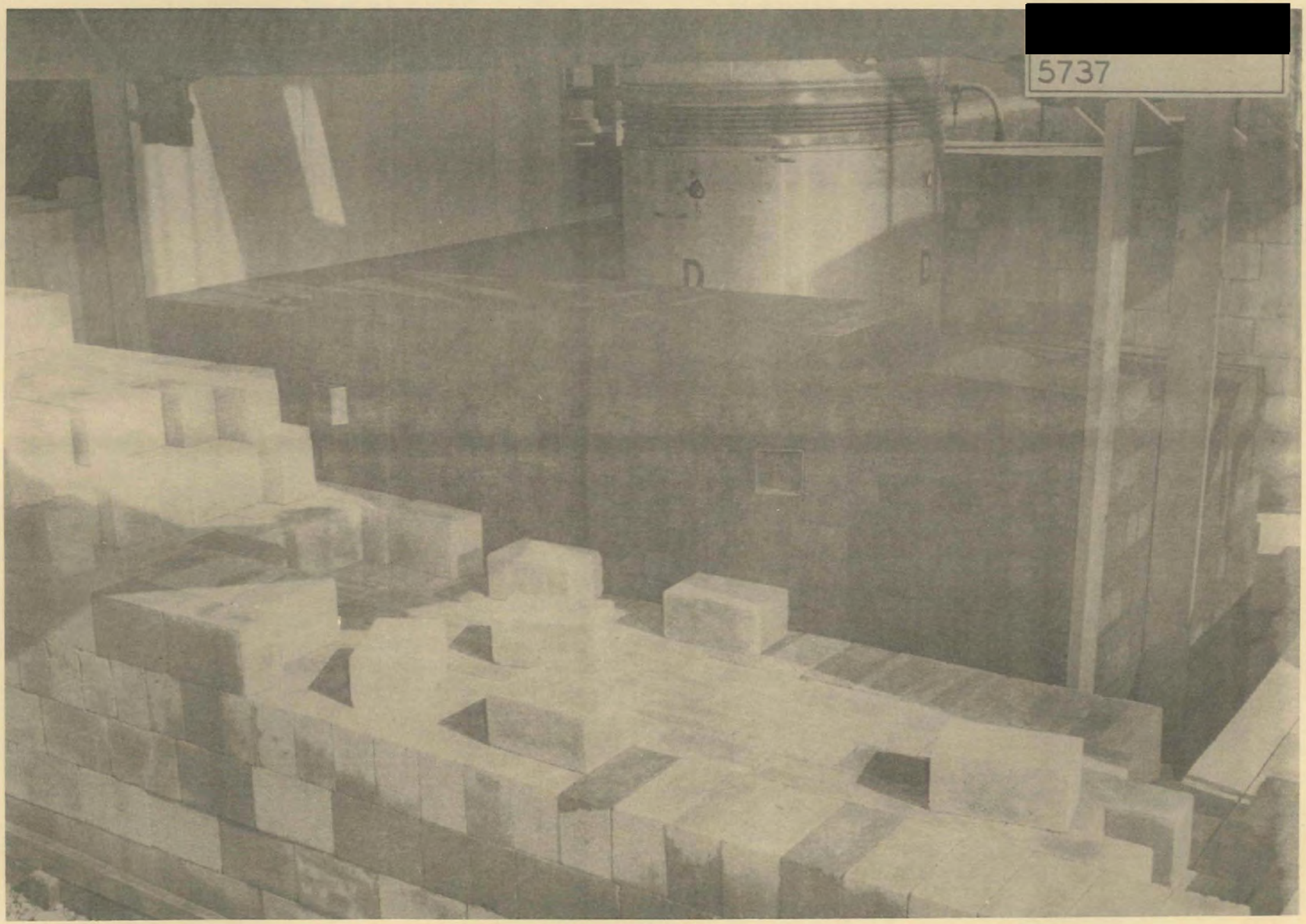

FIG 4-1

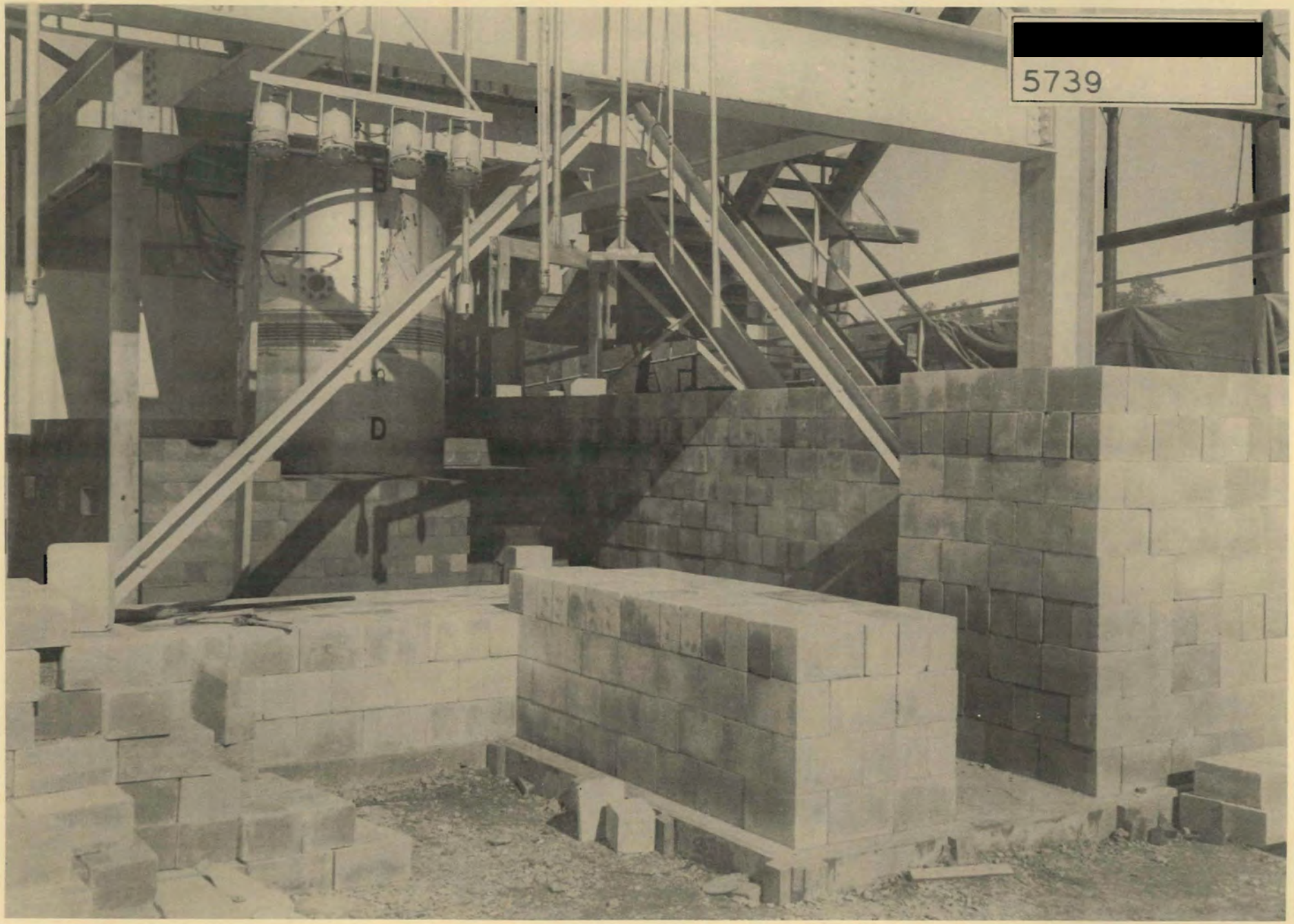

FIG. 4-2

3. *Shielding.* A four-foot thick wall of concrete blocks is being constructed to surround the entire active portion of the structure. A waterproof coating will be provided.

4. *Instrumentation.* The two shim-safety rods and single regulating rod with their existing control instrumentation are being supplemented with the ion chambers, amplifiers and recorders necessary for adequate protection. This portion of the control system will be almost identical to that of a typical channel planned for the MTR. A third safety rod is being prepared for use as a cocked safety during the fuel loading operation.

HYDRAULIC MEASUREMENTS IN REFLECTOR AND FUEL LATTICE

As discussed in the previous quarterly report, initial measurements in the reactor tank indicated 750 to 1000 gpm water wastage in the reflector and that further throttling of flow through the lower support casting would be necessary to reduce the loss. Additional throttling was accomplished by closing the keyway spaces in the removable reflector pieces and by plugging the the weep holes in the casting. The total area closed by these remedies was approximately 7.5 square inches, estimated to be 30% of the total area available for flow. Subsequent velocity data indicated less than a 5% reduction in reflector flow. Additional reduction of the open area of the casting would be difficult in the Mock-Up.

During the past quarter the entire system has been checked for leaks, fuel assemblies recalibrated and additional data taken. Since the resignation of W. K. Stromquist, who began these studies, the work has been continued by W. L. Ross with the cooperation of R. B. Briggs of the Design Section. Although analysis of the data is not yet complete, the following typical figures may be tentatively reported:

Pressure Drops

1.	Overall	33.3 psi
2.	Through upper support casting and grid spacer	3.2 psi
3.	Through fuel assemblies, including end boxes	30.1 psi
4.	Through fuel plates (between end boxes)	17 psi
5.	Velocity through fuel plates	27 ft/sec
6.	Through reflector	6.6 psi
7.	Through lower support casting (beneath reflector)	26.7 psi

Flow Balance - 33 psi

1.	Through reflector	2000 gpm
2.	19 fuel assemblies	8100
3.	14 Be assemblies in lattice	40
4.	8 shim rods or substitutes	3170
5.	Gaps in lattice (between assemblies)	1050
	Total flow	14,360 gpm

It was learned that an overall pressure drop of approximately 45 psi is necessary to maintain the desired velocity of 30 ft/sec in fuel elements. The total water requirement at this pressure may be estimated, based on the above flow balance and assuming a normal slab loading (3.3 kg).

Estimated Flow at 45 psi

1.	Reflector	2300 gpm
2.	23 fuel elements	10925
3.	16 Be assemblies	50
4.	8 shim rods or equivalent	3450
5.	Gaps in lattice between assemblies	2100
6.	By pass	475
	Total	19300 gpm

CONTROL SYSTEM

1. Regulating Rods. (a) Performance in Mock-Up. Two series of response time runs have been made to show the characteristic of the regulating rod motor (1) when running with no load, and (2) when loaded with the reduction gears and rod.

These runs demonstrated that the servo system will respond in 0.1 seconds for both conditions. Tachometer performance must be improved before better response is obtained. The Physics Division has the problem under further study.

(b) Pile simulator. The electronic pile simulator developed by Jordan's Physics group is being installed by that group for further testing of the servo system. This work will continue during preparations for the critical experiment.

(c) Shock absorber for regulating rod drive. The lower shock absorber

housing has been revised to eliminate the upper oil filling hole and to provide a smooth inner wall. A new spring has been made to provide a stronger return action.

Both upper and lower shock absorbers are now considered acceptable.

2. *Shim Safety Rods.* *(a) Drive Mechanisms.* Although the longest single continuous test was only 60 hours, both drives have been operated intermittently over the past six months without difficulty and are considered satisfactory.

The speed of rod withdrawal was measured as 0.080 in. per second, slightly less than the design speed of 0.083 in. per second.

(b) Electromagnets. The two magnets used in the Mock-Up for the past six months have been completely reconditioned for the critical experiment. One of the magnets has now been subjected to 60 hours continuous cycling with no sign of trouble. In a test previous to reconditioning, this magnet failed because of a break in the coil winding near the center of the coil.

The laminations for the test magnet of Hypersil alloy were completed and sent to the Westinghouse Laboratories for heat treatment.

(c) Bearings. The spring-loaded graphite bearings now being used offer approximately 50 lbs resistance to the fall of the safety rods. This resistance in addition to 15-20 lbs buoyancy becomes serious if one considers a safety-rod of less weight than the present one (130 lbs) and falling while no water is flowing. The result is a rod acceleration of considerably less than the 1 g desired for safety. For this reason, graphite pads are being replaced by rollers in a test set of bearings now being fabricated.

MOCK-UP—REMOTE HANDLING TOOLS AND ACCESSORIES

The fabrication and testing of handling tools and accessories was completed during this period. A final report has been drafted and is in the process of being edited.

Tools and accessories not discussed in previous quarterly reports will be described in this report. Photographs and descriptions are given below.

Regulating Rod Tool. The regulating rod bearings are removed with the tool shown in Fig. 4-3. The tool is inserted into the bearing, then by means of a rod extending through the tool shaft a taper wedge is pulled toward the tool handle by the operator. This faces the shoes on the tool against the bearing wall and permits removal. By attaching the pair of shoes shown in the

[redacted]

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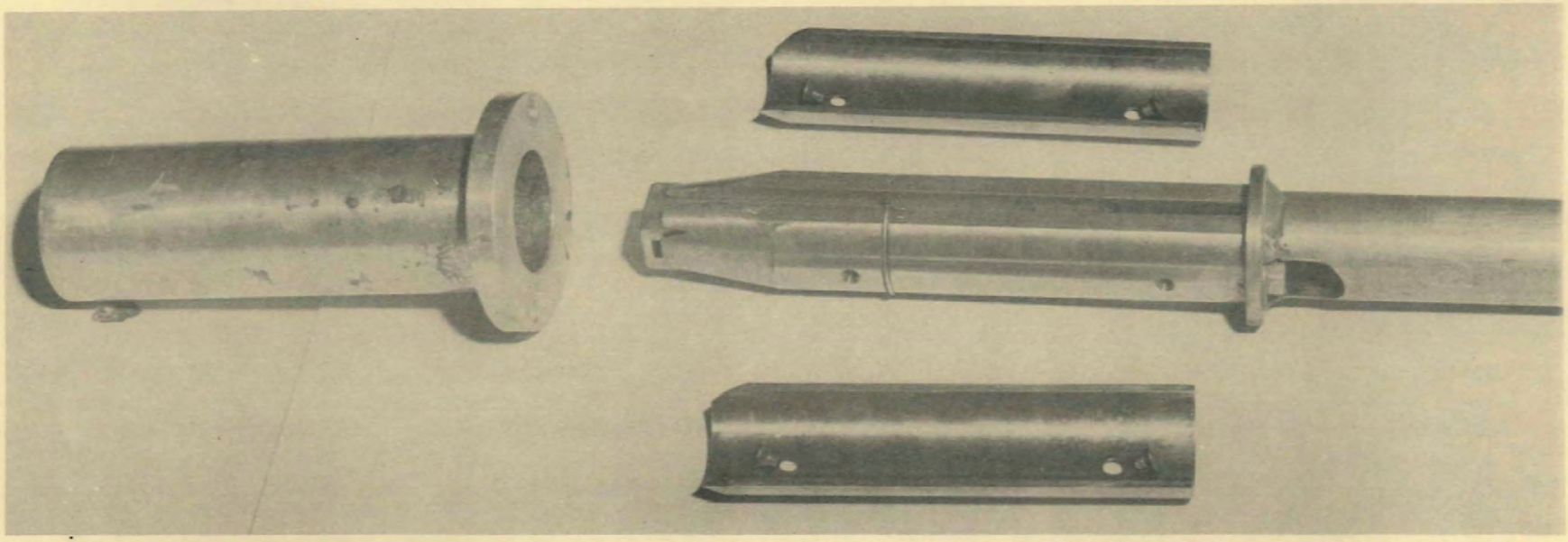

FIG. 4 -3

photograph to the end of the tool, its diameter is increased sufficiently to prevent the removal of the beryllium piece in the reflector region through which the regulating rod extends.

Extension Type Socket Wrench. The socket wrench shown in Fig. 4-4 consists of a one inch drive and standard hexagonal socket welded into the ends of suitable lengths of heavy-wall pipe. The pipe sections are coupled together by flanged joints.

A wire spring has been recessed into the interior castellations of the socket as shown. A plunger supported by means of a spring may be pushed into the castellation region by means of a rod coupled to the plunger and extending to the drive end of the wrench. When the wrench is used it is pushed into the nut, and torque is then applied by an air impact wrench or by hand at the drive end. When the nut is screwed from the thread it remains in the socket, being held by the wire spring. By means of the plunger the nut may be ejected into a container located near the work.

Underwater Lights and Fixtures. Figure 4-5 shows the design of lights used to illuminate the interior of the MTR tank while underwater operations are being performed. Standard 110 V, a-c, 150 watt lamps of the projector flood type provides adequate illumination. Water tight gaskets are used on the aluminum cans. A feature of the lighting fixture is that it is supported on the operating platform in a jig which permits the lamps to be lowered or rotated into a position to provide maximum lighting for the desired area.

Viewing Tube. The accessory shown in Fig. 4-6 is a viewing tube used to provide good vision through the reactor water even with ripples on the water surface. The tube is fastened to the operating platform by means of the clamp device attached to the bolts on the barrel. The tube may be pivoted to view any desired area. A lens of lucite on the tube extends under the water surface and gives good vision (the glass-bottomed boat idea). A telescoping tube permits an operator to adjust the tube to any desired eye level.

Lower Assembly Grid Tool. The lower assembly grid may be removed from the reactor by use of the tool shown in Fig. 4-7. The tool is inserted into a central space in the grid and is stopped by the flange on its shaft. By means of a rod extending through the shaft to the tool handle and attached to the levers in the end, the operator extends the two lifting arms horizontally into a position under the assembly. This allows the assembly to be lifted and guided accurately from the reactor.

[redacted]

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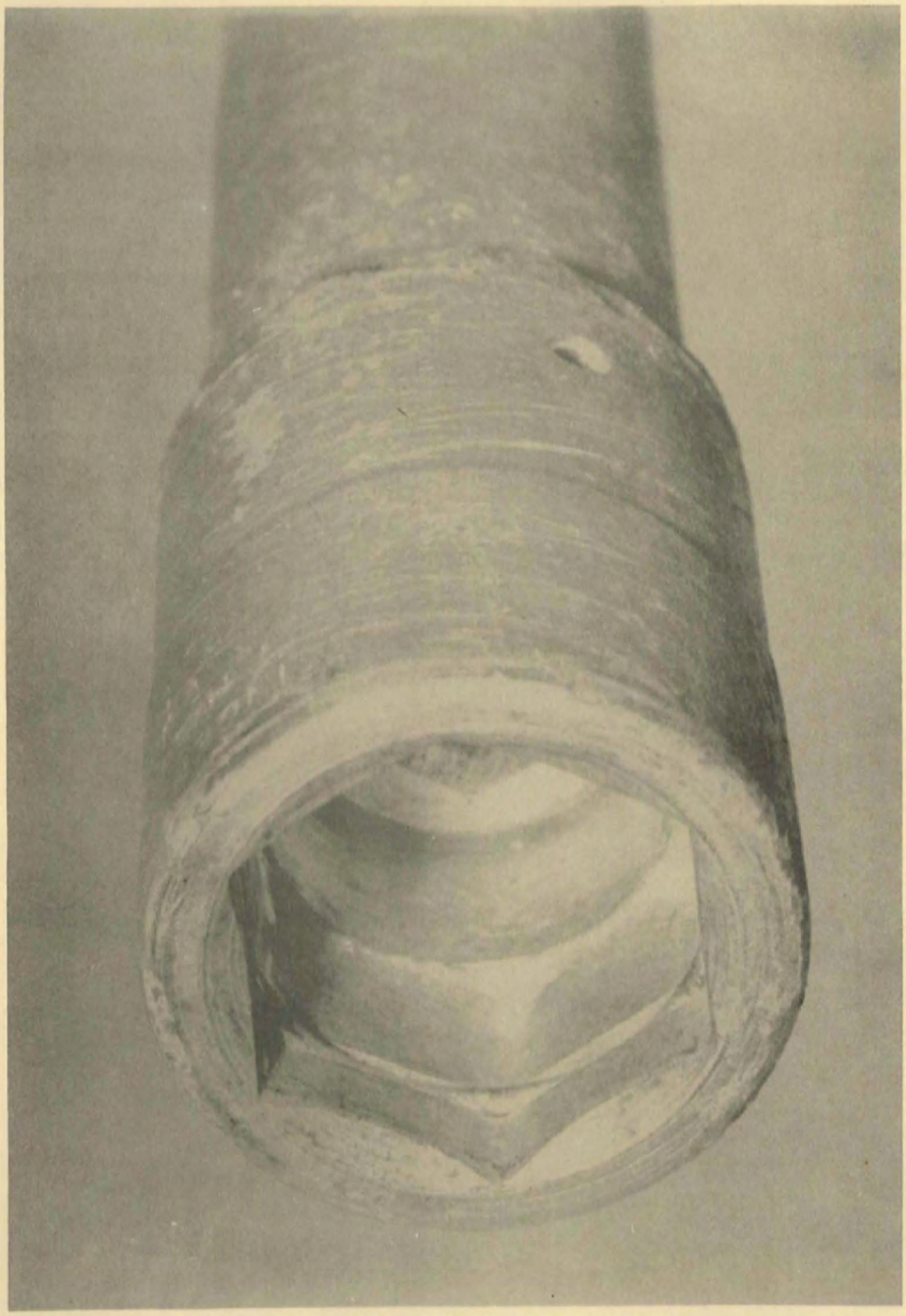

FIG. 4 - 4

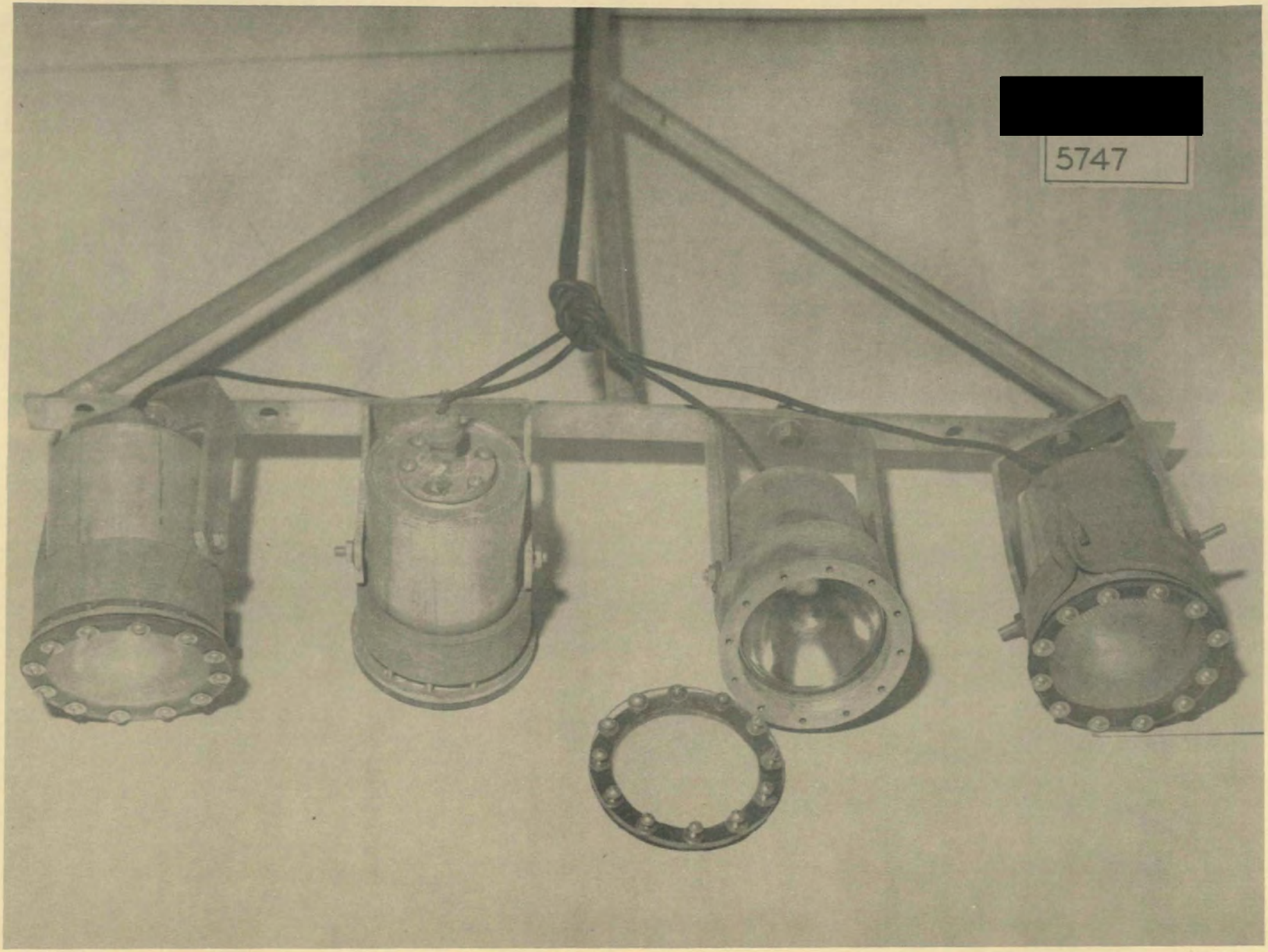

FIG. 4 - 5

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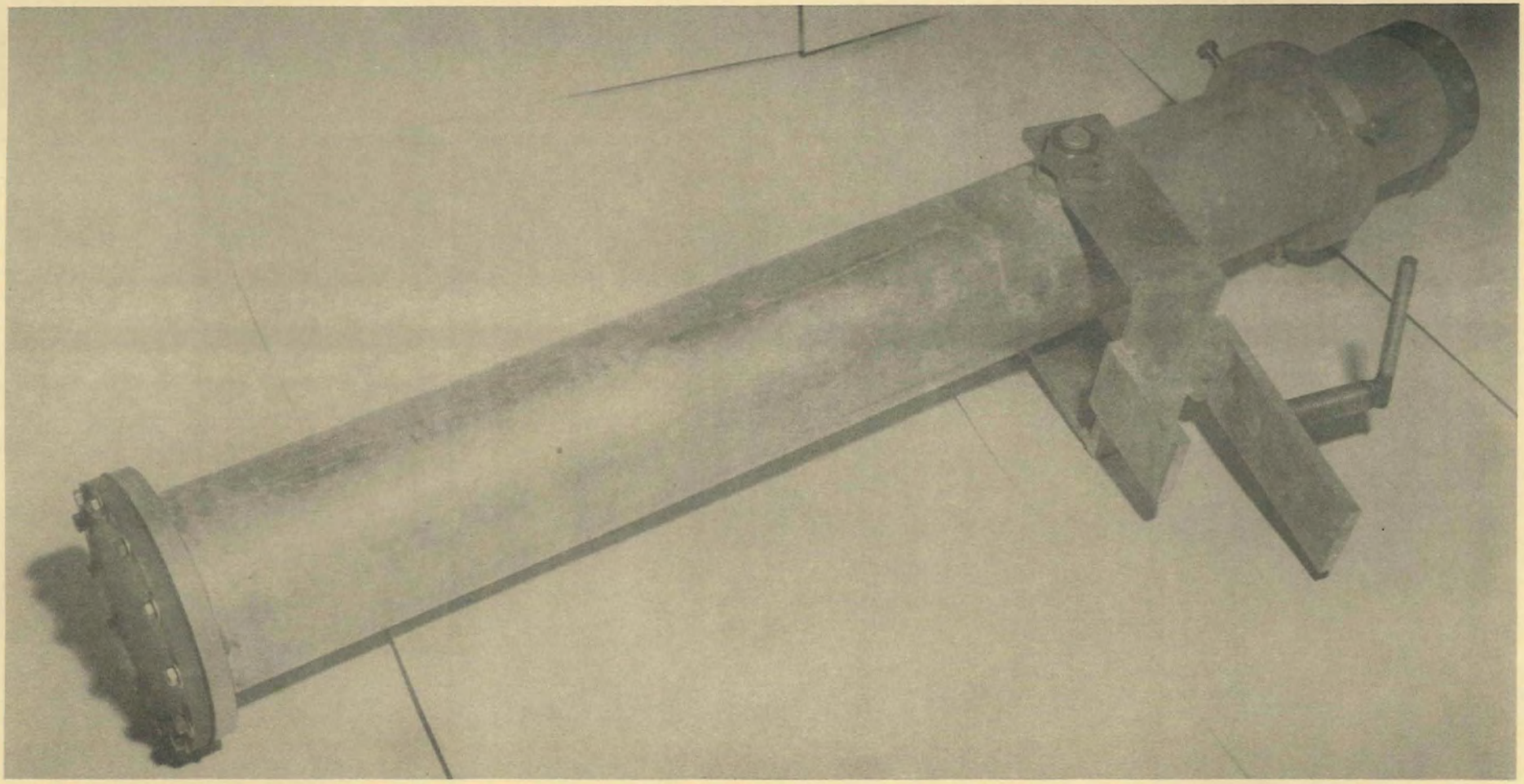

FIG. 4-6

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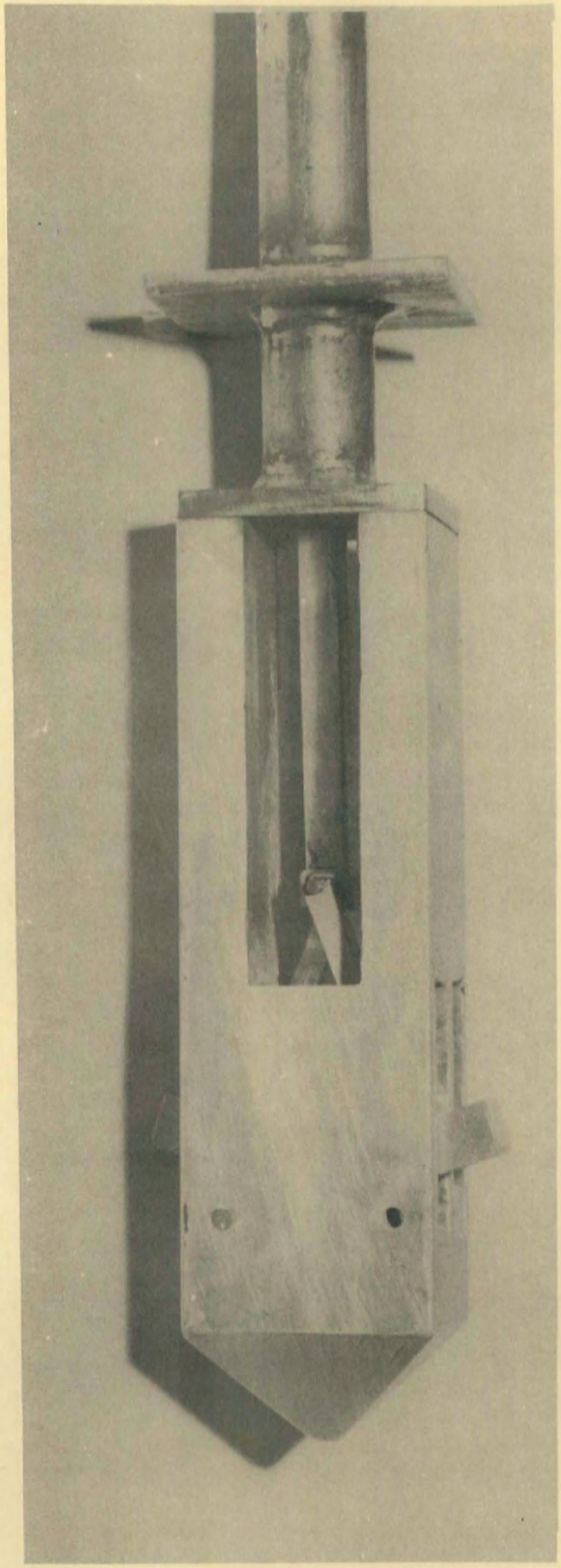


FIG. 4-7

Lower Guide Grid Tool. The tool shown in Fig. 4-8 is designed so that it will extend over the tops of the monitor tubes and attach its four hooks to four eye-bolts provided on the top of the lower-bearing guide grid. Space is available for this operation when the eight shim-safety rod spaces are vacant.

Upper Support Casting Tool. Figure 4-9 shows the specially designed yoke used to remove the upper support casting. Two horizontal pins have been provided in the vertical central ribs of the casting. The yoke is lowered so as to straddle the two ribs, is then slid sideways to engage the pins and is lifted.

STRAIN MEASUREMENTS ON MTR

A final report (ORNL 472) has been issued on this work. Baldin-Southwark "SR-4" resistance wire strain gages were used in making all strain measurements from which values of stress were calculated. The selection of points for strain gage application was made in cooperation with the design section and is believed to represent the points of maximum stress.

The results show that all parts of the MTR tested are satisfactorily designed from a stress standpoint. All castings were made of aluminum alloy 356-T7A with a measured yield strength of 25,000 psi at 0.2% set. The maximum value of stress measured on the castings under operating conditions was 2250 psi on the lower support casting.

Tank sections D and the side skirt plates were made of aluminum alloy 3SF with a minimum yield strength of 6,000 psi. The material used in making tank section D has a tensile strength of 17,850 psi. The maximum value of stress found on these parts was 2150 psi in the wall of tank section D.

It must be pointed out that the design criteria in a large number of cases were not stresses alone, but other considerations such as vibration, alignment and bearing surface area, so most of the stresses were expected to be small.

Fuel Assembly—Thermal Cycling. A "thermal cycling" experiment was conducted on an active fuel assembly (30% U-Al alloy plates) to determine its dimensional stability after repeated heating and cooling. The assembly was subjected to a total of 3925 cycles from approximately 17° C to 100° C with no appreciable change in dimensions or other damage such as warping or blistering. A final report (ORNL 473) has been issued and may be consulted for detailed information.

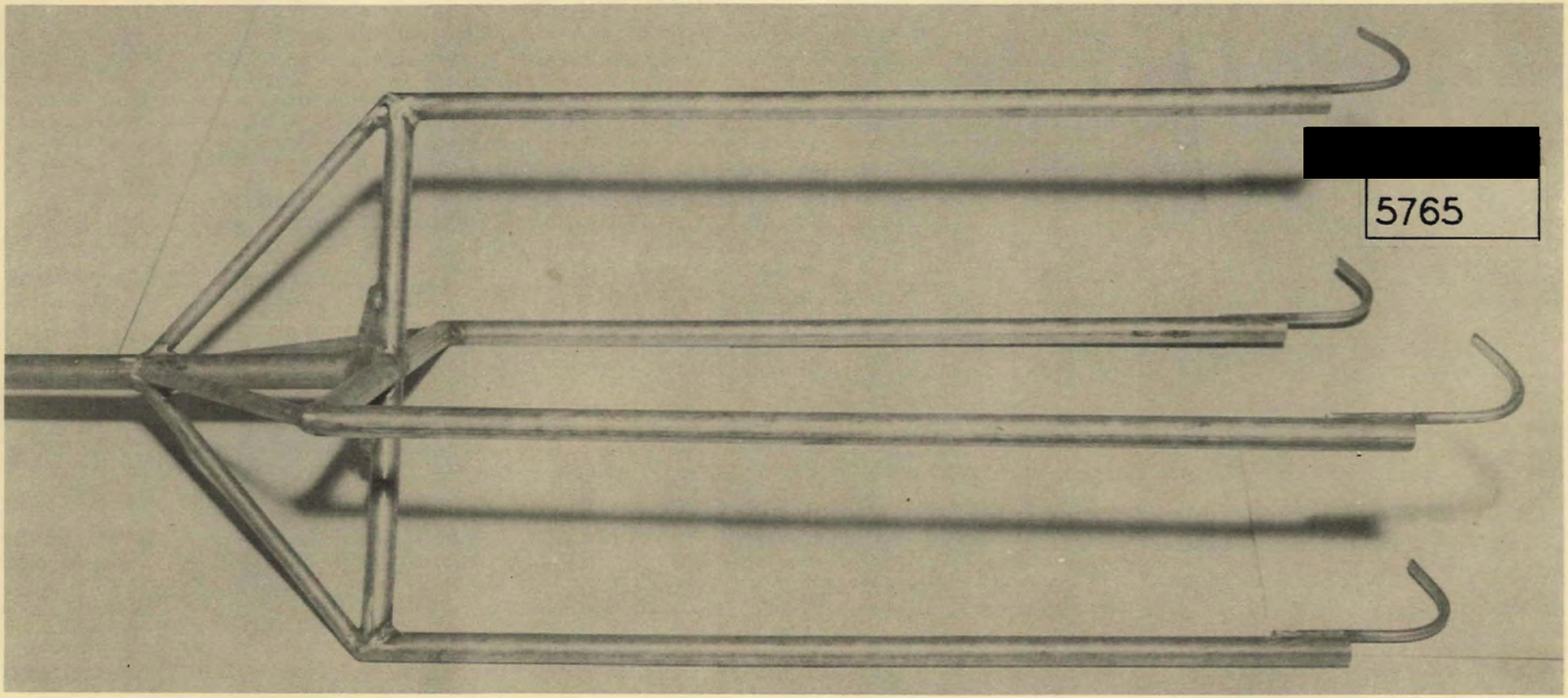


FIG. 4-8

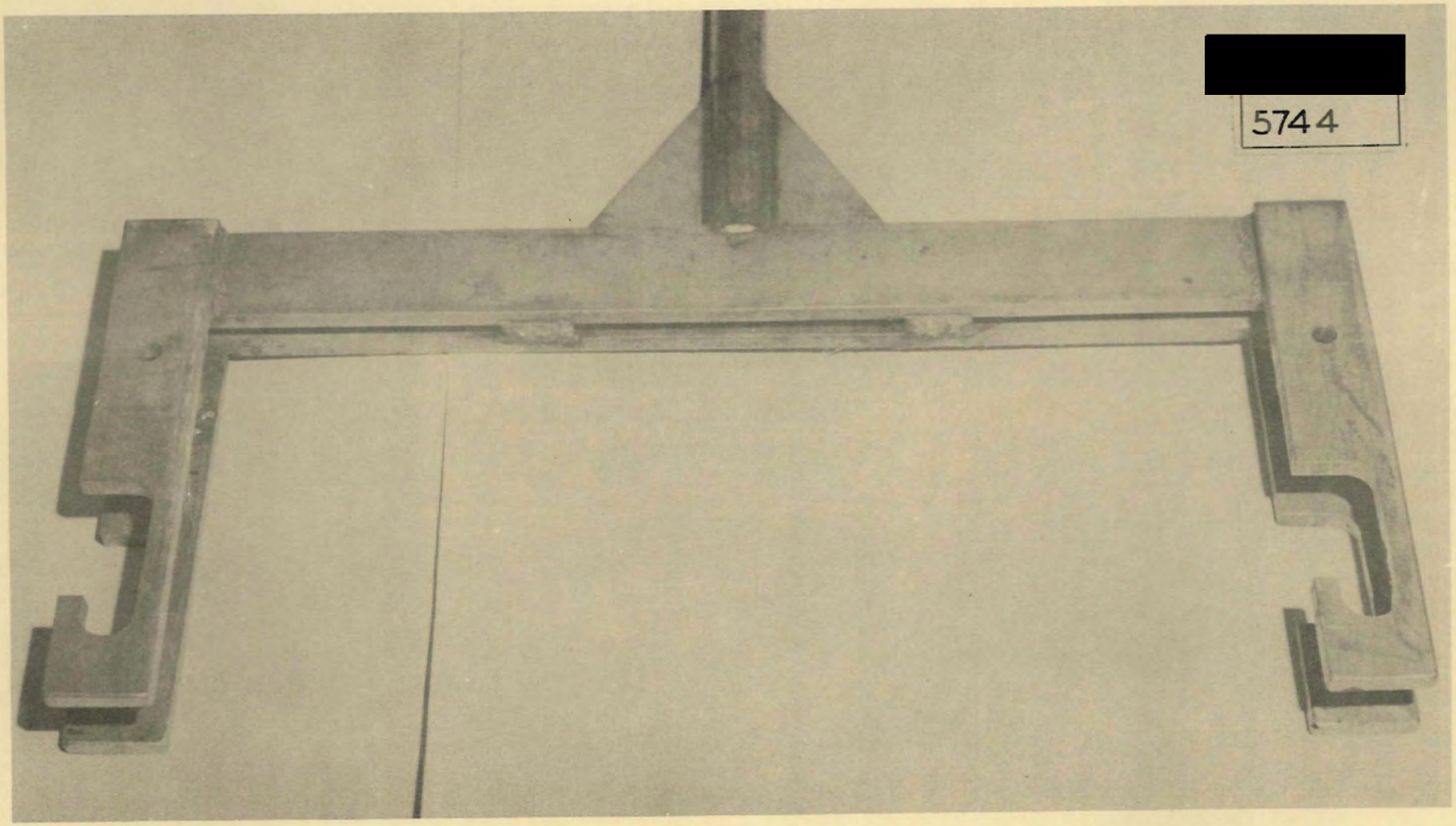


FIG. 4-9

Beryllium—Thermal Stress. A previous investigation (ORNL 73) indicated that beryllium is adequately stable to thermal stress which would result from heat fluxes at the cooling hole 4.7 times greater than will occur in the high flux reactor as presently designed (based on a heat production rate of 4 cal/cm^3 sec).

Additional work has been done on this problem which has resulted in cracking cylindrical specimens of vacuum cast and extruded beryllium. The specimens are heated at the outside surface by means of a high frequency induction coil and cooled at the center by water flow through an axial hole. In these tests temperatures near the inner and outer surfaces were measured and are in good agreement with calculated values.

Cracks at both the inner and outer surface were observed on two cylinders having the longitudinal axis *perpendicular* to the direction of extrusion with a total heat flow of 7.5 KW (1800 cal/sec). Based on the volume of the specimens (86 cm^3), this is an equivalent heat production rate of 20.9 cal/cm^3 sec.

Specimen "failure" was by formation of longitudinal and circumferential cracks at the surface of the axial hole and circumferential cracking at the outer surface. Calculated stresses are 80,000 psi (tension) at the surface of the axial hole and 25,000 psi (compression) at the outer surface. These are both longitudinal and tangential.

One test cylinder having the longitudinal axis *parallel* to the direction of extrusion cracked with the same total heat flow and equivalent heat production rate as above. All cracks were longitudinal and were observed at both inner and outer surface. Calculated stresses were as above.

A second test cylinder having the longitudinal axis *parallel* to the direction of extrusion was subjected to 100 thermal cycles at a heat flow of 7.5 KW (1800 cal/sec). However this specimen had a volume of 172 cm^3 which results in an equivalent heat production rate of only 10.5 cal/sec cm^3. This sample cracked at the inner surface *only*. Cracks were in a longitudinal direction. Calculated stresses produced at the inner and outer surface are 40,000 psi (tension) and 12,500 psi (compression), respectively. [A separate piece of the same material measured 26,500 psi (tension) in the tensile test machine.] A few cycles did not produce visible cracks, but repeating the stress a number of times did. This result seems to validate the calculated stresses, but before a definite conclusion is reached additional samples of QM beryllium will be tested.

Formulae used for stress calculations are those for tubes of infinite

length in which the heat is produced at the outer surface and removed from the surface of an axial hole.

Longitudinal and tangential stress at the surface of axial hole:

$$\sigma = [\, E\, \alpha\, Q\, a^2/8\, k\, (1-\mu)] \quad [\, 4\, P^2 \ln P - 2\, P^2 + 2\,]$$

(R. B. Briggs - memo to file, 12/1/47)

$$\sigma = [\, \Delta\, T\, \alpha\, E/2\, (1-\mu)\, \ln P\,]\, [\, 1 - \{(2\, P^2\, \ln P)/(P^2 - 1)\}\,]$$

(Roark, Formulas for Stress and Strain, p. 325)

Longitudinal and tangential stress at outer surface:

$$\sigma = [\, E\, \alpha\, Q\, a^2/8\, k\, (1 - \mu)\,] \quad [\, 4 \ln P - 2\, p^2 + 2\,]$$

(R. B. Briggs - memo to file, 12/1/47)

$$\sigma = [\, \Delta\, T\, E\, \alpha/2\, (1 - \mu)\, \ln P\,] \quad [\, 1 - \{(2 \ln P)/(P^2 - 1)\}\,]$$

(Roark, Formulas for Stress and Strain, p. 325)

Units:

σ = longitudinal and tangential stress, lbs/in.^{2}

E = Young's modulus = 40 x 10^6 lbs/in.^{2}

α = thermal coefficient of linear expansion = 13 x 10^{-6}/°C

Q = heat removed, cal/cm^3 sec

a = radius of axial hole = 0.238 cm

b = outer radius of cylinder = 1.90 cm

k = thermal conductivity = 0.4 cal/cm^2 sec °C/cm

μ = Poisson's ratio 0.03

P = b/a = 8

ΔT = temperature difference, $T_a - T_b$, °C

Beryllium—Poisson's Ratio. This investigation has been completed and a final report (ORNL 524) prepared. Baldwin Southwark "SR-4" resistance wire strain gages, in conjunction with a universal test machine, were used to meas-

ure lateral and axial strain in specimens of extruded and QM beryllium in tension and compression. For the limited number of pieces tested, the following results are reported:

1. Poisson's ratio was measured as 0.020-0.060 at stresses under 5,000 psi. Up to 12,000 psi (maximum stress used) Poisson's ratio increased to 0.010-0.13.

2. For one sample of QM tested, a value of Poisson's ratio of 0.020-0.050 in both tension and compression was obtained.

In addition to Poisson's ratio, values for Young's Modulus and shear modulus are reported.

CORROSION

Extrusion Beryllium. A full size extruded beryllium assembly which has been exposed to flowing demineralized water containing 0.005 *M* hydrogen peroxide at 90° C for 13 months was dismantled from the test apparatus for inspection. Water velocity through the test cell averaged 10-12 feet per second and the water *p*H was maintained between 5.5-6.5. The assembly was a Type 2-removable unit, machined from three 1 inch by 3 inch extruded billets and bolted together with beryllium bolts. The extrusion ratio of the beryllium was 6:1, and the material was obtained from MIT.

The first 10 months of this dynamic corrosion test were conducted in demineralized water at X-10 under the above conditions. At the end of this period, the unit showed no signs of indications of corrosion damage. The outer surfaces remained bright and clean as did the 1/8 inch diameter cooling channels extending through the core. After the relocation of the Corrosion Group to the Y-12 area, the assembly was placed back into test using demineralized water from a different system. The test was operated under the original condition for a period of three months with intermittent shut-downs.

During the last shutdown (one month prior to the end of the three month's test: a recording *p*H meter was installed into the system. At this time, daily water analyses were made to determine the extent to chloride leakage into the system from the calomel electrode used in the *p*H meter. By the time water analysis were completed, the assembly had been in test for 1 month. The analysis showed a chloride content of 8-9 ppm in the system, and the run was stopped immediately.

An inspection of the assembly disclosed prolific pitting over the surfaces, with the most concentrated area appearing on the top curved surface. The pits were shallow, seldom exceeding 8-10 mils in depth, and were completely black in color. In numerous regions, an extremely adherent black deposit extended 5-10 mils above the surrounding metal surfaces.

The reason for the sudden appearance of the localized corrosion attack was uncertain, although the effect was simultaneous with the large increase in chloride ion concentration in the water.

Conductivity changes above in the water due to the addition of very small quantities of dilute nitric acid or sodium hydroxide for *p*H adjustments may be almost disregarded as the immediate cause for the intensified corrosion attack since the method of *p*H control used at the X-10 area was identical to the system used at the Y-12 installation. No evidences of localized corrosion attack were observed on the unit exposed to X-10 water after 10 months even when the water resistance sometimes dropped as low as 80,000 ohms by the addition of the *p*H control solutions.

Frequent water analyses at the Y-12 area during the three months' exposure showed wide variations in the chloride content. Concentrations of 4.0 to 10.0 ppm were encountered. The system was drained frequently, but the chloride concentration never approached values encountered in the X-10 water. Chloride content in the latter water averaged 1.0 ppm and less.

The beryllium assembly will be placed back into test using low chloride concentration water from another source in order to determine any tendency for the pitting attack to increase or remain unchanged. The presence of chloride ions in the system from the *p*H meter will be eliminated by a rearrangement of the test apparatus.

QM Beryllium. Sixty day corrosion tests in simulated MTR cooling water were completed during this quarter. An interim report on the Results of these tests will be issued shortly. Results may be summarized as follows:

(a) All four lots of Brush QM beryllium showed corrosion resistance equivalent to good vacuum cast and extruded beryllium. Comparative weight losses were

MATERIAL	WEIGHT LOSS RANGE ($Mg/dm^2/mo$)
Vacuum cast and extruded	10-12
QM	5-12

(b) The corrosion resistance of different lots of QM beryllium varies. However, most of the samples from a single lot have comparable resistance.

Corrosion tests on Brush QRM (remelted) and QMV (vacuum melted) powder process beryllium have not been completed. Results of these tests will be issued in the near future.

Effects of Annealing and Machining on the Corrosion Resistance of Extruded Beryllium. In an attempt to evaluate the effects of annealing on the corrosion resistance of extruded beryllium, a single good extruded billet was selected and treated in the following manner:

(a) The billet was cut into three 14 inch lengths and each length then cut into three sections;

(b) One section from each length machined directly to corrosion samples;

(c) The other six sections were annealed in an argon atmosphere for one hour at 830° C and machined into corrosion samples;

(d) The samples from three of the sections treated in (c) were re-annealed under vacuum for one hour at 830° C.

Specimens from all three lengths and in all three conditions were exposed to simulated MTR cooling water, singly and in couples with 356 aluminum for 60 days. Results on these tests are not complete, but indicate that:

(a) The "as machined" beryllium exhibited the best corrosion resistance, although there was little difference between machined and the annealed-machined forms. Annealing after machining was definitely detrimental to the corrosion resistance.

(b) There was no significant variation in corrosion attack from the leading to the trailing edge of the billet.

Note Several QM beryllium samples were annealed after machining and in all cases there was an increased susceptibility to pitting type of corrosion attack.

Corrosion Properties of QM, QRM, and QM-V Beryllium. In an effort to get a quick comparison of the corrosion resistance of powder-process beryllium made by the three processes of the Brush Beryllium Company, short term autoclave tests were run. All QM beryllium metal was tested for 384 hours at 525° F and 840 psi. Both the QRM and QM-V samples were tested for 306 hours at 525° F and 840 psi.

The results of these tests are shown in Table 4-1. These figures indicate that there is very little difference in corrosion resistance under these conditions. Visual inspection of these samples shows that the pitting attack on powder process beryllium tends to spread laterally rather than deepening. QM-V may be the best form of powder process beryllium corrosion wise. Since only a limited number of samples have been tested, no definite conclusion can yet be reached.

TABLE 4-1

The Corrosion of QM, QRM, and QM-V Beryllium

in Distilled Water at 525° F for Two Weeks

SAMPLE NUMBER	WEIGHT LOSS (MDM)*	PIT COUNT (per sq cm)	MAXIMUM PIT DEPTH (mm)
1	16.6	0.0	0.0
2	10.5	4.3	0.165
3	144.5	2.8	0.030
4	1.0	10.6	0.110
5	114.3	17.1	0.340
6	18.1	4.9	0.194
6a	2.3	3.4	0.072
6b	2.3	2.9	0.064
7	0.6	1.5	0.043
7a	0.0	1.6	0.047
7b	5.0	2.0	0.129

* MDM - Milligrams/square decimeter/month

Indentification

(1) QM Beryllium, 1949, Billet Y4382
(2) QM Beryllium, 1949, Billet Y4373
(3) QM Beryllium, 1949, Lot 2 CF 49-4-144
(4) QM Beryllium, 1949, Lot 3 CF 49-4-144
(5) QM Beryllium, 1949, Lot 4 CF 49-4-144
(6 ab) QRM Beryllium, 1949, Billet Y-4411 13P
(7 ab) QM-V Beryllium, 1949, Billet Y-4409 AP

Proposed Beryllium Corrosion Testing Program. In order to evaluate more clearly the present status of the corrosion resistance of extruded and QM types of beryllium, a concentrated program will be started immediately encompassing the following three problems:

(1) Effect of water composition variations with subsequent determination of minimum water specifications for the Materials Testing Reactor;

(2) The effect of chemical and metallurgical variations on the corrosion resistance;

(3) Recommendations for a corrosion acceptance test for beryllium to be used in the Materials Testing Reactor.

Present indications are that approximately 85-90 percent of the beryllium to be used for the reflector will be of a QM type; the remaining 10-15 percent beryllium will be an extruded form. The effort on the proposed corrosion testing program will be proportioned accordingly.

For Part 1 of this program, samples will be obtained which will be representative of the material to be used in the Materials Testing Reactor. For Part 2, an attempt will be made to select material with varying chemical ranges in order to correlate variations in chemical composition with corrosion resistance. Part 3 of the program would entail recommendations for a precorrosion test on the beryllium to determine whether or not the corrosion resistance was satisfactory to permit usage in the reactor. The conditions for such a test may involve exposure in normal operating conditions expected in the MTR, or it may be a form of accelerated corrosion test.

The types of material to be included in this program are extruded beryllium and QM, QRM, and QMV types of powder processed beryllium. Samples will be obtained through the Metallurgy Division.

INDUCED RADIATION IN GRAPHITE

Pile induced activity is being determined for CS and GBF graphite. Figures 4-10 and 4-11 show the gamma intensity of these materials up to 900 hours decay for 2 weeks and 4 weeks irradiation time. Other samples are being irradiated for two months and six months.

Similar tests have been conducted at Hanford and Argonne (see HW-12780 and ANL-JHR-18), but those were carried out with reactor irradiation times of thirty minutes or less and hence were only used for the detection of short

Dwg. 8126

[redacted]

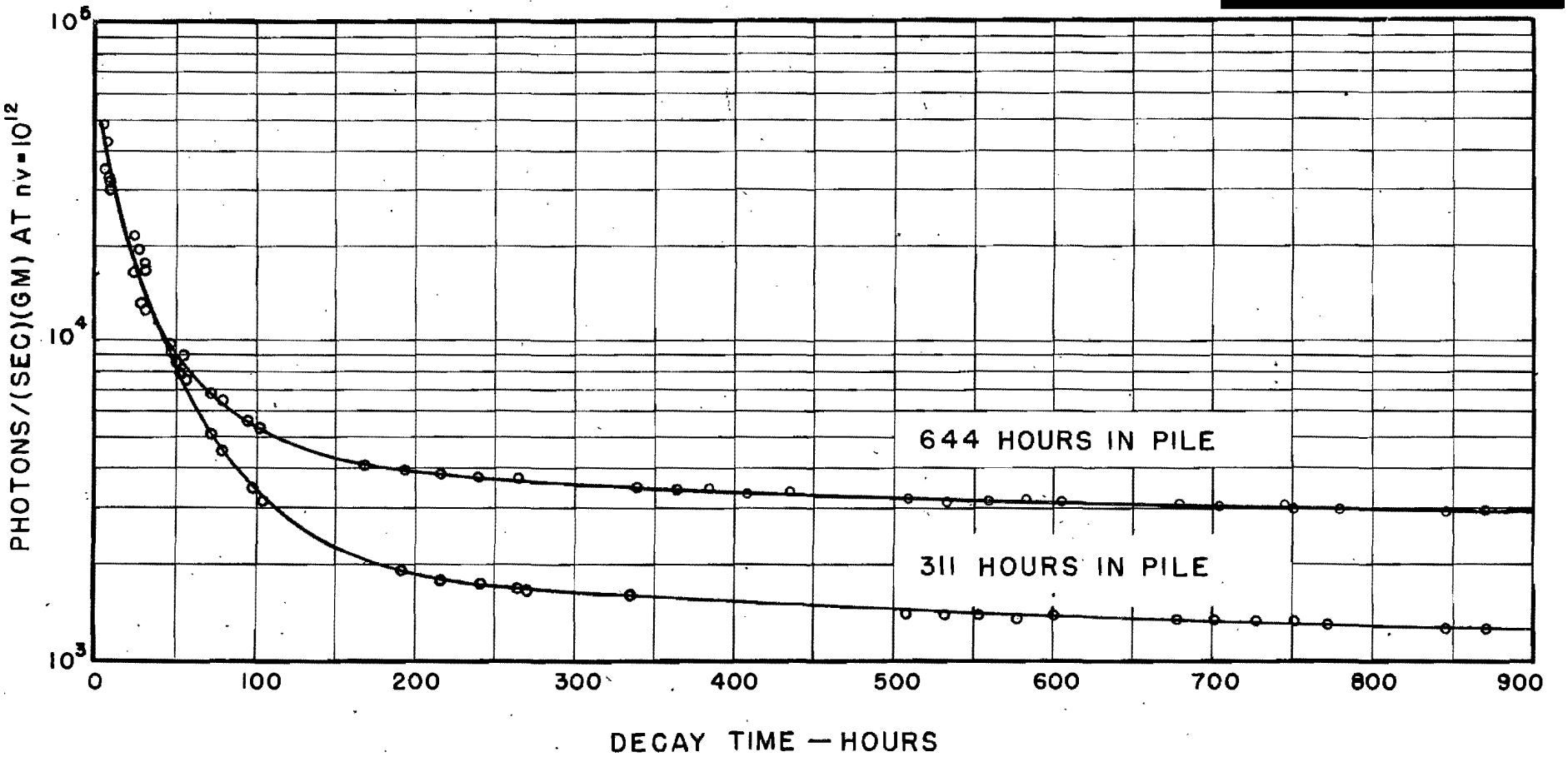


FIGURE 4-10 GAMMA RADIATION FROM C S GRAPHITE

Dwg. 8127 [redacted]

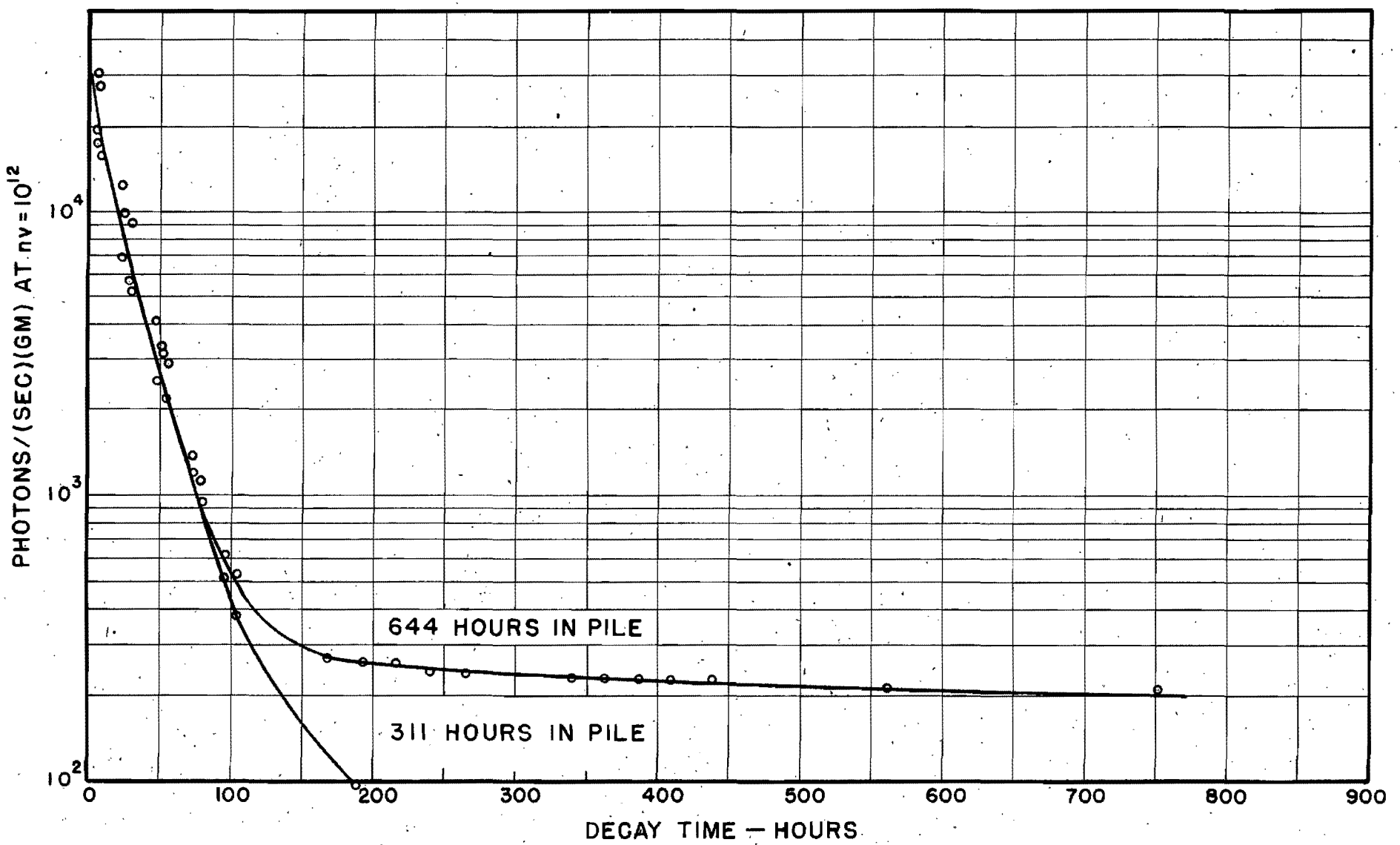


FIGURE 4-11 GAMMA RADIATION FROM G BF GRAPHITE

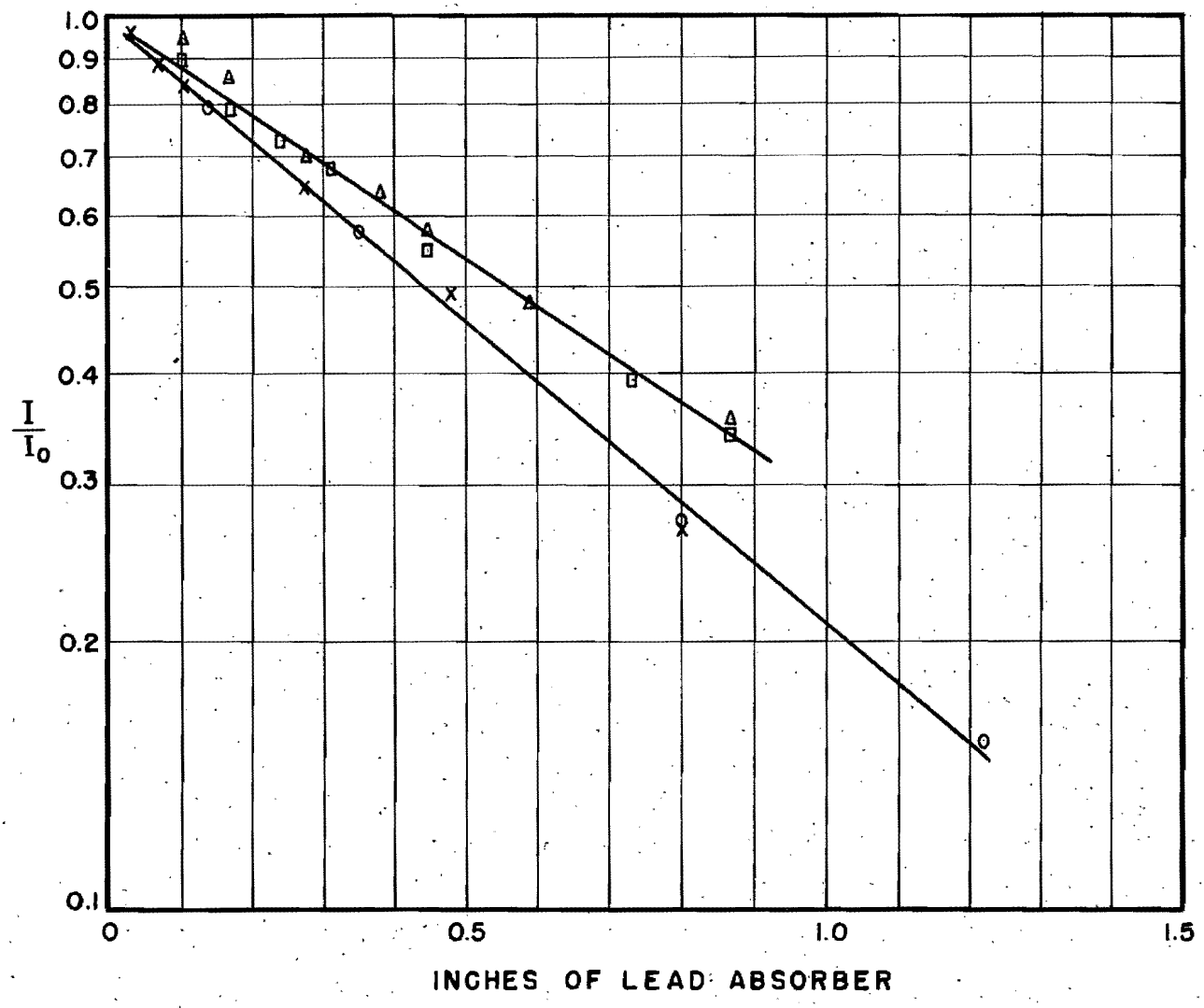
lived induced activity. The program at ORNL has been used to study the long lived induced activity.

The gamma decay curves were taken with an ionization chamber which was calibrated with Co^{60} standards (average gamma energy of 1.2 Mev). Analysis of the curves show three components for both materials. These are shown in Table 4-2 along with a spectrographic analysis.

Figure 4-12 shows the absorption curves for lead for the four week irradiation period. These measurements were made by interposing sheets of lead between the sample and a mica window G-M chamber. Most of the 15 hour component (Table 4-2) had decayed when these measurements were made, which would seem to indicate that the 33 hour component must have a considerably higher energy than the 50 hour component since the curve for the longer decay time has a greater slope.

Dwg. 8128

[redacted]



SYMBOL	TYPE OF GRAPHITE	TIME IN PILE	DECAY TIME
O	G B F	644 HRS	10 DAYS
X	C S	644 HRS	46 DAYS
Δ	G B F	644 HRS	1 DAY
□	C S	644 HRS	1 DAY

FIGURE 4-12 ABSORPTION OF GAMMA RADIATION FROM GBF & CS GRAPHITE

TABLE 4-2

Analysis of GS and GBF Graphite Decay Curves

HALF LIFE FOUND	ELEMENT SUSPECTED
15 hours	12.8 hr. Cu^{64}
33 hours	(34 hr. Br^{82}, 40 hr. La^{140}, 27 hr As^{76})?
50 days	47 d Fe^{59}
>100 days	?

Spectrographic Analysis

ELEMENT	CS GRAPHITE	GBF GRAPHITE
C	very strong	very strong
Ca	trace	very faint trace
Cu	very faint trace	very faint trace
Fe	faint trace	------
Mg	very faint trace	very faint trace
Si	faint trace	faint trace
V	very faint trace	------

METALLURGICAL ENGINEERING DEVELOPMENT

FUEL ASSEMBLIES

Reduction. A major problem in the production of fuel assembly results from the formation of blisters on the surfaces of the 2S aluminum-clad U-Al alloy plates when the plates are heated to 1100° F during the processing cycle. Descriptions of the active plates and assemblies, and the methods for producing them have been described in earlier progress reports of the Metallurgy and Technical Divisions.

Previously reported work (ORNL 407) indicated that the use of a dried air atmosphere in the furnace used to preheat the alloy for rolling and cladding, decreased blister frequency, but subsequent work indicates that this alone will not guarantee freedom from blistering. As a consequence, additional test heats have been made to study the effects of other variables. These are reported in more detailed form in ORNL 511. The variables included the atmospheres used in melting and casting and the moisture content of the cryolite flux. Results indicate the desirability of melting the cryolite and U_3O_8 to drive off water before adding aluminum in the reduction heat, and of vacuum remelting and casting. However, the vacuum remelting and casting operation is time consuming and makes it difficult to sample the melt and to avoid segregation. Optimum conditions were obtained in heats in which the aluminum was added to a mixture of molten cryolite and U_3O_8 followed by vacuum remelting and air casting of the reduced alloy.

Inspection. An inspection procedure has been established to center the U-Al cores in the clad plates. This insures the presence of a sufficient margin of aluminum (at least 0.160 in.) along each edge of the plate to minimize changes of cutting into the core when machining the plates to final dimensions. This location of cores is accomplished with fluoroscopic X-ray examination and steel gages which are shown in detail in ORNL 511.

Rolling and Cladding. To study the effects of core centering in the frame on behavior in rolling and cladding, ten plates were made up with the cores deliberately placed off center by one-half inch. After the initial rolling operation, large voids appeared at the corners of these cores nearest the frame edge, and yields of blister-free plates were significantly lower than in plates with properly centered cores.

Shearing tests have been made in an effort to eliminate machining of clad plates to size. Cold rolled aluminum can be sheared perpendicular to the rolling direction with a variation from straightness of only 0.003 in. per 24 in. length. When clad plates are sheared, the edges curve and variations average 0.010 in. per 24 in. with cold rolled plates, and 0.006 in. with annealed plates.

Critical Experiment Requirements. Twenty fuel assemblies containing enriched uranium are being made for the critical experiment in the MTR Mock-Up. Specifications for active plate thickness have been changed from 0.064 in. ± 0.001 in. to 0.060 in. ± 0.001 in.

Brazing. Inclusions which may result from excessive or improper use of fluxes in the brazing of fuel assemblies present possibilities of corrosion damage and of lithium contamination of the assemblies. Brazing tests have been made to determine the optimum conditions which minimize these flux inclusions. Inclusions are readily observed under 20 X magnification on a machined cross section of a brazed joint after exposure to the air for a time, since the flux is hygroscopic and causes droplets of moisture to collect at the points of flux concentration.

A direct comparison of the two proprietary fluxes, Eutectic 190 and Alcoa 33, has been made by brazing opposite sides of the same assembly. Examination of brazed joint sections showed inclusions in both cases, with some possible slight advantage in favor of the Alcoa flux.

Earlier cleaning practice using etchant type cleaners—acid followed by caustic—have been replaced by vapor degreasing only, because excessive blistering occurred when the etchant cleaners were used.

Brazed joints were made using the two different cleaning techniques on opposite sides of the same assembly to compare possible effects on flux inclusions. Both sides were well brazed, but somewhat fewer flux inclusions occurred when etchant type cleaners were used.

Efforts to formulate fluxes with no lithium were not successful.

Fuel Assembly Uniformity. Although within tolerance, sizeable dimensional variations occur in the machined side plates. By careful inspection and sorting of components and pairing those which most nearly match each other, a greater dimensional uniformity in final assemblies has been obtained. This is described in greater detail in ORNL 511.

Dimensional Changes after Thermal Cycling. The Metallurgy and Technical Divisions have cooperated in the preparation and testing of an assembly by

thermal cycling. Results have been reported in ORNL 473.

BERYLLIUM

Laboratory Melting and Casting. The casting of 4¼ in. diameter ingots of beryllium in the 12 in. diameter induction vacuum furnace has been continued, a total of 30 castings having been made. Previous conclusions about the desirability of casting the billets in an inert atmosphere rather than in a vacuum have been substantiated. The use of argon or helium at a pressure of 11 psi during casting has eliminated the blowing of the liquid metal from the mold. Apparently this blowing from the mold is the result of vaporization of volatile condensate on the mold walls. Such condensate has been observed to oxidize with explosive violence when a furnace in which a melt has been frozen directly in the crucible is opened to the atmosphere.

Attempts were made to refine the grain size of beryllium ingots by additions of Ti and Zr. Alloys containing 0.1% to 1% Ti and 0.1% to 1% Zr were prepared according to standard Be melting practice, and cast into cold graphite molds of 4½ in. diameter. The results are not definite, but it appears that 1% Ti may have been effective as a grain refiner. More work will be necessary before this can be considered conclusive. Analyses and results are tabulated in ORNL 511.

Extrusion. During this period two beryllium extrusion runs were made at the Gerity Michigan Company plant in Adrian, Michigan, and one was made at ORNL.

(1) Extrusion at Adrian. Twelve billets were extruded in the first run. Seven were reported as being sound after having been radiographed in six positions instead of the usual three, while five billets contained internal cracks. Two of the sound billets contained small additions of aluminum, 0.5% in one case, and 1% in the other. All billets were heated to 1800° F and extruded with a ram speed of less than one-half inch per second. The general method of extrusion was that described in ORNL 407.

All extruded bars were cut into six inch lengths to permit investigation of internal cracks. After being chemically polished and etched, all the bars which did not contain aluminum additions were found to contain internal cracks. The sixth section, 36 inches from the leading end, was the first section that was consistently sound. The bars that came from the reportedly good billets

contained fewer and smaller cracks than those from unsound billets, but all showed cracks. No cracks were found, by either fracture or chemical etching, in any section from the extrusions containing aluminum.

During the second extrusion run, billets containing 0.1%, 0.25%, and 0.50% aluminum, respectively, and one plain billet, were extruded. The extrusion conditions were identical with those of the previous run.

The bars were sectioned at the 6, 12, and 56 inch marks and the sections examined. None of the bars contained internal cracks. One did contain fine lines caused by contamination by a non-metallic impurity (beryllium carbide), but there were no cracks.

Corrosion and physical tests are being run on the entire series of aluminum-containing bars. These tests include both material annealed at 1800° F and as-extruded material.

The Instrument Department at ORNL designed and built an instrument for recording instantaneous values of speed and pressure for any position of the ram during the working strokes. The instrument was installed at Adrian and worked very well during these extrusions. The records showed that the operator was holding both the pressure and speed much more uniform than was thought probable.

Of twenty-five billets extruded into 3¾ round-corner squares using the technique mentioned above, only one contained a serious defect that could be traced to the extrusion. This indicates that a technique has been developed that is suitable for extruding the three-inch squares necessary for the MTR.

The major problem now is the procurement of sound eight-inch billets. The status of the problem of casting these billets is not definite. This work is being performed at another site.

(2) Extrusion at ORNL. Since the large 8 in. diameter billets are both expensive and difficult to make, some work was scheduled at ORNL, using 3 in. billets and the same ratio of reduction as employed at Adrian. The major difference in technique between the two operations is that in Oak Ridge a one-piece die is used which eliminates the need for a cone on every billet. The radius on the corners of the die was 3/16 in. instead of the 3/8 in. used on the large press dies. Since the supply of cast billets was limited, some billets were made by machining a previously extruded bar.

The results show that copper cans require a lower extrusion pressure than do copper plated steel cans. At the lower temperatures the copper seemed to deform more easily than the beryllium, since in many cases the bar was

cracked, but the jacket had not ruptured. Within this temperature range beryllium has a zone of maximum plasticity at about 850° F. To take full advantage of both this zone and the plastic properties of the copper jacket, extrusions in the vicinity of this temperature will be attempted.

The corners of the bars were well filled under all conditions. This indicates that the radius of the corners of the large squares can be safely reduced to lessen machining losses.

Testing. During the past quarter, comparative grain size, mechanical strength, and corrosion data were obtained on the metal types actually considered for use in construction of the Materials Testing Reactor reflector. Interest has centered on Adrian extruded (4.1 to 1 reduction) and QM and QRM grades of hot pressed metal.

Review of beryllium corrosion work to date indicates that hot-pressed metal is at least as corrosion resistant in simulated reactor cooling water as is extruded metal, and emphasizes the importance of completing corrosion tests to determine the relative resistance of hot-pressed metals, grades QM, QRM and QMV, and of extrusions containing some aluminum, in simulated reactor cooling water and in autoclave testing.

(1) Grain Structure Comparison. Typical grain structures of hot-pressed and of extruded metal were examined metallographically. Transverse and longitudinal specimens were prepared from the center and edge of an Adrian-extruded billet (extrusion temperature, 1700° F). Grain refinement resulting from one-hour anneals at 1700, 1800, and 1900° F, was also observed. A refined, equiaxed structure resulted in each case. The grains from the 1900° F anneal were slightly larger than those obtained at the two lower temperatures.

(2) Compressive Tests. Room temperature compressive tests were made to obtain information on the effects of extrusion temperature on strength, the degree of anistropy found in extruded metal, and the relative mechanical strength of hot-pressed and extruded metal. Compressive testing was selected in preference to tensile testing because of the large number of samples involved and the greater conveniences of fabrication.

The compressive strength appears to increase in the transverse direction and decrease in the longitudinal direction with an increase in extrusion temperature within the range tested. As expected from examination of the grain structures and from work previously reported by MIT, the extruded sections tested show a high degree of anistropy. Differences of properties were also evident between center and edge samples. These differences were not observed

in the sintered metal. Extruded metal is substantially weaker in compression in the transverse direction, but generally stronger in the longitudinal direction, than in hot-pressed metal. The results are tabulated quantitatively in ORNL 511.

(3) Corrosion. At the request of the Laboratory, Battelle Memorial Institute conducted distilled-water corrosion tests on beryllium metal at 600° F (1500 psi) and 525° F (800 psi). Samples were selected from vacuum cast metal extruded at 1400, 1600, 1700 and 1800° F, and from hot-pressed metal, grades QM and QRM.

Results showed that all samples, regardless of condition (as-machined, annealed, or chemically polished for removal of cold worked metal), were resistant to 525° F water. The hot-pressed beryllium was generally superior to the extruded metal in its resistance to 600° F water.

Deep Hole Drilling. Activity on the problem of drilling long, narrow holes in beryllium metal has increased as a result of the successful extrusion work at Adrian. Stock of the desired length and cross section is now available for fabrication into full-sized reflector units. Success in drilling water cooling channels will permit greater flexibility of design and reduce cost of fabrication of the MTR reflector.

Arrangements were made with the High Standard Manufacturing Company of Hamden, Connecticut, and the Conner Tool and Cutter Company of Detroit, Michigan, to test drill some beryllium metal in their shops. All this work was observed by Mr. P. Kofmehl of the Research Shops to insure proper safety precautions in the drilling of the beryllium.

(1) Work at Conner. A series of 3/16 in. diameter holes was drilled to determine optimum conditions for deep hole drilling of extruded metal. All tests were conducted on a standard deep hole machine using the Conner-fabricated tungsten carbide (Carboloy 883) tipped tools. Speeds and feeds were varied from 800 to 1600 rpm and between 0.24 and 1.40 in. drill depth per minute, respectively. Rotation of both drill and work and of work alone were tried. Optimum conditions for drilling with this tool were:

Work only turning

Spindle speed - 1400 rpm

Feed - 0.006 in. per spindle revolution or 0.840 in. depth per minute

Coolant - 50% paraffin and 50% sulphur-base oil

Coolant pressure - 500 psi

Evaluation of the tests showed that prolonged tool life, greater speeds,

and heavier feeds are possible with the Conner carbide tipped tool. The amount of hole run-out was greater than that found in similar tests conducted with high speed steel tools at Oak Ridge, but additional development work should improve this factor.

(2) Work at High Standard. A similar set of tests was conducted to determine the applicability of hardened high speed steel tools for drilling beryllium. In all test cases, the drill remained stationary while the work revolved. Best conditions determined for use of the HSS tool are:

Spindle speed - 100 rpm

Feed - 0.00028 in. per spindle revolution or 0.280 in. depth per minute

Coolant - "Cutmax"

Coolant Pressure - 700 psi

High speed steel tools require more resharpening than do the carbide tools. One important design feature noted is that the HSS tool bit is considerably longer (2 in. as against 5/8 in.) than the carbide bit, and this is believed to be conducive to greater accuracy in drilling.

(3) Work at ORNL. Lack of adequate deep hole drilling equipment at the Laboratory necessitated outfitting a conventional lathe with an external prime mover to power the drill, a high pressure coolant pump, and special fixtures for holding and rotating the long stock. Drill feeding is accomplished through use of the lathe carriage on which a constant speed motor and drill holding fixtures are mounted.

This improvised set-up is satisfactory except that the present lathe bed is too short and requires drill changes every 8 in. as the hole depth increases. Alterations on the lathe to increase its length to 96 in. are in progress.

The majority of tool failures are a result of motor slow-down under load, which in turn greatly increases the rate of feed. This had been remedied by use of a constant speed motor.

Data for drilling deep holes in beryllium with both tool and work rotating are as follows:

Spindle speed - 146 rpm (counter-clockwise)

Drill speed - 1340 rpm (clockwise)

Coolant — sulphurized oil with 1-2% by volume "oil Dag"

Coolant pressure - 22 psi

Commerical "Oil Dag" (colloidal suspension of graphite in oil) was added to sulphurized oil to improve the lubricating properties of the coolant.

NEUTRON WINDOW FOR MATERIALS TESTING REACTOR

Lead is being considered as a possible material for fabrication into a neutron window for the MTR. No information is available on the creep of lead or lead alloys at 30 psi and 150° C, the probable operating conditions. Tests are in progress to determine this value. Results are not yet available.

CADMIUM SECTION OF MTR SHIM ROD

To obtain an aluminum clad sandwich containing cadmium, a 0.020 in. sheet of cadmium was covered with a layer of 0.010 in. zinc foil and an outer cover of aluminum. Hot rolling at 250° C was tried with reductions of 50% and greater. The zinc layer was intended to promote bonding of the aluminum to the cadmium, but destructive tests showed that the bond was not satisfactory. Further tests are planned.

CHEMICAL PROCESSING PLANTS FOR THE MATERIALS TESTING REACTOR

Chemical and metallurgical reprocessing and reactor component fabrication facilities for the Materials Testing Reactor have been receiving renewed attention during the past three months. It will be necessary to provide fuel and breeder reprocessing plants, metallurgical reprocessing facilities, liquid and gaseous waste disposal facilities, and normal services for all reactors to be erected at the Idaho reactor test site. During the months of October and November two meetings were held to discuss chemical and metallurgical processing for reactors currently being considered for Idaho at which representatives from all groups now engaged in reactor component chemical process development were present. A joint advisory committee, composed of representatives from Oak Ridge National Laboratory, Argonne National Laboratory, Knolls Atomic Power Laboratory, Atomic Energy Commission, and headed by W. Schumacher of Standard Oil of New Jersey, has been established to review and make recommendations concerning the chemical and metallurgical plants for the reactors to be built at the Idaho site.

Oak Ridge National Laboratory has been engaged in the development of chemical and metallurgical process development for fuel and breeder materials in the MTR for approximately four years. Chemical processes for the separation of depleted U^{235} from aluminum clad uranium-aluminum alloy fuel elements, and the separation of U^{233} from irradiated thorium have been studied on laboratory, semi-works, and pilot plant scale.

STATUS OF MTR FUEL ELEMENT PROCESS DEVELOPMENT (25 PROCESS)

The countercurrent extraction of uranium from nitric acid dissolved fuel assemblies with hexone in an acid difficient system, using the fuel element aluminum as a salting agent, has been demonstrated on a pilot plant scale at approximately 10% of the activity level of the MTR fuel assemblies. Runs using 93.5% enriched uranium are now in progress in the ORNL pilot plant and will be complete early in December, 1949. Losses of less than 0.1% uranium have been demonstrated for two cycles of column extraction plus a laboratory scale clean-up cycle. Satisfactory decontamination factors for the separation

of uranium from fission products and plutonium have been demostrated,

viz: Overall 2 cycle Gross βDF = 3×10^6

Overall 2 cycle Gross γDF = 1.4×10^6

Overall 2 cycle Puα DF = 1.5×10^3

This decontamination, plus that obtained in a laboratory clean-up extraction cycle, are sufficient to reduce 12 day irradiated (at 9.15 kw per gram U^{235}), 120 day cooled fuel to background for U^{235} + U^{237}.

The preliminary design for the fuel plant for the MTR must be revised to take into account process changes and equipment development resulting from two years of development work. The coordination of the design of the MTR fuel plant with fuel plants for other reactors to be constructed at Idaho must be accomplished.

STATUS OF U^{233}—THORIUM PROCESS DEVELOPMENT

The irradiation of thorium in the MTR for the production of U^{233} by the successive capture of two neutrons has been considered. Present thinking is that no thorium will be placed in the MTR at start-up; the thorium production rods have been removed from the reflector zone and replaced with beryllium sections to permit an approximate increase of 16% in thermal neutron flux in the beryllium and graphite zone for irradiation experiments. Thorium sections in the shim safety rods have been replaced with cadmium. The removal of thorium production rods is a result of a change in philosophy concerning the use of the reactor primarily for irradiation of various materials at a high neutron flux. Provisions will be made to replace beryllium reflector pieces with thorium rods for intermittent U^{233} breeding experiments.

A countercurrent solvent extraction process for the separation of U^{233} from thorium and fission products has been demonstrated on a laboratory and a countercurrent batch scale using either diisopropyl ether or dibutyl carbitol as the extractant. A pilot plant for a single cycle of countercurrent column extraction has been built and is now starting operation to demonstrate the U^{233} recovery process. Hexone has been substituted for diisopropyl ether and dibutyl carbitol, because it is less volatile (hence less danger from explosion, fire, etc.) than diisopropyl ether, and because it has a lower HETS than

either DIP or DBC, easier to use in column operation than dibutyl carbitol. Experience gained with the hexone system in 25 and Redox development plus laboratory countercurrent batch runs indicated that the demonstration of the 23 process will be easily accomplished. Pilot plant runs on 740 kg of thorium, irradiated in the Hanford piles for ninety days and cooled from nine months to two years, will be completed by March, 1950.

METALLURGICAL FABRICATION OF MTR ASSEMBLIES

Metallurgical alloying, cladding, and assembly of uranium fuel assemblies has been satisfactorily accomplished. Actual fuel assemblies for the MTR are now being fabricated.

SUMMARY OF CHEMICAL-METALLURGICAL PROCESSING MEETING DISCUSSIONS

A meeting to discuss chemical-metallurgical processing plants for the Idaho reactor site were held at Argonne National Laboratory on Oct. 11, 1949, at which representatives from all areas concerned with chemical process development were present. The important considerations and conclusions of this meeting were reported in ANL-SM-50 by Dr. Stuart McLain and in ORNL CF-49-11-140 by F. L. Culler. The following points have been extracted from these documents to present the more important decisions:

1. The reactors which are now being considered for construction in Idaho, plus other reactors whose fuel and breeder materials may be processed there, were listed and discussed. The following table summarizes the limited data presented.

REACTOR	FUEL (yearly)	BREEDER (yearly)	LOCATION
CP4	40 kg 25 (E) Stainless or Zr Alt. 40 kg Pu Be Reflector	4 tons natural U Stainless Clad	Idaho
MTR	70 kg 25(E) Aluminum Alloy Be Reflector	2 tons Th 3 kg 23 Aluminum	Idaho

REACTOR (cont'd)	FUEL (yearly) (cont'd)	BREEDER (yearly) (cont'd)	LOCATION (cont'd)
Navy Phototype	100 kg 25(E) Stainless or Zr Be Reflector	None	Idaho
Knolls	40 kg 25(E) Stainless or Zr Be Reflector	12 tons natural U Stainless Steel	West Milton
CP5	4 kg 25 or equivalent Natural U		Chicago

Other reactors to consider are as follows:

REACTOR	FUEL (yearly)	BREEDER (yearly)	LOCATION
University Research Reactor	10 kg 25(E)	?	?
ORNL Homogeneous	25 kg 25(E)	?	Oak Ridge
Brookhaven	Natural U (?)	?	?

It can be seen from the foregoing table that facilities for chemical and metallurgical processing of the following materials may be required at the Idaho reactor site:

a. U-235 (93.5% down to possible 60% quality)
b. Natural Uranium
c. Plutonium
d. U-233 (and Pa)
e. Thorium
f. Beryllium
g. Zirconium (possibly)
h. Neptunium (of experimental interest, from 25 plant).

There is also a possibility that other chemical separations plants for such materials as Ba-140 and RW irradiation materials may be required at the site because of availability of large neutron fluxes in such units as the MTR.

2. The group favored a combined chemical-metallurgical processing facility for all reactors proposed for the Idaho site, but recommended that this

proposition be studied further by an advisory group familiar with more of the detailed requirements for each reactor.

3. The appointment of an advisory committee for chemical processing was recommended.

4. The design of the MTR and its service facilities was to proceed without consideration of chemical and metallurgical plant requirements.

As a result of the Chicago meeting, an advisory committee, was appointed and a meeting held in Boston on Nov. 1, 1949. The committee, in order to establish the general scope of chemical and metallurgical proceeding, has prepared a survey questionaire to be circulated for answer to various sites where reactor and chemical process development is being carried out. The committee will serve in an advisory capacity only.